FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLA
UNIT NO.2

UPDATED

FINAL SA
ANALYSIS R

VOLUME 6

TWO DIGIT TABLE OF CONTENTS

TWO DIGIT TABLE OF CONTENTS (Cont'd)

TWO DIGIT TABLE OF CONTENTS (Cont'd)

TWO DIGIT TABLE OF CONTENTS (Cont'd)

TWO DIGIT TABLE OF CONTENTS (Cont'd)

TWO DIGIT TABLE OF CONTENTS (Cont'd)

4.4 THERMAL AND HYDRAULIC DESIGN

This section presents the steady-state thermal and hydraulic analysis of the reactor core, the analytical methods, and the experimental work done to support the analytical techniques. Discussions of the analyses of anticipated operational occurrences and accidents are presented in Chapter 15. The prime objective of the thermal and hydraulic design of the reactor is to ensure that the core can meet steady-state and transient performance requirements without violating the design bases.

The St. Lucie Unit 2 UFSAR has been revised to incorporate the Bounding Cycle introduced in the Cycle 12 reload information. Historical information pertaining to Cycles 1 and Cycle 2 has been retained. Bounding Cycle modifications to the Cycles 1 and 2 Design have been incorporated where required.

The St. Lucie Unit 2 UFSAR has been revised to incorporate the reload methodology described in WCAP-9272-P-A/WCAP-9273-NP-A (Reference 41) in support of the 30% steam generator tube plugging (SGTP). The WCAP-9272 bounding cycle information has been incorporated into the appropriate sections.

4.4.1 DESIGN BASES

Avoidance of thermally or hydraulically induced fuel damage during normal steady-state operation and during anticipated operational occurrences is the principal thermal hydraulic design bases. The design bases for accidents are specified in Chapter 15. In order to satisfy the design bases for steady-state operation and anticipated operational occurrences, the following design limits are established, but violation of these will not necessarily result in fuel damage. The Reactor Protective System (RPS) provides for automatic reactor trip or other corrective action before these design limits are violated.

The St. Lucie Unit 2 FSAR has been revised to incorporate the Extended Power Uprate (EPU) analyses. The EPU information has been incorporated into the appropriate sections and historical information has been retained. The EPU uses the same codes and methodology as the WCAP-9272 bounding analysis unless otherwise noted.

4.4.1.1 DNBR Analysis

Steady state DNBR analyses of Cycle 2 at the Stretch power level of 2700 Mwt have been performed using the TORC computer code described in Reference 6, the CE-1 Critical Heat Flux (CHF) correlation described in Reference 1, the simplified modeling methods described in Reference 10, and the CETOP code described in Reference 26.

A variation of TORC called CETOP, optimized for simplified modeling applications, was used in Cycle 2 to develop the "design thermal margin model" described generically in Reference 10. Details of CETOP are discussed in Reference 26, CETOP is used only because it reduces computer costs significantly; no margin gain is expected or credited.

Steady state DNBR analyses of the bounding cycle design at the rated power level of 2700 Mwt have been performed using the TORC computer code, the CE-1 CHF correlation, the simplified TORC modeling methods, and the CETOP code described respectively in References 1, 26, 33, 34 and 38.

Table 4.4-11 presents pertinent thermal-hydraulic design parameters for the current cycle. The reference cycle design is based on a core with all GUARDIAN^{TM} fuel which was previously introduced in Region N. Thermal hydraulic design for transition cycles is discussed in Section 4.4.2.2.2. For the reference cycle, the Extended Statistical Combination of Uncertainties (ESCU) methodology presented in Reference 35 was applied with St. Lucie Unit 2 specific data using the calculational factors listed in Table 4.4-11 and other uncertainty factors at the 95/95 confidence/probability level to define a design limit of 1.28 on the CE-1 minimum DNBR.

The bounding cycle DNBR limit includes the following allowances:

1. NRC specified allowances for TORC code uncertainty and CE-1 CHF correlation cross validation uncertainty as discussed in Reference 36.

2. NRC specified allowances for uncertainty associated with the CE-1 CHF correlation applied to CE HID grid fuel, discussed in Reference 39.

3. Rod bow penalty as discussed in Section 4.4.4.1.

Table 4.4-11 also presents pertinent thermal-hydraulic design parameters for the WCAP-9272 bounding cycle introduced in Cycle 15. The bounding cycle design is based on a core with all GUARDIAN fuel that was previously introduced in Region N. The DNBR calculations are performed using the ABB-NV DNB correlation (References 42 and 43), Westinghouse version of the VIPRE-01 (VIPRE) code (Reference 44 and 45), and the Revised Thermal Design Procedure (RTDP) methodology Reference 46. RTDP defines a Design Limit DNBR based on St. Lucie 2 specific data using the parameters listed in Table 4.4-11 and other uncertainty factors, including the ABB-NV correlation uncertainty, at a 95 probability with a 95% confidence level (95/95).

Table 4.4-11 also presents pertinent thermal-hydraulic design parameters for the EPU. The EPU design is based on the same fuel, codes and methodology as the WCAP-9272 bounding cycle analyses.

Bounding cycle assumptions will be confirmed on a cycle by cycle basis.

4.4.1.1.1 Minimum Departure from Nucleate Boiling Ratio (DNBR)

The minimum DNBR is such as to provide at least a 95 percent probability with 95 percent confidence that departure from nucleate boiling (DNB) does not occur on a fuel rod having that minimum DNBR during steady-state operation and anticipated operational occurrences. A value of 1.13, using the CE-1 correlation coupled with the TORC code, provides at least this probability and confidence. Based on review of CENPD-162(1), the NRC requires use of a minimum DNBR of 1.19. Further adjustments to the 1.19 design DNBR limit are required by the NRC to account for the possibility of the fuel rod bowing (Reference 20) and increased grid spacing, which results in an increase in the DNBR limit from 1.19 to 1.20. These additional penalties are applied to the value of F_r^T. DNBR limit for Cycle 2 was 1.28. For the Bounding Cycle Analyses introduced in Cycle 12, rod bowing is discussed in Section 4.4.4.1. For cycles prior to Cycle 15, ESCU was used and the DNBR limit was 1.28.

The 95/95 ABB-NV correlation DNBR limit is 1.13 coupled with either the TORC code (Reference 42) or the VIPRE code (Reference 43). For the WCAP-9272 bounding cycle analysis, the maximum RTDP Design Limit is 1.30. For the EPU analysis, the maximum RTDP Design Limit is 1.32. The rod bow penalty discussed in Section 4.4.4.1 remains applicable to the minimum DNBR predicted using the ABB NV correlation (Reference 43).

4.4.1.2 <u>Hydraulic Stability</u>

Operating conditions do not lead to flow instability during steady-state operation or anticipated operational occurrences.

4.4.1.3 <u>Fuel Design Bases</u>

a) The peak temperature of the fuel is less than the melting point (5080 F) unirradiated and reduced by 58 F per 10,000 MWD/MTU during steady-state operation and anticipated operational occurrences.

b) The fuel design bases for fuel clad integrity and fuel assembly integrity are given in Subsection 4.2.1. Thermal and hydraulic parameters that influence the fuel integrity include maximum linear heat rate, core coolant velocity, coolant temperature, clad temperature, fuel-to-clad gap conductance, fuel burnup, and UO_2 temperature. Other than the design limits already specified, no limits need be applied to these parameters directly. Conformance with the design limits specified here and conformance with the design bases specified in Subsection 4.2.1, are sufficient to ensure fuel clad integrity, fuel assembly integrity, and the avoidance of thermally or hydraulically induced fuel damage for steady-state operation and anticipated operational occurrences.

4.4.1.4 Coolant Flow, Velocity, and Void Fraction

The reactor coolant flow with all four pumps in operation is neither less than the minimum allowable nor greater than the maximum allowable. These two values for Cycle 1 are shown in Table 4.4-1

The core flow used in the thermal margin analyses is determined by subtracting the maximum core bypass flow from the minimum allowable reactor coolant flow. This is the flow rate that the four reactor coolant pumps will produce at a 97.7 percent probability. A more detailed definition of minimum allowable flow is given in Subsection 4.4.4.5.1.

The core bypass flow is the percentage of the flow entering the reactor vessel which is not effective for cooling the core.

Design of the reactor internals ensures that the coolant flow to the core such that the core is adequately cooled during steady-state operation and anticipated operational occurrences. Therefore, no specific orificing configuration is used.

Although the coolant velocity, its distribution, and the coolant voids affect the thermal margin, design limits need not be applied to these parameters because they are not in themselves limiting. These parameters are included in the thermal margin analyses and thus affect the thermal margin to the design limits.

4.4.2 DESCRIPTION OF THERMAL AND HYDRAULIC DESIGN OF THE REACTOR CORE

4.4.2.1 Summary Comparison

The thermal and hydraulic parameters for the St. Lucie Unit 2 reactor, Cycle 1, are compared to those for St. Lucie Unit 1 in Table 4.4-1.

The principal reason for the differences is that the St. Lucie Unit 2 reactor uses 16 x 16 fuel assemblies and the St. Lucie Unit 1 reactor contains 14 x 14 fuel assemblies.

4.4.2.2 Critical Heat Flux Ratios

4.4.2.2.1 Departure from Nucleate Boiling Ratio

The margin to DNB in the core is expressed in terms of the departure from nucleate boiling ratio (DNBR). The DNBR is defined as the ratio of the heat flux required to produce departure from nucleate boiling at the calculated local coolant conditions to the actual local heat flux.

The DNB correlation used for design of the core is the CE-1 correlation. [1][2] Based on statistical evaluation of the CE-1 correlation and relevant data, it is concluded that the appropriate minimum DNBR is 1.13.[1][2] NRC evaluation of the uniform axial power distribution data results in their concluding that the CE-1 critical heat flux correlation (CHF), when coupled with the TORC code provides an acceptable correlation of uniform axial CHF data and that the minimum acceptable DNBR is 1.19.[4] Therefore, the minimum DNBR used for design is 1.19 for Cycle 1. Cycle 2 uses a DNBR of 1.28. Subsequent cycles with ESCU also use a DNBR of 1.28.

A typical DNB analysis is performed to evaluate the minimum DNBR in the core for the coolant conditions and engineering factors given in Table 4.4-1 for Cycle 1. Adverse radial and axial power distributions in Figures 4.4-1, 4.4-2 and 4.4-3 are chosen to evaluate the minimum DNBR in the core for Cycle 1. The 1.26 peak axial power distribution is chosen for the analysis since it produces the lowest DNBRs compared to the other distributions in Figure 4.4-3. The radial power distributions chosen in Figures 4.4-1 and 4.4-2 are typical adverse distributions but may not be the most limiting in the core. Table 4.4-1 gives the value of minimum DNBR for this DNB analysis.

A comparison of the minimum DNBRs computed using different correlations for the same power, flow, coolant temperature and pressure, and power distribution is presented in Table 4.4-2. The minimum DNBR values in both the limiting matrix subchannel and the limiting subchannel next to the guide tube are presented. The correlations compared are the CE-1 correlation, the original W-3 correlation,[3] the revised W-3 correlation[5] and the B&W-2 correlation.[5] The differences between the original and revised W-3 correlations as used here are in the C-factor and the cold wall correction factor. Further explanation of the C-factor and the cold wall correction factor can be found in References 3 and 5, respectively.

Additional comparisons are contained in CENPD-162A.[1] In general, the CE-1 correlation predicts lower values of critical heat flux than the B&W-2 correlation, with the differences increasing with increasing inlet subcooling. In comparison with the W-3 correlation, the CE-1 correlation tends to predict lower values of CHF with high inlet subcooling and higher values of CHF with low inlet subcooling.

The TORC computer code[6] is used to compute the local coolant conditions in the core and the minimum DNBR. For the WCAP-9272 bounding cycle analysis, the VIPRE-01 code (References 44 and 45) is used to compute the local coolant conditions in the core and the minimum DNBR. A discussion of the CE-1 DNB correlation, the ABB-NV DNB correlation and the analytical methods is presented in Subsections 4.4.4.1 and 4.4.4.5.2, respectively.

4.4.2.2.2 Application of Power Distribution and Engineering Factors

Distribution of power in the core is expressed in terms of factors that define the local power per unit length produced by the fuel relative to the core average power per unit length produced by the fuel. The method used to compute these factors, which describe the core power distribution, is discussed in Section 4.3. The energy produced in the fuel deposits in the fuel pellets, fuel cladding, and the moderator and results in the generation of heat in those places. The fraction of energy deposited in the fuel pellet and cladding is called the fuel rod energy deposition fraction. Accordingly, the core average heat flux from the fuel rods is determined by multiplying the core power by the average fuel rod energy deposition fraction and then dividing by the total heat transfer area. The energy deposition fractions used for DNB analyses for the average and the hot fuel rods are given in Table 4.4-1 for Cycle 1.

The effect on local heat flux and subchannel enthalpy rise of within tolerance deviations from nominal dimensions and specifications are included in thermal margin analyses by certain factors called engineering factors. These factors are applied to increase the local heat flux at the location of minimum DNBR and to increase the enthalpy rise in the subchannel adjacent to the rod with the minimum DNBR. Diversion crossflow and turbulent interchange mixing are not input as factors on subchannel enthalpy rise but are explicitly treated in the subchannel code analytical model.

Uncertainties in the power distribution factors are discussed in Subsection 4.4.2.9.4.

4.4.2.2.2.1 Power Distribution Factors

a) Rod Radial Power Factor

The rod radial power factor is the ratio of the average power per unit length produced by a particular fuel rod to the average power per unit length produced by the average powered fuel rod in the core. The maximum rod radial power factor is the ratio of the average power per unit length produced by the highest powered rod in the core to the average power per unit length produced by the average powered fuel rod in the core. Radial power distributions are dependent upon a variety of parameters (control rod insertion, power level, fuel exposure, etc.). The core wide and hot assembly radial power distributions used for a sample DNB analysis are shown in Figures 4.4-1 and 4.4-2.

The design maximum rod radial power factor selected is 1.70. The actual maximum rod radial power factor in the figures is 1.63. The actual maximum rod radial power factor in the core will normally be lower than the design maximum peak but it is not limited to this maximum value. The only limits are those specified in Subsection 4.4.1. The protective system ensures that those design limits are not violated.

The effects of fuel rod bowing on DNBR margin have been incorporated in the safety and setpoint analyses in the same manner as discussed in Reference 28. The value used for this analysis, 1.75% MDNBR, is valid for bundle burnups up to 30,000 MWD/MTU.

For those assemblies with an assembly average burnup in excess of 30 GWD/T, the minimum best estimate margin available, relative to more limiting peaking values present in other assemblies, exceeds the corresponding rod bow penalties based upon Reference 28. Hence, sufficient available margin exists to offset rod bow penalties for assemblies with burnup greater than 30 GWD/T.

For the Bounding Cycle Analyses introduced in Cycle 12 the effects of fuel rod bowing on DNBR margin have been incorporated in the safety and setpoint analyses in the manner discussed in References 35, 28, 39 and 40. The rod bowing penalty used for this analysis, 1.2% on minimum DNBR, is valid for bundle burnups up to 31,700 MWD/MTU. This penalty is included in the 1.28 DNBR limit. For assemblies with burnups greater than 31,700 MWD/MTU sufficient margin is available due to the reduction in radial power peaking to offset potential rod bow penalties for these higher burnup assemblies. Hence, the rod bow penalty based upon Reference 28 and methodology in References 39 and 40 for 31,700 MWD/MTU is bounding for all assembly burnups expected for the bounding cycle. The rod bow penalty of 1.2% on minimum DNBR remains applicable with the ABB-NV DNB correlation (Reference 42).

b) Axial Power Factor

The axial power factor is the ratio of the local power per unit length produced by a fuel rod to the average power per unit length produced by the same fuel rod. The maximum axial power factor is the ratio of the maximum local power per unit length produced by a rod to the average power per unit length produced by the same fuel rod. The axial power distribution directly affects DNBR.

Typically, the farther the location of the peak heat flux is from the core inlet, the lower the value of the peak heat flux needed to reach the DNBR limit. The axial power index and power density are monitored continuously and processed through the RPS such that the design basis limits are not exceeded. Section 4.3 describes the power distributions and their control. Figure 4.4-3 shows sample axial power distributions. The minimum DNBR for Cycle 1 in Table 4.4-1 is determined using the 1.26 peaked axial power distribution whereas the maximum heat fluxes are determined using 1.47 peaked axial power distributions.

c) Nuclear Power Factor

The nuclear power factor is the ratio of the maximum local power per unit length produced in the core to the average power per unit length produced by the average powered fuel rod in the core. It is conservatively calculated as the product of the maximum axial and radial power factors. The selected value for computing maximum heat fluxes for Cycle 1 is shown in Table 4.4-1. The actual value of the nuclear power factor will normally be lower throughout the cycle; but it is not limited to a maximum value shown in the table. The design limits are those specified in Subsection 4.4.1. The protective systems assure that those design limits are not violated.

d) Total Heat Flux Factor

The total heat flux factor is the ratio of the maximum local fuel rod heat flux to the core average fuel rod heat flux. The effects of fuel densification are not included in this factor. To determine the maximum local heat flux including the effect of gaps occurring between the fuel rod pellets, the augmentation factor should be applied. From this definition, the total heat flux factor is the product of the nuclear power factor, the engineering heat flux factor, and the ratio of the hot to the average rod energy deposition fractions. The total heat flux factor for Cycle 1 is given in Table 4.4-1. 1

e) Augmentation Factor

The densification of the fuel may lead to axial gaps in the fuel pellets stacks and can cause increased localized power peaking. This effect is expressed in terms of the augmentation factor which is defined as the ratio of the local heat flux to the unperturbed heat flux. The axial length over which the localized power perturbation considered to occur is called the gap length. Maximum values of the augmentation factor and gap length for Cycle 1 are given in Table 4.4-1. The effect of this factor on DNBR is discussed 1
in Subsection 4.4.2.2.3.

4.4.2.2.2.2 Engineering Factors

a) Engineering Heat Flux Factor

The effect on local heat flux due to normal manufacturing deviations from nominal design dimensions and specifications is accounted for by the engineering heat flux factor. Design variables that contribute to this engineering factor are initial pellet density, pellet enrichment, pellet diameter, and clad outside diameter.

These variables are combined statistically to obtain the engineering heat flux factor. The design value used for the engineering heat flux factor is based on deviations obtained from fuel manufacturing inspection data for over 25 batches of fuel for previous reactor cores. Similar tolerances and quality control procedures are used for the St. Lucie Unit 2 core, and as-built fuel manufacturing data are used to confirm that the Cycle 1 factors 1
given in Table 4.4-1 is conservative. The engineering heat flux factor is applied to the rod with the minimum DNBR and increases the heat flux only for the calculation of DNBR. This factor does not affect the enthalpy rise in the subchannel; the effect on the enthalpy rise in the subchannel due to normal manufacturing deviations from normal design dimensions and specifications is accounted for by the engineering enthalpy rise factor.

b) Engineering Factor on Linear Heat Rate

The effect of deviations from nominal fuel rod design dimensions and specifications on fuel temperature is accounted for by the engineering factor on linear heat rate. The method used to calculate this factor is described in detail in Appendix B of Reference 24. Since the final value is expected to be less than the value shown in Table 4.4-1, this is the value currently used; it is confirmed to be conservative when dimensional tolerances and specifications are established.

c) Engineering Enthalpy Rise Factor

The engineering enthalpy rise factor accounts for the effects of normal manufacturing deviations in fuel fabrication from nominal dimensions and specifications on the enthalpy rise in the subchannel adjacent to the rod with minimum DNBR. Tolerance deviations (averaged over the length of the fuel rods that adjoin the subchannel) for fuel pellet density, enrichment, and diameter contribute to this factor. As-built fuel manufacturing data will be used to confirm that the factor for Cycle 1 given in Table 4.4-1 is conservative.

The engineering ethalpy rise factor is applied by multiplying the factor times the rod radial power factor of the fuel rods adjacent to the subchannel adjoining the rod with the minimum DNBR. This increases the enthalpy rise in the subchannels which adjoin the same fuel rods.

d) Pitch and Clad Diameter Allowance and Rod Bow Penalty

The pitch and clad diameter allowance and rod bow penalty are allowances for the effect on enthalpy rise of the possible decreased flowrate in the subchannel resulting from a smaller than nominal subchannel flow area. The pitch and clad diameter allowance accounts for the systematic variation, while the rod bow penalty accounts for the random variation, in the fuel rod pitch and clad diameter.

The fractional heat deposited in the limiting subchannel adjacent to the rod with the minimum DNBR is multiplied by the pitch and clad diameter allowance. This increases the enthalpy rise in that subchannel in the same manner as does the engineering enthalpy rise factor, but does not directly affect the heat input into the surrounding subchannels. The combined effects of diversion crossflow and turbulent interchange resulting from the higher heat input and enthalpy rise are computed by the subchannel code.

The rod bow penalty is applied to the measured value of F_RT Discussion of fuel and poison rod bowing are presented in CENPD-225[20] and Subsection 4.4.4.1.

4.4.2.2.3 Fuel Densification Effect on DNBR

The perturbation in local heat flux due to fuel densification is given in Table 4.4-1 as the augmentation factor. As shown in CENPD-207[2] (see Subsection 4.4.4.1), even much larger local heat flux variations have no significant adverse effect on DNB. Therefore, no specific allowance is made or required for the effect on DNBR of local heat flux variations due to densification of the fuel.

The perturbation in core average heat flux due to fuel densification is considered by adjusting the active core length with the fuel densification factor given in Table 4.4-1. This factor reduces the active core length slightly which causes an increase in the core average heat flux.

4.4.2.3 Linear Heat Generation Rate

The core average and maximum fuel rod linear heat generation rates for Cycle 1 are given in Table 4.4-1. The maximum fuel rod linear heat generation rate is determined by multiplying the core average fuel rod linear heat generation rate by the product of the nuclear power factor, the engineering factor on linear heat rate, and the ratio of the hot to the average fuel rod energy deposition fractions. The local effects of fuel densification are not included in the maximum fuel rod linear heat generation rate presented in Table 4.4-1; although, to determine the maximum local linear heat generation rate including the effect of gaps occurring between the fuel pellets, the augmentation factor is applied.

4.4.2.4 Void Fraction Distribution

The core average void fraction and the maximum void fraction are calculated using the Maurer method[7] for Cycle 1. The void fractions discussed below are values for the reactor operating conditions and engineering factors for Cycle 1 given in Table 4.4-1, for the radial power distributions in Figures 4.4-1 and 4.4-2, and for the 1.26 peaked axial power distribution in Figure 4.4-3. For these conditions, only subcooled boiling occurs in the core.

The core average void fraction is less than 0.1 percent. The local maximum void fraction is less than 10 percent and occurs at the exit of the subchannel adjacent to the rod with the minimum DNBR. The average exit void fractions and qualities in different regions of the core are shown in Figure 4.4-4 for the core radial power distribution shown on Figure 4.4-1. The axial distribution of quality in the subchannel adjacent to the rod with the minimum DNBR is shown on Figure 4.4-5. The void fraction in that subchannel is very close to zero and therefore the axial distribution of void fraction is not shown.

4.4.2.5 Core Coolant Flow Distribution

The core inlet flow distribution is required as input to the subchannel margin code (refer to Subsection 4.4.4.5.2). The inlet flow distribution for four pump operation was determined from a 1/5 scale reactor flow model test. Descriptions of the model test and the resulting core inlet flow distribution are given in Subsection 4.4.4.2.1.

Intentional selective orificing is not used in the core design.

4.4.2.6 Core Pressure Drops and Hydraulic Loads

4.4.2.6.1 Reactor Vessel Flow Distribution

The main coolant flow path in the reactor vessel is down the annulus between the reactor vessel and the core support barrel, through the flow skirt, up through the core support region and the reactor core, through the fuel alignment plate, and out through the two reactor vessel outlet nozzles. A portion of this flow leaves the main flow path as shown schematically in Figure 4.4-6. Part of the bypass flow is used to cool the reactor internals in areas not in the main coolant flow path and to cool the CEAs. Table 4.4-3 lists the bypass flow paths and the percent of the total vessel flow that enters and leaves these paths.

The thermal margin calculations conservatively use the maximum bypass flow of 3.7 percent of the total vessel flow as compared to the calculated best estimate bypass flow shown in Table 4.4-3.

4.4.2.6.2 Reactor Vessel and Core Pressure Drops

The irrecoverable pressure losses from the inlet to the outlet nozzles are calculated using standard loss coefficient methods and information from flow model tests (Subsection 4.4.4.2).

Pressure losses (at 100 percent power, minimum allowable reactor coolant flow, and an operating pressure of 2250 psia), are listed in Table 4.4-4 together with the coolant temperature used to calculate each pressure loss. The calculated pressure losses include both geometric and Reynolds number dependent effects.

4.4.2.6.3 Hydraulic Loads on Internal Components

The significant hydraulic loads which act on the reactor internals during steady state operation are listed in Table 4.4-5. These loads are determined from analytical methods and from results of reactor flow model and components test programs (refer to Subsections 4.4.4.2.1 and 4.4.4.2.2, respectively). The design hydraulic loads consist of steady state drag and impingement loads, and the fluctuating loads induced by pump-induced pressure pulsations, vortex shedding, and turbulence.

In evaluating the design hydraulic loads, consideration is given to the particular pump operating configuration and coolant temperature that maximizes the hydraulic load for a given internal component.

All hydraulic loads in Table 4.4-5 are based on a coolant flow rate of 389,760 gpm and a coolant temperature of 500°F. Where conservative and applicable, core pressure drop is increased by six psi in the calculation of design hydraulic loads to account for the possibility of core crudding. The six psi value was chosen to provide a sufficient margin for accommodating core crudding effects. Recent experience at other CE plants indicates that any increases in core pressure drop due to crudding effects are much smaller than the six psi design value.

Finally, an uncertainty allowance of 10 percent in the resulting load is added to arrive at the final design hydraulic loads. When other coolant conditions result in more limiting loading for individual components, the loads in Table 4.4-5 are adjusted in the detailed design analysis.

The 389,760 gpm value represents the original maximum design reactor coolant flow set for the CE 2560 MWt class reactors. Subsequent startup of the first four of these reactors showed that the indicated flow rates exceeded the 389,760 gpm value due to high performance curves for the primary coolant pumps. Re-evaluation of the stress margins at the higher indicated flow rates and subsequent operating experience have shown that operation is satisfactory at those flow rate levels. In the case of St. Lucie Unit 2, the internals are identical to the CE 3400 MWt class reactor internals with the exception of the shorter core length for St. Lucie. The 3400 MWt internals have been designed to withstand hydraulic loads based on a flow rate of 475,200 gpm. This value is significantly larger than the current maximum allowable flow rate of 424,350 gpm set for St. Lucie Unit 2 and shown in Table 4.4-1. The 424,350 gpm value is also consistent with the operating experience from earlier CE 2560 MWt plants and is expected to envelope the indicated flow rates from the St. Lucie Unit 2 flow tests; see Subsection 4.4.4.5.1. The mechanical design flow rate has increased to 431,000 gpm (Reference 47). New calculations have been carried out that demonstrate the acceptability of the increased flow rate for the RSGs.

Hydraulic loads for postulated accident conditions are discussed in Subsection 3.9.2.5.

4.4.2.7 Correlations and Physical Data

4.4.2.7.1 Heat Transfer Coefficients

The correlations used to determine cladding temperatures for non-boiling forced convection and nucleate boiling are discussed here. The surface temperature of the cladding is dependent on the axial and radial power distributions, the temperature of the coolant, and the surface heat transfer coefficient.

The surface heat transfer coefficient for non-boiling forced convection is obtained from the Dittus-Boelter correlation[8] where fluid properties are evaluated at the bulk condition.

$$h_{db} = \frac{0.023k}{De} (N_R)^{0.8} (N_{Pr})^{0.4}$$

where:

h_{db} = Heat transfer coefficient, Btu/h-ft^2 - F

k = Thermal conductivity, Btu/h-ft - F

De = Equivalent diameter = $4A/P_w$, ft

N_R = Reynolds number, based on the equivalent diameter and coolant properties evaluated at the local bulk coolant temperature.

N_{Pr} = Prandtl number, based on coolant properties evaluated at the local bulk coolant temperature.

A = Cross-sectional area of flow subchannel, ft^2.

P_w = Wetted perimeter of flow subchannel, ft.

No specific allowance is made or considered necessary for the uncertainties associated with the Dittus-Boelter correlation because the Dittus-Boelter correlation is not used directly in computing thermal margin, but rather plays a part in determining pressure drop and cladding temperature. The validity of the overall scheme for predicting pressure drop is shown by the excellent agreement between predicted and experimental values obtained during the DNB test program described in Subsection 4.4.4.1. The uncertainty associated with the cladding temperatures calculated for single phase heat transfer is not a major concern because the limiting fuel and cladding temperatures occur where the cladding-to-coolant heat transfer is by nucleate boiling.

The temperature drop across the surface film is calculated from:

$$T_{film} = q''/h_{db} = T_{wall_{db}} - T_{coolant}$$

where:

q" = fuel rod surface heat flux, Btu/h-ft^2

$T_{wall_{db}}$ = fuel rod surface temperature determined by Dittus-Boelter, F

$T_{coolant}$ = reactor coolant temperature, F

The maximum fuel rod heat flux is the product of the core average fuel rod heat flux and the total heat flux factor (refer to Table 4.4-1 and Subsection 4.4.2.2.2). Nucleate boiling may occur on the clad surface. In the nucleate boiling regime, the surface temperature of the cladding for Cycle 1 is determined from the Jens and Lottes correlation[9].

$$T_{wall_{JL}} = T_{sat} + 60\,(q'' \times 10^{-6})^{0.25}\,[\exp(-P/900)]$$

where:

P = Pressure, psia

q" = Defined above

$T_{wall_{JL}}$ = Fuel rod surface temperature determined by Jens and Lottes, F

T_{sat} = Saturation temperature, F

Nucleate boiling is assumed to exist if $T_{wall_{JL}}$ is less than $T_{wall_{db}}$.

The cladding surface temperature is calculated by summing the temperature of the coolant at the particular location and the temperature drop across the surface film; or if nucleate boiling is occurring, it is calculated directly from the Jens and Lottes correlation.

4.4.2.7.2 Core Irrecoverable Pressure Drop Loss Coefficients

Irrecoverable pressure losses through the core result from friction and geometric changes. The pressure losses through the lower and upper end fittings were initially calculated using the standard loss coefficient method and then verified by test (refer to Subsection 4.4.4.2.2). The correlations used to determine frictional and geometric losses in the core are presented in Subsection 4.4.4.2.3.

4.4.2.7.3 Void Fraction Correlations

There are three separate void regions to be considered in boiling flow for Cycle 1. Region 1 is highly subcooled in which a single layer of bubbles develops on the heated surface and remains attached to the surface. Region 2 is a transition region from highly subcooled to bulk boiling where the steam bubbles detach from the heated surface. Region 3 is the bulk boiling regime.

The void fraction in Regions 1 and 2 is predicted using the Maurer method.[7] The calculation of the void fraction in the bulk boiling regime is discussed in Subsection 4.4.4.2.3.

4.4.2.8 Thermal Effects of Operational Transients

Design basis limits on DNBR and fuel temperature are established to assure that thermally induced fuel damage will not occur during steady-state operation and during anticipated operational occurrences. The RPS ensures that the design limits are not violated.

4.4.2.9 Uncertainties in Estimates

4.4.2.9.1 Pressure Drop Uncertainties

The reactor vessel pressure losses in Table 4.4-4 are the values calculated for the best estimate flow with standard loss coefficient methods. The uncertainties in the correlations for the loss coefficients and the dimensional uncertainties on the reactor vessel and internals are accounted for when determining maximum and minimum vessel hydraulic resistance. The uncertainties at the 2 σ level are estimated to be equivalent to approximately ±10 percent of the best estimate vessel pressure loss.

4.4.2.9.2 Hydraulic Loads Uncertainties

The design hydraulic loads are determined at the maximum allowable reactor coolant flow rate (refer to Subsection 4.4.2.6.3); the effects of uncertainties in load determination are represented by a 10 percent increase in the hydraulic loads to arrive at the final design values.

4.4.2.9.3 Fuel and Clad Temperature Uncertainty

Uncertainty in the ability to predict the maximum fuel temperature is a function of gap conductance, thermal conductivities, peak linear heat rate, and heat generation distribution. Uncertainties in gap conductance and thermal conductivity are taken into account in the analytical model.

Uncertainties in the peak linear heat rate are accounted for by including the engineering factor on linear heat rate in the analysis. This factor includes uncertainties in fuel pellet density, enrichment, and pellet diameter.

Uncertainty in predicting the cladding temperature at the location of maximum heat flux is the uncertainty in the film temperature drop, which is minimal at this location where nucleate boiling occurs.

4.4.2.9.4 DNBR Calculation Uncertainties

a) The uncertainty in the calculation of minimum DNBR is divided into:

1) The uncertainty in the input to the core analytical model, the subchannel code. This includes the core geometry, power distribution, inlet flow and temperature distribution, exit pressure distribution, single phase friction factor constants, spacer grid loss coefficients, diversion crossflow resistance and momentum parameters, turbulent interchange constants, and hot fuel rod energy deposition fraction.

2) The uncertainty in the analytical model to compute the actual distribution of flow and the local subchannel coolant conditions.

3) The uncertainty in the DNB correlation to predict DNB.

b) The following paragraphs discuss the above uncertainties and the allowances for them, if needed, in the thermal margin analysis of the core.

1) Uncertainty in the input to the core analytical model:

(a) Uncertainty in core geometry, as manifested by manufacturing variations within tolerances, is considered by the inclusion of engineering factors in the DNBR analyses; see Subsection 4.4.2.2.2.2 for a discussion of the method used to compute conservative values.

(b) The core inlet flow distribution is obtained from flow model testing discussed in Subsection 4.4.4.2. Uncertainties in the core flow distribution are included in the design method for DNBR analyses.

(c) Uncertainties in the core inlet temperature distribution and core exit pressure distribution are included in the design method for DNBR analyses.

(d) The Blasius single-phase friction factor equation for smooth rods is given and shown to be valid in Subsection 4.4.4.2.3. The loss coefficient for the spacer grid is obtained from pressure drop data discussed in Subsection 4.4.4.2.3.

(e) The value of minimum DNBR is relatively insensitive to crossflow resistance and momentum parameters.[6]

(f) Subsection 4.4.4.1 describes the testing to determine the inverse Peclet number which is indicative of the turbulent flow interchange between subchannels. The inverse Peclet number is input to the subchannel code and is used to determine the effect of turbulent interchange on the enthalpy rise in adjacent subchannels. From the testing, a value of 0.0035 is justified.

(g) The same fuel rod energy deposition fraction is used for the hot rod as for the average rod. The hotter the rod, the lower is the actual value of energy deposition fraction with respect to that for the average rod. A lower energy deposition fraction reduces the hot rod heat flux and thereby increases its DNBR. The use of the average rod energy deposition fraction for the hot rod is therefore conservative. See Section 4.3 for a discussion of the calculation of the energy deposition fractions.

2) Uncertainty in the analytical model:

The ability of the TORC code to accurately predict local subchannel conditions in rod bundles is described in CENPD-161.[6] The ability of the code to accurately predict the core wide coolant conditions is described in CENPD-206.[10]

The ability of the VIPRE code to accurately predict reactor core and local subchannel conditions in rod bundles is described in References 44 and 45.

3) Uncertainty in the DNB correlation:

The uncertainty in the DNB correlation is determined by a statistical analysis of DNB test data. For example, a value of 1.13 has been shown to provide a 95 percent probability with 95 percent confidence that DNB will not occur on a fuel rod having that minimum DNBR.[1][2]

4.4.2.10 Flux Tilt Considerations

An allowance for degradation in the power distribution in the x-y plane (commonly referred to as flux tilt) is provided in the protection limit setpoints even though little, if any, tilt in the x-y plane is expected.

The tilt, along with other pertinent core parameters, is monitored during operation. The RPS actuates a trip if limiting safety system settings are reached.

The thermal margin calculations used in designing the reactor core are performed using the subchannel code, such as TORC or VIPRE. The TORC code, which is described in Subsection 4.4.4.5.2, is based on an open core analytical method for performing such calculations and treats the entire core on a three-dimensional basis. Thus, any asymmetry or tilt in the power distribution is analyzed by providing the corresponding power distribution in the TORC input.

4.4.3 DESCRIPTION OF THE THERMAL AND HYDRAULIC DESIGN OF THE REACTOR COOLANT SYSTEM (RCS)

A summary description of the RCS is given in Section 5.1.

4.4.3.1 Plant Configuration Data

An isometric view of the RCS is given in Figure 4.4-7. Dimensions are shown on the general arrangement drawings, Figures 5.1-1 and 5.1-2. Table 4.4-6 lists the valves and pipe-fittings which form part of the RCS.

Table 4.4-7 lists the design flow through each flowpath in the RCS.

Table 4.4-8 provides the volume, flowpath length, height and liquid level of each volume, bottom elevation of each volume, and minimum flow area for each component within the RCS.

The line lengths and sizes of the safety injection lines are listed in Table 4.4-9 and Figures 6.3-1a, 6.3-1b, and 6.3-1c.

Tables 4.4-10 and 5.1-1 list the thermal-hydraulic design parameters for the RCS.

4.4.3.2 Operating Restrictions on Pumps

The minimum RCS pressure for reactor coolant pump (RCP) operation at any given temperature is limited by the required net positive suction head (NPSH) for the RCPs during portions of plant heatup or cooldown. To ensure that the pump NPSH requirements are met under all possible operating conditions, an operating curve is used which gives permissible RCS pressure for RCP operation as a function of temperature (See Subsection 5.4.1.2).

The reactor coolant pump NPSH restriction on this curve is determined by using the most restrictive NPSH requirement for any combination of pump operation and correcting it for pressure and temperature instrument errors and pressure measurement location. The NPSH required versus pump flow is supplied by the pump vendor. RCP operation below this curve is prohibited. At low reactor coolant temperature and pressure, minimum RCP seal pressure may require that the minimum pressure for RCP operation versus temperature curve be above the NPSH curve.

4.4.3.3 Power-Flow Operating Map (BWR)

This subsection is not applicable to St. Lucie Unit 2.

4.4.3.4 Temperature-Power Operating Map (PWR)

Reactor operation at power with one, two or three pumps operating, or while in natural circulation is not allowed. However, decay heat may be transferred to the steam generator in any of the above cases. The temperature - power operating map (temperature control program) is shown in Figure 4.4-10.

The adequacy of natural circulation for decay heat removal after reactor shutdown has been verified analytically and by tests on the Palisades reactor (Docket No. 50-255) and Calvert Cliffs I (Docket No. 50-317). The core ΔT in the analysis has been shown to be lower than the normal full power ΔT; thus the thermal and mechanical loads on the core structure are less severe than normal design conditions. In addition, St. Lucie Unit 1 (Docket No. 50-335) successfully performed a cooldown from full power conditions using only natural circulation cooling following a reactor trip.

Heat removed from the core during natural circulation may be rejected either by dumping steam to the main condenser or to the atmosphere; the rate of heat removal may be controlled to maintain core ΔT within allowable limits.

4.4.3.5 Load Following Characteristics

The design features of the RCS influence its load following and transient response. The RCS is capable of following the normal transients identified in Subsection 3.9.1. These requirements are considered when designing the pressurizer spray and heater systems, charging/letdown system, Reactor Regulating System, turbine bypass control system, and Feedwater Regulating System. Finally, these transients are included in the equipment specification for each RCS component to ensure the structural integrity of the system.

When load changes are initiated, the Reactor Regulating System senses a change in the turbine power and positions CEAs to attain the programmed coolant average temperature. RCS boron concentration can also be adjusted to attain the appropriate coolant temperature. The turbine bypass control system senses changes in steam pressure and steam flow in order to control steam pressure following load reductions. The feedwater system employs a controller which senses changes in steam flow, feedwater flow, and water level and acts to maintain steam generator level at the desired point. The pressurizer pressure and water level control systems respond to deviations from preselected setpoints caused by the expansion or contraction of the reactor coolant and actuate the spray or heaters and the charging or letdown systems as necessary to maintain pressurizer pressure and water level.

4.4.3.6 Thermal and Hydraulic Characteristics Table

Principal thermal and hydraulic characteristics of the RCS components are listed in Table 4.4-10.

4.4.4 EVALUATION

4.4.4.1 Critical Heat Flux

The margin to critical heat flux (CHF) or DNB is expressed in terms of the DNBR. The DNBR is defined as the ratio of the heat flux required to produce DNB at the calculated local coolant to the actual heat flux.

The CE-1 correlation[(1)(2)] was used with the TORC computer code[(6)] to determine DNBR values for normal operation and anticipated operational occurrences. The CE-1 correlation was developed in conjunction with the TORC code specifically for DNB margin predictions for fuel assemblies with standard spacer grids similar to those in St. Lucie Unit 2. Topical Reports CENPD-162[(1)] and CENPD-207[(2)] provide detailed information on the CE-1 correlation and source data, and also provide comparisons with other data and correlations. In brief, the correlation is based on data from tests conducted for CE at the Chemical Engineering Research Laboratories of Columbia University. Those tests used electrically-heated 5 x 5 array rod bundles corresponding dimensionally to a portion of a 16 x 16 or 14 x 14 assembly with standard spacer grids. The test programs conducted for the 16 x 16 and 14 x 14 assembly geometries each included tests to determine the effects on DNB of the CEA guide tube, bundle heated length, axial grid spacing, and lateral and axial power distributions.

The uniform axial power CE-1 correlation[(1)] was developed from DNB data for six test sections with the following characteristics:

Fuel Assembly Geometry	No. Heated Rods	Lateral Power Distr	Heated Length (ft)	Axial Grid Spacing (in.)
16 x 16	25	Uniform	7	16.0
16 x 16	21	Nonuniform	7	18.3
16 x 16	21	Nonuniform	12.5	17.4
14 x 14	25	Uniform	7	14.3
14 x 14	21	Nonuniform	7	14.3
14 x 14	21	Nonuniform	12.5	14.3

Local coolant conditions at the DNB location were determined by using the TORC code in a manner consistent with the use of the code for reactor thermal margin calculations. The uniform axial power CE-1 correlation was developed from 731 DNB data for the following parameter ranges:

Pressure	1785 to 2415 lb/in.^{2}a
Inlet temperature	382 to 644°F
Heat flux	0.213×10^6 to 0.952×10^6 Btu/h-ft^2
Local coolant quality	-.016 to 0.20
Local mass velocity	0.87×10^6 to 3.21×10^6 lb/h-ft^2

The uniform axial power CE-1 correlation predicted the 731 source data with a mean and standard deviation of the ratio of measured and predicted DNB heat fluxes of 1.000 and 0.068, respectively. The validity of the CE-1 correlation for predicting DNB for 16 x 16 fuel assemblies was further verified by the analysis of data obtained by repeating one of the tests for the 16 x 16 assembly geometry at the Winfrith Laboratory of the UKAEA.

For nonuniform axial power distribution, the uniform axial power CE-1 correlation is modified by the F-factor(5). The conservatism of that method of predicting DNB for 16 x 16 fuel assemblies with nonuniform axial flux shapes is demonstrated in CENPD-207(2). CENPD-207(2) presents measured and predicted DNB heat fluxes for a series of tests using nonuniform axial power rod bundles representative of 16 x 16 or 14 x 14 fuel assemblies with standard spacer grids. Those test sections had the following characteristics:

Fuel Assembly Geometry	No. Heated Rods	Lateral Power Distr	Axial Power Distr	Heated Length (ft)	Axial Grid Spacing (ft)
16 x 16	21	Nonuniform	1.46 symmetric	12.5	14.2
16 x 16	21	Nonuniform	1.47 top peak	12.5	14.2
14 x 14	21	Uniform	1.68 top peak	12.5	17.4
14 x 14	21	Nonuniform	1.68 bottom peak	12.5	17.4

The DNB data from those tests were evaluated using the CE-1 correlation modified by the F-factor and the TORC code used in a manner consistent with the use of the code for reactor calculations. The evaluation included DNB data within the following parameter ranges:

Pressure	1745 to 2425 lb/in.^{2}a
Inlet temperatures	333 to 631°F
Local coolant quality	-0.27 to 0.20
Local mass velocity	0.81 x 10^6 to 3.07 x 10^6 lb/h-ft^2

It was found that the mean and standard deviation of the ratio of measured and predicted DNB heat fluxes were 1.229 and 0.125, respectively, for the 369 DNB data within the parameter ranges mentioned above.

Testing was also conducted with rod bundles representative of the 16 x 16 fuel assembly to determine the effect on DNB of local perturbations in heat flux. Results are presented in CENPD-207(2) for two nonuniform axial power rod bundles which were similar except that one test bundle had a heat flux spike (23 percent higher heat flux for a four in. length) at the location where DNB was anticipated. The results show that there is no significant adverse effect on DNB due to that flux spike. Therefore, it is concluded that no allowance is required for the effect on DNB or local heat flux perturbations less severe than that tested.

One important factor in the prediction of DNB and local coolant conditions is the treatment of coolant mixing or turbulent interchange. The effect of turbulent interchange on enthalpy rise in the subchannels of 16 x 16 fuel assemblies with attendant spacer grids is calculated in the TORC code by

$$\hat{Pe} = \frac{\omega'}{\overline{G} D_e} = 0.0035$$

where:

$\hat{P}e$= inverse Peclet number

ω' = turbulent interchange between adjacent subchannels lb/h-ft

D_e = average equivalent diameter of the adjacent subchannels, ft

G = average mass velocity of the adjacent subchannels, lb/h-ft^2

The value of 0.0035 for the inverse Peclet number for use with the 16 x 16 fuel assembly with standard spacer grids was originally chosen based on cold water dye mixing tests conducted for the 14 x 14 assembly and for a "prototype" of the Palisades reactor fuel assembly. The validity of the inverse Peclet number of 0.0035 for the 16 x 16 assembly with standard grids was verified with data obtained in the tests conducted at Columbia University.[1]

The design basis requires that the minimum DNBR for normal operation and anticipated operational occurrences be chosen to provide a 95 percent probability at the 95 percent confidence level that DNB will not occur on fuel rod having that minimum DNBR. Statistical evaluation of the CE-1 correlation and relevant data shows that the appropriate minimum DNBR is 1.13[1][2]. Based on review of CENPD-162[1], the NRC requires use of a minimum DNBR of 1.19. CE maintains that use of the 1.19 DNBR design limit with the St. Lucie Unit 2 15.8-inch grid spacing is bounded by the 14.3-inch and 17.4-inch grid spacing CHF data. However, the NRC has required that, in the absence of specific test data on the St. Lucie Unit 2 bundle geometry, the design limit DNBR should be penalized resulting in a DNBR limit of 1.20 for Cycle 1. The DNBR limit for Cycle 2 was changed to 1.28. Subsequent cycles that used ESCU also used a DNBR limit of 1.28. Commencing with Cycle 15, RTDP (Reference 71) and a DNBR limit of 1.30 is used. The EPU DNBR limit is 1.32.

Although the illustrative DNBR analysis presented here does not include an allowance for rod bow, Reference 25 provided a method for convoluting the CE-1 DNB statistics with rod bowing statistics, which is applied. Reference 20, particularly in its supplements B and C, provides supporting data on the appropriate adjustments to the unbowed design limit DNBR, to produce a burnup dependent design limit DNBR which includes the effects of rod bow. Modifications are made to the value of F_r^T to account for fuel rod bow DNBR penalty and the additional penalty on DNBR due to increased grid spacing.

For the Bounding Cycle analyses introduced in Cycle 12 the effects of fuel rod bowing on DNBR margin have been incorporated in the safety and setpoint analyses in the manner discussed in References 35, 28, 39 and 40. The rod bowing penalty used for this analysis, 1.2% on minimum DNBR, is valid for bundle burnups up to 31,700 MWD/MTU. This penalty is included in the 1.28 DNBR limit. For assemblies with burnups greater than 31,700 MWD.MTU sufficient margin is available due to the reduction in radial power peaking to offset potential rod bow penalties for these higher burnup assemblies. Hence, the rod bow penalty based upon Reference 28 and methodology in References 39 and 40 for 31,700 MWD/MTU is bounding for all assembly burnups expected for the bounding cycle.

For the bounding cycle analysis, the rod bowing effects are extrapolated from the 14X14 channel closure data using the L^2/I dependence as discussed in References 39 and 40. L is the span length between two adjacent grids, and I is the moment of inertia of the fuel rod cladding. A comparative analysis of design features of the 14x14 design (rod bow data in Reference 28), the Arkansas Nuclear One Unit 2 design (rod bow data in Reference 39), and St. Lucie Unit 2 with respect to factors that influence rod bow justifies use of the L^2/I dependence for St. Lucie Unit 2. The bases for that conclusion are summarized below.

The mechanism causing rods to bow is compressive axial loads on the rod induced by the spacer grids restraint of differential growth between the fuel rods and the guide tubes. With this loading, the rod behaves like a column with multiple supports.

The basic design differences pertinent to the evaluation of rod bow are the axial spacing between grids (L), the moment of inertia of the cladding (I), the number of spacer grid assemblies, and the differential growth between the fuel rods and the guide tubes. The comparative analysis evaluated the 14x14 design, the St. Lucie 2 design, and the Arkansas Nuclear One Unit 2 design with respect to these differences.

The St. Lucie 2 design was shown to represent less of an extrapolation from the 14x14 design than the Arkansas design because, relative to the Arkansas design, the St. Lucie 2 design has a longer span length; the same moment of inertia of the cladding, fewer spacer grids, and less differential growth between the fuel rods and the guide tubes. Furthermore, there have been no design or manufacturing changes introduced that increase either the as-fabricated rod bow, or the anticipated rod bow as a function of burnup, relative to the Arkansas fuel measured rod bow.

Since rod bow data has demonstrated the applicability of the L^2/I dependency for Arkansas and the St. Lucie 2 design has been shown to be conservatively represented by the Arkansas design, the L^2/I dependence has been applied in extrapolating from the 14x14 channel closure data to determine the rod bow penalty for St. Lucie 2 fuel.

For the WCAP-9272 bounding cycle analysis, the ABB-NV DNB correlation (Reference 42) was used as the primary correlation with the VIPRE code (References 44 and 45) to determine DNBR values. The ABB-NV correlation was developed for CE14x14 and CE16x16 non-mixing vane fuel designs, based on the Critical Heat Flux (CHF) test data obtained from the Heat Transfer Research facility of Columbia University. The CHF tests simulated 5x5 and 6x6 arrays of the fuel assembly geometry with or without guide tubes, uniform and non-uniform axial power shapes, and different grid designs and heated lengths. The empirical correlation includes the following variables: pressure, local mass velocity, local quality, distance from grid to CHF location, heated length and the heated hydraulic diameter of the CHF subchannel. Special geometry terms are used in the correlation to correct CHF for grid, heated length, cold wall and guide tube effects. The Tong F-factor was also optimized and applied to the correlation to account for the effects of non-uniform axial power shapes. The 95/95 DNBR limit for the ABB-NV correlation is 1.1.3 (Reference 42).

The ABB-NV correlation was developed and first applied with the TORC code. Reference 43 provides justification on use of the ABB-NV correlation with the VIPRE code and demonstrates that VIPRE is equivalent to TORC for the DNBR calculations. The inverse Peclet number of 0.0035 for the CE16x16 fuel assembly remains unchanged with the the use of VIPRE code. The 95/95 DNBR limit for the ABB-NV correlation remains applicable with the VIPRE code under the following parameter ranges:

Pressure	1750 to 2415 psia
Local Mass Velocity	0.8 to 3/16 $Mlbm/hr\text{-}ft^2$
Local Quality	-0.14 to 0.22
Heated Length, Inlet to CHF Location	48 to 150 inches
Grid Spacing	8 to 18.86 inches
Heated Hydraulic Diameter Ratio	0.679 to 1.08

4.4.4.2 Reactor Hydraulics

Design reactor hydraulic parameters include the distribution of flow in the reactor core, the sum and breakdown of irreversible pressure drops, and the forces due to flow acting on various reactor internal components. These parameters are derived for steady state flow conditions from flow test measurements in reactor scale models and full-scale prototype fuel assemblies. These tests are part of the Combustion Engineering reactor development program and are instituted whenever results of prior tests can not be extended with sufficient accuracy for evaluation of a new reactor or component design.

4.4.4.2.1 Reactor Flow Model Tests

Combustion Engineering (CE) reactor scale model flow test programs have supported the progression of seven basic geometric configurations, as follows:

Configuration	Reactor(s)	Distinguishing Hydraulic Features
1	Palisades	Four inlets, two outlets, cruciform control elements, 204 fuel assemblies.
2	Fort Calhoun	Four inlets, two outlets, 5-rod control elements, 133 fuel assemblies.
3	Maine Yankee	Three inlets, three outlets, 5-rod control elements, 217 fuel assemblies 137 inch active core.
4	2560 MWT Class: Calvert Cliffs 1 and 2, St. Lucie Unit 1 and 2, Millstone 2	Four inlets, two outlets, 5-rod control elements, 217 fuel assemblies, 137 inch active core
5	Arkansas Nuclear One (Unit 2)	Four inlets, two outlets, 5-rod control elements, 177 fuel assemblies 150 inch active core
6	3400 MWT Class: San Onofore 2 and 3 Forked River Waterford 3, Pilgrim 2	Four inlets, two outlets, 5-rod control elements, 217 fuel assemblies, 150 inch active core.
7	System 80	Four inlets, two outlets, 4-rod control elements, 241 fuel assemblies modified upper and lower plena design 150 inch active core.

Flow model tests have been conducted for configurations 1 through 4, 6 and 7. The configuration 1 and 2 tests were run under contract to Battelle Memorial Institute using air as the test medium. The configuration 3, 4, 6 and 7 tests were performed in a 15,000 gal/min cold water facility in the CE Nuclear Laboratories. All final tests have been with representation of the entire reactor; generally at 1/5 scale. The final System 80 flow model, configuration 7, was constructed at 3/16 scale. Model components for these tests were constructed geometrically to scale, with the exception of the model core, where individual fuel assemblies were represented by "core tubes".

Core tubes for configuration 1 through 4 were "closed", with a single flow metering orifice sized to represent the fuel assembly axial flow resistance. Core tubes for configurations 6 and 7 were "open", with prototypical flow redistribution permitted through holes in the double-wall boundaries between adjoining core tubes. Fuel assembly axial flow resistance in the open-core tests was represented by a series of six orifice plates.

Hydraulic design analysis methods for St. Lucie Unit 2 are based on appropriate consideration of data from all flow model tests through configuration 6. Test results most directly applicable to St. Lucie Unit 2 are those of configurations 4 and 6. The configuration 4 and 6 models have identical geometry up to the core inlet. Beyond the core inlet, configuration 6 most closely represents St. Lucie Unit 2, since St. Lucie Unit 2 has the 3400-MWT class 16 x 16 fuel rod array and outlet plenum geometry. Except for its shorter core and lower power rating, St. Lucie Unit 2 would be grouped with the 3400 MWT class reactors.

Descriptions of the 1/5 scale configuration 6 flow model, the test facility, and portions of the test information can be found in Appendix A of CENPD-206[10]. An elevation view of the flow model is shown here on Figure 4.4-11. Instrumentation was installed at several points within the reactor flow model to measure the variables of interest. The coolant flow path on the model was segmented by station numbers from inlet to outlet as shown in Figure 4.4-11. The tabulation below describes the measured variables and the location and type of instrumentation.

	Measured Variable	Model Station	Description of Instrumentation
1.	Inlet duct flowrate	Inlet ducts, upstream of station 1	Pressure differential measured across a Gentile-type flow meter in each inlet duct.
2.	Inlet nozzle static pressure	Station 1	Static pressure measured at each inlet nozzle, using four equally spaced pressure taps upstream of inlet elbows.
3.	Downcomer inlet static pressure	Station 5	Static pressure measured at each of 12 pressure taps located equally spaced around the vessel wall.
4.	Downcomer outlet static pressure	Station 7	Static pressure measured at each of 12 pressure taps locations equally spaced around the vessel wall.
5.	Core tube inlet static pressure	Station 13	Static pressure measured in each core tube with a wall static pressure tap at a position of 0.5 inches upstream of the inlet orifice plate.
6.	Core tube inlet flowrate	Station 17	Differential pressure measured across each core tube inlet orifice plate with static pressure taps 0.5 inches upstream and 0.25 inches downstream of the orifice plate.
7.	Core tube outlet static pressure	Station 18	Static pressure measured in each core tube with a wall static pressure tap at a position of 0.5

			in. upstream of the outlet orifice plate. The static pressure at the core tube exit, (station 20) was calculated from static pressures measured at station 18, the measured core tube exit flow rate, and the calibrated core tube exit pressure loss coefficients.
8.	Core tube outlet flowrate	Station 18	Pressure differential measured across each core tube outlet orifice plate with static pressure taps 0.5 in. upstream and 0.25 in. downstream of the orifice plate.
9.	Outlet nozzle static pressure	Station 24	The static pressure measured at each outlet nozzle using four equally spaced pressure taps just downstream of the outlet nozzle.
10.	Outlet nozzle flowrate	Outlet ducts, downstream of station 24	Pressure differential measured with an elbow type flow meter in each outlet duct.
11.	Coolant temperature	Test loop duct	Temperature measured with an iron-constantan thermocouple.

The pressure differentials across the flow meters in the inlet and outlet ducts were set and measured with mercury filled manometers. The remaining flow model pressures and pressure differentials were measured using a system consisting of several diaphragm-type pressure transducers with a scanning valve arrangement.

4.4.4.2.1.1 Core Inlet Flow and Core Outlet Pressure Distributions

The core inlet flow distribution and core outlet pressure distribution serve as boundary conditions for TORC open-core thermal margin analysis (refer to Subsection 4.4.4.5.2). The applicable configuration 6 flow model core flow and pressure distributions are given in Appendix A of CENPD - 206[10].

For application in reactor TORC analyses, the core inlet flow distribution is non-dimensionalized as the ratio of local to average fluid mass velocity. The core outlet pressure distribution is non-dimensionalized as a Euler number describing individual bundle outlet static pressure relative to the planar average. The standard error of estimate determined for the reactor core inlet flow distribution is three percent; for the reactor core outlet pressure distribution standard error of estimate is one percent.

4.4.4.2.1.2 Reactor Pressure Losses

The reactor vessel pressure losses were determined on the basis of data from the 1/5 scale configuration 6 flow model tests and from the components tests on the CE 16x16 fuel assembly (refer to Subsection 4.4.4.2.2). The flow path segments, using station numbers in Figure 4.4-11 as identification:

Region	Station Numbers
Inlet nozzle and 90° turn	1-5
Downcomer annulus	5-7
Lower plenum	7-13
Core	13-20
Upper plenum	20-24

The reactor vessel pressure loss for the core region (given in Table 4.4-4) was determined from the components test results, while the pressure losses for the remaining regions are based on the flow model results. Where appropriate, corrections were made to the flow model test results to account for differences in Reynolds number and surface relative roughness between model and reactor.

4.4.4.2.2 Components Testing

Components test programs have been conducted in support of all CE reactors. The tests subject a full-size reactor core module comprising one to five fuel assemblies, control rod assembly and extension shaft, control element, drive mechanism, and reactor internals to reactor conditions of water chemistry, flow velocity, temperature, and pressure under the most adverse operating conditions allowed by design. Two objectives of the programs are to confirm the basic hydraulic characteristics of the components and to verify that fretting and wear will not be excessive during the component's lifetime. When the reactor design is revised, a new program embodying the important aspects of the latest design is conducted.

Thus, components tests have been run on the Palisades design, with cruciform control rods, on the Fort Calhoun with CEAs and rack-and-pinion control element drive mechanisms (CEDM), on the Maine Yankee design with a dual CEA and a magnetic jack CEDM, and on the Arkansas design with a 16x16 fuel assembly, a CEA, and magnetic jack CEDM.

During the course of the tests, information is obtained on fuel rod fretting, on CEA/CEDM trip behavior, and on fuel assembly uplift and pressure drop. The first two subjects are discussed in Section 4.2. The third is discussed below.

As part of the assessment of fuel assembly margin to uplift in the reactor, measurements are made of the flow rate required to produce fuel assembly lift-off over a temperature range of 150 to 600°F at a system pressure of 350 to 2250 psia. To obtain the desired information, the point of fuel assembly lift-off is determined with load beams or lift-off conductivity probes. With the first appproach, one of the fuel assemblies of the module is mounted on the load beam so that the assembly net weight can be monitored as a function of flow rate and temperature. Fuel assembly lift-off is established when the net weight goes to zero. With the second approach, the lift-off probes are mounted to contact the bottom of the fuel assembly. When the fuel assembly is seated, the contact between the assembly and lift-off is indicated by a large step change in the circuit resistance, caused by the break in contact between the probes and the fuel assembly.

Data reduction involves the calculation of an uplift coefficient, describing the hydraulic uplift force acting on the assembly; the coefficient is defined as follows:

$$K_{up} = Wo/(\rho V^2 A/2 g_c)$$

where:	Wo	=	wet weight of assembly, lb.
	V	=	flow velocity in assembly at the point of lift-off, ft/sec.
	A	=	envelope area of assembly, ft^2
	ρ	=	water density, lb/ft^3.

A plot of the K_{up} data shows that they can be fitted by the relation:

$$K_{up} = \alpha \, N_R^{\;-\beta}$$

where α and β are peculiar to the particular components test being run and the standard error of estimate is typically four percent including replication and instrument error.

The uplift coefficient and its associated uncertainty are employed in the analysis of the uplift forces on the fuel assemblies in the reactor. The force is determined for the most adverse assembly location for startup and normal operating conditions. Additional input to the calculation includes analytical corrections to the coefficient for the absence of the CEA, for crud formation, and for small geometrical differences among the fuel assemblies for the different reactor designs all nominally describable by the same components test.

Pressure drop measurements are also made during the components test to verify the accuracy of the calculated loss coefficients for various fuel assembly components. Direct reduction of the pressure drop data yields the loss coefficients for the lower and upper fitting regions, while the spacer grid loss is evaluated by subtracting a calculated fuel rod friction loss from the measured pressure drop across the fuel rod region.

Experience has shown that the experimental end fitting loss coefficients are essentially independent of Reynolds Number and, with their sample standard deviations, are in reasonable agreement with the predicted values used in the calculation of core pressure drop (Subsection 4.4.2.6).

Evaluation of St. Lucie Unit 2 core pressure drop and fuel assembly uplift relationships is made with components test results for Arkansas fuel. Test results reduced to individual spacer grid and end fitting loss coefficients are directly applicable to the shorter St. Lucie Unit 2 fuel. The reactor fuel assembly uplift coefficient is derived by an analytical model which is matched to test results. The analytical model, proven for the Arkansas fuel, is applicable to St. Lucie Unit 2 fuel, with adjustments for the shorter St. Lucie Unit 2 fuel and the fewer spacer grids.

4.4.4.2.3 Core Pressure Drop Correlations

The total pressure drop along the fuel rod region of the core is computed as the sum of the individual losses resulting from friction, acceleration of the fluid, the change in elevation of the fluid, and spacer grids. The individual losses are computed using the momentum equation and the consistent set of empirical correlations presented in the TORC code(6).

In the following paragraphs, the correlations used are summarized and the validity of the scheme is demonstrated with a comparison of measured and predicted pressure drops for single-phase and two-phase flow in rod bundles with CEA-type geometry.

For isothermal, single-phase flow, the pressure drop due to friction for flow along the bare rods is based on the equivalent diameter of the bare rod assembly and the Blasius friction factor:

$$f = 0.184\ N_R^{-0.2}$$

The pressure drop associated with the spacer grids is computed using a grid loss coefficient (K_{SG}) given by a correlation which has the following form:

$$K_{SG} = D_1\ (N_R)\ D_2 \pm \text{Standard Error of Estimate}$$

The constants, D_n, are determined from pressure drop data obtained for a wide range of Reynolds Numbers for isothermal flow through a CEA-type rod bundle fitted with the high impact spacer grids. The data comes from a components test program on a 16x16 fuel assembly design (Subsection 4.4.4.2.2). The standard error of estimate associated with the loss coefficient relation includes replication and instrument error.

To compute pressure drop either for heating without boiling or for subcooled boiling, the friction factor given above for isothermal flow is modified through the use of the multipliers given in Pyle(13). It is important to recognize that the multipliers were developed in such a way as to incorporate the effects of subcooled voids on the acceleration and elevation components of the pressure drop as well as the effect on the friction losses. Consequently, it is not necessary to compute specifically either a void fraction for subcooled boiling or individual effects of subcooled boiling on the friction, acceleration, or elevation components of the total pressure drop.

The effect of bulk boiling on the friction pressure drop is computed using a curve fit to the Martinelli-Nelson data (14) above 2000 lb/in,^{2}a or the Martinelli-Nelson correlation(14) with the modification given in Pyle(13) below 2000 lb/in^{2}a. The acceleration component of the pressure drop for bulk boiling conditions is computed in the usual manner for the case of two-phase flow where there may be a nonunity slip ratio(15). The elevation and spacer grid pressure drops for bulk-boiling are computed as for single phase flow except that the bulk coolant density($\bar{\rho}$) is used, where:

$$\bar{\rho} = \alpha \rho_V + (1 - \alpha) \rho_\ell$$

and

α = bulk boiling void fraction

ρ_v = density of saturated vapor, lb/ft^3

ρ_ℓ = density of saturated liquid, lb/ft^3

The bulk boiling void fraction used in computing the elevation, acceleration, and spacer grid losses is calculated by assuming a slip ratio of unity if the pressure is greater than 1850 lb/in^2 absolute or by using the Martinelli-Nelson void fraction correlation[14] with the modifications presented in Pyle[13] if the pressure is below 1850 lb/in^2 absolute.

To verify that the scheme described above accurately predicts pressure drop for single-phase and two-phase flow through the 16x16 assembly geometry comparisons have been made of measured pressure drop and the pressure drop predicted by TORC,[6] for the rod bundles used in the DNB test program at Columbia University (refer to Subsection 4.4.4.1). Figure 6.7 of CENPD-161[6] shows some typical results for a 21 rod bundle of the 16x16 fuel assembly geometry (5x5 array with four rods replaced by a control rod guide tube). The excellent agreement demonstrates the validity of the methods described above.

4.4.4.2.4 Hydraulic Loads on Reactor Internals Components

Hydraulic loads were estimated on the basis of both experimental data from flow model tests on reactor configurations one thorough four and by analytical means. When experimental data are used, they are first reduced to dimensionless forms of a pressure difference coefficient,

$$E = \frac{P_{local} - P_{ref}}{\rho V^2_{ref}/2g_c}$$

a force coefficient,

$$C_F = \frac{F}{\rho V^2_{ref} A/2g_c}$$

or a velocity ratio,

$$\frac{V_{local}}{V_{ref}}$$

The quantities with subscript "ref" represent appropriate reference values: for example, the average velocity or pressure at the particular flow path station of interest. These dimensionless quantities are then converted to absolute quantities by multiplying by the appropriate reference quantity (i.e., by $V^2_{ref}\,2g_c$ or V_{ref}) for the reactor of interest.

Adjustment to the resulting absolute quantities are made by analytical means if there are substantial differences in geometry between the reactor configuration for which the test data were derived and the reactor configuration of interest.

4.4.4.2.5 Effect of Partial or Total Isolation of a Loop on Core Hydraulics

Power generation is only permitted with 4-pump operation and the "low flow trip" will shutdown the reactor if flow to the vessel is reduced due to a loss of one of the coolant pumps (See Section 7.2). However, part-loop operation is permitted during plant startup and shutdown. Hydraulic loads on various reactor internal components were computed for all part-loop configurations to assure the integrity of the components. These part-loop loads are included in the analysis of Subsection 4.4.2.6.3.

4.4.4.3 Influence of Power Distributions

The reactor operation is such that power distributions which are permitted to occur will have adequate margin to satisfy the design bases during anticipated operational occurrences. A discussion of the methods of controlling the power distributions is given in Subsection 4.3.2.4.2. A discussion of the expected power distributions is given in Subsection 4.3.2.2.3, and typical planar rod radial power factors and axial shapes are given in Figures 4.3-2 through 4.3-18. The full-power maximum rod radial power factor is taken as 1.63 and is used in the calculations of the core thermal margins which are given here in Section 4.4. Comparison with expected power distributions, discussed in Section 4.3, shows that this integrated rod radial power factor is at least five percent higher than all the calculated values and, therefore, is a meaningful value for thermal margin analyses.

If CEAs are inserted in the core, the same planar radial power distribution does not exist at each axial elevation of the core, nor does the same axial power distribution exist at each radial location in the core. From the analysis of many three-dimensional power distributions, the important parameters which establish the thermal margin in the core are the maximum rod power and its axial power distribution.[10] Examination of many axial power distributions shows the 1.26 peaked axial power distribution in Figure 4.4-3 to be among those giving the lowest DNBRs. The combination of that axial shape and the maximum rod radial power factor is a meaningful combination for DNB analyses. The maximum linear heat rate at a given power is determined directly from the core average fuel rod linear heat rate and the nuclear power factor. The value of 2.5 for the nuclear power factor is selected and corresponds to the 1.7 design maximum radial power factor combined with the 1.47 peaked axial shape shown on Figure 4.4-3. The RPS monitors the maximum rod radial power factor and the axial power distribution in the core and ensures that the design limits specified in Subsection 4.4.1 are not violated.

When fuel from regions prior to Region N are present in the core, more favorable power distributions than those used in the bounding thermal-hydraulic design will compensate for the effect of a reduction in inlet flow due to the GUARDIAN™ fuel. Evaluation of power distributions will be performed each transition cycle to confirm the more favorable power distribution assumption. Therefore, the bounding design is expected to be applicable to transition cycles. If the more favorable power distributions do not entirely compensate for the reduction in inlet flow to the GUARDIAN™ fuel for a transition cycle, an additional penalty will be applied to offset the uncompensated adverse effect due to the inlet flow reduction.

4.4.4.4 Core Thermal Response

Steady-state core parameters for Cycle 1 are summarized in Table 4.4-1 for normal four-pump operation. Figure 4.4-12 shows the sensivity of the minimum DNBR to small changes in pressure, inlet temperature, and flow from the conditions specified in Table 4.4-1. The 1.26 peaked axial power distribution and 1.63 maximum rod radial power factor are used.

The response of the core to plant transients is discussed in Chapter 15.

4.4.4.5 Analytical Methods

4.4.4.5.1 Reactor Coolant System Flow Determination

Upon completion of the manufacturing and testing of the pumps, the characteristic pump head or performance curve is established. The maximum allowable, best estimate, and minimum allowable Reactor Coolant System flow rates are determined as follows:

a) Best Estimate Expected Flow

The best estimate expected RCS flow is determined by equating the head loss around the reactor coolant flow path to the head rise supplied by the reactor coolant pumps (Subsection 5.4.1 has a description of the pumps).

b) Maximum Allowable Flow

The maximum allowable flow is defined as the upper flow rate limit on the expected flow rate probability distribution above which the actual flow rate has only a 2.3 percent probability of existing. The expected flow rate probability distribution is determined from the statistical combination of the respective pump curve uncertainties, the system resistance uncertainties and the flow measurement uncertainties.

c) Minimum Allowable Flow

The minimum allowable flow is also determined by using the expected flow rate probability distribution discussed above. The minimum allowable flow rate is defined as the lower flow rate limit on the expected flow rate probability distribution below which the actual flow rate has only a 2.3 percent chance of existing. This minimum allowable flow is used for the thermal margin analyses. The minimum allowable reactor coolant flow rate is 369,947 gpm as given in Table 4.4-1 for Cycle 1. This value corresponds to 139.4 x 10^6 lbs/Hr at the reactor inlet coolant temperature of 548°F. The minimum design reactor coolant flow rate of 122 x 10^6 lbs/Hr given in Table 5.1-1 is based upon four times the minimum pump design capacity or 4 x 81,200 = 324,800 gpm.

Upon installation of the pumps in the Reactor Coolant System, the operating flow is determined by one or more of the following flow measurement techniques:

a) Pump casing differential pressure method, using a correlation between pump casing differential pressure and flow rate

b) Calorimetric methods

The uncertainties included in the calculation of the operating flow are those uncertainties associated with the measurement technique or techniques used above. These uncertainties are statistically combined to give the overall uncertainty in reactor coolant flow as determined from onsite tests.

Any significant formation of crud buildup is detected by continuous monitoring of the Reactor Coolant System flow. A significant buildup of crud is not anticipated however, due to maintenance of proper water chemistry.

4.4.4.5.2 Thermal Margin Analysis

Discussions of methodologies within this section are written from an historical perspective and may have been superseded by newer methods as discussed in Section 4.3, Reference 65.

Thermal margin analyses of the reactor core are performed using the TORC code or the VIPRE-01 code which is an open core analytical method based on the COBRA-IIIC code[16]. A complete description of the TORC code and application of the code for detailed core thermal margin analyses is contained in CENPD-161.[6] A simplified procedure used to apply the TORC code for design thermal margin calculations is described in detail in CENPD-206[10]. A brief description of the code and its use is given here.

The COBRA-IIIC code solves the conservation equations for mass, axial and lateral momentum, and energy for a collection of parallel flow channels that are hydraulically open to each other. The size of a channel in the model can vary from the size of one or more fuel assembly down to the size of a subchannel within a fuel assembly. Therefore, certain modifications were necessary to enable a realistic analysis of thermal-hydraulic conditions in both geometries. The principal revisions to arrive at the TORC code, which leave the basic structure of COBRA-IIIC unaltered, are in the following areas:

a) Modification of the lateral momentum equation for core wide calculations where the smallest channel size is typically that of a fuel assembly.

b) Addition of the capability for handling non-zero lateral boundary conditions on the periphery of a collection of parallel flow channels. This capability is particularly important when analyzing the group of subchannels within the hot fuel assembly.

c) Addition of the capability to handle nonuniform core exit pressure distributions.

d) Insertion of standard CE empirical correlations and the ASME fluid property relationships.

Details of the lateral momentum equations and the empirical correlations used in the TORC code are given in CENPD-161[6].

The application of the TORC code for detailed core thermal margin calculations typically involves three stages. The first stage consists of calculating coolant conditions throughout the core on a coarse mesh basis. The core is modelled such that the smallest unit represented by a flow channel is a single fuel assembly. The three-dimensional power distribution in the core is superimposed on the core coolant inlet flow and temperature distributions. The core inlet flow and core exit static pressure distribution are obtained from flow model tests discussed in Subsection 4.4.4.2, and the inlet temperature for normal four-loop operation is assumed uniform. The axial distributions of flow and enthalpy in each fuel assembly are then calculated on the basis that the fuel assemblies are hydraulically open to each other. Also determined during this stage are the transport quantities of mass, momentum, and energy which cross the lateral boundaries of each flow channel.

In the second stage, typically the hot assembly and adjoining fuel assemblies are modelled with a coarse mesh. The hot assembly is typically divided into four to five partial assembly regions. One of these regions is centered on the subchannels adjacent to the rod having the minimum DNBR. The three-dimensional power distribution is superimposed on the core coolant inlet flow and temperature distributions. The lateral transport of mass, momentum, and energy from the stage one calculations is imposed on the peripheral boundary enclosing the hot assembly and the neighboring assemblies. The axial distributions of flow and enthalpy in each channel are calculated as well as the transport quantities of mass, momentum, and energy which cross the lateral boundary of each flow channel.

The third stage involves a fine mesh modelling of the partial-assembly region which centers on the subchannels adjacent to the rod having the minimum DNBR. All of the flow channels used in this stage are hydraulically open to their neighbors. The output from the stage two calculations, in terms of the lateral transport of mass, momentum, and energy is imposed on the lateral boundaries of the stage three partial assembly region. Engineering factors are applied to the minimum DNBR rod and subchannel to account for uncertainties on the enthalpy rise and heat flux due to manufacturing tolerances. The local coolant conditions are calculated for each flow channel. These coolant conditions are then input to the DNB correlation and the minimum value of DNBR in the core is determined.

A more detailed description of this procedure with example is contained in CENPD-161[6]. This procedure is used to analyze in detail any specific three-dimensional power distribution superimposed on an explicit core inlet flow distribution. The detailed core thermal margin calculations are used primarily to develop and to support the simplified design core thermal margin calculational scheme discussed below.

The method used for design calculations is discussed in detail in CENPD-206[10]. In summary, the method is to use one limiting hot assembly radial power distribution for all analyses, to raise or lower the hot assembly power to provide the proper maximum rod radial power factor, and to use the core average mass velocity in all fuel assemblies except the hot assembly. The hot assembly mass velocity is varied until the simplified thermal margin calculations produce DNBR results which are slightly conservative with respect to the detailed thermal margin calculations. This methodology is used in the thermal margin analyses.

The WCAP-9272 thermal margin analyses are performed using the VIPRE-01 (VIPRE) code (References 44 and 45) VIPRE is a three-dimensional subchannel code the accounts for hydraulic and nuclear effects on enthalphy rise in the core and hot channels. Conservation equations of mass, axial and lateral momentum, and energy are solved for the fluid enthalpy, axial flow rate, lateral flow and pressure drop. The NRC-approved VIPRE code is described in detail in Reference 44, including discussions on code validation with experimental data.

VIPRE modeling of a PWR core is based on the NRC-approved one-pass (or one-stage) modeling approach. In the one-pass modeling, hot channels and their adjacent channels are modeled in detail, while the rest of the core is modeled simultaneously on a relatively coarse mesh. The behavior of the hot assembly is determined by superimposing the power distribution upon inlet flow distribution while allowing for flow mixing and flow distribution between flow channels. Local variations in fuel rod power, fuel rod and pellet fabrication, and turbulent mixing are also considered in determining conditions in the hot channels. The VIPRE modeling method is discussed in Reference 45, including empirical models and correlations used.

The VIPRE core model is used with the applicable DNB correlations to determine DNBR distributions along the hot channels of the reactor core. The effect of crud on flow and enthalpy distribution in the core is not directly accounted for in the VIPRE evaluations. However, conservative treatment by the VIPRE modeling method has been demonstrated to bound this effect in DNBR calculations (Reference 45).

VIPRE is capable of transient DNB and/or fuel temperature analysis. The conservation equations in the VIPRE code contain the necessary accumulation terms for transient calculations. The VIPRE conduction rod model, as discussed in Reference 45, calculates transient temperature distributions of fuel rods and heat flux at the rod surface, based on core power and the local fluid conditions. The input to the conduction rod model includes pellet power profile, heat transfer correlations, and pellet-clad gap conductance. For post-DNB applications, heat addition due to zirconium-water reaction is also considered. The input description can include one or more of the following time-dependent parameters:

- Core inlet flow
- Core power
- Core pressure
- Core inlet temperature or enthalpy
- Core power distributions.

The VIPRE solution methods are generally fully implicit and have no time step size limitations for stability. However, solution instability could occur in transient calculations using a subcooled void model that was developed from steady state data. In order to avoid any numerical instability, appropriate time steps and axial nodes were selected in the transient DNBR calculations to ensure the Courant number greater than one when a subcooled void model is used. The Courant number is defined as:

$$Nc = u\Delta t/\Delta x$$

where Nc = Courant number
u = flow velocity
Δt = time step
Δx = axial node size.

4.4.4.5.3 Hydraulic Instability Analysis

Flow instabilities leading to flow excursions or flow oscillations have been observed in some boiling flow systems containing one or more closed, heated channels. Flow instability phenomena are a concern primarily because they may lead to a reduction in the DNB heat flux relative to that observed during a steady flow condition. Flow instabilities are not, however, expected to reduce thermal margin during normal operation or anticipated operational occurrences. This conclusion is based upon available literature, experimental evidence and the results of core flow stability analyses.

Review of the available information on boiling systems has resulted in the following qualitative observations. Flow instabilities which have been observed have occurred almost exclusively in closed channel systems operating at pressures low relative to PWR operating pressures. Increasing pressure has been found to have a stabilizing influence in many cases where flow instabilities have been observed[18] and the high operating pressure characteristic of PWR's minimizes the potential for flow instability. For PWR operating pressures, experimental results[19] have shown that, even with closed channel systems, operating limits due to the occurrence of critical heat flux (CHF) are encountered before the flow stability threshold is reached. It would be expected that the low resistance to coolant crossflow among subchannels of fuel assemblies would have a stabilizing effect, and that expectation is confirmed by experimental results[17][21][22] which show that flow stability in parallel heated channels is enhanced by cross connections between the channels.

Experimental evidence that flow instabilities will not adversely affect thermal margin is provided by the data from the rod bundle DNB tests conducted by CE[1][2]; many rod bundles have been tested over wide ranges of operating conditions with no evidence of premature DNB or of inconsistent data which might be indicative of flow instabilities in the rod bundle.

Analytical support for the conclusion that flow instabilities will not reduce the thermal margin is provided in Reference 23. That document presents an assessment of core flow stability. The assessment was made using the CE-HYDNA code, the CE version of the HYDNA flow stability code presented in Reference 12. In addition to the flow stability assessment, Reference 23 contains:

a) A description of the CE-HYDNA flow stability code.

b) a user's manual and Fortran listing of the CE-HYDNA code.

c) Results of sensitivity studies and of code verification through comparison with experimental data.

The CE-HYDNA code provides the fundamental analytical tool for the assessment of flow stability. The code has the capability of analyzing transient one-dimensional flow phenomena in several groups of laterally closed channels with common entrance and exit plenums. The use of CE-HYDNA for analysis of open-array cores is conservative because the stabilizing effects of inter-channel communication[17,21,22] are neglected.

The results presented in Reference 23 are for a 3450 MWt class reactor but those results are representative for St. Lucie Unit 2. It was found that, for nominal coolant conditions, the flow is stable throughout the range of reactor power levels examined (100 percent - 250 percent rated power). Additional calculations were performed covering a wide range of operating conditions. These calculations showed that, even under severely adverse operating conditions, the flow is stable at greater than 100 percent of rated power. These results provide additional evidence that flow instabilities will not adversely affect core thermal margin during normal operation or anticipated operational occurrences.

4.4.5 TESTING AND VERIFICATION

Data descriptive of thermal and hydraulic conditions within the reactor vessel will be obtained as part of the startup program. These will include hot and cold leg temperature, loop flowrates, and core power distributions. The data will be evaluated and compared with design calculations and parameters to assure that the reactor thermal and hydraulic behavior is as predicted.

4.4.6 INSTRUMENTATION REQUIREMENTS

The incore instrumentation is used to confirm core power and temperature distributions and to perform periodic calibrations of the excore flux measurement system. Further descriptions are contained in Section 7.7.

4.4.6.1 <u>Loose Parts Monitoring System</u>

4.4.6.1.1 Design Bases

St. Lucie Unit 2 is provided with a loose parts monitoring system (LPMS) that is permanently installed to fulfill the in-service monitoring function during plant operation.

The LPMS monitors the reactor coolant system (RCS) for internal loose parts. The system is designed to detect a loose part striking the internal surface of RCS components with an energy level of one-half foot pound or more within 3 feet of one of the eight sensors.

4.4.6.1.2 Design and Operation

The LPMS consists of transducers, preamplifiers, a computer and readout instrumentation, including a flat panel display to automatically detect and record the occurrence of a loose part within the RCS.

Eight piezoelectric accelerometers are installed externally on the Reactor Coolant System:

a) Two at each steam generator
b) Two at the head of the reactor vessel
c) Two at the bottom of the reactor vessel

The accelerometer signals are individually processed within the containment for wire transmission to an alarm and recording console located in the control instrumentation area. When a loose part is detected, the data acquisition system automatically activates and an audible as well as visual alarm to alert the control room operator. Alarm indication for each accelerometer, and simultaneous recording of signals from all accelerometers is provided at the console. All channels can be recorded and compared to historical data.

Additionally, an alarm indication of a loose part or critical-failure of the system at the loose parts monitor console is also indicated on a control room annunciator. The system will activate the annunciator alarm channel whenever a critical failure or loose part is detected.

A training program is established to train operators in the Operations Department as well as the Instrumentation and Controls Maintenance Department in operation of the Loose Parts Monitoring System.

4.4.6.1.3 Guidelines of Regulatory Guide 1.133

The LPMS is able to detect a metallic loose part that weighs from 0.25 lb to 30 lb and impact with the kinetic energy of 0.5 ft-lb on the inside surface of the reactor coolant pressure boundary within 3 feet of a sensor. Additionally, the system employs High Pass, Low Pass and Algorithms to filter out non-metal to metal impact to minimize the alarms from vibrations that are not loose parts.

Two sensors in each location provide separate signals to one charge amp. Two separate outputs from the charge amp, located outside the Bio-Shield, are transmitted through one cable to the LPMS Cabinet. Separation is required only to a point that is accessible during full power operation which is maintained from the preamp to the sensors.

The LPMS has both an automatic and manual startup to data acquisition equipment. When a loose part is detected, the data acquisition system automatically activates an audible as well as visual alarm to alert the control room operator. This system also provides for recording of all sensor signal waveforms in digital form. Upon alarm, the system defaults to recording the alarming channel as well as the three nearest sensors. The system is capable of immediate visual and audio monitoring of all signals.

The LPMS system provides an alarm for a loose part and alert level for any noise or adverse system condition that is not a loose part. The capability of the system exceeds the Regulatory Guide requirements by using alternate means of loose parts determination. All channels can be recorded and compared to historical data using the Waveform Software.

A test signal is provided to the preamp to initiate an alarm condition for checking operation of the system to the maximum extent practicable.

The portion of the system located inside the containment is designed for high temperature and humidity, is resistant to the effects of radiation and will function after a design basis earthquake. The processor is a heavy duty, industrial chassis built to withstand vibration and the environment of the Control Room.

The components supplied with the LPMS are of the highest quality available.

Self diagnostics that are a design feature of the LPMS assist in finding problems quickly. Access to the control room equipment is from the rear of the cabinet and any component can be repaired or replaced easily with plug in cables and rack style screw mounting. There are two sensors in each location, one is a backup for more reliable system operation since there is no access to the sensors during operation. The pre-amps are located outside the Bio-Shield which allows access under some at power conditions.

to effectively operate the system.

To insure that each channel of the loose parts monitoring system is operable, a channel check and channel functional test are performed on a periodic basis as described in Section 13.7 of the UFSAR. The channel check provides a qualitative assessment of channel behavior during operation by observation. The channel functional test provides the means for verifying the operability of the channel including alarm functions by injection of a simulated signal into the channel as close as practicable to the accelerometer. In addition, channel calibration is performed periodically on the loose parts monitoring system as described in Section 13.7 of the UFSAR.

Finally the plant operations personnel are provided with instruction in the operation of the loose parts monitoring system during their formal on-shift training watches. This instruction familiarizes the control room licensed operators with the fundamentals of operation of the equipment. With this instruction, the operators are prepared to respond to system alarms in a manner consistent with safe operation.

Plant operating procedures govern the actions of the control room operators when an alarm occurs on the loose parts monitor. The plant annunciator procedure requires the control room operator to acknowledge any alarm on the loose parts monitor and then notify the Instrumentation and Control department of the alarm. The I & C department, when notified by operations of a loose parts monitor alarm, investigates whether the alarm is spurious or valid. If the alarm is valid, as certified by vendor manual instructions, the I & C Department will notify the NSSS vendor and request assistance to diagnose the problem. The I & C department action to resolve identified problems with the loose parts monitor is governed by site procedures, which require that the I & C department correct the identified discrepancy.

This comprehensive program of supervised on-the-job training and vendor manual instruction provides assurance that the loose parts monitoring system will be tested, aligned and operated properly. The assistance of specialized offsite personnel will provide St. Lucie Unit 2 with the technical capability to identify and locate potential problems involving loose parts in the reactor coolant system.

SECTION 4.4: REFERENCES

1. "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," September, 1976, CENPD-162-P-A (proprietary), CENPD-162-A (non-proprietary).

2. "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 2, Non-Uniform Axial Power Distributions," June 1976; CENPD-207-P (proprietary), CENPD-207 (non-proprietary).

3. Tong, L. S., "An Evaluation of the Departure from Nucleate Boiling in Bundles of Reactor Fuel Rods," Nuclear Science and Engineering, Vol 33, pp. 7 to 15, 1968.

4. Letter from Olan O. Parr of NRC to A. E. Scherer of C-E dated May 5, 1976 concerning NRC review of CENPD-162.

5. Tong, L. S., "Boiling Crisis and Critical Heat Flux," AEC Critical Review Series, 1972.

6. "TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core," July 1975; CENPD-161-P (proprietary), CENPD-161 (non-proprietary).

7. Maurer, G. W., "A Method of Predicting Steady-State Boiling Vapor Fractions in Reactor Coolant Channels," Bettis Technical Review, WAPD-BT-19, Reactor Technology, June 1960.

8. Dittus, F. W. and Boelter, L. M. K., "Heat Transfer in Automobile Radiators of the Tubular Type," University of California Publication in Engineer Vol 2, No. 13, pp 443-461, 1930.

9. Jens, W. R. and Lottes, P. A., "Analysis of Heat Transfer, Burnout, Pressure Drop, and Density Data for High Pressure Water," ANL-4627, May 1, 1951.

10. "TORC Code Verification and Simplified Modeling Methods," CENPD-206-P (Proprietary), CENPD-206 (Non-Proprietary), January, 1977.

11. "Coast Code Description," approval letter from O. Parr (NRC) to F. Stern (C-E), December 4, 1974, C-E Topical Report, CENPD-98 April 1973.

12. Currin, H. B., et al, "HYDNA-Digital Computer Program for Hydrodynamic Transient in a Pressure Tube Reactor or a Closed Channel Core," Report CVNA-77, 1961.

13. Pyle, R. S., "A Program for the Thermal Analysis of a Pressurized Water Nuclear Reactor During Steady-State Operation, STDY-3," WAPD-TM-213, June 1960.

SECTION 4.4: REFERENCES (Cont'd)

14. Martinelli, R. C. and Nelson, D. B., "Prediction of Pressure Drop During Forced Circulation Boiling of Water," ASME Transactions, August, 1948.

15. Mendler, O. J., et al., "Natural Circulation Tests with Water at 800 to 2000 psia under Non-Boiling, Local Boiling, and Bulk Boiling Conditions," Journal of Heat Transfer, August 1961.

16. Rowe, D. S., "COBRA-IIIC: A Digital Computer Program for Steady-State and transient Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements," BNWL-1695, March 1973.

17. Veziroglu, T. N. and Lee, S. S., "Boiling-Flow Instabilities in a Cross-Connected Parallel-Channel or Flow System," ASME Paper 71-HT-12, August 1971.

18. Boure, J. A., Bergles, A. E., and Tong, L. S., "Review of Two-Phase Flow Instability," ASME Paper 71-HT-42, August 1971.

19. Kao, H. S., Morgan, C. D., and Parker, M. B., "Prediction of Flow Oscillation in Reactor Core Channel," ANS Transactions, Vol 16, P 212, 1973.

20. "Fuel and Poison Rod Bowing," Combustion Engineering Topical Report CENPD-225, October 1976.

21. Kakac, S., et al., "Transient Boiling Flow Instabilities in a Four Parallel Channel Upflow System," Proceedings of Condensed Papers of the Two-Phase Flow and Heat Transfer Symposium-Workshop, October 18-20, 1976, pp. 175-181.

22. RESAR-3S, Volume 2, Section 4.4.3.5, July 1975.

23. Letter to J. F. Stolz, USNRC, from D. H. Williams, Arkansas Power & Light Co., January 16, 1978, and enclosure, "Assessment of Core Flow Stability for C-E PWR's," CEN-64(A)-P (Proprietary and CEN-64(A), (Non-proprietary), July, 1977.

24. "C-E Fuel Evaluation Model Topical Report," Combustion Engineering, Inc., CENPD-139-A (Proprietary), CENPD-139-A (Non-Proprietary) CENPD-139 Supplement 1 (Proprietary), CENPD-139 Supplement 1, Rev. 01 (Non-Proprietary), CENPD-139-P-A (Proprietary), CENPD-139-A (Non-Proprietary) July, 1974.

25. Letter A. E. Scherer, Combustion Engineering, Inc. from D. B. Vassallo US NRC, June 12, 1978, and enclosure, "Staff Review of Fuel Rod Bowing Models".

26. CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981.

27. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology," Parts 1, 2, and 3, December 1979, January 1980, February 1980, respectively.

28. CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1983.

29. CEN-124(B)-P, "Statistical Combination of Uncertainties, Part 1," January 1980.

30. CEN-124(B)-P, "Statistical Combination of Uncertainties, Part 2," January 1980.

31. CEN-124(B)-P, "Statistical Combination of Uncertainties, Part 3," March 1980.

32. CEN-371(F)-P, "Extended Statistical Combination of Uncertainties (ESCU)," July 1989.

33. CENPD-161-P-A. "TORC Code. A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986.

34. CENPD -206-P-A. "TORC Code. Verification and Simplified Modeling Methods," June 1981.

35. CEN-348 (B)-P-A, Supplement 1-P-A, "Extended Statistical Combination of Uncertainties," January, 1997.

36. Letter, Robert a. Clark (NRC) to William Cavanaugh III (AP&L), "Operation of ANO-2 During Cycle 2," July 21, 1981 (Safety Evaluation Report and License Amendment No. 26 for ANO-2).

37. NUREG-0787, Supplement 1, "Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station, Unit No. 3," Docket No. 50-382, October 1981.

38. CENPD-207-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 2, Non-uniform Axial Power Distribution," December 1984.

39. Docket No. 50-368, CEN-289(A)-P, "Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2," December 1984.

40. Letter, Robert S. Lee (NRC) to John M. Griffin (AP&L), Enclosure 2, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 66 to Facility Operating License No. NPF-6, Arkansas Power & Light Company, Arkansas Nuclear One, Unit 2, Docket No. 50-368," May 7, 1985.

41. WCAP-9272-P-A/WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

42. CENPD-387-P-A, Rev. 0, "ABB Critical Heat Flux Correlations for PWR Fuel," May 2000.

43. WCAP-14565-P-A, Addendum 1, "Addendum 1 to WCAP-14565-P-A Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," May 2003.

44. Stewart, C.W. et al., "VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores," Volume 1-3 (Revision 3, August 1989), Volume 4 (April 1987), NP-2511-CCM-A, Electric Power Research Institute.

45. WCAP-14565-P-A/WCAP-15306-NP-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," October 1999.

46. WCAP-11397-P-A, "Revised Thermal Design Procedure," April 1989.

47. AREVA Document 77-5069878-01, Replacement Steam Generator Report for Florida Power and Light St. Lucie Unit 2, February 2007.

TABLE 4.4-1

THERMAL AND HYDRAULIC PARAMETERS

Reactor Parameters	St. Lucie Unit 2, Cycle 1	St. Lucie Unit 1, Cycle 3*
Core Average Characteristics at Full Power.		
Total core heat output, MWt	2560	2560
Total core heat output, million Btu/h	8737	8737
Average fuel rod energy deposition fraction	0.975	0.975
Hot fuel rod energy deposition fraction	0.975	0.975
Pressurizer pressure, PSIA	2250	2250
Reactor inlet coolant temperature, F	548	542
Reactor outlet coolant temperature, F	596	590
Core exit average coolant temperature, F	598	592
Average core enthalpy rise, Btu/lbm	65.1	64.6
Minimum allowable reactor coolant flow rate, gal/min	369947	369947
Maximum allowable reactor coolant flow rate, gal/min	424350	424350
Maximum core bypass flow, % of reactor coolant flowrate	3.7	3.7
Minimum allowable core flow rate, gal/min	356,259	356,259
Hydraulic diameter of nominal sub-channel, in.	0.4714	0.5334

* Source of data: Letter from RE Uhrig (FP&L) to V Stello (US NRC) dated February 22, 1979, L-79-45, "Proposed Amendment to Facility Operating License."

TABLE 4.4-1 (Cont'd)

Reactor Parameters	St. Lucie Unit 2, Cycle 1	St. Lucie Unit 1, Cycle 3*
Core flow area, ft^2	54.70	53.5
Core avg mass velocity, million $lbm/h\text{-}ft^2$	2.45	2.53
Core avg coolant velocity, ft/s	15.1	15.4
Core avg fuel rod heat flux $Btu/h\text{-}ft^{2}$ (1)	151,300	174,400
Total heat transfer area, ft^2	56,315	48,860

(1) Includes factor on fuel densification, does not include heat generated in moderator.

* Source of data: Letter from RE Uhrig (FP&L) to V Stello (US NRC) dated February 22, 1979, L-79-45, "Proposed Amendment to Facility Operating License".

TABLE 4.4-1 (Cont'd)

Reactor Parameters	St. Lucie 2 Cycle 1	St. Lucie Unit 1*
Average fuel rod linear heat rate kW/ft	4.43	6.00
Power density, kW/liter	78.7	9.3
No. of active fuel rods	49,580	36,896
Power Distribution Factors:		
Rod radial power factor	1.70	1.70
Nuclear power factor	2.50	2.71
Total heat flux factor	2.57	2.85
Maximum augmentation factor	1.05	1.06
Maximum gap length, in.	1.80	2.67
Engineering Factors:		
Engineering heat flux factor	1.03	1.05
Engineering enthalpy rise factor	1.03	1.05
Pitch and clad diameter allowance	1.05	1.065
Engineering factor on linear heat rate	1.03	1.03
Fuel densification factor	1.002	1.01
Characteristics of Rod and Channel with Minimum DNBR:		
Maximum fuel rod heat flux, Btu/h-ft^2	388,800	507,000
Maximum fuel rod linear heat rate, kW/ft	11.4	17.1
UO_2 maximum steady state temperature, ^oF	2986	3805
Outlet temperature, ^oF	622	639
Outlet enthalpy, Btu/lbm	646	674
Minimum DNBR at nominal conditions	2.64 (CE-1)	2.37 (orginal W-3)

* Source of data: Letter from RE Uhrig (FP&L) to V Stello (US NRC) dated February 22, 1979, L-79-45, Proposed Amendment to Facility Operating License".

TABLE 4.4-2

COMPARISON OF THE DEPARTURE FROM NUCLEATE BOILING RATIOS COMPUTED WITH DIFFERENT CORRELATIONS

	DNBRs for Nominal Reactor Conditions		DNBRs for Reactor Conditions Giving a 1.19 CE-1 Minimum DNBR in Matrix Subchannel	
Correlation	Matrix Subchannel	Subchannel Next to Guide Tube	Matrix Subchannel	Subchannel Next to Guide Tube
CE-1	3.06	2.64	1.19	1.15
Original W-3 (3)	3.47	3.59	1.18	1.25
Revised W-3 (5)	3.47	2.95	1.17	1.25
B&W-2 (5)	3.79	3.94	1.47	1.71

TABLE 4.4-3

REACTOR COOLANT FLOWS IN BYPASS CHANNELS

Bypass Route	Percent of Total Reactor Flow
Outlet nozzle clearances	0.53
Alignment keyways	0.09
Core shroud annulus	0.63
Guide tubes	1.29
Total bypass	2.54

Note: Bypass flow rate calculations are based on the minimum design reactor flow.

TABLE 4.4-4

REACTOR VESSEL BEST ESTIMATE
PRESSURE LOSSES AND COOLANT TEMPERATURES

Component	Pressure Loss Psi	Temperature (F)
Inlet nozzle and 90° turn	5.0	546
Downcomer, lower plenum, and support structure	10.4	546
Fuel assembly	13.4	574
Fuel assembly outlet to outlet nozzle	6.7	600
Total Pressure Loss	35.5	

TABLE 4.4-5

DESIGN STEADY STATE HYDRAULIC LOADS ON VESSEL INTERNALS AND FUEL ASSEMBLIES

Load Values at 500 F and at Original Maximum Design Reactor Coolant Flow

Component	Steady State Load Description	Load Value
1. Core support barrel	Radial pressure differential directed inward opposite inlet nozzle	58 lb/in^2
	Uplift load	744000 lb
	Lateral load	189000 lb
2. Upper guide structure	Uplift load	358000 lb
	Lateral load	59000 lb
3. Combined core support barrel and upper guide structure	Uplift load	865000 lb
4. Flow skirt	Radial load directed inward	6000 max. lb/ft of circ 2900 avg. lb/ft of circ
	Axial load directed downward	1200 max. lb/ft of circ 600 avg. lb/ft of circ
5. Bottom plate	Drag load directed upward	49000 lb
6. Core support plate	Drag load directed upward	58000 lb
7. Fuel alignment plate	Drag load directed upward	99000 lb
8. Upper guide plate	Load directed downward	53000 lb
9. Fuel assembly	Uplift load	1900 lb
10. Core shroud	Radial pressure differential directed outward	22 lb/in^2 at bottom 0 lb/in^2 at top
11. CEA shrouds	Lateral drag load	4500 lb, max. tube
12. Instrumentation tube	Lateral drag load	750 lb, max. tube

TABLE 4.4-6

RCS VALVES AND PIPE FITTINGS

PRESSURE BOUNDARY VALVES

Valve	Valve No.	Size In.	Type
Pressurizer safety valves	V1200, V1201 V1202	3 x 6	Safety
Pressurizer power operated relief valves	V1402, V1404	2.5 x 4	Relief
Pressurizer power operated relief isolation valves (upstream/downstream)	V1403, V1405/ V1406, V1407	2.5/4	Gate/Gate
Pressurizer spray valves	PCV1100 E, PCV1100 F	3	Angle
Pressurizer spray bypass valves	V1453, V1454	0.75	Angle
Pressurizer spray isolation valves	V1441, V1442, V1443, V1444	3	Gate
Pressurizer spray check valves	V1248, V1249	3	Check
Pressurizer vapor space sample isolation valve	V1238	0.75	Globe
Pressurizer vent isolation valve	V1239	0.75	Globe
Pressurizer auxiliary spray line isolation valve	V2483	2	Gate
Surge line sample isolation valve	V1210	0.75	Globe
Refueling water level indicator connection isolation valve	V1214	0.75	Globe
Reactor vessel O-ring leak detector connection isolation valve	V1211	0.75	Globe
Reactor leak detector line drain valves	V1392, V1393	0.75	Globe
Reactor vessel head vent isolation valve	V1212	0.75	Globe

TABLE 4.4-6 (Cont'd)

Valve	Valve No	Size In.	Type
Hot leg sample isolation valve	V1213	0.75	Globe
Letdown line isolation valve	V2593	3	Gate
Charging line isolation valves	V2484, V2485	2	Gate
Hot leg shutdown cooling line isolation valves	V3480, V3652	10	Gate
Hot leg injection isolation valves	V3525, V3527	3	Check
Cold leg safety injection and shutdown cooling line isolation valves	V3217, V3227, V3237, V3247	12	Check
Hot leg drain isolation valves	V1215, V1247	2	Globe
Cold leg drain isolation valves	V1449, V1233, V1450, V1235	2	Globe

RCS PIPE FITTINGS

Elbows	Size (In.)	Radius (In.)	Quantity
35°	42 I.D.	63	2
45°	30 I.D.	45	4
60°	30 I.D.	45	6
90°	30 I.D.	45	4
32° 47'-25"	30 I.D.	45	2
90°	12 Sch. 160	Long Radius	2
80° 38'-4"	12 Sch. 160	Long Radius	1

TABLE 4.4-6 (Cont'd)

RCS PIPE NOZZLES

Nozzles	Size (In.)	Schedule	Quantity
Surge	12	160	1
Shutdown cooling outlet	12	160	2
Safety injection and shutdown cooling inlet	12	160	4
Pressurizer Spray	3	160	2
Charging inlet	2	160	2
Letdown	2	160	1
Drain	2	160	4
Sampling	3/4	160	2
Flow measurement	3/4	160	8
RTD penetration	---	---	23

TABLE 4.4-7

RCS FLOWRATES

Cycle 1 1

Flow Path	Flow (lbm/hr)*
Total RCS flow	139.4×10^6
Core bypass flow	5.1×10^6
Core flow	134.3×10^6
Hot leg flow	69.7×10^6
Cold leg flow	34.85×10^6

*T inlet = 548 F

TABLE 4.4-8

REACTOR COOLANT SYSTEM GEOMETRY

Component	Flow Path Length (ft)	Top Elevation (ft) (d)	Bottom Elevation (ft) (d)	Minimum Flow Area (ft^2)	Volume (ft^3)
Hot Leg	14.53	2.38	-1.75	9.62	139.81
Suction Leg	22.83	1.04	-7.25	4.91	112.07
Discharge Leg					
Parallel	16.39	1.25	-1.25	4.91	80.46
Non-parallel	16.42	1.25	-1.25	4.91	80.52
Reactor Coolant Pump	22.81	1.25	-1.79	4.91[f]	112
Pressurizer	-----	47.20	10.83	----	1500
Liquid level	-----	30.66	10.83	50.07[a]	800
Surge Line	54.51	10.83	1.75	0.56	29.30
Steam Generator					
Inlet nozzle (ea)	2.23	2.24	0.95[e]	9.62	21.77
Inlet plenum	4.64	6.91	0.36	61.04	342.94
Tubes (active and passive)	56.81	37.65	6.91	0.0024[c]	1247.71
Outlet plenum	5.50	6.91	0.36	61.04	337.95
Outlet nozzle (ea)	1.72	1.39	0.16[e]	4.91	8.58
Reactor Vessel					
Inlet nozzles	3.6	1.5	-1.5	4.9	78
Downcomer	20.9	1.5	-20.9	30.3	674
Lower plenum	6.4[b]	-20.9	-27.0	43.7	702
Lower support structure and inactive core	3.5	-17.4	-20.9	28.0	473
Active core	11.4	-6.0	-17.4	54.8	669
Upper inactive core	1.5	-4.5	-6.0	47.1	85
Outlet plenum	11.0[g]	2.0	-4.5	23.45	524
CEA shroud	12.2[g]	9.8	2.0	-----	430
UGS annulus, outside CEA shroud	4.5	6.5	2.0	-----	122
Top Head	-----	13.0	6.5	-----	753
Outlet nozzles	4.1	2.0	-2.0	9.62	105

TABLE 4.4-8 (Cont'd)

Notes:

(a) For the cylinder

(b) Represents a geometrical rather than an actual flow path length

(c) Flow path area per tube

(d) Reactor vessel nozzle centerline is the reference elevation; it has an elevation of 0.0 ft.

(e) Nozzle centerline

(f) RCP outlet

(g) Approximate flow path length

TABLE 4.4-9

SAFETY INJECTION LINE LENGTHS AND SIZES

(x) - (y)[a]	Length (ft.)	Size (in., nominal)
1 - 2	62	12
2 - 3	62	12
2 - 4	240	6
5 - 6	52	12
6 - 7	69	12
6 - 8	140	6
9 - 10	64	12
10 - 11	85	12
10 - 12	87	6
13 - 14	55	12
14 - 15	53	12
14 - 16	34	6
17 - 18	291	3
19 - 20	90	3
4 - 23	106	6
23 - 24	8	3
8 - 21	108	6
21 - 22	8	3
12 - 27	140	6
27 - 28	8	3
16 - 25	143	6
25 - 26	9	3
21 - 29	90	6
23 - 29	59	6

TABLE 4.4-9 (Cont'd)

(x) - (y)[a]	Length (ft.)	Size (in., nominal)
29 - 30	71	10
25 - 31	62	6
27 - 31	57	6
31 - 32	77	10
28 - 33	49	2
26 - 34	23	2
22 - 35	17	2
24 - 36	20	2
22 - 39	68	2
24 - 40	71	2
26 - 41	20	2
28 - 42	12	2
33 - 34	23	4
39 - 40	20	4
34 - 35	2	6
35 - 36	3	6
36 - 37	14	6
37 - 38	9	3
40 - 41	1.5	6
41 - 42	1.5	6
42 - 43	20	6
43 - 44	9	3
18 - 37	136	3
20 - 43	162	3
45 - 50	42	14

TABLE 4.4-9 (Cont'd)

(x) - (y)[a]	Length (ft.)	Size (in., nominal)
46 - 51	67	14
47 - 52	27	6
48 - 49	31	6
49 - 53	6	8
50 - 57	36	24
57 - 58	55	24
57 - 59	203	24
51 - 53	23	24
53 - 55	50	24
55 - 54	190	24
55 - 56	110	24
60 - 62	6	3
61 - 40	10	3

a. The designation (x) - (y) indicates the sections 1-2, 21-22, etc. for the line lengths and sizes as indicated on Figures 6.3-1a and 6.3-1b.

TABLE 4.4-10

REACTOR COOLANT SYSTEM COMPONENT THERMAL AND HYDRAULIC DATA[a]

Cycle 1 | 1

Component	Data
Reactor Vessel	
Rated core thermal power, MWt	2560
Design pressure, psia	2500
Operating pressure, psia	2250
Design temperature, F	650
Nominal Coolant outlet temperature, F	596
Nominal Coolant inlet temperature, F	548
Coolant outlet state	Subcooled
Minimum Allowable Reactor Coolant Flow, lbm/hr	139.4×10^6
Core average coolant enthalpy	
Inlet, Btu/lb	545
Outlet, Btu/lb	610
Average coolant density	
Inlet, lb/ft^3	47.0
Outlet, lb/ft^3	43.4
Steam Generators	
Number of units	2
Reactor Coolant side (tube side)	
Design pressure/temperature, psia/F	2500/650
Operating pressure, psia	2250
Design Inlet temperature, F	604
Design Outlet temperature, F	550
Design Flow rate, 10^6 lb/hr	61
Main Steam side (shell side)	
Design pressure/temperature, psia/F	1000/550
Operating pressure/temperature, psia/F (warranted)	815/520.3
Total steam flow per generator, 10^6 lb/hr	5.603

TABLE 4.4-10 (Cont'd)

Component	Data
Steam quality, %	99.8
Feedwater temperature, F	435
Design tube sheet differential pressure, psi	2250
Pressurizer	
Design pressure/temperature, psia/F	2500/700
Operating pressure/temperature, psia/F	2250/652.7
Internal volume, ft^3	1500
Operating water volume, ft^3	800
Heater capacity, KW	1500
Spray rate	
continuous, gpm	1.5
maximum, gpm	375
Reactor Coolant Pumps	
Number of units	4
Design pressure/temperature, psia/F	2500/650
Operating pressure/temperature, psia/F	2250/550
Rated capacity, gpm	81,200
Rated head (hot, 550 F), ft	310
Pump type	Centrifugal
Motor type	Squirrel cage induction
Power requirement, hp (cold)	6500 (1.15 service factor)
Reactor Coolant Piping	
Design Flow per loop, 10^6 lb/hr	
Hot leg	61
Cold leg	30.5

TABLE 4.4-10 (Cont'd)

Component	Data
Pipe size (inside diameter), in.	
Hot leg	42
Suction leg (cold leg)	30
Discharge leg (cold leg)	30
Design pressure/temperature, psia/F	2500/650
Operating pressure/temperature, psia/F	
Hot leg	2250/604
Cold leg	2250/550

a. Full power conditions

Table 4.4-11

St. Lucie Unit 2
Thermal Hydraulic Parameters at Full Power*++++@

General Characteristics	Units	EPU	30% SGTP & WCAP-9272+++	Reference++ or Bounding Cycle
Total Heat Output (Core Only)	MWt	3030	2700	2700
	10^6 Btu/hr	10339	9215	9215
Fraction of Heat Generated in Fuel Rod	--	0.975	0.975	0.975
Primary System Pressure (Nominal)	psia	2250	2250	2250
Inlet Temperature (Maximum Indicated)	°F	551	549	550
Total Reactor Coolant Flow (Minimum Steady State)	gpm	375,000	335,000	363,000
	10^6 lbm/hr	140.7	126.1	136.4
Bypass Flow (Maximum for Minimum Core Flow)	10^6 lbm/hr	5.2	4.7	5.0
Coolant Flow Through Core (Minimum)	10^6 lbm/hr	135.5	121.4	131.4
Hydraulic Diameter (Nominal Channel)	ft	0.039	0.039	0.039
Average Mass Velocity	10^6 lbm/hr-ft^2	2.47	2.21	2.40
Pressure Drop Across Core (Minimum Steady State Flow Irreversible Over Entire Fuel Assembly)	psi	14.3	11.6	13.4
Total Pressure Drop Across Vessel (Based on Nominal Dimensions And Minimum Steady State Flow)	psi	37.8	30.1	35.4
Core Average Heat Flux (Accounts for Fraction of Heat Generated in Fuel Rod and Axial Densification Factor)	Btu/hr-ft^2	173,634***	154,515**	154,723***
Total Heat Transfer Area (Accounts for Axial Densification Factor)	ft^2	58,055***	58,131**	58,055***
Film Coefficient at Average Conditions	Btu/hr-ft^2 - °F	5984	5700	5800
Average Film Temperature Difference	°F	29.0***	27.1**	26.7***
Average Linear Heat Rate of Undensified Fuel Rod (Accounts for Fraction of Heat Generated in Fuel Rod)	kw/ft	5.07***	4.52**	4.52***
Average Core Enthalpy Rise	Btu/lbm	76.3	75.9	70.1
Maximum Clad Surface Temperature	°F	656.7	656.5	656.6
Engineering Heat Flux Factor	----	1.032+	1.032+	1.032+
Engineering Factor on Hot Channel Heat Input	----	1.030+	1.030+	1.030+
Rod Pitch, Bowing and Clad Diameter Factor	----	1.05+	1.05+	1.05+
Fuel Densification Factor (Axial)	----	1.003	1.003	1.003

Table 4.4-11 (cont'd)

St. Lucie Unit 2
Thermal Hydraulic Parameters at Full Power*

Notes:

(*) Due to the statistical combination of uncertainties described in References 35 and 46, the nominal inlet temperature and nominal primary system pressure were used to calculate some of these parameters.

(**) Based on a core containing 33 non-fuel rods (12 shims in St. Lucie Unit 2 Cycle 14, and an allowance for up to 21 stainless steel rods).

(***) Based on a core containing a total of 100 shims and stainless steel rods, the maximum number supported in the St. Lucie Unit 2 reference analysis.

(+) These factors have been combined statistically with other uncertainty factors at 95/95 confidence/probability level and included in the design limit DNBR.

(++) The Bounding Cycle values were first introduced for Cycle 12.

(+++) The WCAP-9272 Bounding Cycle values account for 30% steam generator tube plugging.

(++++) The RSGs are warranted and analyzed to an upper limit of Steam Generator Tube Plugging of 20%.

(@) The 30% SGTP analyses remain the analysis of record with minimum RCS flow of 335,000 gpm. However, partial credit for flow increase from 335,000 gpm to 375,000 gpm is taken for the Locked Rotor and Pre-trip steam line break with LOOP at time = 0.

ASSY. AVG. ROD RADIAL POWER FACTOR

ASSY. MAXIMUM ROD RADIAL POWER FACTOR

						.8686 1.4420	1.1803 1.6311	
				.6844 1.1304	.9774 1.3627	1.1755 1.5123	1.4086 1.6185	1.2982 1.4708
			.6875 1.0678	1.0072 1.1893	1.0561 1.2350	1.1068 1.1893	1.2228 1.3900	1.1667 1.3556
		.6876 1.0679	.7655 .8901	.8474 1.0785	.9093 1.0785	1.0521 1.2356	1.0015 1.1770	1.0299 1.2410
	.6845 1.1305	1.0073 1.1894	.8474 1.0230	.5320 .6628	.8589 1.0043	.7897 .8428	.9294 1.0665	.6041 .7471
	.9776 1.3629	1.0362 1.2352	.9094 1.0786	.8589 1.0044	.8863 1.0460	1.0319 1.2090	.9874 1.1574	1.0171 1.2233
.8688 1.4423	1.1757 1.5126	1.1070 1.2896	1.0523 1.2358	.7898 .8429	1.0319 1.2090	1.0797 1.2554	1.2182 1.3973	1.1538 1.3386
1.1806 1.6315	1.4089 1.6188	1.2231 1.3903	1.0017 1.1772	.9295 1.0667	.9874 1.1575	1.2182 1.3973	1.2279 1.4172	1.3429 1.4968
	1.2985 1.4711	1.1670 1.3558	1.0301 1.2412	.6042 .7472	1.0172 1.2234	1.1538 1.3387	1.3429 1.4968	1.2879 1.4784

AMENDMENT NO. 1 (4/86)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CORE WIDE PLANAR POWER DISTRIBUTION FOR SAMPLE DNB ANALYSIS

CYCLE 1

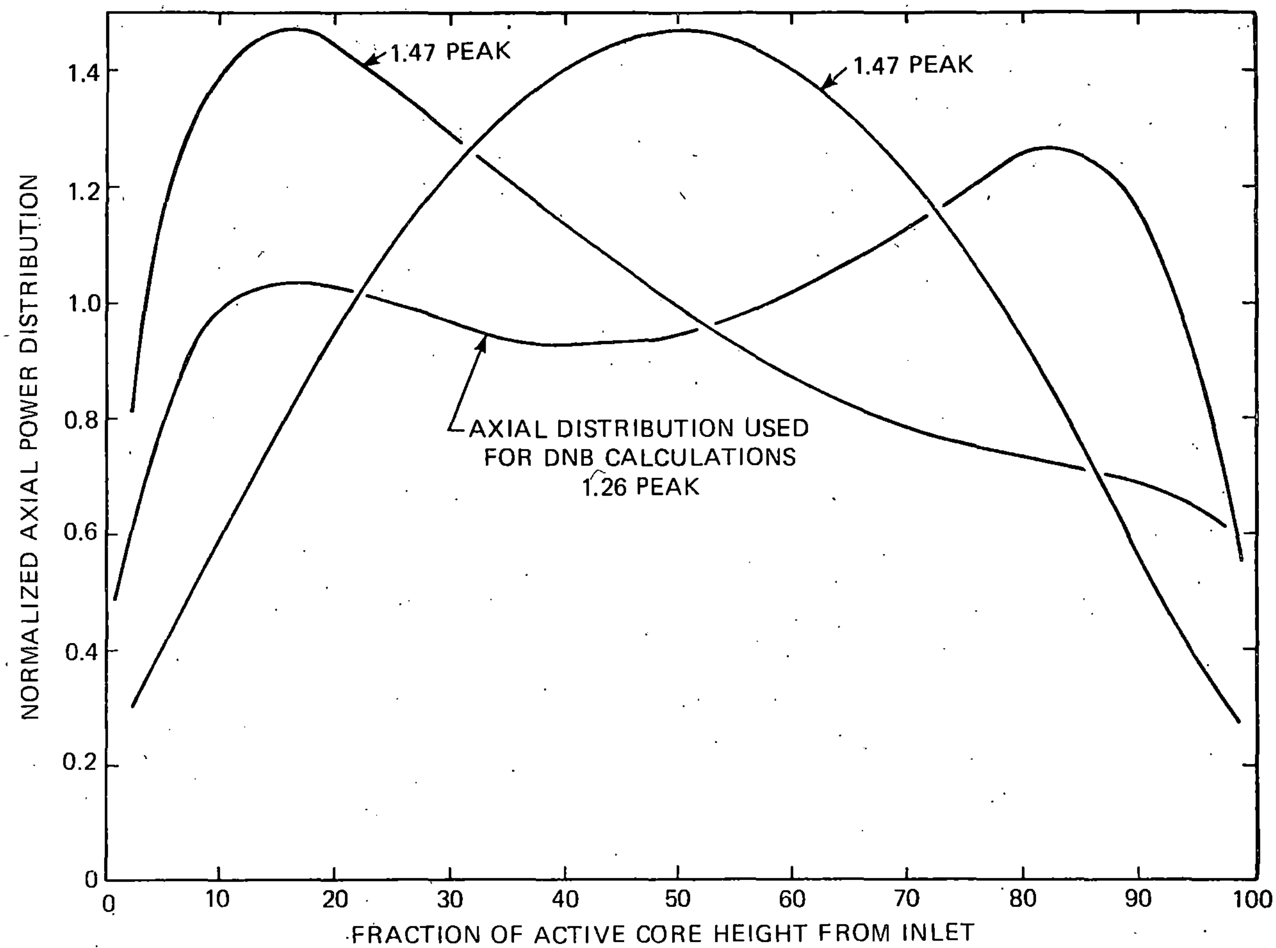
FIGURE 4.4-1

.585	.592	.601	.611	.617	.618	.619	.621	.626	.633	.641	.649	.652	.652	.653	.657
.687	.696	.709	.726	.732	.728	.722	.721	.727	.738	.754	.770	.774	.768	.765	.770
.780	.794	.824	.916	.924	.845	.823	.817	.824	.842	.875	.970	.975	.891	.872	.872
.864	.886	.998			1.024	.921	.909	.916	.941	1.061			1.076	.970	.963
.937	.959	1.079			1.113	1.006	.997	1.005	1.027	1.151			1.161	1.048	1.042
.999	1.014	1.049	1.165	1.180	1.091	1.082	1.092	1.100	1.104	1.126	1.233	1.233	1.126	1.105	1.107
1.055	1.064	1.080	1.104	1.122	1.137	1.172	1.294	1.303	1.193	1.172	1.170	1.166	1.156	1.156	1.165
1.109	1.115	1.125	1.142	1.164	1.197	1.346			1.369	1.231	1.211	1.203	1.201	1.207	1.220
1.163	1.169	1.179	1.198	1.219	1.251	1.404			1.426	1.284	1.265	1.258	1.255	1.261	1.274
1.218	1.227	1.246	1.273	1.290	1.301	1.332	1.465	1.473	1.352	1.334	1.336	1.332	1.320	1.317	1.326
1.271	1.289	1.331	1.475	1.489	1.369	1.348	1.353	1.360	1.368	1.401	1.538	1.539	1.404	1.376	1.376
1.317	1.345	1.508			1.535	1.379	1.361	1.368	1.398	1.569			1.584	1.429	1.419
1.352	1.378	1.544			1.562	1.399	1.376	1.382	1.416	1.593			1.615	1.457	1.447
1.375	1.390	1.431	1.579	1.585	1.444	1.404	1.390	1.396	1.419	1.470	1.624	1.631	1.491	1.463	1.464
1.396	1.401	1.415	1.436	1.441	1.427	1.414	1.410	1.415	1.428	1.451	1.474	1.479	1.469	1.469	1.480
1.426	1.426	1.430	1.443	1.449	1.445	1.442	1.450	1.455	1.456	1.470	1.485	1.486	1.482	1.493	1.511

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

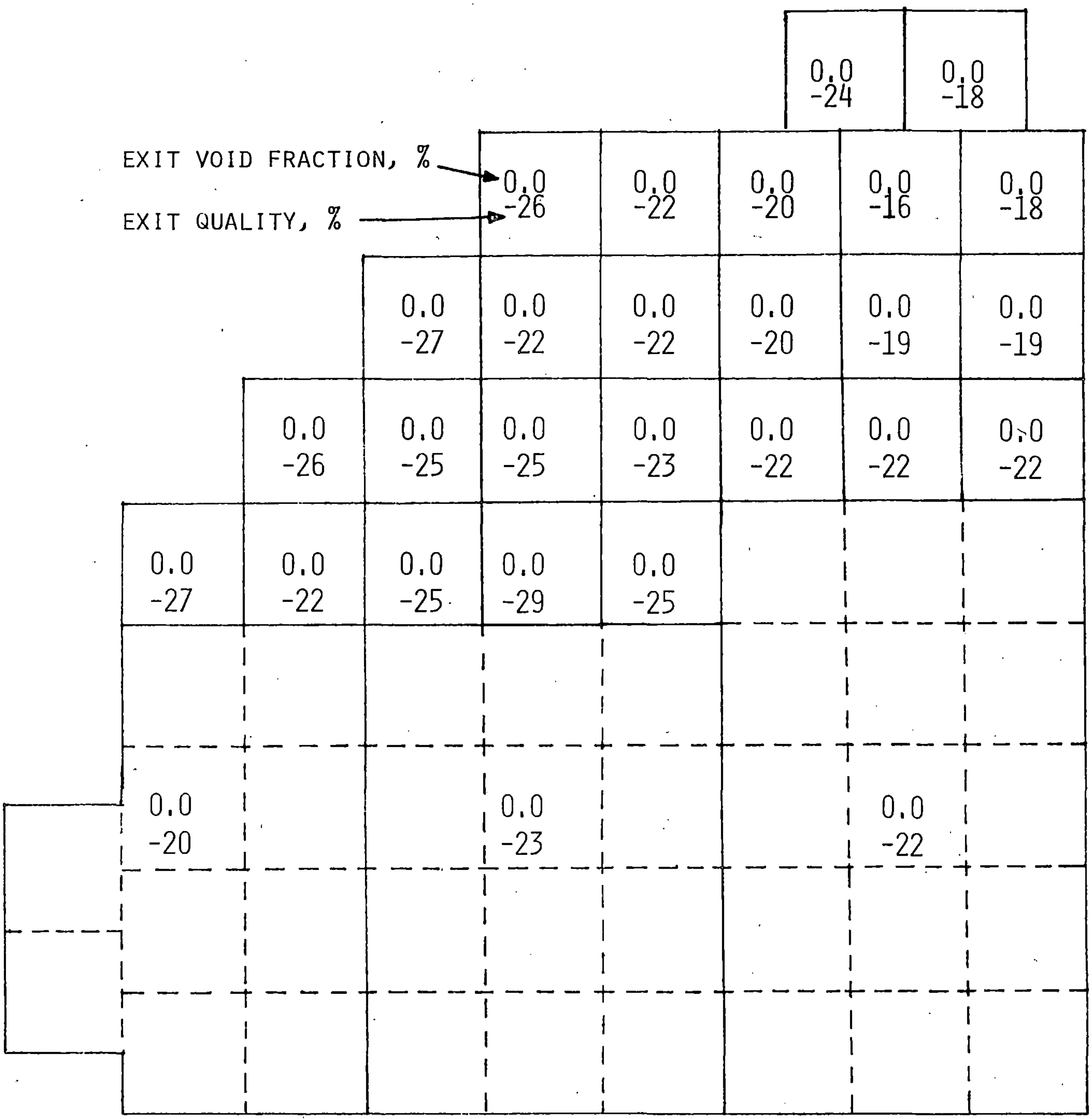
ROD RADIAL POWER FACTORS IN HOT ASSEMBLY FOR SAMPLE DNB ANALYSIS

FIGURE 4.4-2



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

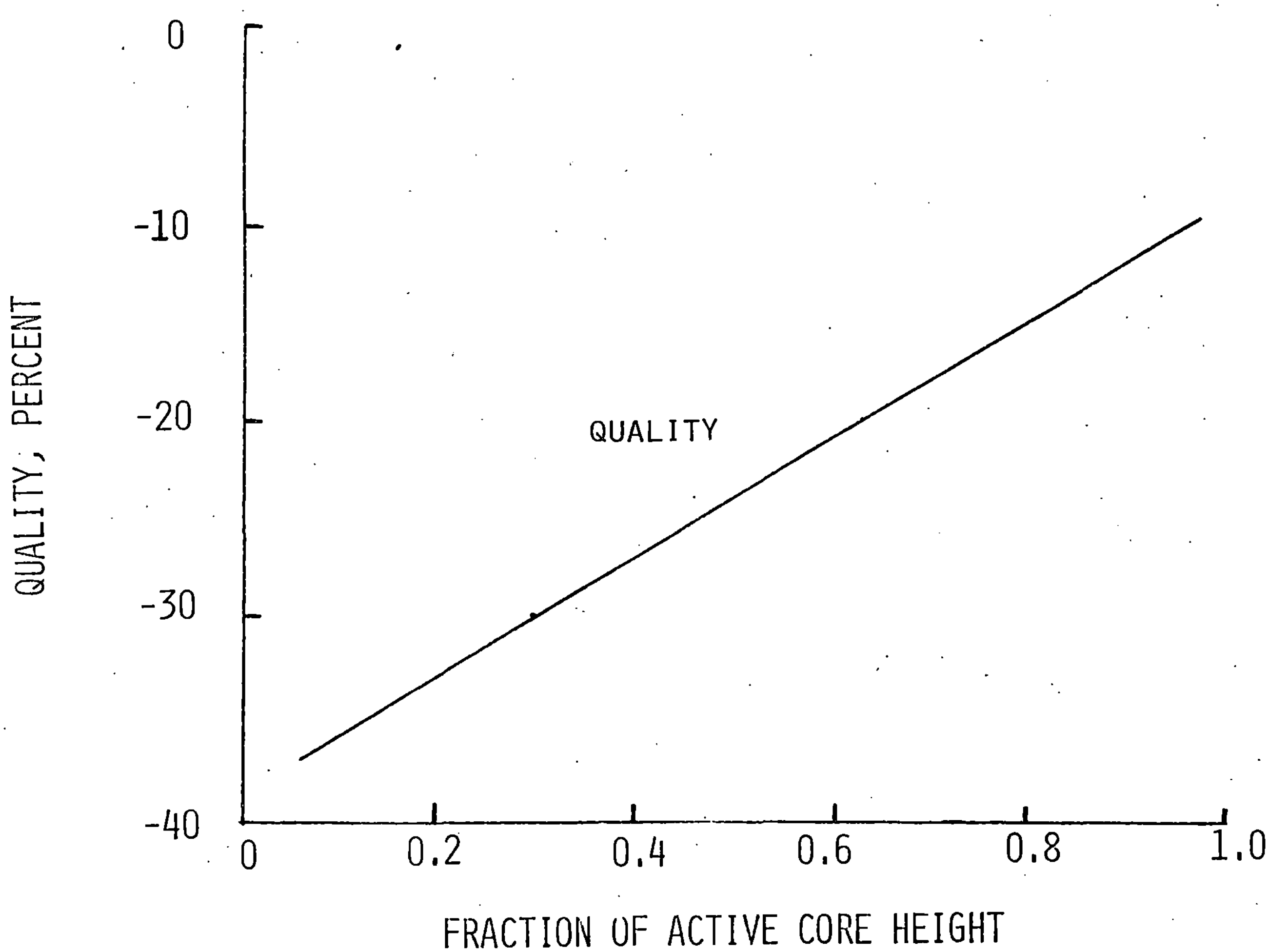
TYPICAL AXIAL POWER DISTRIBUTIONS
FIGURE 4.4-3



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

AVERAGE VOID FRACTIONS AND QUALITIES AT THE EXIT OF DIFFERENT CORE REGIONS

FIGURE 4.4-4

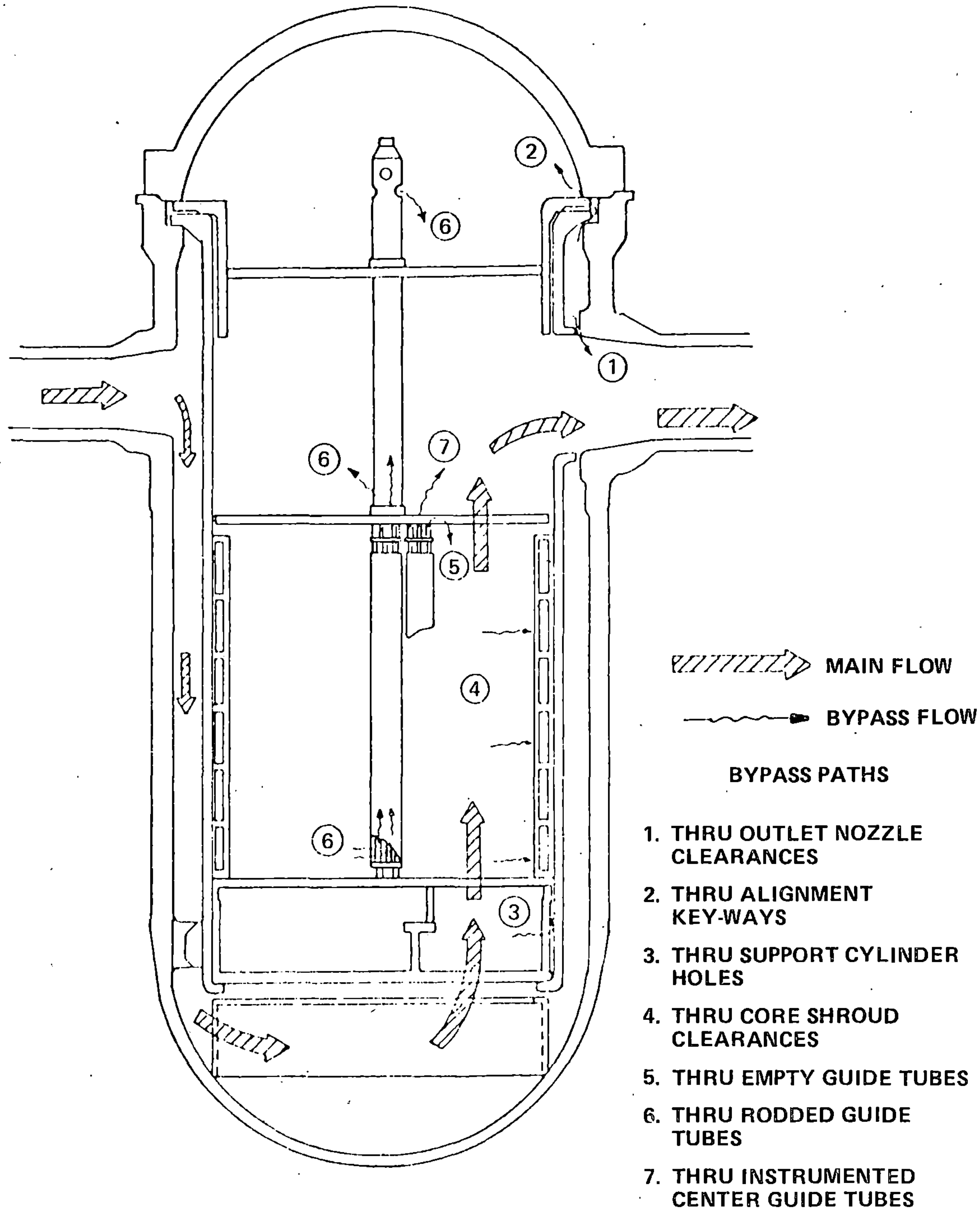


NOTE: VOID FRACTION IS ESSENTIALLY ZERO ALONG THE ENTIRE CORE HEIGHT.

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

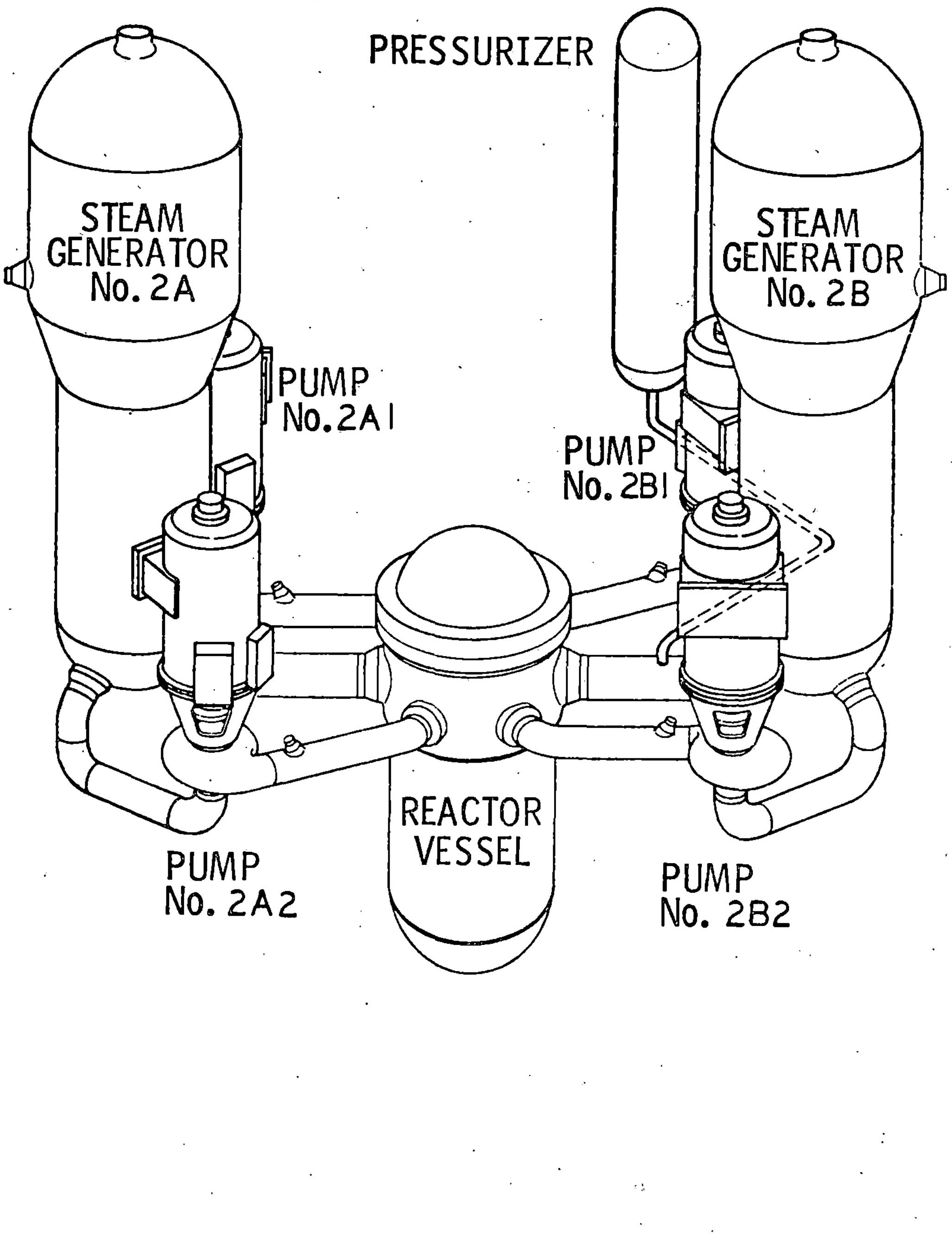
AXIAL DISTRIBUTION OF VOID FRACTION AND QUALITY IN THE SUBCHANNEL ADJACENT TO THE ROD WITH MINIMUM DNBR

FIGURE 4.4-5



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR VERTICAL ARRANGEMENT
SHOWING BYPASS FLOW PATHS
FIGURE 4.4-6



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ISOMETRIC VIEW OF THE REACTOR COOLANT SYSTEM
FIGURE 4.4-7

Deleted

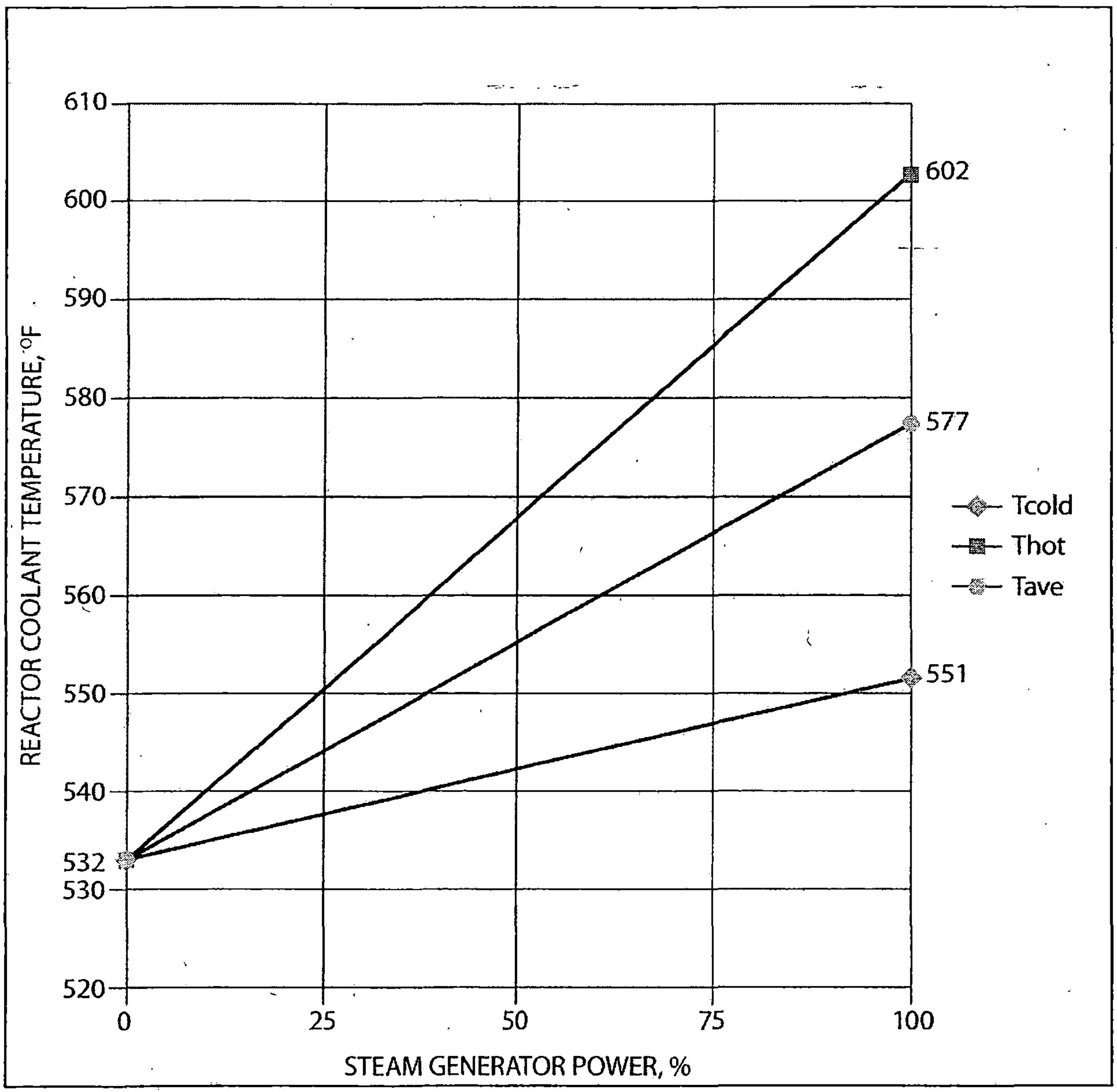
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FIGURE 4.4-8

Deleted

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FIGURE 4.4-9

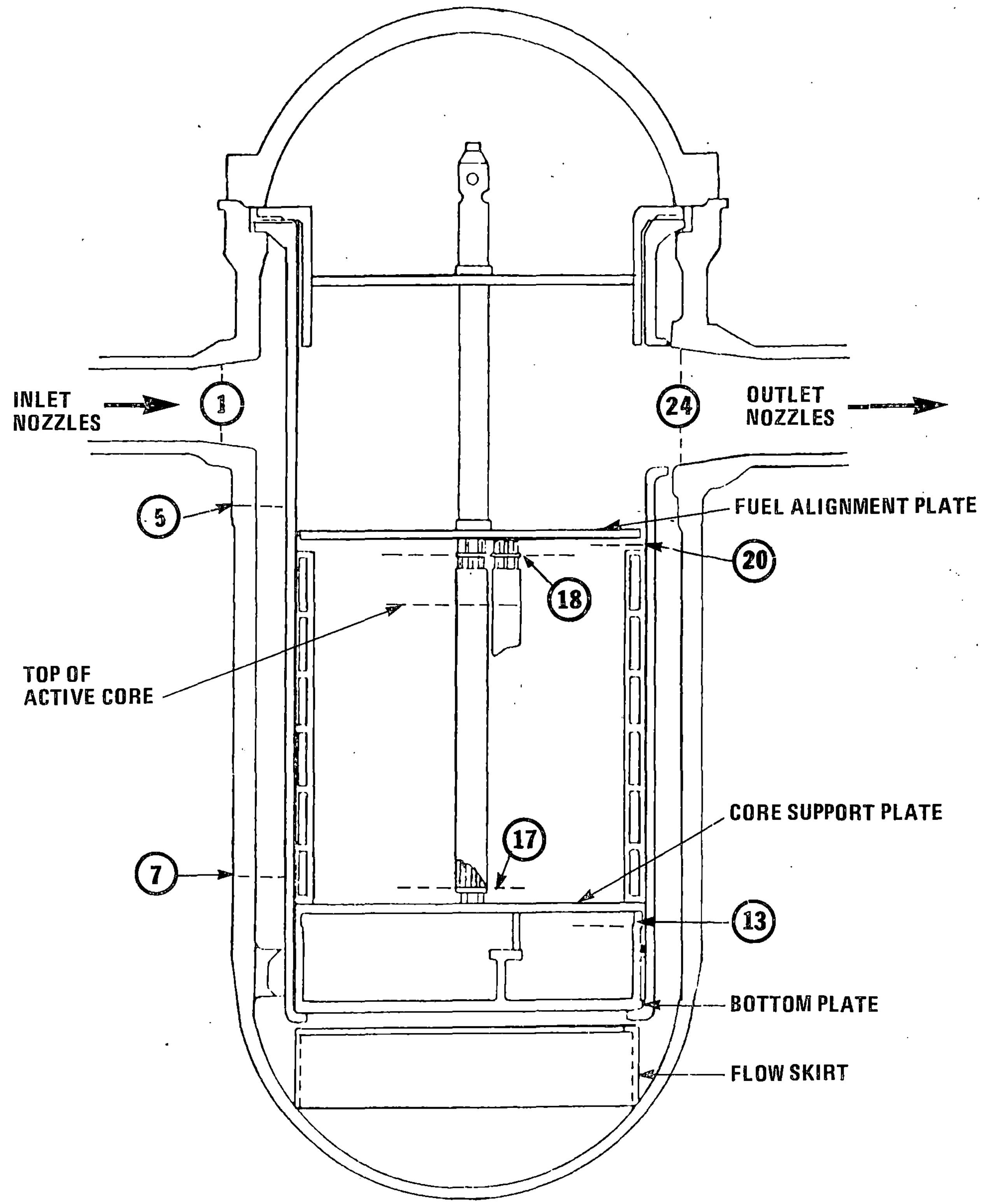


FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

TEMPERATURE CONTROL PROGRAM

FIGURE 4.4-10

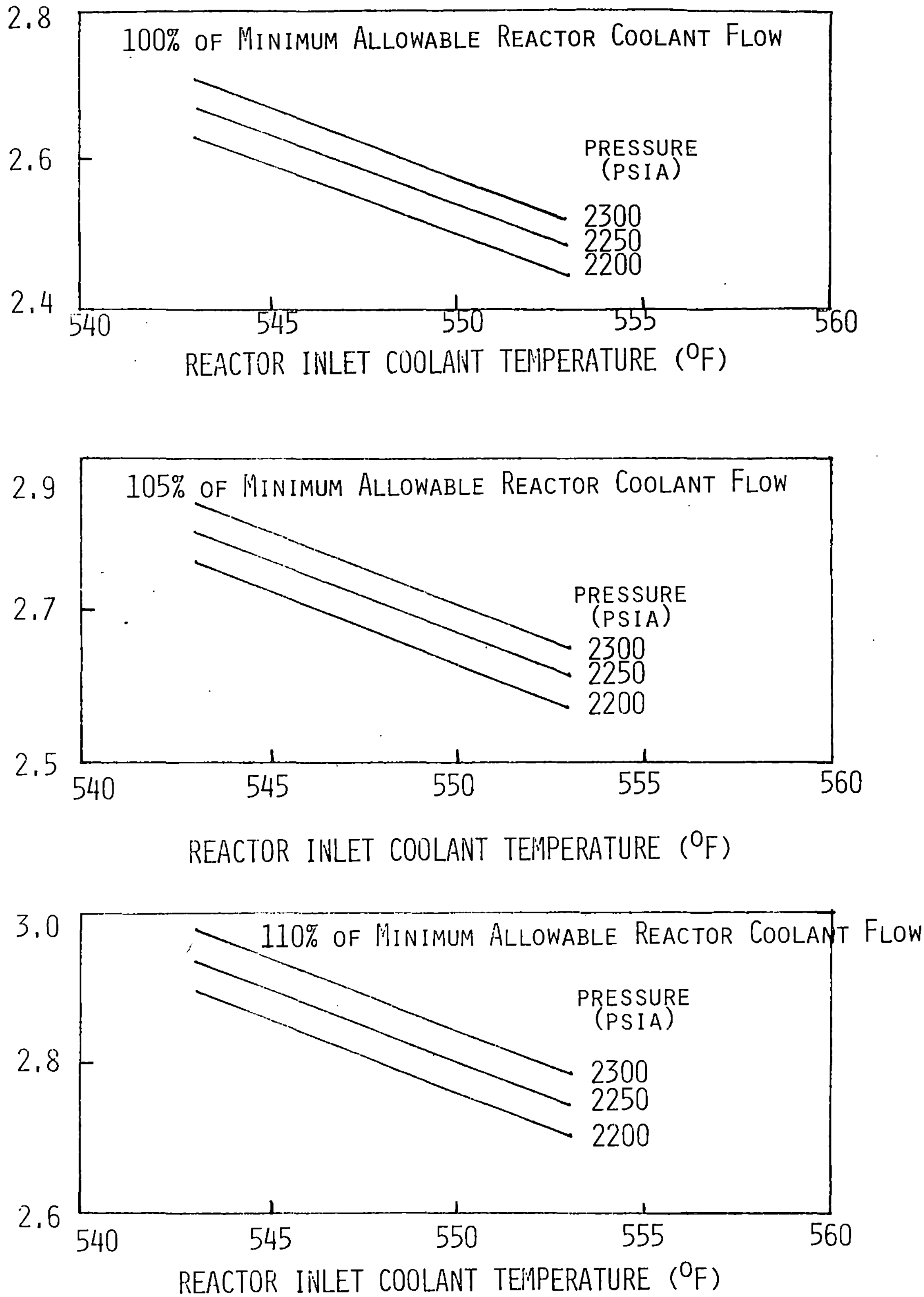
Amendment No. 21 (11/12)



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR STATIONS

FIGURE 4.4-11



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

SENSITIVITY OF MINIMUM DNBR TO SMALL CHANGES IN REACTOR COOLANT CONDITIONS

FIGURE 4.4-12

4.5 REACTOR MATERIALS

4.5.1 CONTROL ELEMENT DRIVE STRUCTURAL MATERIALS

4.5.1.1 Material Specifications

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

1) Motor housing assembly

~~SA 182, Type 348 (austenitic stainless steel)~~

SA 182, Type 403 Modified, and Code Case N-4-11 (martensitic stainless steel)

SB 166 (Alloy 690) (nickel-chromium-iron alloy)

2) Upper pressure housing

SA 213, Type 316 (austenitic stainless steel)

SA 479, Type 316 (austenitic stainless steel)

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

The above listed materials with the exception of the ASTM A270, Type 440 material complay with the following ASME Code Sections and Editions:

Section II, 1998 Edition through 2000 Addenda
Section XI, 1989 Edition, No Addenda
Section XI, 1998 Edition through 2000 Addenda

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

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

1) Latch guide tubes

ASTM A269, Type 316 (austenitic stainless steel)

Chrome Oxide (plasma spray treatment)

2) Magnet and spacer

ASTM A276, Type 410 (martensitic stainless steel)

3) Latch and magnet housing

ASTM A276, Type 316 (austenitic stainless steel)

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

4) Spacer

ASTM A240, Type 304 (austenitic stainless steel)

5) Alignment Button

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

6) Spring

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

7) Pin

Haynes Stellite No. 6B (cobalt base alloy)

8) Dowel pin

ASTM A314, Type 410 (martensitic stainless steel)

9) Spacer and screw

ASTM A276, Type 321 (austenitic stainless steel)

10) Stop

ASTM A276, Type 304 (austenitic stainless steel)

11) Latch and pin

Haynes Stellite No. 36 (cobalt base alloy)

12) Locking cup and screws

300 Series austenitic stainless steel

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

c) The materials in contact with the reactor coolant used in the extension shafts are listed below:

1) Shafts, rod, and plunger

ASTM A276, Type 304 (austenitic stainless steel)

ASTM A264, Type 304 (austenitic stainless steel)

2) Gripper

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

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

3) Spring

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

4) Pin

304 austenitic stainless steel

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

d) The weld rod filler materials used with the above listed components are 308 stainless steel, 316 stainless steel and Inconel 52 or 52M.

All of the material except Alloy 690 listed in the above listings, A through D, were used in an extensively tested CEDM assembly that exceeded lifetime requirements, as described in Subsection 3.9.4.4.1. Also, all of the materials have performed satisfactorily in service in the Maine Yankee (Docket 50-309), Millstone II (Docket 50-236), Calvert Cliffs (Docket 50-317), in addition to other CE designed reactors. The CEDM assembly that was tested had Alloy 600 parts and weld material as part of the assembly. For the replacement CEDMs the Alloy 600 materials have been replaced with Alloy 690 materials. Alloy 600 materials have proven to exhibit PWSCC tendencies under some service environments. Alloy 690 materials have proven to be an acceptable material for the replacement of Alloy 600 materials because they have proven to be less susceptible to PWSCC under the same service environments. Therefore, Alloy 690 materials have become the accepted substitute for Alloy 600 materials.

4.5.1.2 Control of the Use of 90 ksi Yield Strength Material

The only control element drive structural material identified in Subsection 4.5.1.4 which has a yield strength greater than 90 ksi is ASTM A276, Type 440, martensitic stainless steel. Its usage is limited to the steel ball in the vent valve on the top of the CEDM. The ball is used as a seal and is not a primary load bearing member of the pressure boundary. This material was tested and exceeded lifetime requirements. Also, this material is presently being used in operating reactors such as Maine Yankee (Docket 50-209), Millstone II (Docket 50-236), and Calvert Cliffs (Docket 50-317) and has performed satisfactorily for the same application.

4.5.1.3 Control of the Use of Sensitized Austenitic Stainless Steel

Control of the use of sensitized austenitic stainless steel is consistent with the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel" May 1973 (R0) as described in Subsections 4.5.1.3.1 through 4.5.1.3.3, except for the criteria used to demonstrate freedom from sensitization. The ASTM A708 (Strauss Test) is used in lieu of the the ASTM A262 Method (A or E) (Modified Strauss Test) to demonstrate freedom from sensitization in fabricated unstabilized austenitic stainless steel. The former test has shown, through experimentation, excellent correlation with the type of corrosion observed in severely sensitized austenitic stainless steel.

4.5.1.3.1 Solution Heat Treatment Requirements

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

4.5.1.3.2 Material Inspection Program

Extensive testing on stainless steel mockups, fabricated using production techniques, was conducted to determine the effect of various welding procedures on the susceptibility of unstabilized 300 series stainless steels to sensitization-induced intergranular corrosion. Only those procedures and/or practices demonstrated not to produce a sensitized structure were used in the fabrication of central element drive system structural components. The ASTM Standard A708 (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 A708 is utilized as a go/no-go standard for acceptability.

4.5.1.3.3 Avoidance of Sensitization

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

Weld heat affected zone sensitized austenitic stainless steel (which will fail the Strauss Test, ASTM A708) is avoided in control element drive system structural components by careful control of:

a) Weld heat input to less than 60 kJ/in.

b) Interpass temperature to 350 F maximum.

c) Carbon Content.

4.5.1.4 <u>Control of Delta Ferrite in Austenitic Stainless Steel Welds</u>

The austenitic stainless steel, primary pressure retaining welds in the control element drive system structural components are consistent with the recommendations MTEB 5-1, "The Interm Position of Regulatory Guide 1.31," as described in Subsection 4.5.2.4.3.

4.5.1.5 <u>Cleaning and Contamination Protection Procedures</u>

The procedure and practices followed for cleaning and contamination protection of the control element drive system structural components are as described in Subsection 4.5.2.4.1 and Subsection 4.5.2.5.

4.5.2 REACTOR INTERNALS MATERIALS

4.5.2.1 <u>Material Specifications</u>

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

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

a) Core Support Barrel Assembly

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

(a) ASTM-A-182

(b) ASTM-A-240

(c) ASTM-A-479

2) Precipitation hardening stainless steel to the following specifications:

(a) ASTM-A-453, Grade 660

(b) ASTM-A-638, Grade 660

b) Upper Guide Structure Assembly

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

(a) ASTM-A-182

(b) ASTM-A-240

(c) ASTM-A-249

(d) ASTM-A-269

(e) ASTM-A-479

(f) ASTM-A-451, Grade CPF8

2) Type 304L austenitic stainless steel to the following specification:

(a) ASTM-A-312

c) Core Shroud Assembly

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

(a) ASTM-A-182

(b) ASTM-A-240

d) Holddown Ring

ASTM-A-182, modified to ASME Code Case 1747

e) Incore instrument support system

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

(a) ASTM-A-193

(b) ASTM-A-194

(c) ASTM-A-213

(d) ASTM-A-240

(e) ASTM-A-249

(f) ASTM-A-312

(g) ASTM-A-473

(h) ASTM-A-479 or ASTM-A-276

(i) ASTM-B-446

(j) ASTM-A-269

2) Zircaloy-4

(a) ASTM-B-351

(b) ASTM-B-353

3) Inconel 625

(a) ASTM-B-446

f) Threaded Structural Fastener and Pin Material

ASTM-A-453 and ASTM-A-638, Grade 660 material (trade name A-286) is used for bolting and pin applications. This alloy is heat treated to a minimum yield strength of 85,000 lb/in. Its corrosion properties are similar to those of the 300 series austenitic stainless steel. It is austenitic in all conditions of fabrication and heat treatment. This alloy was used for bolting in previous reactor systems and test facilities in contact with primary coolant and has proven completely satisfactory.

g) Chrome Plating and Hardfacing

Chrome plating or hardfacing are employed on reactor internals components or portions thereof where required by function. Chrome plating complies with Federal Specification No. QQ-C-320a, Class 2B. The hardfacing material employed is Stellite 25.

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

4.5.2.2 Welding Acceptance Standards

Welds employed on reactor internals and core support structures-meet the acceptance standards delineated in article NG-5000 Section III, 1974 Edition, and control of welding has been performed in accordance with Sections III and IX of the applicable ASME Code. In addition, consistency with the recommendations of Regulatory Guides 1.31 "Control of Ferrite Content in Stainless Steel Weld Metal" May 1977 (R2) and 1.44 "Control of the Use of Sensitized Stainless Steel" May 1973 (R0) is described in Subsection 4.5.2.4.

4.5.2.3 Nondestructive Examination of Wrought Seamless Tubular Products and Fittings

Quality Group A components in the reactor internals which are wrought seamless tubular products or fittings are consistent with the recommendations of Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products" (since withdrawn 9/77).

The nondestructive examination procedures used for the examination of tubular products and fittings used as elements of the reactor internals are in compliance with the Class 1 requirements of Section III of the ASME Code, 1971 issue, including 1972 Summer Addenda.

4.5.2.4 Fabrication and Processing of Austenitic Stainless Steel

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

4.5.2.4.1 Cleaning and Contamination Protection Procedures

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

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

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

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

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

To prevent halide-induced, intergranular corrosion that could occur in an aqueous environment with significant quantities of dissolved oxygen, flushing water is inhibited via additions of hydrazine. Experiments have proven this inhibitor to be effective.[1] Operational chemistry specifications preclude halides and oxygen, (both prerequisites of intergranular attacks), and are shown in Subsections 9.3.4 and are a part of the Technical Specifications.

4.5.2.4.2 Control of the Use of Sensitized Austenitic Stainless Steel

The recommendations of Regulatory Guide 1.44 (RO), as described in Subsections 4.5.2.4.2.1 through 4.5.2.4.2.5, were followed except for the criteria used to demonstrate freedom from sensitization. The ASTM A708 Strauss Test was used in lieu of the ASTM A262 Method E Modified Strauss Test to demonstrate freedom from sensitization in fabricated unstabilized austenitic stainless steel, since the former test has shown, through experimentation, excellent correlation with the type of corrosion observed in severely sensitized austenitic stainless steel NSSS components. ASTM A262 Method E is used as the acceptance criteria for raw austenitic stainless steel method.

4.5.2.4.2.1 Solution Heat Treatment Requirements

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

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

4.5.2.4.2.2 Material Inspection Program

Extensive testing of stainless steel mockups, fabricated using production techniques, was conducted to determine the effect of various welding procedures on the susceptibility of unstabilized 300 series stainless steels to sensitization-induced intergranular corrosion. Only those procedures and/ or practices demonstrated not to produce a sensitized structure were used in the fabrication of reactor internals components. The ASTM Standard A708 (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 A708 is utilized as a go/no-go standard for acceptability.

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

4.5.2.4.2.3 Unstabilized Austenitic Stainless Steels

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

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

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

a) CF8M Cast stainless steels (delta ferrite controlled to 5-25 volume percent)

b) CF8

c) Type 308 Singly and combined

d) Type 309 Stainless steel weld filler metals. (Delta ferrite controlled to 5-18 volume percent as deposited)

e) Type 312

f) Type 316

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

4.5.2.4.2.4 Avoidance of Sensitization

Exposure of unstabilized austenitic 300 stainless steels to temperatures ranging from 800 to 1600F will result in carbide precipitation. The degree of carbide precipitation, or sensitization, depends on the temperature, the time at that temperature, and also the carbon content. Severe sensitization is defined as a continuous grain boundary chromium-iron carbide network. This condition induces susceptibility to intergranular corrosion in oxygenated aqueous environments, as well as those containing halides. Such a metallurgical structure will readily fail the Strauss Test, ASTM A708. Discontinuous precipitates (i.e., an intermittent grain boundary carbide network) are not susceptible to intergranular corrosion in a PWR environment.

Weld heat affected zone sensitized austenitic stainless steels were avoided (which will fail the Strauss Test, ASTM A708) by careful control of:

a) Weld heat input

b) Interpass temperature

c) Carbon content

A weld heat input of less than 60 kJ/in. is used during most fabrication states of the Type 304 stainless steel core support structure. Higher heat inputs are used in same heavy section weld joints. Freedom from weld heat-affected zone sensitization in these higher heat input weldments is demonstrated with weld runoff samples produced at the time of component welding in material having a carbon content equal to or greater than the highest carbon content of those heats of steel being fabricated. Specimens so provided are subjected to the Strauss Test, ASTM A708.

4.5.2.4.2.5 Retesting Unstabilized Austenitic Stainless Steels Exposed to Sensitizing Temperature

Sensitization which may be susceptible to intergranular corrosion, is avoided during welding as described in Subsection 4.5.2.4.2.4. Homogeneous or localized heat treatment of unstabilized stainless steels in the temperature range 800 to 1500° F is prohibited except in the case of the core support structure. This complex substructure is thermally stabilized at 900 ± 25° F for seven hours after fabrication and prior to final machining. Such treatment produces only minor, discontinuous precipitates. In addition to thermocouple records during this heat treatment, a sample of Type 304 stainless steel having a carbon content equal to or greater than the highest carbon heat of material present in the structure is included as a monitor sample. After heat treatment, the monitor sample is subject to the Strauss Test, ASTM A708, as well as a metallographic examination to verify freedom from sensitization.

4.5.2.4.3 Control of Delta Ferrite In Welds

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

a) The analysis of the filler metal is adjusted to contain a ferrite content of at least five percent and no greater than 15 percent as determined by the modified DeLong Diagram (1973) in the as-deposited condition, and additionally by the use of a calibrated Severn Gauge (or equivalent) measurement on a weld pad. This delta ferrite requirement is met for each heat or lot of filler metal, heat of consumable inserts, and for each combination of heat of bare electrodes and lot of dry blend of flux mix to be used for welding.

   All production welds have to be capable of passing five to 15 percent by volume delta ferrite requirement as determined by a calibrated Severn (or equivalent) Gauge. The vendor performs tests on an audit basis.

4.5.2.4.4 Control of Electroslag Weld Properties

The electroslag process was not utilized to fabricate reactor internal components, and therefore Regulatory Guide 1.34 is not applicable.

4.5.2.4.5 Welder Qualification for Areas of Limited Accessibility

The specific recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," December 1973 (R0) were not followed. However, performance qualifications, for personnel welding under condition of limited accessibility, are conducted and maintained in accordance with the requirements of ASME Code Sections III and IX. A requalification is required when:

a) Any of the essential variables of Section IX are changed

b) Authorized personnel have reason to question the ability of the welder to satisfactorily perform to the applicable requirements.

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

4.5.2.4.6 Non Metallic Thermal Insulation

Nonmetallic thermal insulation is not used on the reactor internals, and therefore Regulatory Guide 1.36 is not applicable.

4.5.2.5 Contamination Protection and Cleaning of Austenitic Stainless Steel

St. Lucie Unit 2 is 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," March 1973 (R0). The Quality Assurance Program for safety-related items during onsite cleaning of materials and components cleanliness control, and preoperational cleaning and layup of nuclear fluid systems is in accordance with ANSI N45.2.1-1973 as interpreted by Regulatory Guide 1.37 (R0). ASME NQA-1-1994, Subpart 2.1 was substituted for ANSI N45.2.1 as described in the FPL Quality Assurance Topical Report discussed in Section 17.2.

SECTION 4.5: REFERENCES

1. Habicht, P.R. and Bryant, P.E.C., Fluoride Induced Intergranular Corrosion of Sensitized Austenitic and Austenoferritic Stainless Steel, presented at the International Atomic Energy Authority Workshop on Stress Corrosion Cracking, San Francisco, California, March 1976.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

4.6.1 INFORMATION FOR CEA CONTROL SYSTEMS

Component diagrams, descriptions, and characteristics of the Control Element Drive Mechanisms (CEDMs) are presented in Subsection 3.9.4. Figure 4.6-1 shows the reactor vessel closure head plan view detailing the CEDM layout. The instrumentation is described in Section 7.2 (Reactor Protection System) and Section 7.7 (Control System Not Required For Safety).

4.6.2 EVALUATIONS

Control element assemblies (CEAs) are inserted into the reactor core when electrical power is removed from the coils of the CEDMs by the Reactor Protective System (RPS). A failure modes and effects analysis is presented in Section 7.2 which demonstrates compliance with IEEE Standard 279-1971 and shows that no single failure can prevent electrical power from being removed from the CEDM coils when required. For the trip function, the CEDMs are essentially passive devices. When power is removed from the CEDM coils, the armature springs automatically cause the driving and housing latches to be withdrawn from the CEDM drive shaft, allowing insertion of CEAs by gravity. Actuation of trip breakers is independent of any existing control signals.

For the trip function, all CEDMs are independent of one another. That is, the failure of one CEDM to trip would have no effect on the operability of other CEDMs. Sufficient shutdown margin is always provided to assure that the safety function can be performed assuming the failure of a CEDM.

RPS design criteria are discussed in detail in Subsection 7.2.2.3.

The CEDMs are located where they are protected from common mode failure due to missiles and failure of moderate and high energy pipes. Sections 3.5 and 3.6 discuss protection of essential systems against missiles and pipe breaks. A potential source of common mode failure is loss of air cooling to the CEDM coils. Worst case analysis indicates that there would be adequate mechanical clearances, to permit CEDM trip, at temperatures well above normal operating temperatures of the reactor(1). Testing was performed to determine the maximum CEDM temperature under conditions that simulated loss of air cooling. With the upper gripper coil energized, which is the normal operating mode, and with a reactor coolant loop temperature of 600° F, the maximum CEDM temperature was 535° F. An analysis of worst case tolerance stack-up within the CEDM indicated adequate clearances to assure scram at 650° F.

Analyses of possible malfunctions are discussed in Section 15.4.

4.6.3 TESTING AND VERIFICATION

The functional testing program for the CEDMCS is described in Subsection 3.9.4.4. (Testing of the CEAs is described in Subsection 4.2.4.4).

As discussed in Subsection 4.6.2, upon reactor trip all CEDMs are independent of one another. Thus, the worst single failure is one that prevents one CEDM from tripping. This failure mode was considered and included in the accident analysis presented in Chapter 15.

Under large break LOCA conditions (Chapter 6) where severe loads may be applied to CEAs, no credit is taken for CEDM functioning.

Testing was performed on a prototype CEDM to verify insertion time, assuming worst case plant operating conditions. Insertion time was verified by dropping the minimum effective (dry) weight of 86 pounds. This weight was calculated to be the minimum effective dry weight, assuming maximum delta-P across the core due to crudding and high reactor coolant density due to operating at a loop temperature of 475 F.

4.6.4 INFORMATION FOR COMBINED PERFORMANCE OF REACTIVITY SYSTEMS

Figures 1.2-3 through 1.2-22 provide plant and elevation layout drawings. These figures show that the CEA Control Systems, Safety Injection System (SIS), and Chemical and Volume Control System (CVCS) are located in the Reactor Building and the Reactor Auxiliary Building. The physical arrangement insures that no single occurrence can affect two or more reactivity control systems concurrently.

Table 4.6-1 lists the postulated accidents evaluated in Chapters 6 and 15 that take credit for two or more reactivity control systems for preventing or mitigating each accident. The related reactivity systems are also tabulated.

The maximum rate of reactivity addition that may be produced by the CVCS is too low to induce any significant pressure forces that might rupture the reactor coolant pressure boundary or disturb the reactor vessel internals.

Inadvertent startup of the Safety Injection System during normal plant operation would have no effect since Reactor Coolant System pressure is higher than the shutoff head of the high pressure safety injection pumps.

4.6.5 EVALUATIONS OF COMBINED PERFORMANCE

Since the CEA Control Systems and the CVCS/SIS are separated and totally diverse in design and operation, with no common link, and since the CEDMCS is protected from the effects of failure of high and moderate energy piping, there are no credible potential common mode failures that could cause the CEDMCS to fail in combination with CVCS or SIS.

Missile protection, pipe rupture and fire hazard analyses are provided in Sections 3.5, 3.6 and Subsection 9.5.1, respectively.

SECTION 4.6: REFERENCES

1. "Review of Reactor Shutdown System (PPS Design) for Common Mode Failure Susceptability," Combustion Engineering Topical Report, CENPD-148, November 1974.

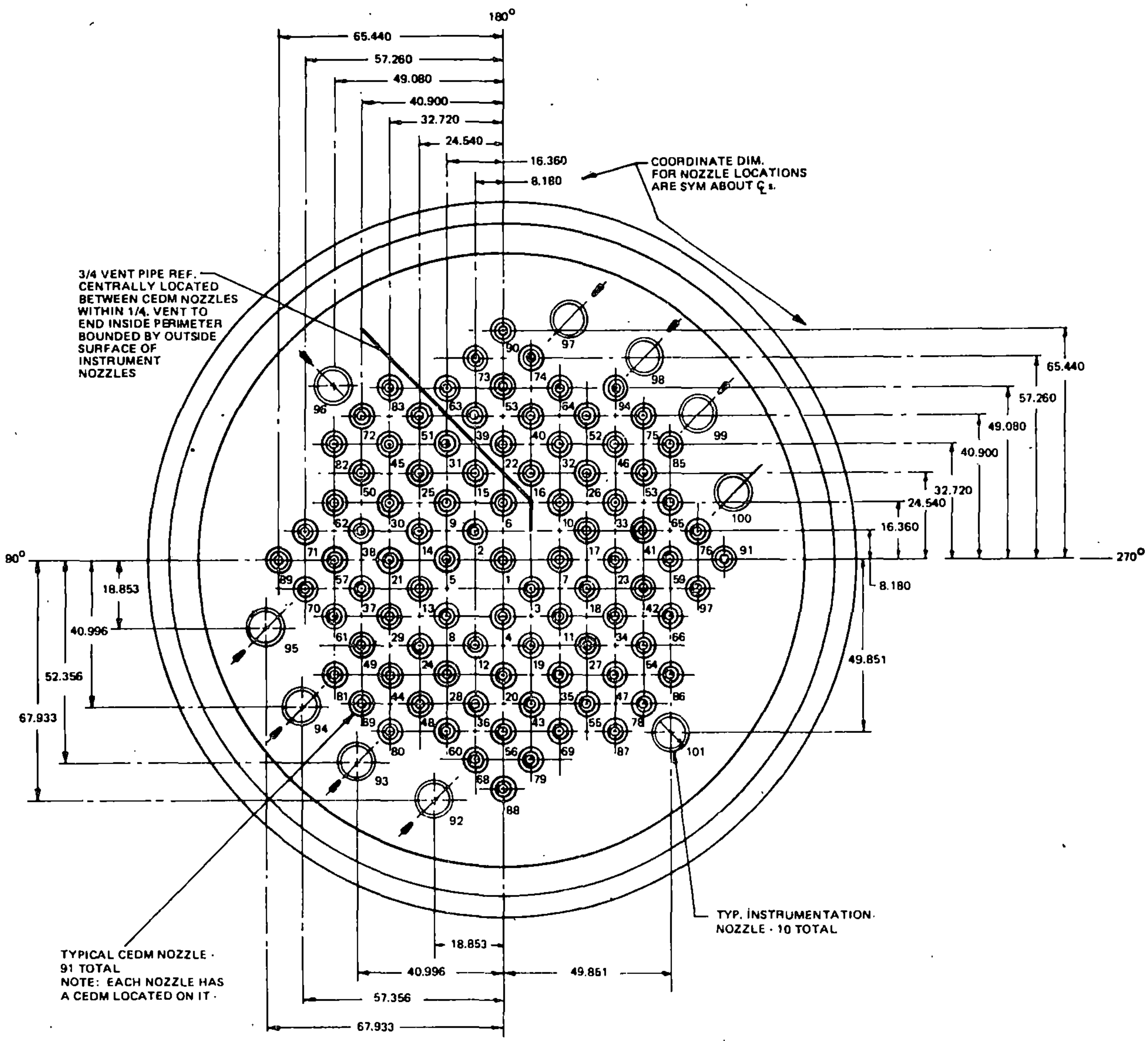
TABLE 4.6-1

POSTULATED EVENTS REQUIRING OPERATION
OF TWO OR MORE REACTIVITY CONTROL SYSTEMS
(Information retained for historical purposes)

Subsection*	Event Name	CEDMCS	CVCS	SIS
15.1.4.3	Loss of main steam-large, inside containment and failure to fast transfer one 6.9 kV bus	X	X	X
15.1.5.1	Loss of main steam-large, outside containment and loss of offsite power at turbine trip	X	X	
15.1.5.3	Loss of main steam, large, inside containment, and loss of offsite power at turbine trip	X	X	X
15.2.2.1	Isolation of turbine (20 percent power) and failure of one MSSV to close	X	X	X
15.2.5.2	Loss of feedwater-large, inside containment, downstream of check valve and loss of offsite power at turbine trip	X	X	X[a]
15.4.2.4	Boron Dilution	X	X	
15.4.5.3	CEA ejection and failure to fast transfer one 4.16 kV bus	X		X
15.6.2.1	Steam generator tube rupture	X	X	X
15.6.3.1	Small loss of RCS fluid-outside containment	X	X	X
6.3.3.3	Small loss-of-coolant accident	X		X
6.3.3.2	Large loss-of-coolant accident	X		X

[a] Actuated on high containment pressure but not required.

* The subsection numbers may refer to earlier FSAR Amendments and may not exist in the current amendment.



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR VESSEL CLOSURE HEAD
PLAN VIEW CEDM LAYOUT
FIGURE 4.6-1

FSAR User Comment Form

FSAR errors or improvement suggestions should be identified below by FSAR Users and forwarded to the appropriate Nuclear Engineering Project Licensing Supervisor.

Originator ____________ Dept ____________ Location ____________ Phone ____________

Plant PTN _____ PSL 1 _____ PSL 2 _____

FSAR Areas Affected

Sections		Figures	
________________	________________	________________	________________
________________	________________	________________	________________

Comments Attached _______ Below _______

____
____
____
____
____
____
____

Engineering Review (To be completed by Project Licensing)

Accepted _______ Insufficient Information _______ No Change Required _______

Disposition: ____
____
____
____
____
____

Assigned User Comment #__________________ Reviewing Engineer __________________

LIST OF EFFECTIVE PAGES

CHAPTER 5

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

Page	Amendment
5-1	23
5-2	23
5-3	23
5-4	23
5-i	21
5-ii	18
5-iii	18
5-iv	23
5-iva	
5-v	23
5-vi	6
5-vii	21
5-viii	21
5.1-1	21
5.1-1a	1
5.1-2	23
5.1-2a	21
5.1-2b	21
5.1-2c	21
5.1-2d	21
5.1-2e	21
5.1-2f	21
5.1-3	21
5.1-4	18
5.1-5	1
F5.1-1	18
F5.1-2	18
F5.1-3	18
F5.1-4	18
F5.1-4a	18
F5.1-4b	18
F5.1-5	18
F5.1-6	18
F5.1-6a	23
F5.1-7	12
5.2-1	18
5.2-2	16
5.2-3	-
5.2-4	18
5.2-5	18
5.2-6	19
5.2-6a	18
5.2-7	18
5.2-7a	18
5.2-8	21
5.2-9	21
5.2-10	20
5.2-11	20
5.2-12	18
5.2-13	18
5.2-14	21
5.2-15	12
5.2-16	21
5.2-17	18
5.2-18	12
5.2-19	20
5.2-19a	18
5.2-20	17
5.2-21	21
5.2-22	15
5.2-23	19
5.2-23a	20
5.2-23b	19
5.2-23c	21
5.2-23d	19
5.2-23e	21
5.2-23f	20
5.2-24	18
5.2-25	18
5.2-26	21
5.2-27	18
5.2-27a	18
5.2-28	21
5.2-29	21
5.2-30	21
5.2-31	20
5.2-32	18
5.2-33	18
5.2-33a	20
5.2-33b	20
5.2-34	18
5.2-34a	-
5.2-35	14
5.2-36	12
5.2-37	18
5.2-37a	20
5.2-37b	-
5.2-37c	11
5.2-37d	11
5.2-37e	11
5.2-38	18
5.2-38a	18
5.2-39	-
5.2-40	18
5.2-41	18
5.2-42	-
5.2-43	-
5.2-43a	-

LIST OF EFFECTIVE PAGES

CHAPTER 5

Page	Amendment
5.2-44	18
5.2-44a	-
5.2-44b	-
5.2-44c	-
5.2-44d	-
5.2-44e	18
5.2-45	21
5.2-46	14
5.2-47	13
F5.2-1	18
F5.2-2	-
F5.2-3	-
F5.2-4	-
F5.2-5	9
F5.2-6	9
F5.2-7	9
F5.2-8	9
F5.2-9	9
F5.2-10	9
F5.2-11	-
F5.2-12	-
F5.2-13	18
F5.2-14	18
F5.2-15	-
F5.2-16	-
F5.2-17	-
F5.2-18	-
F5.2-19	-
F5.2-20	-
F5.2-21	-
F5.2-22	-
F5.2-23	18
F5.2-23a	-
F5.2-24	19
F5.2-25	19
F5.2-26	-
F5.2-27	19
F5.2-28	19
F5.2-29	19
5.2A-1	21
5.2A-2	21
5.2A-3	21
5.2A-4	21
5.2A-5	21
5.2A-6	21

Page	Amendment
5.2A-7	-
F5.2A-1	-
F5.2A-2	-
F5.2A-3	-
F5.2A-4	-
F5.2A-5	21
F5.2A-6	21
5.2B-i	18
5.2B-1	21
5.2B-2	18
5.2B-2a	21
5.2B-2b	21
5.2B-3	-
5.2B-4	-
5.2B-5	-
5.2B-6	21
F5.2B-1	-
F5.2B-2	-
F5.2B-3	-
F5.2B-4	-
5.2C-i	18
5.2C-1	-
F5.2C-1	-
5.3-1	-
5.3-2	-
5.3-3	18
5.3-3a	18
5.3-4	18
5.3-5	18
5.3-5a	18
5.3-6	21
5.3-7	21
5.3-8	-
5.3-9	21
5.3-10	14
5.3-10a	18
5.3-11	18
5.3-12	20
5.3-13	19
5.3-14	12
5.3-15	19
5.3-16	18
5.3-17	21
5.3-18	21

LIST OF EFFECTIVE PAGES

CHAPTER 5

Page	Amendment
5.3-19	12
5.3-20	12
5.3-21	-
5.3-22	-
5.3-23	-
5.3-24	-
5.3-25	-
5.3-26	12
5.3-27	21
5.3-27a	18
5.3-27b	12
5.3-28	-
5.3-29	18
F5.3-1	-
F5.3-2	-
F5.3-3	-
F5.3-4	-
F5.3-5	12
F5.3-6	19
F5.3-7	19
F5.3-8	-
5.4-1	-
5.4-2	10
5.4-3	18
5.4-3a	-
5.4-4	19
5.4-5	21
5.4-6	3
5.4-7	-
5.4-8	23
5.4-9	20
5.4-10	21
5.4-11	21
5.4-12	18
5.4-13	18
5.4-13a	15
5.4-14	12
5.4-15	21
5.4-16	-
5.4-17	20
5.4-18	18
5.4-19	13
5.4-19a	13
5.4-20	21
5.4-21	18
5.4-22	13

Page	Amendment
5.4-23	18
5.4-24	18
5.4-24a	20
5.4-24b	17
5.4-25	12
5.4-26	21
5.4-27	21
5.4-27a	-
5.4-28	21
5.4-28a	23
5.4-28b	18
5.4-28c	20
5.4-28d	20
5.4-28e	20
5.4-28f	21
5.4-28g	20
5.4-28h	-
5.4-28i	18
5.4-28ia	19
5.4-28j	13
5.4-28k	18
5.4-28l	20
5.4-28m	21
5.4-28n	21
5.4-28o	21
5.4-28p	21
5.4-29	12
5.4-30	-
5.4-30a	-
5.4-31	12
5.4-31a	21
5.4-32	21
5.4-33	21
5.4-34	18
5.4-35	12
5.4-36	20
5.4-36a	20
5.4-36b	-
5.4-36c	-
5.4-36d	21
5.4-37	23
5.4-38	23
5.4-39	21
5.4-40	-
5.4-41	-

LIST OF EFFECTIVE PAGES

CHAPTER 5

Page	Amendment	Page	Amendment
5.4-42	13		
5.4-43	20		
5.4-43a	20		
5.4-43b	21		
5.4-43c	21		
5.4-44	18		
5.4-45	-		
5.4-46	21		
5.4-47	19		
5.4-48	6		
F5.4-1	18		
F5.4-2	-		
F5.4-3	18		
F5.4-4	18		
F5.4-5	-		
F5.4-6	22		
F5.4-7	21		
F5.4-8	21		
F5.4-9	18		
F5.4-10	-		
F5.4-11	21		
F5.4-12	13		
F5.4-13	21		
F5.4-14	18		
F5.4-15	-		
F5.4-16	22		
F5.4-17	22		

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

CHAPTER 5

TABLE OF CONTENTS

CHAPTER 5

TABLE OF CONTENTS (Cont'd)

CHAPTER 5

LIST OF TABLES

LIST OF TABLES (Cont'd)

CHAPTER 5

LIST OF TABLES (Cont'd)

LIST OF FIGURES

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

This chapter was originally prepared to describe the reactor coolant system during the initial fuel cycle. Much of the original text is retained for historical record. However, where applicable, changes have been made to reflect the uprating of the unit to a stretch power rating of 2700 Mwt and then to an EPU power rating of 3020 MWt. The corresponding EPU normal operating NSSS power is 3034 MWt, which accounts for net heat addition to the NSSS from other sources, such as the reactor coolant pumps. This increase has a negligible impact on the structural evaluations of the RCS. Consequently the EPU does not have a significant impact on the design bases of the RCS. Where information associated with the higher power level is not available the existing information is identified as Cycle 1. Replacement Steam Generators of type 89/19TI were installed in Unit 2 in the fall of 2007.

5.1 SUMMARY DESCRIPTION

The reactor is a pressurized water reactor with two coolant loops. The Reactor Coolant System (RCS) circulates water in a closed cycle, to remove heat from the reactor core and transfers it to a secondary (steam generating) system. 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 main steam by the steam generator tubes and the steam generator tube sheet. The RCS is a closed system thus forming a barrier to the release of radioactive materials.

The arrangement of the RCS is shown on Figures 5.1-1 and 5.1-2. 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 containment.

Reactor Coolant 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 a 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 chemical content 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; two outlet nozzles to the Shutdown Cooling System, one in each reactor vessel outlet pipe; two pressurizer spray nozzles; vent and drain connections; and sample and instrument connections.

Overpressure protection for the reactor coolant pressure boundary is provided by three spring-loaded ASME Code pressurizer safety valves connected to the top of the pressurizer. Two power operated relief valves are provided to minimize the opening of the pressurizer safety valves. All 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 via the quench tank relief valve and/or a rupture disc installed in the tank.

Overpressure protection for the secondary side of the steam generators is provided by 16 spring-loaded ASME Code safety valves located in the Main Steam System upstream of the steam line isolation valves (see Section 10.3).

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.

Design parameters of the RCS are listed in Table 5.1-1. Table 5.1-2 lists RCS volumes.

Shielding requirements of the surrounding concrete structures are described in Chapter 12. Reactor Coolant System shielding permits limited personnel access to the containment during power operation. The reactor vessel is enclosed by a primary shield wall. 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 flowrates at major components are listed in Table 5.1-3. These parameters are referenced to Figures 5.1-3 and 5.1-4, 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 Figures 5.1-3, 5.1-4, 5.1-4a and 5.1-4b.

5.1.2 PIPING AND INSTRUMENT DIAGRAM

Figures 5.1-3, 5.1-4, 5.1-4a and 5.1-4b comprise the piping and instrumentation diagram of the RCS. Figures 5.1-3 and 5.1-4 depict the reactor coolant pressure boundary and illustrate the use of the isolation design features utilized between the RCPB and connected systems. The entire RCS is located within the containment. Fluid systems which are connected to the RCS and which include the limits of the reactor coolant pressure boundary (as defined in 10 CFR 50.2 (V)) are identified and the appropriate piping and instrumentation diagrams are referenced. Figure 5.1-6 is the piping and instrumentation diagram for the reactor coolant pumps.

5.1.3 ELEVATION DRAWINGS

Reactor Coolant System plan and elevation drawings are provided as Figures 5.1-1 and 5.1-2.

Major components of the RCS are surrounded to the extent possible by concrete structures which provide support, shielding and missile protection. Elevation drawings illustrating principal dimensions of the RCS and its relationship to the surrounding concrete structures are provided as Figures 1.2-8 through 1.2-11.

5.1.4 OPERATION WITH THE RCS AT REDUCED INVENTORY OR MID-LOOP CONDITIONS

Background

Following reactor shutdown ($k_{eff} < 1.0$) irradiated fuel continues to produce substantial quantities of heat due to the decay of fission products, primarily through the emission of gamma (γ) rays. Most of this decay heat is deposited in the reactor coolant and is subsequently removed from the RCS by the shutdown cooling heat exchangers. Sometimes routine reactor maintenance, such as the replacement of a reactor coolant pump seal, can require opening the cold leg while the RCS is partially drained. For this work, the RCS is drained to approximately the mid-plane of the hot leg piping; this condition is referred to as "mid-loop". Reducing the RCS water inventory has the effect of decreasing the system response time to any loss of decay heat removal capability. The plant is considered to be in mid-loop conditions when the reactor vessel water level is below the top of the hot leg and at or above the mid-plane of the hot leg piping. The term "reduced inventory" refers to a water level beginning 3 feet below the reactor vessel flange and continuing down to the top of the hot leg. This definition is consistent with that used by the NRC in Generic Letter 88-17, Loss of Decay Heat Removal (Reference 1).

In the event of a loss of shutdown cooling while at a reduced inventory or mid-loop condition, the deposited γ-ray energy would heat the core coolant inventory to saturation and begin to boil-off the water remaining in the reactor vessel above the core. If this boiling condition persists, reactor fuel will be uncovered from the loss of inventory. Substantial quantities of steam will be evolved during any inventory boil-off.

Reference 2EPU calculations have shown that the boil-off of RCS inventory at low pressures is a relatively slow evolution that requires two hours or more to reduce the RCS water level below the top of the active fuel. However, if the steam generated by the boil-off process is not effectively vented from the system, the pressure within the reactor vessel upper plenum may increase, depressing the reactor vessel water level such that active fuel is exposed. This scenario, which also requires the presence of an opening in the RCS cold leg and installation of both hot leg nozzle dams, leads to a more rapid core uncovery than does the low pressure boil-off scenario.

For example, if the pressurizer manway is open and the reactor vessel water level is below the top of the hot leg piping, a direct steam vent pathway is available. However, water levels above the mid-plane of the hot leg increase the k_{90} loss factor, leading to an increase in the loss (or resistance) coefficient K and lower steaming rates, until pressurization forces additional inventory into the cold legs.

To preclude rapid core uncovery following any loss of shutdown cooling, the timing of the RCS drain down to mid-loop conditions is constrained by the following factors: 1) sequencing of nozzle dam installation; 2) the available RCS vent area; 3) the steam production rate due to inventory boil-off; and 4) the containment closure capability. The steam production rate after a loss of shutdown cooling is proportional to the decay heat generation rate. The relationship between hot side vent area and the steam evolution rate is critical to maintaining a stable (and acceptably small) pressurization of the core and upper plenum region in the event of a loss of shutdown cooling when steam generators are not available for heat removal. The ability to close containment openings quickly aids in minimizing offsite radiological dose.

Reference 4 derives the vent area required to avoid excessive pressurization of the RCS following any loss of shutdown cooling.

REGULATORY BASIS

Initially, concerns related to operation while at reduced inventory arose following a loss of shutdown cooling event at the Diablo Canyon site in 1987. As a result of the event, the NRC prepared regulatory guidance in the form of Generic Letters (GL), including GL 88-17, Loss of Decay Heat Removal to discuss a number of phenomena recognized as affecting nuclear plant operation when these plants are operating in a non-power condition. Some of the phenomena identified in GL 88-17 can cause the time between the loss of decay heat removal and the onset of severe core damage to be as short as one hour. The common industry understanding prior to issuance of GL 88-17 was that core damage required a loss of shutdown cooling of about 4 hour duration. Enclosure 1 of GL 88-17 went on to discuss the six areas of concern listed below:

1) Pressurization
2) Vortexing
3) Draining the S/G U-tubes
4) RCS level differences
5) Design characteristics of decay heat removal systems
6) Instrumentation

Of these six items, the NRC's principal concern was that pressurization could occur as a result of conditions unique to operation with a reduced RCS inventory – and that excessive pressurization could adversely affect plant safety. After shutdown cooling was lost at Diablo Canyon, the inventory of core coolant began to boil in 30 to 45 minutes. More importantly, the boiling led to pressurization of the RCS, which was an unanticipated outcome. As previously noted, in the presence of a cold leg opening co-incident with installed hot leg nozzle dams pressurization of the RCS can depress the water level in the reactor vessel, leading to a loss of inventory and an early core uncovery.

These areas of concern formed the basis for the expeditious and programmatic action requirements imposed on licensees, including St. Lucie. Section 2.2.2 of Enclosure 2 to GL 88-17 provided guidance for licensees on establishing interim restrictions for draining the RCS. The goal of the interim guidance was to have licensees implement procedures and administrative controls that would reasonably assure containment closure prior to a core uncovery resulting from a loss of decay heat removal (DHR) and the coincident failure of alternate makeup capability. The NRC anticipated that this interim guidance would later be supplanted by site-specific guidance developed by licensees and industry owners groups. NRC also envisioned that periodic updates and revisions to this guidance would be developed using 10 CFR 50.59 criteria.

For reduced inventory and mid-loop operations with irradiated fuel present in the reactor vessel, an open vent path through the pressurizer manway is required. Under some conditions it requires opening an additional vent pathway through ICI Quickloc assemblies. The RCS venting calculation quantifies the available ICI vent capacity with and without upper flange thread protection devices installed. The available ICI vent capacity without thread protectors installed is slightly greater than the ICI vent capacity available when thread protectors are installed. To credit this ICI vent path, removal of the ICI Quickloc seal assemblies is required prior to entry into reduced inventory conditions. With a vent pathway available through the pressurizer manway and at least six ICI Quickloc assemblies, adequate venting is present for all times after shutdown greater than 81 hours.

Time to Boil

Based on revised EPU decay heat loads, times to boil and the time required to boil-off enough coolant inventory to uncover the core have been determined. The input assumptions used will bound both current and future operating cycles.

When the RCS is drained to mid-loop and the initial coolant temperature is 120°F, the time to boil is 10.2 minutes at 72 hours after shutdown and 11.2 minutes at 90 hours after shutdown. Both of these values assume an equivalent operating cycle length of 16000 EFPH for previous fuel cycles. If the initial coolant temperature is assumed to be 110°F, the time to boil increases to 11.3 minutes at 72 hours after shutdown and 12.4 minutes at 90 hours after shutdown. At RCS temperatures above ~90°F, the time to boil from mid-loop conditions is less than 26.4 minutes for the first 360 hours after reactor shutdown.

Increasing the water level in the reactor vessel above the core has a minor effect on the time to boil. At 90 hours after shutdown, increasing the RV water volume by 100 ft^3 (748 gallons) delays the onset of core boiling by 0.857 minutes, if the initial temperature is 120°F.

Time to Core Uncovery

Assuming the RCS pressure remains at approximately 15 psia (i.e. depressurized), the time to core uncovery from mid-loop conditions as a function of initial coolant temperature and time after shutdown has been evaluated for EPU conditions. No inventory makeup is assumed available. At 72 hours after shutdown and with an initial temperature of 120°F, core uncovery occurs 110 minutes (1.83 hours) after a loss of shutdown cooling. Decreasing the initial temperature to 110°F increases the time to core uncovery by 2 minutes to 112 minutes. At 90 hours after shutdown the corresponding the corresponding values are 121 minutes (2.02 hours) and 123 minutes, respectively. If the reactor has been shutdown for only 60 hours, water level in the core will drop below the top of the active fuel 103 minutes (1.72 hours) after shutdown cooling is lost from mid-loop conditions when the initial temperature is 120°F.

Steaming Rates and Required Vent Area

The rate of steam generation following any loss of shutdown cooling has been determined for EPU conditions for reduced inventory and mid-loop conditions. Steam production rates in the range of 10.2 lbm/sec to 13.9 lbm/sec could be expected following a loss of shutdown cooling from hot mid-loop conditions at Unit 2.

The upper limit on acceptable pressurization can be determined by considering the elevation difference between the RCS cold legs and the top of the core. Pressurization that will raise a water column by more than this amount could fill the cold legs with liquid inventory that should be covering the core and lead to a loss of inventory through any hole in the cold leg.

For Unit 2, the maximum permissible pressurization of the core/upper plenum region that avoids inventory loss to the cold legs is on the order of 3.0 psig. This is the basis for the 17.7 psia value (14.7 + 3.0) discussed in Reference 6.

Vent Area and Time after Reactor Shutdown

The cross sectional area of the pressurizer surge line is the limiting dimension constraining steam flow through an open pressurizer manway. The Unit 2 pressurizer surge line has a cross-sectional area of 0.5592 ft^2. The EPU analysis uses a K-factor of 3.38 for the surge line/pressurizer manway combination. Results from the orifice flow equation used to calculate the steaming rate through the open manway are sensitive to the flow resistance (K) factor used. The K-factor used in the EPU analysis is explicitly calculated through review of the appropriate drawings. This was utilized in the analysis that determined the required hot side vent area as a function of cycle length and time after shutdown.

The analysis determined a required cooling time of 110 hours after shutdown if the equivalent operating cycle length for past operation was ≤16000 EFPH (this bounds 18 month fuel cycles) and 114 hours if the equivalent operating cycle length for past operation was characterized by 24-month cycles. These cooling times will ensure that the pressurizer manway vent path alone has sufficient capacity to maintain the core and upper plenum region pressure not greater than 3.0 psig.

To reduce these required cooling times, an additional vent path on the reactor head must be opened.

Each ICI Quickloc flange assembly through which a vent path is opened provides an additional 0.286 lbm/sec of steam venting capacity at an assumed 3.0 psid. The venting capacity of an ICI assembly is greater if the upper flange thread protector is not installed. If a vent path is established through any six of the ICI Quickloc assemblies and the pressurizer surge line/manway during mid-loop operations, then the combined vent capability is 12.41 lbm/sec. Therefore, removing the pressurizer manway and removing at least six of the ICI Quickloc seal assemblies will provide a vent path adequate to prevent pressurization above 3.0 psig in the event of a complete loss of shutdown cooling for times of 81 hours or greater after shutdown.

Containment Closure

The undesirable containment environment that will rapidly ensue following a loss of shutdown cooling requires that containment closure be promptly initiated on any loss of shutdown cooling. The containment closure requirements for St. Lucie Unit 2 ensure that potentially adverse environmental factors present in the containment atmosphere do not effect the closure evolution.

Radiological Consequences

The radiological consequences of a sustained loss of shutdown cooling with an open containment equipment hatch have not been evaluated for either St. Lucie unit. This event is not a design basis event at St. Lucie.

A qualitative assessment of the dose consequences inside and outside containment was performed for Ft. Calhoun in 1988 by the CEOG (see Reference 7). These calculations used conservative inputs, such as a 1% failed fuel fraction, and concluded the site boundary whole body doses would be in the tens of millirem range. This analysis also concluded that the in-containment isotopic concentrations of many species would exceed 10 CFR Part 20 limits once core boil-off has begun.

Pump Makeup to Compensate for Inventory Boil-off
An EPU analysis determined that two charging pumps can provide sufficient makeup capability to compensate for inventory boil-off if the reactor has been shutdown for at least 72 hours. In addition to charging pumps other makeup sources, including a high pressure safety injection (HPSI) pump, are available. HPSI pump makeup capability is greater than the boil-off rate from any credible mid-loop condition, so depending on the location of any RCS openings the added makeup will either flow out through a hole or increase the level of the RCS. If makeup flow is not required at a time when low elevation openings are present on the hot side of the RCS, it may be desirable to throttle HPSI pump flow some time after initiation to match the boil-off rate so to avoid spilling injection flow on the containment floor.

PLANT RESTRICTIONS

St. Lucie Unit 2 Plant procedures include the required restrictions and operational guidance for reduced inventory and mid-loop conditions.

REFERENCES FOR SECTION 5.1.4

1) Loss of Decay Heat Removal (Generic Letter No. 88-17, including Enclosures), U.S. Nuclear Regulatory Commission, October 17, 1988.

2) Deleted

3) Deleted

4) Attachments to CEOG letter CEOG-88-599, Task 555 Draft Final Report, "Loss of RHR Scenarios, Detailed Qualitative Assessment", CE-NPSD-421, Revision 01, October 17, 1988.

5) NRC Generic Letter 87-12, Loss of Residual Heat Removal (RHR) While the Reactor Coolant System (RCS) is Partially Filled, July 9, 1987.

6) Deleted

7) Engineering Evaluation of Fort Calhoun Station, Loss of SDC at Mid-Loop Conditions, prepared for the Omaha Public Power District, Final Report, September 1988.

TABLE 5.1-1

DESIGN PARAMETERS OF REACTOR COOLANT SYSTEM

	Cycle 1	EPU
Design Thermal Power, Mwt (Including net heat addition from RCPs)	2570	3050
Thermal Power, Btu/hr	8.77×10^{9}	1.03×10^{10}
Design Pressure, psia	2500	2500
Design Temperature (except Pressurizer), °F	650	650
Pressurizer Design Temperature, °F	700	700
Reactor Coolant System Design Flow Rate, lb/hr	122×10^{6}	140.8×10^{6}
Cold Leg Operating Temperature, °F	550	551
Coolant Average Operating Temperature, °F	577	578.5
Hot Leg Operating Temperature, °F	604	606
Normal Operating Pressure (psia)	2250	2250

TABLE 5.1-2

REACTOR COOLANT SYSTEM VOLUMES

Component	Volume (Ft^3)
Reactor Vessel	4615
Steam Generator	1900.22 (each)
Reactor Coolant Pump	112 (each)
Piping	
Hot Leg	140 (each)
Suction Leg	112 (each)
Discharge Leg	81 (each)
Pressurizer	1509
Surge Line	29

TABLE 5.1-3

PROCESS DATA POINT TABULATION*

Cycle 1 1

Parameter	Pressurizer	S.G. 2A1 Midpoint	RCP 2A1 Outlet	R.V. Midpoint	RCP 2A2 Outlet	S.G. 2B1 Midpoint	RCP 2B1 Outlet	RCP 2B2 Outlet
Data Point Figures 5.1-3 and 5.1-4	1	2	3	4	5	6	7	8
Pressure, psia	2250	2237	2296	2277	2296	2237	2296	2296
Temperature, F	653	572	548	572	548	572	548	548
Mass Flow Rate, lb/hr	—	69.7×10^6	34.85×10^6	139.4×10^6	34.85×10^6	69.7×10^6	34.85×10^6	34.85×10^6
Volumetric Flow Rate, gpm	—	191,850	92,500	383,700	92,500	191,850	92,500	92,500

* Full power operating parameters corresponding to the minimum allowable Reactor Coolant Flow Rate, defined in Table 4.4-10.

Refer to Drawing
2998-3793

FLORIDA POWER & LIGHT COMPANY **ST. LUCIE PLANT UNIT 2**
REACTOR COOLANT SYSTEM ARRANGEMENT - PLAN **FIGURE 5.1-1**

Refer to Drawing
2998-3794

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR COOLANT
SYSTEM ARRANGEMENT -
ELEVATION
FIGURE 5.1-2

Refer to Drawing
2998-G-078 SH 110

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
REACTOR COOLANT SYSTEM

FIGURE 5.1-3

Amendment No. 18 (01/08)

Refer to Drawing
2998-G-078 SH 109

FLORIDA POWER & LIGHT COMPANY **ST. LUCIE PLANT UNIT 2**
FLOW DIAGRAM REACTOR COOLANT SYSTEM **FIGURE 5.1-4**

Amendment No. 18 (01/08)

Refer to Drawing
2998-G-078 SH 108

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
REACTOR COOLANT SYSTEM

FIGURE 5.1-4a

Refer to Drawing
2998-G-078 SH 107

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
REACTOR COOLANT SYSTEM

FIGURE 5.1-4b

Amendment No. 18 (01/08)

DELETED

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FIGURE 5.1-5

Refer to Drawings
2998-G-078 SH 111A, B, C, D

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

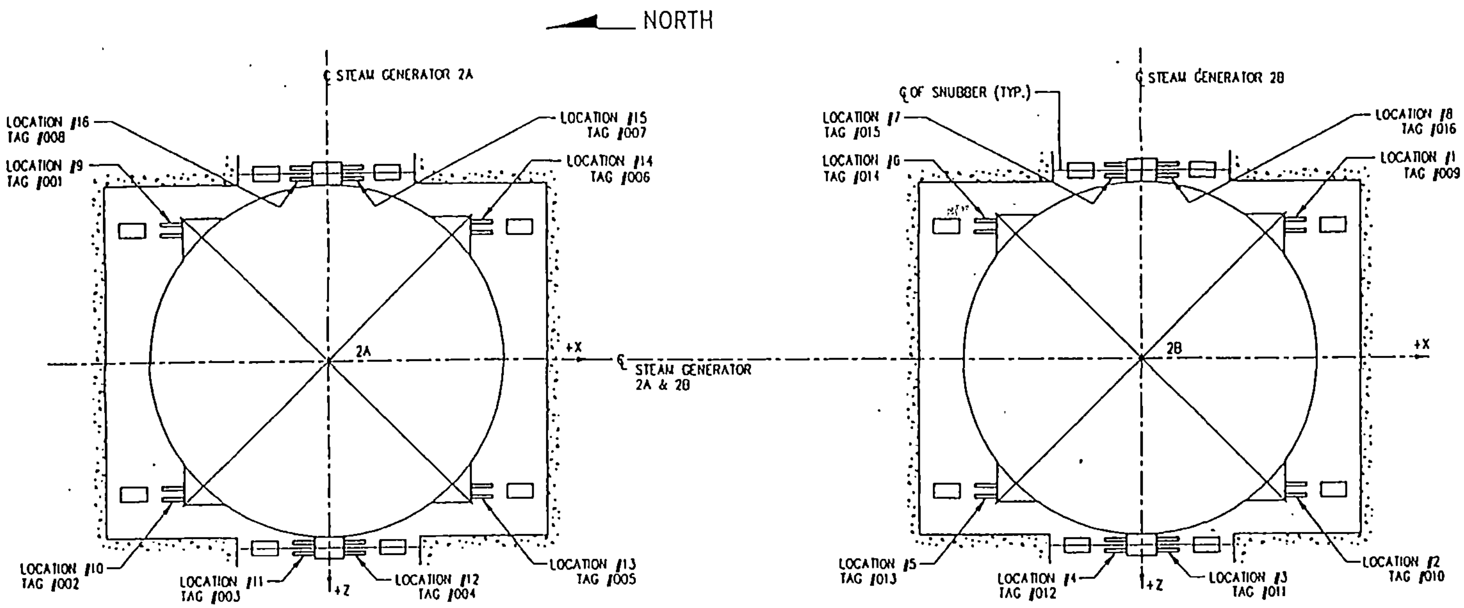
FLOW DIAGRAM
REACTOR COOLANT PUMPS

FIGURE 5.1-6

DELETED

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
REACTOR COOLANT
SYSTEM
FIGURE 5.1-6a



PLAN AT EL. 76.00"
STEAM GENERATOR UPPER SUPPORT

FLORIDA POWER & LIGHT
ST. LUCIE UNIT 2

STM. GEN. SNUBBER SYSTEM
VALVE IDENTIFICATION

DWG. CE-068-292-35 REV. 0

LOCATION #	TAG #	MARK #
9	001	SS-1-2A
10	002	SS-2-2A
11	003	SS-3-2A
12	004	SS-4-2A
13	005	SS-5-2A
14	006	SS-6-2A
15	007	SS-7-2A
16	008	SS-8-2A
1	009	SS-1-2B
2	010	SS-2-2B
3	011	SS-3-2B
4	012	SS-4-2B
5	013	SS-5-2B
6	014	SS-6-2B
7	015	SS-7-2B
8	016	SS-8-2B

Amendment No. 12 (12/98)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CE STEAM GENERATOR SUPPORT
SNUBBER SYSTEM

FIGURE 5.1-7

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY (RCPB)

5.2.1 COMPLIANCE WITH CODES AND CODE CASES

5.2.1.1 Compliance with 10 CFR 50.55a

The code class, edition and addenda for the RCPB are listed in Table 5.2-1 and are in accordance with the provisions of 10 CFR 50.55a.

5.2.1.2 Applicable Code Cases

The Code Cases applied in the construction of the reactor coolant pressure boundary components and those used for modifications to instrument nozzles are listed in Table 5.2-2. Except as noted, all of the code cases listed have been included in Regulatory Guides 1.84, "Code Case Acceptability ASME Section III Design and Fabrication," March 1977 (R9) or 1.85, "Code Case Acceptability ASME Section III Materials," March 1977 (R9) as approved cases.

The original Reactor Vessel Closure Head and the original CEDMs have been replaced. The Code Cases used in the replacement components are listed in Table 5.2-2.

5.2.2 OVERPRESSURIZATION PROTECTION

5.2.2.1 Design Bases

Appendix 5.2A presents the design bases for sizing the overpressurization protection system. The loss of load transient which is used to size the pressurizer safety valves is not a design transient for any other component in the reactor coolant pressure boundary.

5.2.2.2 Design Evaluation

Section 15.2 provides an evaluation of the functional design of the overpressure protection system. In this analysis, the capability of the overpressure protection system to maintain secondary and primary operating pressures within 110 percent of design is clearly demonstrated.

The analytical model and assumptions used in the analysis are discussed in Section 15.2.

The analysis demonstrates that sufficient relieving capacity has been provided so that when acting in conjunction with the Reactor Protective System the safety valves prevent exceeding 110 percent of the design pressure.

5.2.2.3 Piping and Instrumentation Diagram

The piping and instrumentation diagram showing the primary safety valves and their associated discharge lines are shown on Figures 5.1-3 and 5.1-4. The secondary safety valves are shown on Figure 10.1-1.

5.2.2.4 Equipment and Component Description

5.2.2.4.1 Pressurizer Safety Valves

See Subsection 5.4.13 for a description on pressurizer safety valves. The pressurizer safety valves discharge into the pressurizer quench tank. The pipe diameter, pipe length and routing is shown on Figure 5.2-1.

5.2.2.4.2 Main Steam Safety Valves

See Subsection 10.3.2 for a description of main steam safety valves.

5.2.2.5 Mounting of Pressure-Relief Devices

Mounting of the pressurizer safety valves and main steam safety valves are described in Subsections 5.4.13 and 10.3.2 respectively.

5.2.2.6 Applicable Codes and Classification

The applicable codes and classification for the overpressure protection system are contained in Subsections 3.2.1 and 3.2.2. Additional component information can be found in Subsections 5.2.1, 5.4.11, 5.4.13 and Section 10.3.

5.2.2.7 Material Specification

Material specifications for the overpressure protection system are given in Subsections 5.2.3, 5.4.13, and 10.3.6.

5.2.2.8 Process Instrumentation

Process instrumentation for the overpressure protection system is shown on Figures 5.1-3 and 5.1-4 and described in Chapter 7. Instrumentation associated with pressurizer relief discharge is described in Subsection 5.4.11.

5.2.2.9 System Reliability

Reliability of the main steam safety valves is discussed in Subsection 10.3.3. The pressurizer safety valves are spring actuated mechanisms, and cannot close when setpoint pressure is exceeded. The operational reliability of the pressurizer safety valves is assured by:

a) Compliance with ASME Code, Sections III and XI for safety valves

b) Conservative design criteria

c) Selection of a vendor with proven experience and expertise

d) Accounting for thermal cycling during valve operation

e) Technical Specifications

5.2.2.10 Testing and Inspection

Each safety valve undergoes initial testing by the valve vendor. Subsequent testing and inspection of the pressurizer safety valves is governed by ASME Code, Section XI, Subsection IWV. Testing and inspection of the main steam safety valves is discussed in Subsection 10.3.4.

5.2.3 REACTOR COOLANT PRESSURE BOUNDARY MATERIAL

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 5.2-4.

To reduce sensitivity to neutron-induced changes in service, low residual requirements for copper, phosphorous, and vanadium are imposed on plate and weld materials in the reactor vessel belt-line. The core beltline region as defined by Appendix G of 10 CFR 50 includes the intermediate and lower shell courses and their longitudinal weld seams. Also included is the girth seam joining the intermediate to lower shell courses. The chemical content of the reactor vessel beltline plate and weld material as determined by chemical analysis is given in Tables 5.2-5 and 5.2-6.

5.2.3.2 Compatibility With Reactor Coolant

5.2.3.2.1 Reactor Coolant Chemistry

Controlled water chemistry is maintained within the Reactor Coolant System. 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 Reactor Coolant System are given in Subsection 9.3.4.

5.2.3.2.2 Materials Compatibility

The materials of construction used in the reactor coolant pressure boundary and for modifications to instrument nozzles which are in contact with reactor coolant are designated by an "a" in Tables 5.2-3 and 5.2-4. 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

5.2.3.2.3.I NSSS Components

The possibility of leakage of reactor coolant onto the parts of the reactor coolant pressure boundary causing corrosion of the pressure boundary has been investigated for reactors similar to St Lucie Unit 2. Tests have shown that Reactor Coolant System leakage onto surfaces of the reactor coolant pressure boundary does not affect the integrity of the pressure boundary.

The reactor vessel closure head dome and lower portion of the reactor vessel are insulated with stainless steel reflective insulation to minimize insulation contamination in the event of a spillage. Metal encapsulated fiberglass insulating wool blanket is used to insulate the reactor vessel closure head flange.

The steam generators are insulated with stainless steel reflective insulation in response to NRC Generic Letter 2004-02 "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," which satisfies the requirements provided in the guidance report, NEI-04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology."

The quantity of leachable halogens is in accordance with Regulatory Guide 1.36, "Non-metallic Thermal Insulation for Austenitic Stainless Steel," February, 1973 (R0).

The other NSSS components are insulated with non-metallic insulation which is in accordance with the requirements of Regulatory Guide 1.36 (R0).

5.2.3.2.3.2 AE-Supplied Components

The piping within the RCPB is insulated for thermal, personnel or anti- sweat requirements. All insulation used inside the containment consists primarily of fiberglass or metallic reflective insulation. The quantities and locations of each type of insulation are provided in Table 6.2-40; the materials of construction of each type of insulation are listed in Table 6.2-39.

The insulation on stainless steel components is in accordance with the requirements of Regulatory Guide 1.36 (R0).

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

The tests and acceptance requirements of 10 CFR 50, Appendix G, are applied to the reactor coolant pressure boundary ferritic materials, bolting and weld materials used for fabrication of the reactor vessel, steam generators (primary side), pressurizer, and the reactor coolant piping.

An alternate procedure is used for tributory nozzles (spray and let down/ drain) in the reactor coolant piping. This material is tested in accordance with the ASME Code, Section III, 1971 edition through Winter 1972 addenda to, Paragraph NB-2332.a, which requires three Charpy V-notch specimens tested at a single temperature for pumps, valves, and fittings with pipe connection of nominal wall thickness 2-1/2 inches or less. The material must exhibit a minimum of 40 mils lateral expansion at or below the lowest service temperature. Drop weight specimens are not required, therefore no T_{NDT} has been determined for this material. However, the lateral expansion requirement is more severe than that required by 10 CFR 50 Appendix G and a conservative RT_{NDT} has been estimated by use of the Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements for Older Plants," as shown in Table 5.2-12. These RT_{NDT} values were estimated using the CVN values presented in Table 5.2-12A. The lowest service temperature for these nozzles is 40°F.

The material for bolting and other fasteners greater than one inch diameter is purchased to the ASME Code Edition and Addenda specified for the component in which they are used. Table 5.2-1 lists Code Editions and Addenda for the components of the reactor coolant pressure boundary. These materials meet the requirements of the Code Edition and Addenda to which they are ordered.

Although these materials are ordered prior to the publication of 10 CFR 50 Appendix G, sufficient testing to demonstrate compliance with Appendix G is performed in all but two cases. The acceptability of these two cases can be demonstrated, however, as follows:

a) The 2-3/8 inch hex bar of SA-193 Gr. B7 used for pressurizer manway nuts, material ID No. C-5364, displays 25-53 ft-lbs absorbed energy and 18-33 mils lateral expansion at 10°F (see Table 5.2-13a). Since full curves are not required for this material, testing over a range of temperatures is not normally done. Examination of the percent shear results show that the tests were done in the transition region of the Charpy curve; the range of shear values was 40-80 percent and two of the three specimens exhibited over 50 percent shear. At the 10°F test temperature two specimens met the 25 mils lateral expansion requirement. At a temperature 30°F higher (40°F), it is expected that all specimens would exceed 25 mils lateral expansion. The data for SA-193 Gr. B-7 material, given in Table 5.2-13c, exhibits in excess of 30 mils lateral expansion at a minimum of 66 percent shear. Thus, had testing been done at 40°F, the requirements of 10 CFR 50 Appendix G would have been met.

   The heat treatment for material ID. No. C-5364 is given in Table 5.2-13b; the heat treatment for the Table 5.2-13c data is given in Table 5.2-13d. Each of these heat treatments produce similar metallurgical structures in this alloy.

   The preload temperature for these fasteners, equivalent to their lowest service temperature, is 70°F.

b) No data on lateral expansion are reported for material ID. No. C-5365, SA540 Gr. B-24, a 1.656 inch diameter bar used for pressurizer manway studs. This material exhibited absorbed energy values of between 54 and 58 ft-lbs 10°F, well in excess of the required 45 ft-lbs. WRC Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," August 1972, contains energy (ft-lbs) versus lateral expansion (mils) data for bolting steels of the 4340 (SA-540 Gr. B-23 and B-24) composition. Based on this data, at least 30 mils lateral expansion was obtained for bolting steels exhibiting a minimum of 54 ft-lbs absorbed energy. Therefore, the lateral expansion requirement (25 mils) for the SL2 material (ID. No. C-5365) would be met at 10°F.

Fracture toughness data as required by 10 CFR 50, Appendix G is presented in Tables 5.2-7 through 5.2-13. Charpy V-notch test results are shown on Figures 5.2-2 through 5.2-23c.

All ferritic reactor coolant pressure boundary (RCPB) welds were made using submerged arc or covered electrode weld process. The fracture toughness data for the beltline welds are included in Table 5.2-7a. Impact test data for all weld materials used in the beltline region were taken from the weld metal certification tests of NB-2400. The highest RT_{NDT} for the beltline region weld materials is -40°F. In all cases RT_{NDT} is fixed by the NDT temperature.

The test specimens for core beltline welds are made with the same pre, interpass, and post-weld temperature requirements as the reactor vessel welds. The specimens receive a 40 hour, 1150 ± 25°F stress relief treatment equivalent to the stress relief given to the vessel. The test specimens are in the same metallurgical condition as are the welds in the core beltline region.

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

The personnel performing impact testing were qualified in accordance with the ASME Boiler and Pressure Vessel Code. Compliance with the ASME B&PV Code is verified by Combustion Engineering quality assurance procedures, and reviewed by ASME and NRC audits.

As required by the Code, the personnel performing impact testing are certified by qualified supervisory personnel. Records of this certification are maintained in accordance with NA-4900, "Records and Data Reports," and are available for review at Combustion Engineering's Chattanooga facility. Combustion Engineering training methods comply with the revision of 10 CFR 50 Appendix G published November 14, 1980 "for comment".

All of the ferritic pressure retaining materials used in the fabrication of the replacement Reactor Vessel Closure Head (RVCH) 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 ASME Code Section III, Division 1, 1989 Edition No Addenda.

5.2.3.3.2 Control of Welding

5.2.3.3.2.1 Avoidance of Cold Cracking

5.2.3.3.2.1.1 NSSS Components

St. Lucie Unit 2 components conform to NRC Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," May 1973 (R0) 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 steels 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 inch and 30 inch Reactor Coolant System piping and pressurizer.

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 (R0) 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, "Control of Stainless Steel Weld Cladding of Low Alloy Steel Components," May 1973 (R0), major Reactor Coolant System 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.

Moisture Control for low hydrogen covered arc welding electrodes is consistent with Subparagraph NB-2440 of ASME Code, Section III and the requirements of SFA 5.1, "Specification for Mild Steel Covered Arc Welding Electrodes," of ASME Code, Section II.

The replacement Reactor Vessel Closure Head is SA-508, Class 3 forging material. All high heat input welding procedures used to apply the cladding were first qualified by undergoing Intergranular Separation Test as part of the weld procedure qualification. The base materials for the procedure qualification were of the same specification and grade as were used in the cladding of the Reactor Vessel Closure Head.

5.2.3.3.2.1.2 AE-Supplied Components

Low-alloy steels are not utilized in any AE-Supplied RCPB components, therefore, Regulatory Guides 1.50 and 1.43 are not applicable.

5.2.3.3.2.2 Conformance to Regulatory Guide 1.34

Regulatory Guide 1.34, "Control of Electroslag Weld Properties," December, 1972 (R0), addresses controls to be applied during welding using the electroslag process. The electroslag process has not been used in the fabrication of any reactor coolant pressure boundary components. Therefore, the recommendations of this guide are not applicable.

5.2.3.3.2.3 Conformance to Regulatory Guide 1.71

5.2.3.3.2.3.1 NSSS Components

St. Lucie Unit 2 does not comply with the specific requirements of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," December, 1973 (R0). 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 procedure parameters and welding qualification requirements are certified in accordance with Sections III and IX. Weld quality is verified by the performance of the required nondestructive examination.

5.2.3.3.2.3.2 AE-Supplied Components

Conformance to Regulatory Guide 1.71 (R0) is provided in Subsection 5.2.3.4.2.1.2.

5.2.3.3.3 Nondestructive Examination of Tubular Products

5.2.3.3.3.1 NSSS Components

All tubular products used for components of the reactor coolant pressure boundary are nondestructively examined in accordance with the requirements of the ASME Code, Section III, Division 1, with the applicable edition and addenda as listed in Table 5.2-1. In addition the non-destructive examination requirements of all these tubular products (except the two components noted below) are consistent with the recommendations of Regulatory Guide 1.66, "Nondestructive Examination of Tubular Products," October 1973 (R0).

The two components (pressurizer heater tubing and heater sleeve tubing) are ultrasonically tested in accordance with the requirements of the ASME Code Addenda for the 1971 Edition Summer 1972 Addendum.

Reactor vessel instrument tubing and CEDM nozzle housings for the replacement RVCH were ultrasonically examined in accordance with NB-2552.2, in addition to the examinations required by NB-2551(b) of ASME Code Section III, Division 1, 1989 Edition, No Addenda.

5.2.3.3.3.2 AE-Supplied Components

Class 1 and 2 tubular products of the reactor coolant pressure boundary are examined in accordance with the requirements of ASME III, NB 2552 and NB 2560. See Table 5.2-1 for code dates. All pressure retaining forgings associated with inlet piping connections of 2-1/2 inches nominal size and over, are 100 percent liquid penetrant examined. All bevelled weld ends and longitudinal welds are liquid penetrant examined.

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

St. Lucie Unit 2 is consistent with the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," May 1973 (R0), as described in items a) through e), except for the criterion used to demonstrate freedom from sensitization. The ASTM A708 Strauss Test was used in lieu of the ASTM A262 Practice E, Modified Strauss Test, to demonstrate freedom from sensitization in fabricated, unstabilized, stainless steel.

a) Solution Heat Treatment Requirements

All raw austenitic stainless steel material, both wrought and cast, used in the fabrication of the major NSSS components in the reactor coolant pressure boundary, was supplied in the annealed condition as specified by the pertinent ASTM or ASME Code; viz, 1900-2050°F for 1/2 to one hour per inch of thickness and water quenched to below 700°F. The time at temperature was determined by the size and type of component.

Solution heat treatment was 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, was conducted to determine the effect of various welding procedures on the susceptibility of unstabilized 300 series stainless steels to sensitization-induced intergranular corrosion. Only those procedures and/or practices demonstrated not to produce a sensitized structure were used in the fabrication of these reactor coolant pressure boundary components. The ASTM standard A708 (Strauss test) was 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 A708 was utilized as a go/no-go standard for acceptability.

As a result of the above tests, a relationship was established between the carbon content of 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 steels with carbon contents of more than 0.03 percent used for components of the reactor coolant pressure boundary are Type 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.

Duplex, austenitic stainless steels, containing more than 5 FN 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:

CF8M CF8	Cast Stainless Steel	Delta ferrite controlled to 5 FN-28 FN
308 309 312 316	-	Single and combined stainless steel weld filler metals. Delta ferrite controlled to 5 FN-18 FN as deposited.

In duplex austenitic/ferritic alloys, chromium-iron carbides are precipitated preferentially at the ferrite/austenite interfaces during exposure to temperatures ranging from 800°F to 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

Exposure of unstabilized austenitic 300 series stainless steels to temperatures ranging from 800°F to 1500°F results in carbide precipitation. The degree of carbide precipitation or sensitization depends on the temperature, the time at the temperature, and also the carbon content.

Weld heat affected zone sensitized austenitic stainless steels were avoided by careful control of weld heat input to less than 60 kJ/inch, interpass temperature to a maximum of 350°F and carbon content.

Homogeneous or localized heat treatment in the temperature range 800 °F to 1500 °F was prohibited for unstabilized austenitic stainless steel with a carbon content greater than 0.03 percent used in components of the reactor coolant pressure boundary. When stainless steel safe ends were required on component nozzles or piping, fabrication techniques and sequencing required 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 component, the stainless steel safe end is welded to the Inconel overlay, using Inconel weld filler metal.

PC/M 09078M implemented during SL2-19, mitigated cold leg spray and intermediate drain nozzles alloy 600 dissimilar metal (DM) welds by repair and replacement of the DM welds and safe ends. The replacement materials are 304L/316L stainless steel with welding materials of 309L and 316L. Prior to welding the safe end to the nozzle, two layers of ER309L were deposited on the ID and face of the weld prep, followed by two layers of ER316L. Once the nozzle buttering is complete, the 316L safe end was welded to the buttering using a groove weld of ER316L filler.

e) Cleanliness and Contamination Protection

The procedures and practices followed for cleaning and contamination protection of the Reactor Coolant Pressure Boundary components during fabrication, shipment and storage, construction, testing and operation are discussed in Subsection 5.2.3.4.1.2.

5.2.3.4.1.1.2 AE-Supplied Components

The conformance with Regulatory Guide 1.44 (R0) for AE-supplied components is provided in Subsection 6.1.1.

5.2.3.4.1.2 Avoidance of Contamination Causing Stress Corrosion Cracking

5.2.3.4.1.2.1 NSSS Components

Specific requirements for cleanliness and contamination protection are included in the equipment specifications for components fabricated with austenitic stainless steel. The provisions described below indicate the type of procedures utilized for NSSS supplied components to provide contamination control as required by Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," March, 1973 (R0). (See Section 17.2)

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 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 were cleaned to produce an item which is 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). Cleaning water conformed to the following requirements:

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

Prior to shipment, reactor coolant pressure boundary components were packaged in such a manner that they were protected from 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.

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 or phosphate. Results of tests have proven these inhibitors to be completely effective. Operational chemistry specifications restrict concentrations of halide and oxygen, both prerequisites of intergranular attacks (refer to Subsection 9.3.4).

A zinc injection system has been added to allow injection of the zinc acetate solution into the RCS. The concentration of depleted zinc is maintained in the reactor coolant between 5 and 10 ppb. The depleted zinc in the reactor coolant is used primarily as a means to reduce radiation dose rates, but it also will mitigate the occurrence or severity of primary water stress corrosion cracking of Alloy 600.

5.2.3.4.1.2.2 AE-Supplied Components

The specific cleanliness requirements and contamination protection are in accordance with Regulatory Guide 1.37 (R0).

5.2.3.4.1.3 Characteristics and Mechanical Properties of Cold-Worked Austenitic Stainless Steels for Reactor Coolant Pressure Boundary Components

Cold-worked austenitic stainless steel is not utilized for components of the reactor coolant pressure boundary.

5.2.3.4.2 Control of Welding

5.2.3.4.2.1 Avoidance of Hot Cracking

5.2.3.4.2.1.1 NSSS Components

In order to preclude microfissuring in austenitic stainless steel welds, reactor coolant pressure boundary components are consistent with the recommendations of the Interim Position (Branch Technical Position MTEB 5-1) Regulatory Guide 1.31, Control of Stainless Steel Welding, except for the differences noted below.

a) Major Reactor Coolant Pressure Boundary Components, Excluding Reactor Coolant Pumps

The delta ferrite content of A-No. 8 austenitic stainless steel filler metal, except for 16-8-2, in the fabrication of components of the reactor coolant pressure boundary has been controlled to 5-18 FN. Delta ferrite content was determined by magnetic measurement or chemical analysis in conjunction with the Schaeffler or McKay Diagram, performed on undiluted weld deposits. In the case of filler metal used with a non-consumable electrode process, the delta ferrite content may have been determined by chemical analysis of the rod, wire or consumable insert in conjunction with the stainless steel constitution diagram.

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

b) 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 Code referenced in Table 5.2-1.

Conformance to Regulatory Guide 1.34, "Control of Electroslag weld Properties," December 1972 (R0) is discussed in Subsection 5.2.3.3.2.2.

Conformance to Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," December, 1973 (R0) is discussed in Subsection 5.2.3.3.2.3.

5.2.3.4.2.1.2 AE-Supplied Components

Conformance to Interim Position MTEB 5-1 on Regulatory Guide 1.31 is provided in Subsection 6.1.1.

Regulatory Guide 1.34, (R0) does not apply since electroslag welding is not utilized on non-NSSS Components.

Conformance to Regulatory Guide 1.71 (R0) is as follows:

Welder qualification under simulated limited-access conditions is not performed. However, the objective of the Regulatory Guide is adhered to by using welding supervisors to monitor welders and place an experienced welder(s) at limited-access locations.

5.2.3.4.3 Nondestructive Examination

Nondestructive examination of tubular products is discussed in Subsection 5.2.3.3.3.

5.2.4 INSERVICE INSPECTION AND TESTING OF REACTOR COOLANT PRESSURE BOUNDARY

An inservice inspection program is provided for the examination of the Quality Group A reactor coolant pressure boundary (RCPB) components and supports as defined by Code Class 1 in ASME Code, Section XI. The purpose of the inservice inspection program is to periodically monitor the systems or components requiring inservice inspection program in order to identify and to repair those indications which do not meet acceptance standards. The acceptance standards are provided in the ASME Code Section XI. The initial inservice inspections conducted during the first 120 month period following commercial plant operation will be developed to the requirements of 10 CFR 50.55a(g), to the extent practical. Where it becomes impractical to meet this criteria, relief from requirements, on a case-by-case basis, will be requested. References 1, 2, 3, 4. A list of these relief requests is provided in the inservice inspection program.

5.2.4.1 System Boundary Subject to Inspection

The system boundary subject to inspection is defined in 10 CFR 50, Section 50.2(v). The reactor pressure vessel, pressurizer, primary side of the steam generator and associated piping, pumps, valves, bolting, and component supports are subjected to inspection.

5.2.4.2 Arrangement of Systems and Components to Provide Accessibility

Provisions are made in the plant design for access to permit the conduct of preoperational and inservice inspections as specified in the ASME Code, Section XI. The inservice inspection program shall be updated periodically to meet the 10 CFR 50.55a(g) requirements. The design and arrangement of the components are such that space is provided to conduct examinations either from the interior or the exterior or a combination of both.

The use of conventional nondestructive, and visual test techniques, both direct and remote, can be applied to the Reactor Coolant System components. The high radiation levels and remote underwater accessibility of the reactor vessel present special problems. In order to facilitate an inservice inspection of the vessel from the internal surfaces during refueling, the vessel internals and the core barrel are removable. During refueling, the reactor vessel head, closure seal surfaces and studs may be examined. This allows the internal parts of the vessel which are visible, including the cladding and components, to be visually checked, as well as allowing access to the vessel wall for volumetric examinations.

The design considerations which have been incorporated into the system and plant layout to permit the required examinations are as follows:

a) Storage space is provided for the reactor vessel internals and core barrel in the refueling cavity, which permits internal and external examinations of these components.

b) The reactor vessel head is stored dry on the containment operating floor during refueling to facilitate direct visual inspection.

c) Reactor vessel studs, nuts and washers can be removed to dry storage during refueling.

d) Limited clearance is provided around the reactor coolant piping penetrating the primary shield which permits access to at least one of the reactor vessel cold leg nozzle welds.

e) Limited access is provided to the external surface of the reactor vessel lower head through the reactor cavity drain tunnel.

f) Access is provided via manways into the primary water box side of the steam generator. An access opening in the support skirt provides for inspection of the staywell welds.

g) A manway is provided in the pressurizer to allow access for internal inspection.

h) The reactor coolant pumps can be disassembled and inspected internally. Also, access is provided for the volumetric examination of the motor flywheels.

i) Insulation on Reactor Coolant System components and piping is removable where necessary.

j) Portions of the auxiliary systems piping, and Emergency Core Cooling System piping are arranged for maximum accessibility inside the containment. Access is not available to the segments of these systems where the piping penetrates the Reactor Coolant System shield wall.

k) Portions of the auxiliary system piping and emergency core cooling piping external to the containment are accessible for inspection at any time except where the piping penetrates the concrete floors and walls.

l) The piping supports and restraints are designed to facilitate accessibility for examination to the maximum practicable extent commensurate with other design requirements. Certain welds in pipe restraint structures will not be available for inspection after installation. An example of this inaccessibility is the welds attaching shear keys to the base or anchor plates. The shear keys are embedded in cement grout during installation and are therefore inaccessible for visual examination.

m) Safety and relief valves are flanged and can be removed from the system for disassembly and internal inspection.

5.2.4.3 Examination Techniques and Procedures

Examinations include liquid penetrant or magnetic particle techniques when surface examination is specified, ultrasonic, eddy current or radiographic techniques when volumetric examination is specified, and visual inspection techniques are used to determine surface condition of components and for evidence of leakage. Specific techniques, procedures, and equipment are defined in the inservice inspection program.

5.2.4.4 Inspection Intervals

The inspection interval for the examination program is in accordance with the ASME Code, Section XI requirements and is defined in the inservice inspection program.

5.2.4.5 Categories and Requirements

The inservice inspection program category and examination requirements for the reactor coolant pressure boundary components, including their supports, comply with the ASME Code, Section XI.

5.2.4.6 Evaluation of Results

The evaluation of nondestructive examination results, acceptance standards and documentation is in accordance with the ASME Code, Section XI.

5.2.4.7 System Leakage and Hydrostatic Pressure Tests

Code Class 1 systems and components are subjected to (a) a system leakage test prior to startup following each reactor refueling outage, and (b) a system hydrostatic pressure test at or near the end of each inspection interval. The system temperature-pressure relationship is in agreement with Section XI requirements except as limited by the Technical Specifications. Operational limitations during heatup, cooldown, and system hydrostatic pressure testing, are provided in the Technical Specifications.

Table 5.2-15 shows the pressure isolation valves under the scope of Generic Letter 87-06, "Periodic Verification of Leaktight Integrity of Pressure Isolation Valves." The test frequency and leakage limits of these pressure isolation valves are provided in the Technical Specifications.

As an alternative to hydrostatic testing for welded repairs or installation of replacement items by welding in Class 1, 2, and 3 piping systems, a system leakage test using the 1992 edition of Section XI, paragraph IWA-5000, is acceptable when the rules of code case N-416-1 are applied to the repair or replacement activity. Reference 4 grants approval. Also, see Table 5.2-2.

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 RCPB leak detection system is capable of detecting unidentified leakage as low as 1.0 gpm within a reasonable time period. The instruments in the RCPB leakage detection system are calibrated at least once per 18 months.

The RCPB leakage detection system meets the intent of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973 (R0). Reference 5 documents NRC acceptance of the containment radiation monitoring system with respect to leak-before-break and Regulatory Guide 1.45.

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 subsections.

5.2.5.1.1 Sump Level Monitoring

Collection of water in the reactor cavity 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.

All drains entering the sump are routed first to a measurement tank. A triangular notch weir is machined on the side of the measurement tank. The level of the flow through the weir corresponds to the flow of water into the tank. A water level switch is installed on the tank at the level that corresponds to a flow of one gpm into the tank; an alarm is annunciated whenever the flow rises to one gpm. A water level transmitter with a recorder in the control room is used to monitor the actual flow through the weir. The recorder has a range of 0 to 12 gpm.

5.2.5.1.2 Containment Airborne Particulate, Iodine and Gaseous Radioactivity Monitoring.

As described in Subsection 12.3.4.2.3.1, redundant seismic Category I containment atmosphere radiation monitors are designed to provide a continuous indication in the 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.

The monitor draws a sample of containment air through a continuous sampler assembly located outside the containment (see Figure 12.3-14). The sampler assembly is described in Subsection 11.5.2.1.2. The sampler assembly contains a three stage gas, particulate and iodine monitor as described in Subsection 11.5.2.1.3c. Alarms, data display and recorder are located on a seismic panel in the control room.

High radiation level and alert status alarms are provided in the control room. Leakages occurring under conditions having smaller percentages of failed fuel are better detected by the particulate detector. Listings of time rate of change in noble gas concentration and time for 10 percent deviation from normal are shown in Table 5.2-14 which are based on a postulated step increase in direct leakage from 0.1 gpm to one gpm at 85 percent thermal rating, 0.1 percent failed fuel, at the end of a 90 day purge cycle. The response times indicated represent the worst case.

5.2.5.1.3 Pressurizer Power Operated Relief Valves and Pressurizer Safety Valve Leakage

Leakage through the pressurizer power operated relief valves and pressurizer safety valves is detected by an increase in temperature in the valve discharge lines (Figure 5.1-4) and rising water level in the quench tank. These parameters are monitored as follows:

a) Discharge Line Temperature - Each of the pressurizer safety valve discharge lines contain a temperature detector (TE-1107, 1108, 1109) for monitoring valve leakage. The discharge lines from the power operated relief valves also contain temperature detectors (TE-1106 and 1110). Control room temperature monitoring instrumentation consist of indicator/alarm units (TIA-1107, 1108, 1109, 1106 and 1110) for each of these detectors.

b) Acoustic Monitor – Each pressurizer safety and PORV discharge line contain an accelerometer used for detection of flow noise through the valve. The noise signal is converted to a voltage proportional to the flow detected and is indicated in the control room. A common control room annunciator will alarm beyond a specific threshold. The control room operator will confirm the alarm by checking the individual monitors.

c) Quench Tank Water Level - Since the pressurizer safety valve or power operated relief 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 indicator alarm unit LIA-1116 detects this increasing water level change and TIA-1116 detects corresponding increase in water temperature due to steam entry into the tank.

   Pressure does not significantly increase due to leakage of steam into the quench tank. Pressure measurement is used to measure quench tank pressure after a valve discharge.

5.2.5.1.4 Safety Injection Tank Check Valves

Leakage of reactor coolant through the safety injection tank check valves (V3215, 3225, 3235, 3245) shown on Figure 6.3-1c can be detected by:

a) Safety Injection Tank Water Level - In-leakage of reactor coolant to the safety injection tank produces a rising water level in the tank. This is detected by the tank level indicator/high-low alarm (LIA-3311, 3321, 3331 and 3341) and the level indicator/high high low low alarm (LIA-3312, 3322, 3332 and 3342) which actuate high water level alarms.

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 in-flow is accompanied by an increasing tank pressure. Pressure indicator alarm units (PIA-3311, 3321, 3331 and 3341) on the control board monitor the tank pressure and annunciate alarms on high tank pressure as well as pressure switches PIS-3313, -3323, -3333 & -3343 which also actuate a high-high pressure alarm.

c) Safety Injection Tank Sample - RCS inleakage can be confirmed by sampling.

The Safety Injection Tank check valves are within the Scope of Generic Letter 87-06, "Periodic Verification of Leaktight Integrity of Pressure Isolation Valves." The test frequency and leakage limits of these pressure isolation valves are provided in the Unit 2 Technical Specifications.

5.2.5.1.5 Heat Exchanger Leakage

Leakage of reactor coolant through the letdown heat exchanger, reactor coolant pump seal heat exchanger, or sample heat exchangers can be detected by either of the following:

a) Component cooling system radiation - Heat exchanger leaks produce leakage of reactor coolant and fission products into the Component Cooling Water System. Such leakage increases radiation levels in the system and can be detected by the two component cooling water monitors described in Subsection 11.5.2.2.1. These monitors alarm and indicate in the control room.

   Complete dispersion of only one gallon of reactor coolant throughout the volume of the Component Cooling Water System is sufficient to cause early detectable rapid change in detector scale provided there is no residual radioactivity already present in the component cooling water. In this case the limit on detection is the transport time around the Component Cooling Water System loop. 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.

b) Component cooling surge tank water level - Inleakage of reactor coolant increases the inventory in the Component Cooling Water System, causing a rising surge tank water level. A one gpm leak into the component cooling water surge tank can be detected and alarmed within eight hours. Level switch LS-14-5 and local gage glasses (LG-14-2 A and B) mounted on the surge tank provide control room high water level alarms and local indication of tank water level, respectively.

5.2.5.1.6 Steam Generator Tube Leakage

Leakage of reactor coolant through the steam generator tubing is indicated by secondary side radioactivity. The following are methods for detecting the resulting radiation levels.

a) Blowdown line radiation - Increasing radiation levels due to dissolved and estrained fission products in the secondary side water can be detected by the radiation monitors in the steam generator blowdown lines. Remote readout and high radiation alarms are provided. These monitors are described in Subsection 11.5.2.2.4.

b) Off gas radiation - Increasing off gas radiation due to gaseous and volatile fission products in the Main Steam System can be detected by the radiation monitor in the condenser air ejector discharge or the atmospheric steam dump exhaust monitor. These monitors are described in Subsections 11.5.2.2.7 and 11.5.2.2.12.

c) Blowdown sample analysis Steam generator water samples are taken periodically. The sample is analyzed for dose equivalent I-131 and gross activity. The analyses provide the capability of detecting reactor coolant leakage into the steam generator secondary side.

The time to detect a one gpm leak depends on reactor coolant activity and previous leakage. A change from 0.1 gpm to one gpm leak can be detected in approximately two hours by the steam generator blowdown radiation monitor. A one gpm leak with no previous leakage can be detected in less than 20 minutes.

Calculation PSL-2FJI-99-001 (Reference 6), "Steam Generator Blowdown Radiation Monitor Response Time," determined that the time to reach the activity concentration corresponding to the radiation monitor alarm setpoint of two times background is less than 5 seconds. These results confirmed the validity of the value listed in Table 5.2-14. However, since this response time was calculated assuming different conditions than those assumed in Table 5.2-14, Table 5.2-14 was not revised to reflect the calculated value. Calculation PSL-2FJI-99-001 also determined the transit time from the steam generator to the radiation monitor to be 520 seconds, for a total radiation monitor response time of approximately 9 minutes which confirms the validity of the value cited above of less than 20 minutes to detect a one gpm leak with no previous leakage.

5.2.5.1.7 Reactor Coolant Makeup

An important means of detecting abnormal leakage from the Reactor Coolant System 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 reactor coolant pump controlled bleedoff are collected and recycled back into the Reactor Coolant System by the Chemical and Volume Control System described in Subsection 9.3.4, the net inventory in the Reactor Coolant System and Chemical and Volume Control System under normal operating conditions is constant. Transient changes in letdown flow rate or Reactor Coolant System inventory can be accommodated by changes in the volume control tank water level. The net makeup to the system under zero leakage steady state conditions should be essentially zero. The makeup flow rates from the makeup water system and boric acid makeup tanks are continuously monitored and recorded. 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. Additionally, a RCS inventory balance is performed per Technical Specifications using the makeup volume as input.

The maximum capacity of the Chemical and Volume Control System for reactor coolant makeup is 132 gpm (three 44 gpm charging pumps) which gives a ratio of maximum allowable leakage to makeup to 1/132.

5.2.5.1.8 Reactor Coolant Pump Seals

Instrumentation shown on Figure 5.1-6 detects abnormal seal operation. The reactor coolant pumps are equipped with three stages of seals plus a vapor backup seal as described in Subsection 5.4.1. During operation the Reactor Coolant System operating pressure is decreased through the three seals to approximately volume control tank pressure. The vapor 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 seal is designed to withstand full Reactor Coolant System pressure in the event of failure of any or all of the three primary seals, provided the pump is not rotating.

Referring to Figure 5.1-6, the reactor coolant pump P&ID, there is sufficient instrumentation on the reactor coolant pump to detect seal degradation and leakage. All these reactor coolant pump control grade instruments have appropriate high or low range alarms to alert the operator to seal malfunction. The DCS was expanded to include the reactor coolant pump monitoring and display system. A more detailed discussion of the DCS can be found in Subsection 7.5.1.4a. The flat panel display was installed to integrate the RCP monitoring and display system into the DCS in addition to the equipment discussed in Subsection 7.5.1.4a.

5.2.5.1.9 Reactor Vessel Head Closure Leakage

Reactor Vessel Head Closure Flange sealing is accomplished by a double-seal arrangement utilizing two silver jacketed Ni-Cr-Fe alloy spring-energized O-ring seals. The space between the double O-ring seal is monitored by a local pressure gage (PI-1118) and pressure switch (PS-1118) shown on Figure 5.1-3 to detect an increase in pressure which indicates a leak past the inner O-ring. A high pressure alarm actuated by pressure switch PS-1118 alerts the operator to the presence of leakage past the inner seal.

5.2.5.1.10 Reactor Coolant Pump Flange Closure Leakage

This system is essentially the same as the one for the reactor vessel head closure described in Subsection 5.2.5.1.9. The local indicators (PI-1150, 1160, 1170 and 1180) and pressure switches (PS-1150, 1160, 1170 and 1180), shown on Figure 5.1-6, provide the leak detection monitoring system with control room via the annunciator window and a Distributed Control System (DCS) driven Flat Panel Display annunciation for the reactor coolant pump closures.

5.2.5.1.11 Containment Atmosphere Temperature Indication

Temperature indication is not an absolute method of detecting leakage. However, temperature indication is provided for monitoring the containment after a leak has occurred.

5.2.5.1.12 Safety Injection and Shutdown Cooling Isolation and Check Valves

Certain Safety Injection and Shutdown Cooling Valves as identified in Table 5.2-15 are within the scope of Generic Letter 87-06, "Periodic Verification of Leaktight Integrity of Pressure Isolation Valves." The test frequency and leakage limits of these pressure isolation valves are provided in the Unit 2 Technical Specifications.

5.2.5.2 Indication in Control Room

The primary indications of reactor coolant leakage are:

a) Containment sump flow indication and alarm
b) High containment particulate radioactivity indication and alarm
c) High containment gaseous radioactivity indication and alarm
d) High containment iodine radioactivity indication and alarm

Other control room instrumentation used in detecting and identifying reactor coolant leakage includes:

a) Temperature indication downstream of power operated relief valve and pressurizer safety valves and high temperature alarm.
b) Acoustic monitoring downstream of Pressurizer safeties & PORVs.
c) Quench tank temperature and water level indication and alarm.
d) Safety injection tank water level indication
e) High and high-high safety injection tank levels alarm
f) Safety injection tank pressure indication and high pressure alarm
g) Component cooling water radiation indication.

h) Component cooling water surge tank high and low water level alarms
i) Steam generator blowdown radiation indication and alarm
j) Condenser steam jet air ejector exhaust radiation indication
k) Atmospheric steam dump exhaust monitor

5.2.5.3 Limits for Reactor Coolant Leakage

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

5.2.5.4 Differentiation Between Identified and Unidentified Leaks

Reactor Coolant System leakage is categorized as identified and unidentified leakage. Identified leakage is:

a) leakage (except seal water flow from the RCP seals) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b) leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary leakage, or

c) reactor coolant system leakage through a steam generator to the secondary system.

All other leakage is unidentified leakage. Since identified leakage is known, its effect upon the various leakage detection systems also is known. An increase in leakage, resulting from unidentified leakage, is detected by the leakage detection systems. The systems are capable of responding to a one gpm leakage within a reasonable time period (refer to Table 5.2-14).

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

In order to identify leaks from the RCPB to the shell side of the steam generators, each steam generator has a sampling system. The sampling system is tapped off the blowdown line of each steam generator. Samplings 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 Reactor Coolant System into the Component Cooling Water System is identified by an increase in water level in the component cooling water surge tanks and radioactivity in the system.

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

5.2.5.6 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 are 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.

b) The visual examinations above are conducted in conformance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, subject to the modifications established by 10 CFR 50.55a.

   The source of any reactor coolant leakage detected by the examinations of (a) above will be located and evaluated for corrective measures as described in the ASME Code, subject to the modifications established by 10 CFR 50.55a.

5.2.6 LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP)

Overpressure protection of the Reactor Coolant System (RCS) during low temperature conditions is provided by power operated relief valves (PORVs) V1474 and V1475 connected to the pressurizer steam space and when the shutdown cooling system (SDCS) is in operation also by the SDCS relief valves. During hot standby and power operations, the PORVs open at a high pressure setpoint of 2370 psia. When the RCS is at low temperatures the operator realigns the PORV setpoints to a low pressure value of 490 psia for low temperature overpressure protection. The PORVs are shown on the piping and instrumentation diagram of Figure 5.1-4a and described in Subsection 5.4.13. The protection provided by the PORVs and SDCS relief valves precludes any overpressurizing transient from exceeding the technical specification pressure-temperature (P-T) operating limits.

Each PORV solenoid is supplied from a separate IE electrical source. For high pressure service the actuating signal is provided from the Reactor Protection System (RPS) high pressure trip. A separate-train signal is supplied to each PORV. For LTOP service the actuation of each PORV is provided by separate Class 1E channels of RCS temperature and pressure instrumentation.

The low temperature overpressure protection provided by the relief valves is required during heatup and cooldown and during extended periods of cold shutdowns. The low pressure setpoint of 490 psia is incorporated in the LTOP mode since each PORV is designed to accommodate the most limiting transients as described in Subsection 5.2.6.2.1. Per the Technical Specifications, the PORVs are used for LTOP during heatup when any RCS cold leg temperature is in the range from 80-246°F and during cooldown when any RCS cold leg temperature is 224°F and below to 132°F. For temperatures below 132°F during cooldown, the SDCS relief valves are the only means of LTOP. In addition to administrative controls and procedures, a low temperature alarm is provided to alert the operator to align the LTOP 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 2,500 psia. At temperatures above the temperature limit which corresponds to the pressurizer safety valve setpoint, the "normal" PORV setpoint of 2370 psia is selected. The PORVs provide backup overpressure protection for the pressurizer safety valves as described in Subsection 5.4.13.

5.2.6.1 Design Criteria

A discussion follows the criteria considered in determining the adequacy of the PORVs and the SDCS relief valves to provide low temperature overpressure protection (LTOP) of the RCS.

5.2.6.1.1 Credit for Operator Action

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

5.2.6.1.2 Single Failure Criteria

In the LTOP mode, the PORV overpressure protection system and the SDCS relief valves are designed to protect the reactor vessel given a single failure in addition to a failure that initiated the pressure transient. The event initiating the pressure transient is considered to result from either an operator error or equipment malfunction. The PORV system has redundancy in actuation channels and functions during a loss of offsite power. When the PORVs are used to limit safety valve opening, at the high setpoint, redundancy is not required; therefore, only one PORV is aligned to the RCS, since this is an equipment protective rather than a safety-related function. When the SDCS relief valves are the means of LTOP protection, redundancy is provided by the system containing two relief valves.

5.2.6.1.3 Testability

To assure operational readiness in the LTOP mode, the PORV protection system is tested in the following manner:

a) A test is performed to assure operability of the system electronics prior to each planned shutdown.

b) A test for valve operability is, as a minimum, conducted as specified by the ASME Code, Section XI.

c) Subsequent to system, valve, or electronics maintenance, a test on those portion(s) of the systems maintained is performed prior to declaring the system operational.

The SDCS relief valves are passive components of the shutdown cooling system and have surveillance and operability requirements designated in the Unit 2 Technical Specifications.

5.2.6.1.4 Seismic Design and IEEE 279 Criteria

The PORVs, isolation valves, associated interlocks and instrumentation are designed to Quality Group A, seismic Category I requirements. The interlocks and instrumentation associated with the PORVs satisfy the appropriate portions of IEEE 279-1971 criteria.

The SDCS relief valves are Quality Class B, they are included as Class 2 equipment and meet the specifications of ASME Code Section XI.

5.2.6.1.5 Reliability

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

5.2.6.2 <u>Design and Analysis</u>

In demonstrating that use of the LTOP system meets the criteria listed in Subsection 5.2.6.1, the following additional information is provided.

5.2.6.2.1 Limiting Transients

Transients during the low temperature operating mode are more severe when the RCS is operated in the water-solid condition. Addition of mass or energy to an isolated water-solid system produces increases in system pressure. The severity of the pressure transient depends upon the rate and total quantity of mass or energy addition. The choice of the limiting LTOP transients was based on evaluation of potential transients for St. Lucie Unit 1. These transients are shown on Figure 5.2-26 and are applicable to St. Lucie Unit 2.

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

a) An inadvertent safety injection actuation (mass input).

b) A reactor coolant pump start when a positive steam generator to reactor vessel ΔT exists (energy input).

The transients were determined as most limiting by conservative analyses which maximize mass and energy additions to the RCS. In addition, the RCS is assumed to be in a water-solid condition at the time of the transient; such a condition has been noticed to exist infrequently during plant operation since the operator is instructed to avoid water-solid conditions whenever possible. (Note: water-solid conditions may exist during normal plant heatup and cooldown.)

The mass addition due to the simultaneous operation of two HPSI and three charging pumps was considered, along with the simultaneous addition of energy from decay heat and the pressurizer heaters.

The energy addition due to a reactor coolant pump start when a steam generator to reactor vessel ΔT of 40°F exists was considered. In addition to considering the energy addition to the RCS from the steam generator secondary side, energy addition from decay heat, reactor coolant pump, and pressurizer heaters was also included.

LTOP transients have not been analyzed for the simultaneous startup of more than one reactor coolant pump (RCP). Such operation is procedurally precluded since the operator starts only one RCP at a time and a second RCP is not started until system pressure is stabilized. RCP motor amperage is used to establish nominal pump performance and operation prior to starting a second RCP. Additionally, there is an LTOP transient alarm that indicates that a pressure transient is occurring.

A Technical Specification requires that the operator not start an RCP if the ΔT exceeds 40°F. However, administrative procedures will ensure that the ΔT is maintained below 30°F. A separate Technical Specification ensures that the appropriate action is taken if one PORV or one SDCS relief valve is out of service during the LTOP mode of operation.

An analysis of P/T limits and LTOP protection for the period ending at 55 EFPY was performed. The LTOP enable temperatures were determined by following the guidance of the ASME Boiler and Pressure Vessel Code Section XI, Appendix G. They were calculated to be less than or equal to 246°F during heatup and less than or equal to 224°F during cooldown. Since the 55 EFPY pressure-temperature limits are less restrictive than the 21.7 EFPY limits, the PORV setpoint was raised to 490 psia from 470 psia for increased operational flexibility. A change to the SDCRV setpoint of 350 psia was not practical or required.

The P/T limits, LTOP requirements, setpoints and controls were evaluated for operation at EPU conditions. The only update for EPU conditions was a reduction in the period of applicability to approximately 47 EFPY from 55 EFPY.

5.2.6.2.2 Provision for Low Temperature Overpressure Protection

During heatup, RCS pressure is maintained below the PORV pressure setpoints until after the PORVs are re-set to the "normal" high setpoint. The PORVs can be re-set to the normal setpoint when cold leg temperature increases above the maximum temperature for LTOP during heatup (nominally T_c = 246°F), defined in the Technical Specifications.

During cooldown, RCS pressure is decreased to below the PORV low pressure setpoint before cooling the plant to below the maximum temperature for LTOP during cooldown (nominally T_c = 224°F), defined in the Technical Specifications. Once temperature is lowered, the PORV control switch must be aligned in the LTOP mode.

The LTOP mode applies for all temperatures within the maximum LTOP temperature range. The PORVs will remain in the LTOP mode until the RCS is opened during refueling. The LTOP system is designed to be aligned during all heatup and cooldown operations.

5.2.6.3 Equipment Parameters

Each PORV is actuated from a 90-140v dc solenoid valve which is energized automatically from a pressurizer pressure transmitter. Each PORV is designed to close on interruption of power to the solenoid valve.

Pertinent PORV operational and design requirements are presented in Table 5.4-9. The PORVs are sized based on a transient (simultaneous operation of two HPSI pumps and three charging pumps) initiated from a water-solid condition. The analysis of the Subsection 5.2.6.2.1 limiting transients for the period ending 55 EFPY demonstrated that the PORV water relieving capacity is adequate to prevent violation of the pressure-temperature limits for the temperature ranges where a PORV can be relied upon for LTOP protection, in accordance with the Technical Specifications. Moreover, this analysis assumed that only one of the two available PORVs was in operation. Therefore, both PORVs together are designed to have more than twice the capacity needed to mitigate the most severe expected transient.

The flow capacity requirement for steam (153,000 lb/hr) was chosen to avoid unnecessary lifting of the safety valves, based on an analysis of the control rod withdrawal transient. Analysis has shown that each PORV, sized for LTOP as described above, will have approximately twice the required steam flow capacity. For at-power operation (PORV setpoint of 2370 psia) only one PORV need be in service to provide the required flow capacity. However, it is noted that design criteria on reactor coolant system pressure would still be met if the PORVs were not in service. The pressurizer safety valves have sufficient relieving capacity to mitigate the most severe event (see Appendix 5.2A).

The PORVs are not specifically designed for two-phase conditions at the PORV inlet since it is expected that either steam or subcooled water conditions will exist, but not a two-phase mixture. (See Appendix 1.9A [item II.D.1]) and Subsection 5.4.13.4 for further discussion of PORV testing.

The SDCS relief valves are sized for a flow capacity of 2300 gpm at a lift set pressure of 350 psia. Each of the two SDCS relief valves is sufficient to provide LTOP during low temperature operation when the SDCS relief valves are aligned to the RCS during modes 4-6.

5.2.6.4 Administrative Controls

Administrative controls necessary to provide LTOP are limited to those controls that align the PORVs when Reactor Coolant System parameters are satisfied. Before entering the low temperature region for which overpressure protection is necessary, RCS pressure is decreased to below the maximum pressure required for LTOP operation. The technical specifications define when the PORVs are required to be operable, when used for LTOP, and the number of block valves required to be open. The PORVs will remain in the LTOP mode while the RCS is at low temperature and the reactor vessel head is secured. Control room indication is provided to give the operator PORV position indication and mode of operation.

Unit 2 Technical Specifications define the configuration of the Shutdown Cooling System for cooldown and heatup in modes 4-6.

5.2.6.4 Summary of Electrical Controls and Circuitry

This section summarizes the electrical controls and circuitry for the PORVs. The SDCS relief valves are passive devices which have no indication or control features.

PORV Actuation

Each PORV is actuated from a 90-140V dc solenoid pilot valve. This valve is energized automatically from a pressurizer pressure transmitter for LTOP; normal actuation is from the RPS high pressure trip. Each PORV is designed to close on loss of power to the solenoid valve. Figure 5.2-27 illustrates the PORV actuation logic.

PORV Switches

Each PORV is provided with a mode selector switch which is a two position manual control switch with a "NORMAL" AND "LTOP" position. Each PORV is also provided with a three position "Off/Override/Test" Selector Switch. The "OFF" position of this switch is used normally when PORV control circuitry testing is not desired. The "OVERRIDE" position shuts the PORV if it is open and overrides any signal to open the valve. The "TEST" position simulates an open signal to the PORV without physically opening the valve.

LTOP Interlock

An LTOP interlock is provided to prevent PORV actuation when the cold leg temperature is greater than 246°F and the mode selector switch is inadvertently positioned to "LTOP". A "PORV Normal Condition" Alarm will actuate in the control room alerting the operator to select "NORMAL" on the Mode Selector Switch when cold leg temperature is greater than 246°F. The interlock will reset when cold leg temperature drops below 246°F.

LTOP System Alarms (see Figure 5.2-28)

a) PORV LTOP Condition Alarm

During cooldown when the "Mode Selection Switch" is in the "Normal" position a "PORV LTOP Condition" alarm will alert the operator to select "LTOP" on the "Mode Selector Switch" prior to the cold leg temperature reaching a value 224°F. Thus, changing the PORV actuation setpoint from 2370 psia to 490 psia for both PORVs (V1474 and V1475). This alarm will not reset unless the PORV mode selector switch is in the "LTOP" position and the motor operated isolation valves of both PORVs are "OPEN".

b) PORV Normal Condition Alarm

During heatup when the "Mode Selector Switch" is in the "LTOP" position a "PORV Normal Condition" alarm will alert the operator to select "Normal" on the "Mode Selector Switch" after the cold leg temperature reaches a value greater than 246°F. Thus, changing the PORV actuation setpoint from 490 psia to 2370 psia for both PORVs (V1474 and V1475). This alarm will not reset unless the PORV Mode Selector Switch is in the "Normal" position and at least one of the PORV isolation valves is "OPEN".

c) LTOP Transient Alarm

When RCS temperature (cold leg) is less than or equal to 224°F or the mode selector switch is in "LTOP" and pressurizer pressure is greater than or equal to 490 psia, a potential LTOP transient is sensed, actuating the "LTOP Transient" alarm. Alarm values may be set conservative to these values. The LTOP B pressure measurement channel has separate alarm and actuating devices, and the alarm may be set conservative to the actuating device.

d) PORV Test Condition Alarm

A "PORV Test Condition" alarm will alert the operator whenever the PORV protective system is in the "Override" or "Test" position, bypassing all setpoints. The PORV will remain closed in this condition until the selector switch is placed in the "OFF" position at which time the alarm will reset.

e) Indicating Lights

Open and close position indicating lights have been located in the control room, informing the operator of the actual stem position of each PORV. Indicating lights are not energized or deenergized from the solenoid valve circuitry. Indicating lights are controlled from the valve stem position.

Each isolation valve is provided with a two-position open-close switch for valve operation. Open and close position indicating lights are also provided for each motor operated isolation valve.

A test indicating light will be provided for each PORV and lit whenever the PORV protective system control circuitry has been bypassed (in "OVERRIDE") or has been tested satisfactorily (see Figure 5.2-29).

NOTE: Each PORV (V1474 and V1475) and actuation channel (channel A and B) will have their own independent alarms as indicated by parentheses on Figure 5.2-28.

Section 5.2: REFERENCES

1) Letter, A. C. Thadani (NRC) to C. O. Woody (FPL), "Relief from Parts of ASME Code Section XI," dated January 13, 1986.

2) Letter, A. C. Thadani (NRC) to C. O. Woody (FPL), "Relief from Parts of ASME Code Section XI," dated October 10, 1986.

3) Letter, H. N. Berkow (NRC) to J. H. Goldberg (FPL), "St. Lucie Unit 2 - Reliefs from Parts of ASME Code Section XI," dated October 2, 1989.

4) Letter, Eugene V. Imbro (NRC) to Thomas F. Plunkett (FPL), "St. Lucie Units 1 and 2 - Request for Approval of ASME Code Case N-416-1 As An Alternative to the Required Hydrostatic Pressure Test," dated April 29, 1996.

5) Safety Assessment by Plant Systems Branch Division of Systems Safety and Analysis Office of Nuclear Reactor Regulation Region II Concerns (TIA 96-019) Regarding the Containment Radiation Monitoring Systems at St. Lucie Units 1 and 2, and Turkey Point Units 3 and 4, May 27, 1999.

6) Calculation PSL-2FJI-99-001, “Steam Generator Blowdown Radiation Monitor Response Time.”

TABLE 5.2-1

REACTOR COOLANT SYSTEM PRESSURE BOUNDARY CODE REQUIREMENTS

Component	Codes and Classes
Reactor Vessel, Pressurizer	1. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 1971 Edition through Summer 1972 Addenda.
Replacement Reactor Vessel Closure Head	1. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Div. 1 Class 1, 1989 Edition, No Addenda.
Steam Generator	1. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components Class 1 and 2, 1998 Edition through 2000 Addenda.
Reactor Coolant Pump	1. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 1971 Edition through Summer 1973 Addenda.
Pressurizer Spray, Safety, Power Operated Relief, and Power operated relief isolation valves	1. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 1971 Edition through Winter 1972 Addenda. 2. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components Class 1, 1974 Edition. 3. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components Class 1, 1974 Edition through Winter 1974 Addenda. 4. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components Class 1, 1974 Edition through Summer 1974 Addenda. 5. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 1977, Edition Through Summer 1979 Addenda.

set pressure of 2485 psig and a capacity of five gpm. Relief fluid from these valves is collected in the quench tank. These valves are designed to 1974 ASME, Section NB, Quality Group A (see Figure 6.3-1b).

Valves V-3483 and V-3468 have a set pressure of 335 psig and a capacity of 155 gpm. Relief fluid from these valves is collected in a holdup tank in the Waste Management System. These valves are designed to 1974 ASME, Section NC, Quality Group B (see Figure 6.3-1b).

In addition to protecting the components and piping from overpressure due to the thermal expansion of the fluid, valves V-3666 and V-3667 are sized to protect the components and piping from overpressure due to inadvertent starting of the charging pumps, HPSI pumps, and pressurizer heaters. These valves have a set pressure of 335 psig and a capacity of 2300 gpm. Relief fluid collected from these valves is collected in the containment sump. These valves are designed to 1974 ASME, Section NC (to 1975 summer addenda) Quality Group B (see Figure 6.3-1b). When calculating the capacity of valves V-3666 and V-3667, the capacity of each valve was taken to be greater than the combined flow rate of two HPSI pumps, three charging pumps and the fluid forced out of the pressurizer when the backup heaters are actuated. This is a very conservative sizing, as the inadvertent actuation of the HPSI pumps, the charging pumps, and simultaneous energization of the pressurizer heaters is an unlikely coincidence. The largest flow rate which would be expected is from two safety injection pumps delivering approximately 1100 gpm at the Reactor Coolant System at 335 psig.

For the Shutdown Cooling mode, the LPSI pump suction is aligned to the hot leg of the RCS. The flow from the discharge side of the pump goes to the cold leg of the RCS. Due to this arrangement, the LPSI pump would not be dead headed by an RCS pressure surge. Relief valves in the shutdown cooling suction lines prevent isolation of the line from any anticipated overpressure events.

Valves V-3666 and V-3667 are located inside containment in close proximity of the containment sump. The discharge fluid from these relief valves is routed directly to the containment sump in order to prevent the accidental spread of Reactor Coolant System inventory outside the containment. The discharged liquid, which is collected and contained in the containment sump, will then be transported to the Liquid Waste Management System for processing. The containment sump is a large collecting reservoir provided to supply water to the Containment Spray and Safety Injection System for long term recirculation. The design of the containment sump is described in FSAR Subsection 6.2.2.2.3.

The use of the containment sump as an atmospheric collection tank minimizes the possibility of introducing any backpressure on the relief valve discharge piping. The relatively short piping runs in the discharge piping reduces the potential for any downstream pressure buildup or pressure fluctuations. This design assures maximum relief valve capacity at design conditions.

Since the SDCS is not designed to accommodate full Reactor Coolant System pressure, isolation of the system suction lines is assured by interlocks on the four suction line isolation valves inside containment. An independent interlock, utilizing pressurizer pressure is provided for each of the valves. For each SDCS train, power is supplied to the interlocks on these valves by two independent power supplies (one power supply for each interlock). This ensures isolation of the SDCS from the Reactor Coolant System should a spurious signal inadvertently open one of these valves. Each interlock is designed to prevent opening of its associated valve whenever pressurizer pressure is greater than 276 psia. An audible alarm sounds in the control room whenever any of the valves are off the full closed position. The interlock also provides automatic closure of the valves prior to an actual or simulated pressurizer pressure signal exceeding 515 psia. When power to an interlock is removed, the interlock fails as is. This precludes the loss of LPSI pump suction flow and, thereby, damage to the pump is avoided. As there are two isolation valves with interlocks in each LPSI pump suction train it is still possible to close off the LPSI pump line, as required, by both automatic and manual means. It should be noted that the basic purpose of the interlock is to provide double barrier protection between the Reactor Coolant System and the SDCS before the Reactor Coolant System returns to normal operating temperature and pressure. Each of the seven suction line isolation valves is equipped with open/close position indication in the control room.

Valve control circuitry is covered in Section 7.6. Interlocks are discussed further in Section 7.6. Interlocks are provided on the safety injection tank isolation valves for overpressure protection of the SDCS. The SIT interlocks are discussed in Section 7.6 and Subsection 6.3.2.2.1.

TABLE 5.2-1 (Cont'd)

Component		Codes and Classes
Reactor Coolant Piping	1.	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 1971 Edition through Winter 1972 Addenda. (NSSS)
RCPB Piping	2.	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, 1971 Edition through Summer 1973 Addenda (A/E).
Miscellaneous NSSS Valves	1.	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 1971 Edition through Winter 1972, Summer 1973, Winter 1973 Addenda.
	2.	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 1974 Edition through Summer 1974, Winter 1974 and Summer 1975 Addenda.
	3.	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 1977 Edition through Summer 1978 Addenda.
Control Element Drive Mechanisms	1.	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1 1998 Edition through 2000 Addenda.
Miscellaneous Non-NSSS Valves	1.	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Class 1, 1974 Edition (Summer 75 Addenda).

Codes listed above are construction codes. In addition, all these components are designed and constructed to permit the performance of the tests and inspections required by Section XI, Rules for In-Service Inspection.

TABLE 5.2-2

CODE CASE INTERPRETATIONS

Code Case Number	Title	Component(s) Affected
1141-1	Foreign Produced Steel; Annulled 7/23/76	Reactor Vessel, Pressurizer
1332-6	Requirements for Steel Forgings	Reactor Vessels, Reactor Coolant Pipe
1334-3 (N-2)	Requirements for Corrosion Resisting Steel, Steel Bars and Shapes, Section III	CEDMs
1361-2	Socket Welds	Pressurizer
1492	Post-Weld Heat Treatment; Annulled 3/3/75	Reactor Vessel, Reactor Coolant Pipe, Pressurizer
1519*	Use of A105-71 in lieu of SA-105 The ASTM A105-71 material provided for more desirable chemistry than SA-105. This code case was incorporated into the Code in the Summer 1973 Addenda.	Reactor Coolant Pipe
1553-1	Pressure temperature ratings of SA-351 Grades CF8A, CF3 and CF3M, Section III.	Miscellaneous Valves
1580-1	Buttwelded alignment tolerance and acceptable slopes for concentric centerlines for Section III, Class 1, 2 and 3 construction.	Miscellaneous Valves
1581	Power operated pressure relief valves, Section III.	Pressurizer power operated Relief Valve
1590	Chemical analysis variations Section III, construction	Miscellaneous Valves

Code Case Number	Title	Component(s) Affected
1649	Modified SA453-GR660 for class 1, 2, 3 and CS construction	Miscellaneous Valves
1661	Post-Weld Heat Treatment for P-1 Materials	Pressurizer, Reactor Coolant Pipe
1698	Waiver of Ultrasonic Transfer Method This case is used for ferritic safe-end and piping welds. It permits elimination of the transfer method which is ineffective in compensating for attenuation differences between the calibration block and the component materials. The transfer methods used comply with the condition of acceptance given in Regulatory Guide 1.85.	Reactor Coolant Pipe
1731*	Basic Calibration Blocks for Section IX, Division 1, Ultrasonic Examination of Welds 10 inches to 14 inches thick. This case is used for reactor vessel upper shell calibration blocks, since the basic Code does not provide for thicknesses greater than 10 inches.	Reactor Vessel
1769	Qualification of NDE Level III Personnel, Section III, Division I	Miscellaneous Valves
N-432**	Repair Welding Using Automatic or Machine Gas Tungsten-Arc Welding (GTAW) Temperbead Technique Section XI, Division I	Reactor Coolant Pipe and Pressurizer
N-474-1***	Design Stress Intensities and Yield Strength Values For UNS N06690	Pressurizer and Reactor Coolant Pipe Instrument Nozzles
2142*	F-Number Grouping for Ni-Cr-Fe, Classification UNS N06052 Filler Metal, Section IX	Pressurizer and Reactor Coolant Pipe Instrument Nozzles
N-416-1	Alternative Pressure Test Requirements For Welded Repairs or Installation of Replacement Items By Welding Class 1, 2, and 3 Section XI, Division 1	Class 1, 2, and 3 Piping Systems
N-20-4	SB-163, Cold Worked UNS N08800, and SB-163 UNS N06600, UNS N06690, and UNS N08800 to Supplementary Requirements S2 of SB-163, Section III, Division 1, Class 1.	Steam Generator
N-740-1**** (DRAFT)	Dissimilar Metal Weld Overlay for Repair of Class 1, 2, and 3 Items Section XI, Division 1	Class 1, 2 and 3 Piping Systems

TABLE 5.2-2 (Cont'd)

* Not included in Regulatory Guides 1.84 or 1.85

** Included in Regulatory Guide 1.147 Rev. 6 dated May, 1988 and 1.85, Rev. 25, dated May, 1988

*** Included in Regulatory Guide 1.85, Rev. 28 dated April, 1992

**** Not listed in Regulatory Guide 1.147, Rev. 14, dated August 2005

For the Reactor Vessel Head Penetrations (CEDM) nozzle and weld repairs, the following Code Cases were used as permitted by NRC Regulatory Guides 1.85 and 1.147:

Code Case Number	Title
N-416-1, -2	Alternative Pressure Test Requirements for Welded Repairs or Installation of Replacement Items By Welding Class 1, 2, and 3 Section XI, Division 1
N-2142-1	F-Number Grouping for Ni-Cr-Fe, Classification UNS W06052 Filler Metal Section IX
N-2143-1	F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode Section IX
N-474-2	Design Stress Intensities and Yield Strength Values for UNS N06690 with Min. Specified Yield Strength of 35 ksi
N-638	Similar & Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique

The original Reactor Vessel Closure Head has been replaced. The Information above, concerning the Reactor Vessel Head Penetrations (CEDM) nozzle and weld repairs is historical.

For the replacement Reactor Vessel Closure Head the following Code Cases were used as permitted by NRC Regulatory Guide 1.84:

Code Case Number	Title
N-2142-1	F-Number Grouping for Ni-Cr-Fe, Classification UNS W06052 Filler Metal Section IX
N-2143-1	F-Number Grouping for Ni-Cr-Fe, Classification UNS W86152 Welding Electrode Section IX
N-474-2	Design Stress Intensities and Yield Strength Values for UNS N06690 with Min. Specified Yield Strength of 35 ksi
N-416-1	Alternative Pressure Test Requirements for Welded Repairs or Installation of Replacement Items By Welding Class 1, 2, and 3 Section XI, Division 1

TABLE 5.2-3

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

Component	Material Specification
Reactor vessel	
Shell	SA-533 Grade B, Class 1 Steel
Forgings	SA-508 Class 1 and 2
Cladding[(a)]	Weld deposited austenitic stainless steel with greater than 5% delta ferrite (Equivalent to SA-240 Type 304) or NiCrFe alloy (equivalent to SB-168)
Replacement Reactor Vessel Closure Head (RVCH)	
Forging	SA-508, Class 3
Replacement RVCH Cladding	Weld Deposited austenitic stainless steel:
	First layer is 309L with delta ferrite number acceptable range of 5FN to 20FN. Subsequent layers are 308L with delta ferrite number acceptable range of 5FN to 17.6FN.
CEDM Nozzles	Nozzle: SB-167 (Alloy 690)
	Adapter: SB-166 (Alloy 690)
	Weld Filler: 52 or 52M (Alloy 690)
Instrument Nozzle	Nozzle: SB-167 (Alloy 690)
	Adapter: SA-479 Type 304 Stainless Steel
	Weld Filler: 52 or 52M (Alloy 690)
Vessel Internals[(a)]	Austenitic Stainless Steel and NiCrFe alloy
Fuel Cladding[(a)]	Zircaloy-4
Control element drive mechanism housings	
Lower	SA-182 Type 403 Modified stainless steel Special Code Case N-4-11 with end fittings to SB-166 (Alloy 690)
Upper	SA-479 and SA-213 Type 316 stainless steel with end fitting of SA-479 and vent valve seal of Type 440 stainless steel seat to SA-479
Closure head bolts & Nuts	SA-540 B23 and B24, Class 3
Support (on Nozzles)	SA-508, Class 2
Pressurizer -	
Shell	SA-533 Grade A Class 1
Upper Head Instrument Nozzle	SA-533 Grade B Class 1
Penetration Bore[(a)]	
Cladding[(a)]	Weld deposited austenitic stainless steel with greater than 5% delta ferrite or NiCrFe alloy (equivalent to SB-168)

(a) Materials exposed to reactor coolant

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

(c) The four (4) one-inch instrument nozzles in the upper head have SA-182, Type 316L safe ends.

TABLE 5.2-3 (Cont'd)

Component	Material Specification
Pressurizer (Cont'd)	
Forged nozzles	SA-508 Class 2
Instrument nozzles[(a)]	SB-166
Surge and PORV nozzle safe ends[(a)]	SA-351, Gr CF8M
Spray and instrument nozzle safe ends[(a)(c)]	SA-182, Type 316, except upper head instrument nozzles have Type F316L
Studs and nuts	SA-540 Grade B24 and SA-193 Gr. B7
Steam generator	
Primary head, nozzles and manways	SA-508 Grade 3, Class 2
Primary divider plate	SB-168 UNS N06690
Primary nozzle safe ends	SA-105
Primary head cladding[(a)]	Weld deposited austenitic stainless steel (Type 308L and 309L)
Tubesheet	SA-508 Grade 3, Class 2
Tubesheet cladding	Weld deposited NiCrFe (Alloys 52 / 152)
Tube[(a)]	NiCrFe Alloy (SB-163 UNSN 06690)
Secondary shell and head	SA-508 Grade 3, Class 2
Secondary nozzles	SA-508 Grade 3, Class 2
Steam nozzle venturis	SB-166 UNS N06690
Secondary instrument nozzles	SA-105
Studs / Nuts	SA-193 B16 / SA-194 Grade 7
Support Skirt	SA-508 Grade 3, Class 2
Sliding base support studs and nuts	SA-540 Grade B23 Class 2

TABLE 5.2-3 (Cont'd)

Component	Material Specification
Reactor coolant pumps	
Casing[(a)]	A-351 Grade CF8M
Internals[(a)]	Austenitic stainless steel (SA-351 Grade CF8M ASTM A79 Type 316 ASTM A240 Type 316, SA-182 GR F 304)
Studs and Nuts	SA-540 B23 Cl. 4 and SA-194 Gr. 2H
Reactor coolant piping	
Pipe (30 in. and 42 in.)	SA-516 Grade 70
Cladding[(a)]	SA-240 - 304L
Surge Line (12 in.)[(a)]	SA-351 - CF8M
Spray Line Pipe	SA-312, Type 316
	SA-312, 304L
Spray Line Fittings	SA-403, Type WP 316 SA-182, F 316 SA-376, TP 316 SA-182(M), TP 316/316L SA-182, F304L SA-182, F316/F316L
Piping safe ends (30 in.)[(a)]	SA-351 - CF8M
Surge nozzle forging	SA-105 Grade II
Surge nozzle safe end[(a)]	SA-351 - CF8M
Shutdown cooling outlet nozzle forgings	SA-105 Grade II
Shutdown cooling outlet[(a)] nozzle safe ends	SA-351 - CF8M
Safety injection nozzle forgings	SA-182 - F1
Safety injection nozzle[(a)] safe ends	SA-351 - CF8M
Charging inlet nozzle forging	SA-182 - F1
Charging inlet nozzle[(a)] safe end	SA-182 - F316
Spray nozzle forgings	SA-105 Grade II
Letdown and drain pipe	SA-312, 304L

TABLE 5.2-3 (Cont'd)

Component	Material Specification		
Reactor coolant piping (Cont'd)			
Spray nozzle safe ends [a]	SA-479, 316L or SA-182, F316L		
Letdown and drain or drain nozzle forgings	SA-105 - Grade II		
Letdown and drain or drain nozzle safe ends[a]	SA-479, 316L or SA-182, F316L		
Sampling or pressure[a] measurement nozzles	SB-166		
Sampling or pressure measurement nozzle safe ends[a]	SA-182 - F316 or SA-479 TP-316		
RTD nozzles[a]	SB-166		
Hot Leg RTD Split Nozzle Penetration Bore[a]	SA-516 Grade 70		
Sampling nozzle (surge line)[a]	SA-182 - F316		
RTD nozzle (surge line)[a]	SA-182 - F316		
Nozzle Thermal Sleeves[a]	SB-166 or SB168		
Valves[a]	SA-351 - CF8M, SA-182-F316		
AE - Supplied Components			
Valves	ASME	SA-182 SA-479	(F-316), SA-564 (Type-630) (Type-347, 348, 316L)
Pipes	ASME	SA-312	(GR TP 304)
Fittings	ASME	SA-182 SA-403 SA-351	(F-304) (GR-WP 304W) (GR CF8)
Flanges	ASME	SA-182 SA-351	(F-304) (GR-CF8)
Restrictors	ASME	SA-182	F-316
Bolts, Nuts	ASME	SA-193 SA-194	GR B7 GR 2H

TABLE 5.2-4

REACTOR COOLANT PRESSURE BOUNDARY WELDING MATERIAL

Material Specification	Base Material	Weld Material
1. SA-533 Gr. B C1.1	SA-533 Gr. B C1.1	a. SFA 5.9, E-8018, C3 b. MIl-E-18193, B-4
2. SA-508 C1.2	SA-533 GR. B C1.1	a. SFA 5.5, E-8018, C3 b. MIL-E-18193, B-4
3. SA-508 C1.1	SA-508 C1.2	a. SFA 5.5, E-8018, C3
4. SA-516 Gr. 70	SA-516 Gr. 70	a. SFA 5.1. E-7018[e(3)]
5. SA-182 FI	SA-516 Gr. 70	a. SFA 5.1, E-7018
6. SA-105 Gr.II	SA-351 CF8M	a. SFA 5.14, ERNiCr-3 c. ErNiCrFe-7A
7. SA-182 FI	SA-351 CF8M	a. SFA 5.11, ENiCrFe-3
8. SA-105 Cr. II	SA-182 F316	a. SFA 5.14, ERNiCr-3 c. ErNiCrFe-7A
9. SB-166	SA-182 F316	a. SFA 5.14, SFA 5.11, Root ERNiCr-3 Remaining ENiCrFe-3
10. SA-167	SA-182 F304	a. Root SFA 5.14, ERNiLCr-3 Remaining 5.11, ENiCrFe-3
11. SA-516 Gr. 70	SA-351 CF8M	a. SFA 5.11, ERNiCr-3
12. SA-182 F1	SA-182 F316	a. SFA 5.11, ENiCrFe-3
13. SB-166	SA-533 GR. B C1.1	a. SFA 5.11, ENiCrFe-3
14. SA-182 Code Case 1334	SB-167	a. SFA 5.14, ERNiCr-3
15. SA-516 Gr. 70	SA-508 C1.2	a. SFA 5.5, E-8016, C3

TABLE 5.2-4 (Cont'd)

Material Specification	Base Material	Weld Material
16. Austentic[a] stainless steel cladding		a. SFA 5.9, ER-308[e(2)] SFA 5.9, ER-309[e(1)] SFA 5.9, ER-312
17. Inconel	Inconel	a. ENiCrFe-3 ERNiCr-3
18. Inconel	SA-516 Gr. 70	a. ERNiCr-3 UNS N06052 Ni-Cr-Fe
19. Inconel	SA-533 Gr. B C1.1	a. ERNiCr-3
20. SB-166 Alloy[a] 690 (Inconel)	UNS N06052 Ni-Cr-Fe (Inconel Pad)	UNS N06052 Ni-Cr-Fe
21. Inconel Pad[a]	SA-533 GR. B C1.1	UNS N06052 Ni-Cr-Fe
22. SA 508 Gr 3 Cl 2	SA 508 GR 3 Cl 2	covered electrode: SFA-5.5 E9018-G wire/flux: SFA 5.23 F9P4-EG-G
23. SA 105	SA 508 GR 3 Cl 2	covered electrode (repair): SFA-5.1 E7018 wire/flux: SFA 5.23 F10P2-EG-G
24. Inconel 690 Clad[a]	SA 508 GR 3 Cl 2	wire: SFA-5.14 ERN:CrFe-7 covered electrode: SFA-5.11 ENiCrFe-7
25. Austenitic SS clad[a][c]	SA 508 GR 3 Cl 2	strips - trade designation: WEL ESS 309L and WEL ESS 308L flux - trade designation: WEL BND F-8 covered electrode: SFA-5.4 E309L-16 and E308L-16 wire - SFA-5.9 ER309L and ER308L
26. Austenitic SS clad[a][c]	SA 105	strips - trade designation: WEL ESS 309L and WEL ESS 308L flux - trade designation: WEL BND F-8 covered electrode: SFA-5.4 E309L-16 and E308L-16

TABLE 5.2-4 (Cont'd)

	Material Specification	Base Material	Weld Material
27.	SB 163 Alloy 690 (tubes)[(a)(d)]	Inconel 690 Clad	wire (repair): SFA 5.14 ERNiCrFe-7
28.	Inconel 690 Partition Plate[(a)]	Austenitic SS clad and Inconel 690 Clad	wire: SFA-5.14 ERNiCrFe-7 covered electrode: SFA-5.11 ENiCrFe-7

The original Reactor Vessel Closure Head and CEDMs have been replaced. The following weld materials were used in fabricating the pressure boundary of the replacement RVCH:

29.	SB-166 (Alloy 690)	SA-479, F-304	ERNiCrFe-7
30.	SA-508, Class 3	SB-167 (Alloy 690)	ERNiCrFe-7 or EniCrFe-7
31.	SB-167 (Alloy 690)	SB-166 (Alloy 690)	ERNiCrFe-7 or EniCrFe-7
32.	Austentic SS Cladding	SA-508, Class 3	SFA-5.9, ER-308L SFA-5.11, ER-309L
33.	Inconel Pad	SA-533, GR. B C1.1	SFA-5.14 ERNiCrFe-7/7A
34.	SB-166 (Alloy 690)	Inconel Pad	SFA-5.14 ERNiCrFe-7/7A
35.	SA-105, Gr. II	SA-312, 304L	ER-309L/316L
36.	SA-312, 304L	SA-182, F304L	ER-309L/316L
37.	SA-312, 304L	SA-479, 316L	ER-309L/316L

Note: (a) Materials exposed to reactor coolant

(b) Deleted

(c) Filler materials are not classified.

(d) Current welding is performed without filler material but wire may be used for repair.

(e) During replacement of the steam generators, the following welding materials were used for attachment of the RCS piping to the RSG nozzles.

(1) ER309L / E309L
(2) ER308L / E308L
(3) ER70S-6 / E-7018

(f) PC/M 07003M applied ErNiCrFe-7A UNS N06054 Structural Weld Overlay (SWOL) material to the following nozzle to safe-end welds - Hot Leg (HL) Shutdown Cooling Nozzles A&B, HL Surge Nozzle and HL Drain Nozzle

(g) PC/M 08180M replaced the original Alloy 600 Pressurizer heater sleeves with new Alloy 690 heater sleeves. The new heater sleeves are welded to an SFA-5.14 ERNiCrFe-7/7A (UNS N06052/N06054) weld pad applied to the external surface of the SA-533, GR. B C1.1 Pressurizer bottom head using SFA-5.14 ERNiCrFe-7/7A (UNS N06052/N06054) weld material.

Note (continued)

(h) PC/M 09078M mitigated the PWSCC of the RCS drain and spray nozzles by cut and replace. The piping / fittings were replaced with 304L/316L stainless steel and the weld material used for all welds from nozzle to safe-end was ER309L/E316L.

TABLE 5.2-5

CHEMICAL ANALYSES OF PLATE MATERIAL IN ST. LUCIE UNIT 2
REACTOR VESSEL BELTLINE

	Lower Shell Plate			Intermediate Shell Plate		
Heat #	B-8307-2	A-3131-1	A-3131-2	A-8490-2	B-3416-2	A-8490-1
Code #	M-4116-1	M-4116-2	M-4116-3	M-605-1	M-605-2	M-605-3
Element	(wt. %)					
Si	0.24	0.26	0.26	0.23	0.23	0.23
S	0.010	0.009	0.008	0.012	0.014	0.017
P	0.007	0.007	0.008	0.008	0.008	0.009
Mn	1.37	1.44	1.47	1.39	1.40	1.39
C	0.20	0.23	0.23	0.23	0.24	0.23
Cr	0.02	0.03	0.03	0.08	0.13	0.08
Ni	0.57	0.60	0.60	0.61	0.62	0.61
Mo	0.55	0.60	0.61	0.56	0.55	0.57
B	0.001	<.001	<.001	<.001	<.001	<.001
Cb	<.01	<.01	<.01	<.01	<.01	<.01
Ti	<.01	<.01	<.01	<.01	<.01	<.01
Co	0.011	0.011	0.012	0.015	0.016	0.015
Cu	0.06	0.07	0.07	0.11	0.13	0.11
Al	0.025	0.019	0.018	0.028	0.025	0.027
N_2	0.007	0.008	0.008	0.009	0.008	0.009
V	0.004	0.004	0.004	0.004	0.004	0.003
W	<.01	<.01	<.01	<.01	<.01	<.01

Notes: ND = Not Detected
NA = Not Analyzed

TABLE 5.2-5 (Cont'd)

	Lower Shell Plate			Intermediate Shell Plate		
As	0.003	0.004	0.005	0.007	0.006	0.008
Sn	0.003	0.003	0.004	0.011	0.013	0.012
Zr	<.001	<.001	<.001	<.001	<.001	<.001
Sb	NA	NA	NA	0.0032	0.0036	0.0030
Pb	ND	ND	ND	<0.001	<0.001	<0.001

Notes: ND = Not Detected
NA = Not Analyzed

TABLE 5.2-6

CHEMICAL ANALYSES OF WELD MATERIAL
IN ST. LUCIE 2 REACTOR VESSEL BELTLINE

Weld	Lower Shell Long. Welds			C1. Girth Seam Weld	Inter. Shell Long. Welds		
Seam #	101-142	101-142B	101-142C	101-171	101-124A	101-124B	101-124C
Element (Wt. %)							
Si	0.15	0.16	0.13	0.43	0.14	0.10	0.14
S	0.009	0.009	0.008	0.011	0.010	0.014	0.011
P	0.008	0.008	0.008	0.009	0.009	0.011	0.009
Mn	1.37	1.40	1.44	1.30	1.14	0.98	1.19
C	0.13	0.13	0.13	0.10	0.12	0.12	0.12
Cr	0.02	0.02	0.02	0.09	0.03	0.03	0.03
Ni[a]	0.10	0.09	0.09	0.08	0.06	0.06	0.07
Mo	0.58	0.56	0.60	0.53	0.54	0.33	0.58
B	<.001	<.001	<.001	.001	.0005	.0005	.0005
Cb	<.01	<.01	<.01	<.01	<.01	<.01	<.01
Ti	<.01	<.01	<.01	<.01	<.01	<.01	<.01
Co	0.005	0.005	0.005	0.009	0.012	0.012	0.012
Cu[a]	0.04	0.05	0.04	0.07	0.04	0.03	0.04
Al	0.005	0.002	0.002	0.003	0.001	<0.001	0.001
N_2	0.004	0.005	0.005	0.011	0.004	0.006	0.006
V	0.006	0.006	0.006	0.004	0.004	0.002	0.005
W	<.01	<.01	<.01	<.01	0.01	0.01	0.01
As	0.010	0.010	0.010	0.003	0.013	0.012	0.013
Sn	0.005	0.005	0.005	0.003	0.004	0.004	0.004
Zr	<.001	<.001	<.001	<.001	0.002	0.002	0.002
Sb	NA	NA	NA	0.0012	NA	NA	NA

TABLE 5.2-6(Cont'd)

Weld Seam #	Lower Shell Long. Welds 101-142	101-142B	101-142C	C1. Girth Seam Weld 101-171	Inter. Shell Long. Welds 101-124A	101-124B	101-124C
Pb	ND	ND	ND	<.001	ND	ND	ND

ND = Not Detected

NA = Not Analyzed

(a) The weld metal copper and nickel chemical analyses represents a single test result and should be maintained for historical purposes. Embrittlement predictions should be based on the "best estimate" copper and nickel values for a specific weld wire heat using all industry available data. These "best estimate" values were determined in response to NRC GL 92-01, Rev. 1, Supplement 1 and were submitted to the NRC in FPL Letter L-97-233.

TABLE 5.2-7
REACTOR VESSEL TOUGHNESS PROPERTIES

Location	Place Number	Code Number	Material	Drop Weight NDTT(°F)	RT_{NDT} (°F)	Minimum Transverse Charpy USE(1) (ft+lb)
Upper Shell Plate	122-102A	M-604-1	SA 533B C1 1	0	+50	-
Upper Shell Plate	122-102B	M-604-2	SA 533B C1 1	+10	+50	-
Upper Shell Plate	122-102C	M-604-3	SA 533B C1 1	-10	+10	-
Intermediate Shell Plate	124-102B	M-605-1	SA 533B C1 1	0	+30	105
Intermediate Shell Plate	124-102C	M-605-2	SA 533B C1 1	-10	+10	113
Intermediate Shell Plate	124-102A	M-605-3	SA 533B C1 1	-20	0	113
Lower Shell Plate	142-102C	M-4116-1	SA 533B C1 1	-30	+20	91
Lower Shell Plate	142-102B	M-4116-2	SA 533B C1 1	-50	+20	105
Lower Shell Plate	142-102A	M-4116-3	SA 533B C1 1	-40	+20	100
Closure Bead (3)	102-101	M-4110-1	SA 533B C1 1	-10	+30	-
Closure Bead Flange (3)	106-101	M-4101-1	SA 508 C1 2	0	0	-
Inlet Nozzle	128-101A	M-4102-1	SA 508 C1 2	-20	-20	-
Inlet Nozzle	128-101D	M-4102-2	SA 508 C1 2	-20	-20	-
Inlet Nozzle	128-101B	M-4102-3	SA 508 C1 2	-	0	-
Inlet Nozzle	128-101C	M-4102-4	SA 508 C1 2	-10	-10	-
Outlet Nozzle	128-301B	M-4103-1	SA 508 C1 2	-20	-20	-
Outlet Nozzle	128-301A	M-4103-2	SA 508 C1 2	-30	-30	-
Vessel Flange	126-101	M-602-1	SA 508 C1 2	-30	-10	-
Inlet Nozzle Safe End	131-102A	M-4104-1	SA 508 C1 1	-20	+20	-
Inlet Nozzle Safe End	131-102D	M-4104-2	SA 508 C1 1	-20	+20	-
Inlet Nozzle Safe End	131-102B	M-4104-3	SA 508 C1 1	-20	+20	-
Inlet Nozzle Safe End	131-102C	M-4104-4	SA 508 C1 1	-20	+20	-
Outlet Nozzle Safe End	131-101B	M-4105-1	SA 508 C1 1	-10	0	-
Outlet Nozzle Safe End	131-101A	M-41105-2	SA 508 C1 1	-10	0	-
Bottom Head Dome	152-101	M-4112-1	SA 533B C1 1	-50	-40	-
Bottom Head Torus	104-102 (A to F)	M-4111-1	SA 5335 C1 1	-40	+40	-
Closure Head Torus (3)	104-102 (A to D)	M-4109-1	SA 533B C1 1	-60	-10	

(1) Reported only for beltline region plates

(2) Not Used

(3) This is historical data for the original Closure Head. The Closure Head has been replaced. The replacement Closure Head Forging is SA 508, Class 3 Low Alloy Steel. From six Charpy Impact tests conducted at Tndt +60 degree F (20 deg F), the minimum absorbed energy was 139 ft-lbs, which is above the required 50 ft-lbs and the minimum lateral expansion was 79 mils, which is above the required 35 mils minimum. Based on these requirements, RTndt = -40 degree F.

TABLE 5.2-7a

IMPACT TEST DATA FOR ST. LUCIE 2 BELTLINE WELD MATERIALS

Wire Heat No.	Electrode or Flux Lot No.	NDT °F	Temp. °F	CVN Data Absorbed Energy ft-lbs	Shear %	Lat. Exp.	
101-124A	83642	3536 (Type 0091)	-80**	-20	96		66
			-56**		81		58
					72		51
				+10	118		78
					116		79
					115		78
		E8018-LOHB	-60	0	140		73
					135		71
					122		70
				+10	143		78
					137		73
					142		77
		E8018-IAOCE	-80	-100	8	0	3
					6	0	1
					8	0	3
				-80	21	15	12
					26	20	17
					15	15	8
				-40	45	30	30
					65	40	42
					87	50	52
				-20	100	60	60
					87	50	55
					60	40	40
				+60	127	90	70
					136	90	75
					142	90	79
				+100	150	100	81
					165	100	89
					149	100	80
		E8018-BABEF	-70	-80	13	0	6
					8	0	5
					11	0	5
				-40	83	30	52
					87	30	54
					50	20	33

**Note: The conservative generic value of the initial RT(NDT) of -56°F for weld heat 83642 is used for future predictions of adjusted RT(NDT) at the suggestion of the NRC, due to scatter in the available industry initial RT(NDT) data for this non limiting heat as noted in evaluation PSL-ENG-SESJ-97-026 and FPL letter L-97-136.

TABLE 5.2-7a (Cont'd)

Seam No.	Wire Heat No.	Electrode or Flux Lot No.	NDT °F	CVN Data: Temp. °F	Absorbed Energy ft-lbs	Shear %	Lat. Exp. Mils.
				-10	114	60	73
					83	40	51
					116	60	69
				+60	145	80	80
					148	80	80
					163	100	83
				+160	167	100	84
					161	100	88
					153	100	87
		E8018-HABJC	-70	-10	127		72
					129		75
					132		76
				+10	158		73
					167		82
					160		85
		E8018-FAAFC	-60	0	107		68
					103		66
					124		75
				+10	121		75
					118		69
					117		71
		E8018-HAGB	-40	+10	180		90
					146		83
					157		86
				+20	196		87
					210		88
					177		81
101-124B	83642	3536 (Type 0091)	see seam 101-124A				
		E8018 HABJC	see seam 101-124A				
		E8018 GACJC	-80	-30	66		44
					68		43
					53		37
				+10	109		78
					118		81
					112		75

TABLE 5.2-7a (Cont'd)

				CVN Data			
					Absorbed		
Seam No.	Wire Heat No.	Electrode or Flux Lot No.	NDT °F	Temp. °F	Energy ft-lbs	Shear %	Lat. Exp. Mils.
101-124C	83642	3536 (Type 0091)	see seam 101-124A				
		E8018-IAOCE	see seam 101-124A				
		E8018-BABEF	see seam 101-124A				
		E8018-HABJC	see seam 101-124A				
		E8018-FAAFC	see seam 101-124A				
		E8018-HAGB	see seam 101-124A				
	83637	1122 (Type 0091)	-50	+10	153		85
					131		81
					125		77
		E8018 JAHB	-40	+10	165		86
					138		82
					139		76
				+20	157		86
					137		82
					157		85
		E8018 BAOED	-50	+10	156		84
					149		79
					145		77
		E8018 AACJD	-60	0	119		73
					101		62
					152		90
				+10	126		74
					149		91
					127		77
101-142A	83637	1122 (Type 0091)	see seam 101-124C				

TABLE 5.2-7a (Cont'd)

Seam No.	Wire Heat No.	Electrode or Flux Lot No.	NDT °F	CVN Data Temp. °F	Absorbed Energy ft-lbs	Shear %	Lat. Exp. Mils.
101-142B	83637	1122 (Type 0091)	see seam 101-124C				
		E8018 FAOED	-60	-80	9	0	4
					19	5	10
					16	5	7
				-40	76	40	52
					70	40	44
					89	50	55
				0	98	60	60
					102	60	61
					86	50	52
				+100	157	100	90
					152	100	86
					153	100	86
		E8018 EACAE	-80	-100	21	5	12
					19	0	11
					27	5	16
				-20	83	40	52
					89	50	55
					85	40	54
				+100	123	100	78
					124	100	81
					136	100	91
101-142B		E8018 GABID	-50	+10	104		65
					105		64
					101		64
				+100	153		89
					156		91
					136		86
101-142C	83637	1122 (Type 0091)	see seam 101-124C				
101-171 *See surveillance weld seam 101-194, representative of 101-171	83637	0951 Type 124	-70	-10	82	60	61
					99	80	72
					94	70	66
				+100	112	100	83
					116	100	84
					116	100	85

TABLE 5.2-7a (Cont'd)

Seam No.	Wire Heat No.	Electrode or Flux Lot No.	NDT °F	CVN Data Temp. °F	Absorbed Energy ft-lbs	Shear %	Lat. Exp. Mils.
	3P7317	0951 Type 124	-80	-20	51	30	41
					50	30	39
					52	30	41
				+100	94	100	64
					93	100	63
					100	100	70
		E8018 IAOCE	see seam	No. 101-124A			
		E8018 FAAFF	-70	-10	66	30	43
					52	25	37
					74	35	48
				+160	127	100	81
					132	100	80
					122	100	80
		E8018 KABIF	-50	+10	97	50	62
					93	50	57
					96	50	60
				+160	132	100	75
					139	100	79
					156	100	85
101-194*	83637	0951 Type 124	-50	-40	10	20	16
				-30	41	30	31
*Surveillance weld representative of seam 101-171					19	30	16
					62	50	49
				-25	62	40	54
					38	20	29
*Full Charpy curve from baseline testing with Capsule W-83 BAW 1880					29	30	27
				-20	50	30	45
				-1	93	80	73
				+20	61	50	53
				+40	69	60	60
				+70	74	80	67
				+85	118	100	96
				+105	95	90	88
				+150	124	100	93
				+200	106	100	91
				+250	123	100	92
				+300	98	100	80
				+350	127	100	88
				+400	117	100	86
				+450	108	100	87

TABLE 5.2-8a

STEAM GENERATOR (PRIMARY SIDE) TOUGHNESS PROPERTIES (SG A)

Name of Part	Part Number	Heat Number	Material	Drop Weight NDTT (°F)	^{RT}NDT (°F)
Channel Head	GV/SL313 FS/001	04W81-1-1	SA 508 Gr 3 Cl 2	-20	-20
Tubesheet	GV/SL313 PT/001	04W24-1-1	SA 508 Gr 3 Cl 2	-10	-10
Inlet Nozzle Safe End	GV/SL313 ERP1	51049	SA 105	-18	-18
Outlet Nozzle Safe End - 1	GV/SL313 ERP2-1	51049	SA 105	-18	-18
Outlet Nozzle Safe End -2	GV/SL313 ERP2-2	51049	SA 105	-18	-18
Partition Plate	GV/SL313 PLPA		SB-168 UNS N06690	N/A	N/A

TABLE 5.2-8b

STEAM GENERATOR (PRIMARY SIDE) TOUGHNESS PROPERTIES (SG B)

Name of Part	Part Number	Heat Number	Material	Drop Weight NDTT (°F)	^{RT}NDT (°F)
Channel Head	GV/SL314 FS/001	04W84-1-1	SA 508 Gr 3 Cl 2	-20	-20
Tubesheet	GV/SL314 PT/001	04W25-1-1	SA 508 Gr 3 Cl 2	-40	-40
Inlet Nozzle Safe End	GV/SL314 ERP1	51049	SA 105	-50	-50
Outlet Nozzle Safe End - 1	GV/SL314 ERP2-1	51049	SA 105	-50	-50
Outlet Nozzle Safe End -2	GV/SL314 ERP2-2	51049	SA 105	-50	-50
Partition Plate	GV/SL314 PLPA		SB-168 UNS N06690	N/A	N/A

TABLE 5.2-9

PRESSURIZER TOUGHNESS PROPERTIES

Location	Piece Number	Code Number	Material	Drop Weight NDTT (F)	RT_{NDT} (F)
Upper Shell	622-102A	M-8906-1	SA 533B Cl 1	-40	-30
Upper Shell	622-102B	M-8906-2	SA 533B Cl 1	-40	-30
Lower Shell	642-102B	M-8906-3	SA 533B Cl 1	-40	-40
Lower Shell	642-102A	M-8906-4	SA 533B Cl 1	-30	-30
Manway Cover	676-102	M-2137-1	SA 533B Cl 1	-10	-10
Surge Nozzle	658-101	M-8902-1	SA 508 Cl 2	-40	-40
Spray Nozzle	608-101	M-8920-1	SA 508 Cl 2	-30	-30
Bottom Head	650-101	M-8901-1	SA 533B Cl 1	+10	+10
Closure Head	601-102	M-8901-2	SA 533B Cl 1	+10	+10

TABLE 5.2-10

STEAM GENERATOR (SECONDARY SIDE) TOUGHNESS PROPERTIES (SG A)

Name of Part	Part Number	Heat Number	Material	Drop Weight NDTT (°F)	RT_{NDT} (°F)
Lower Shell	GV/SL313 VI/001	04D358-1-1	SA 508 Gr 3 Cl 2	-20	-20
Middle Shell	GV/SL313 VI/002	04D477-1-1	SA 508 Gr 3 Cl 2	-20	-20
Conical Shell	GV/SL313 VI/003	04W110-1-1	SA 508 Gr 3 Cl 2	-10	-10
Intermediate Shell	GV/SL313 VI/004	04D593-1-1	SA 508 Gr 3 Cl 2	-20	-20
Upper Shell	GV/SL313 VI/005	04W92-1-1	SA 508 Gr 3 Cl 2	-10	-10
Elliptical Head	GV/SL313 FE/001	04W92-1-2	SA 508 Gr 3 Cl 2	-20	-20
		04D518-1-1		-10	-10
Feed Water Nozzle	GV/SL313 TE/001	5-0025	SA 508 Gr 3 Cl 2	0	0
Recirculation Nozzle	GV/SL313 TE/002	5-0025	SA 508 Gr 3 Cl 2	-18	-18
Secondary Manways	GV/SL313 TE/003	5-0025	SA 508 Gr 3 Cl 2	-9	-9

TABLE 5.2-11

STEAM GENERATOR (SECONDARY SIDE) TOUGHNESS-PROPERTIES (SG B)

Name of Part	Part Number	Heat Number	Material	Drop Weight NDTT (°F)	^{RT}NDT (°F)
Lower Shell	GV/SL314 VI/001	04D359-1-1	SA 508 Gr 3 Cl 2	-10	-10
Middle Shell	GV/SL314 VI/002	04D478-1-1	SA 508 Gr 3 Cl 2	-10	-10
Conical Shell	GV/SL313 VI/003	04W111-1-1	SA 508 Gr 3 Cl 2	-20	-20
Intermediate Shell	GV/SL314 VI/004	04D663-1-1	SA 508 Gr 3 Cl 2	0	0
Upper Shell	GV/SL314 VI/005	04W144-1-1	SA 508 Gr 3 Cl 2	-20	-20
Elliptical Head	GV/SL314 FE/001	04W144-1-2	SA 508 Gr 3 Cl 2	-20	-20
		04D515-1-1		-10	-10
Feed Water Nozzle	GV/SL314 TE/001	5-0025	SA 508 Gr 3 Cl 2	0	0
Recirculation Nozzle	GV/SL314 TE/002	5-0025	SA 508 Gr 3 Cl 2	-18	-18
Secondary Manways	GV/SL314 TE/003	5-0025	SA 508 Gr 3 Cl 2	-9	-9

TABLE 5.2-12

PIPING TOUGHNESS PROPERTIES

Location	Piece Number	Code Number	Material	Drop Weight NDTT (^{o}F)	RT_{NDT} (^{o}F)
Spray Nozzle	728-401	M-9213-1	SA 105	NA	+20^{A}
Spray Nozzle	728-401	M-9213-2	SA 105	NA	+20^{A}
Let Down Drain Nozzle	728-601	M-9214-1	SA 105	NA	+20^{A}
Let Down Drain Nozzle	728-601	M-9214-2	SA 105	NA	+20^{A}
Let Down Drain Nozzle	728-601	M-9214-3	SA 105	NA	+20^{A}
Let Down Drain Nozzle	728-601	M-9214-4	SA 105	NA	+20^{A}
Charging Inlet Nozzle	728-501	M-9217-1	SA 182 Gr F1	+10	+10
Charging Inlet Nozzle	728-501	M-9217-2	SA 182 Gr F1	+10	+10
Safety Injection Nozzle	728-301	M-9218-1	SA 182 Gr F1	+20	+20
Safety Injection Nozzle	728-301	M-9218-2	SA 182 Gr F1	+20	+20
Safety Injection Nozzle	728-301	M-9218-3	SA 182 Gr F1	+20	+20
Safety Injection Nozzle	728-301	M-9218-4	SA 182 Gr F1	+20	+20
Surge Nozzle	728-101	M-9211-1	SA 541-1	-10	0
Shutdown Coolant Outlet Noz.	728-201	M-9212-1	SA 541-1	-10	-10
Shutdown Coolant Outlet Noz.	728-201	M-9212-2	SA 541-1	-10	-10
Drain Nozzle	728-701	M-9215-1	SA 105	NA	+20^{A}
Straight Segment	722-102A	M-9201-1	SA 516 Gr 70	-10	+60
Straight Segment	722-102B	M-9201-2	SA 516 Gr 70	-10	+60
Straight Segment	722-102A	M-9201-3	SA 516 Gr 70	-30	+30
Straight Segment	722-102B	M-9201-4	SA 516 Gr 70	-30	+30
Straight Segment	722-106A	M-9202-1	SA 516 Gr 70	-30	0
Straight Segment	722-106B	M-9202-2	SA 516 Gr 70	-20	+20

NA - Not Available

A-MTEB position 5-2, "Fracture Toughness Requirements for Older Plants," Paragraph 1.1(3)(b)

TABLE 5.2-12 (Cont'd)

Location	Piece Number	Code Number	Material	Drop Weight NDTT (^{o}F)	RT_{NDT} (^{o}F)
Straight Segment	722-106A	M-9202-3	SA 516 Gr 70	-20	+20
Straight Segment	722-106B	M-9202-4	SA 516 Gr 70	-30	0
Straight Segment	722-108A	M-9203-1	SA 516 Gr 70	-30	0
Straight Segment	722-108B	M-9203-2	SA 516 Gr 70	-20	-10
Straight Segment	722-110A	M-9204-1	SA 516 Gr 70	-20	+40
Straight Segment	722-110B	M-9204-2	SA 516 Gr 70	-20	+40
Straight Segment	722-110A	M-9204-3	SA 516 Gr 70	-10	+30
Straight Segment	722-110B	M-9204-4	SA 516 Gr 70	-10	+30
Straight Segment	722-104A	M-9205-1	SA 516 Gr 70	-20	+30
Straight Segment	722-104B	M-9205-2	SA 516 Gr 70	-20	+30
Straight Segment	722-104A	M-9205-3	SA 516 Gr 70	-40	+20
Straight Segment	722-104B	M-9205-4	SA 516 GR 70	-10	+30
Elbow	742-102A&B	M-9206-1	SA 516 Gr 70	-30	+20
Elbow	742-108A&B	M-9207-1	SA 516 Gr 70	-30	+30
Elbow	742-108A&B	M-9207-2	SA 516 Gr 70	-10	+40
Elbow	742-104A&B	M-9208-1	SA 516 Gr 70	-20	+20
Elbow	742-110A&B	M-9209-1	SA 516 Gr 70	-30	-30
Elbow	742-110A&B	M-9209-2	SA 516 Gr 70	-30	+20
Elbow	742-112A&B	M-9209-3	SA 516 Gr 70	-30	+30
Elbow	742-106A&B	M-9210-1	SA 516 Gr 70	-30	0

TABLE 5.2-12a

CHARPY V-NOTCH IMPACT DATA FOR NB-2332.a COMPONENTS

		CVN DATA									
Component Code No.	Heat No.	Temp °F	Energy Absorbed (ft-lb)			Lat. Exp. (Mils)			Shear (%)		
			1	2	3	1	2	3	1	2	3
M-9213-1		-100	2	3	2	2	4	1	1	1	1
M-9213-2		-40	40	12	11	44	11	9	1	1	1
M-9214-1	A422QT	0	21	30	33	20	30	30	10	20	20
-2		+20	240	240	65	90	88	46	100	100	30
-3		+40	133	240	240	88	86	90	80	100	100
-4		+100	240	240	240	81	88	88	100	100	100
M-9215-1											

All seven nozzles were manufactured from three forgings taken from the same heat.

TABLE 5.2-13

BOLTING MATERIALS TOUGHNESS PROPERTIES

Component	Piece Number	Code Number	Location	Material	Temperature for 45 ft-lb and 25 mils Lat. Exp. (°F)
Reactor Vessel	179-101	M-4114-1	Studs	SA 540 Gr B-24	+10
Reactor Vessel	179-102	M-4115-1	Nuts	SA 540 Gr B-23	+10
Reactor Vessel	179-103	M-4115-1	Washers	SA 540 Gr B-23	+10
Steam Generator A	GV/SL313 GJTHP	Heat 28853	Pri. Manway Studs	SA 193 Gr B16	+60
Steam Generator A	GV/SL313 BL/120	Heat 3093	Pri. Manway Nuts	SA 194 Gr B7	+60
Steam Generator B	GV/SL314 GJTHP	Heat 28853	Pri. Manway Studs	SA 193 Gr B16	+60
Steam Generator B	GV/SL314 BL/120	Heat 3093	Pri. Manway Nuts	SA 194 Gr B7	+60
Pressurizer	676-108	C-5364	Manway Nuts	SA 193 Gr B-7	+60 [A]
Pressurizer	674-106	C-5365	Manway Studs	SA 540 Gr B-24	+10 [A]
Pressurizer Safety Valves	674-106	C-5365	Flange Bolts	SA 193 Gr B-7	+40
Power Operated Relief Valves	676-106	C-5365	Flange Bolts	SA 193 Gr B-7	+40

A - Estimated based on extrapolation of available data (reference Subsection 5.2.3.3.1)

TABLE 5.2-13a

CVN DATA FOR PRESSURIZER MANWAY NUTS, CODE NO. C-5364

Temp F	Ft-lbs	% Shear	Mils Lat Exp
+10	53	80	33
+10	25	40	18
+10	41	60	27

TABLE 5.2-13b

HEAT TREATMENT FOR PRESSURIZER MANWAY NUTS, CODE NO. C-5364

Austenitized	1550F, oil quenched
Tempered	1000F
Stress Relieved	1000F

TABLE 5.2-13c

IMPACT DATA FOR SA-193 GR. B-7 MATERIAL

Heat No.	Piece No.	Temp.	Absorbed Energy (ft-lbs) 1	2	3	Shear (%) 1	2	3	Lateral Exp. (Mils.) 1	2	3
85689	876	+10	69	68	69	100	100	100	45	45	46
	876-1	+10	69	69	70	100	100	100	48	48	46
	831	+10	65	65	65	100	100	100	44	46	43
	831-1	+10	66	68	68	100	100	100	43	43	45
	832	+10	60	60	59	100	100	100	41	38	37
	832-1	+10	60	60	60	100	100	100	39	38	38
	907	+10	59	60	61	100	100	100	36	38	38
	907-1	+10	64	66	65	100	100	100	43	45	45
42540		+10	65	66.5	53	62	64	53	47	50	37
215272		+40	61	74	64	100	100	100	38	44	45
			63	59	61	100	100	100	43	33	36
			59	68	49	100	100	73	31	30	34
216276		+40	60	63	61	100	100	100	42	37	39
215272		+40	75	90	86	100	100	86	40	73	54
215272		+40	64	55	53	79	66	69	39	32	30

TABLE 5.2-13d

HEAT TREATMENT FOR SA-193 GR. B-7 MATERIALS

Heat No.			
85689	Austenitized	1550 F, 1-1/2 hrs at heat,	water quenched
	Tempered	1100 F, 8 hrs. ,	air cooled
	Stress Relieved	1100 F, 5 hrs. ,	air cooled
42540	Austenitized	1550 F, 2-1/2 hrs. ,	oil quenched
	Tempered	1100 F, 5 hrs.	
215272	Austenitized	1580 F, 8 hrs. ,	oil quenched
	Tempered	1120 F, 6 hrs.	
215276	Austenitized	1570 F, 6 hrs. ,	water quenched
	Tempered	1140 F, 7 hrs.	
215272	Austenitized	1560 F, 5 hrs. ,	oil quenched
	Tempered	1130 F, 7 hrs.	
215272	Austenitized	1580 F, 8 hrs. ,	oil quenched
	Tempered	1170 F, 8 hrs.	

TABLE 5.2-13e

STEAM GENERATOR PRIMARY MANWAY STUD AND BOLT MATERIAL IMPACT AND HEAT TREATMENT DATA

Part Number	Heat Number	Material	Temp.	Absorbed Energy (ft/lb)			Lateral Expansion (Mils.)			Austenitized (°F)	Tempering (°F)	Quenching
				1	2	3	1	2	3			
GV/SL313 GJTHP	28853	SA 193 Gr B16	+60	86	93	87	62	64	62	1706	1238	Oil
GV/SL313 BL/120	3093	SA 194 Gr B7	+60	89	90	92	62	65	67	1562	1202	Water
GV/SL314 GJTHP	28853	SA 193 Gr B16	+60	86	93	87	62	64	62	1706	1238	Oil
GV/SL314 BL/120	3093	SA 194 Gr B7	+60	89	90	92	62	65	67	1562	1202	Water

TABLE 5.2-14
REACTOR COOLANT LEAK DETECTION SENSITIVITY

Leakage Source	Detection Instrumentation	Instrument Range(1)	Normal Reading	Average Rate of Change for 1.0 gpm Leak	Time for Scale to Move 10% from Normal Reading for 1.0 gpm Leak
1. Direct	Sump input flow alarm Monitoring Recorder Containment Radiation		NA 0 1.8E-10μCi/cc 9.5E-5 μCi/cc	NA 10 min for first 1 gpm indication 1.5E-7μCi/cc/min	NA NA >62 min
2. Safety Valves and PORV	Discharge Line Temperature Quench Tank Water Level Acoustic Monitoring Alarm		90-120° F 29 in. NA	- - NA	NA NA NA
3. S.I. Tank Check Valves	S.I. Tank Water Level S.I. Tank Pressure		80-88% 579-621 psi	0.0083%/min. 0.36 psi/min.	17.7 hr. 2.87 hr.
4. Heat Exchanger	CCW Radiation CCW Surge Tank Water Level Alarm		2.1E-6μCi/cc NA	2.0E-6μCi/cc/min NA	<1 min. NA
5. Steam Generator Tubing	Blowdown Line Radiation Condenser vacuum pumps exhaust Radiation		5.3E-6μCi/cc 5E-6μCi/cc	2.0E-7μCi/cc/min 2.0E-7μCi/cc/min	<1 min. <1 min.
6. Reactor Vessel Closure Head	O-Ring Space Pressure		0 psig	-	NA
7. Reactor Coolant Pump Closure Cover	Flange Gasket leak of pressure		0 psig	-	Less than 1 min
8. Steam Generator Tube Rupture	Main steam line radiation detectors		Background		NA

TABLE 5.2-14 (Cont'd)

(1) Instrument ranges are selected in accordance with standard Engineering practices.

TABLE 5.2-15

PRESSURE ISOLATION VALVES

Addressed by Generic Letter 87-06

VALVE NUMBER	VALVE FUNCTION
V3227	2A1 SI HDR. CK. VALVE
V3217	2A2 SI HDR. CK. VALVE
V3237	2B1 SI HDR. CK. VALVE
V3247	2B2 SI HDR. CK. VALVE
V3225	2A1 SIT DISCH. CK. VALVE
V3215	2A2 SIT DISCH. CK. VALVE
V3235	2B1 SIT DISCH. CK. VALVE
V3245	2B2 SIT DISCH. CK. VALVE
V3258	2A1 SI HDR. CK. VALVE
V3259	2A2 SI HDR. CK. VALVE
V3260	2B1 SI HDR. CK. VALVE
V3261	2B2 SI HDR. CK. VALVE
V3525	2A HOT LEG INJ. CK. VALVE
V3524	2A HOT LEG INJ. CK. VALVE
V3527	2B HOT LEG INJ. CK. VALVE
V3526	2B HOT LEG INJ. CK. VALVE
V3480	2A SDC RTN. ISOL. VALVE
V3481	2A SDC RTN. ISOL. VALVE
V3652	2B SDC RTN. ISOL. VALVE
V3651	2B SDC RTN. ISOL. VALVE

ABBREVIATION INDEX:

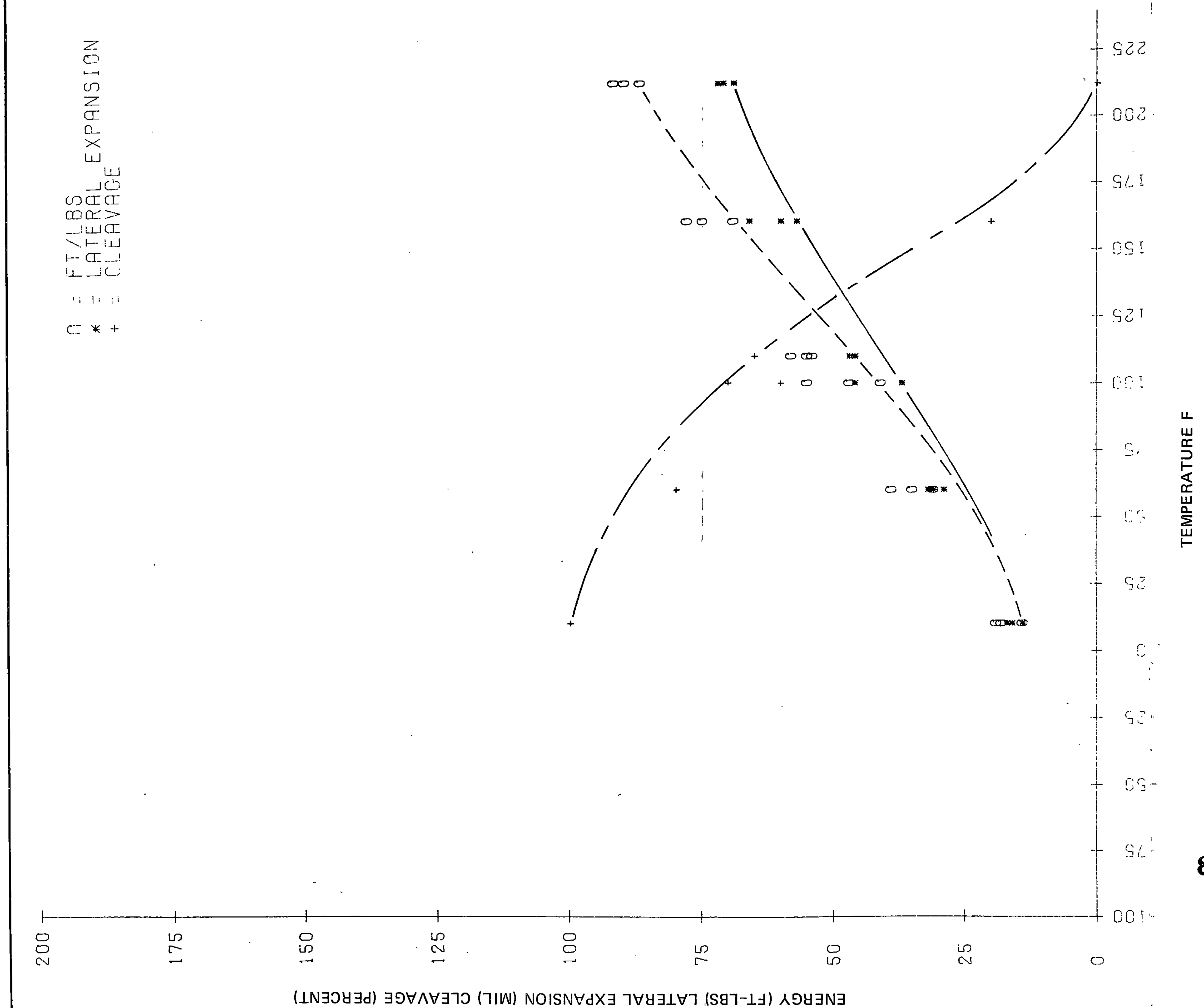
SI	= Safety Injection
HDR	= Header
CK	= Check
SIT	= Safety Injection Tank
DISCH	= Discharge
INJ	= Injection
SDC	= Shut Down Cooling
RTN	= Return
ISOL	= Isolation

Refer to Drawing
2998-G-211

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MISCELLANEOUS PIPING REACTOR
AREA SH. 1

FIGURE 5.2-1



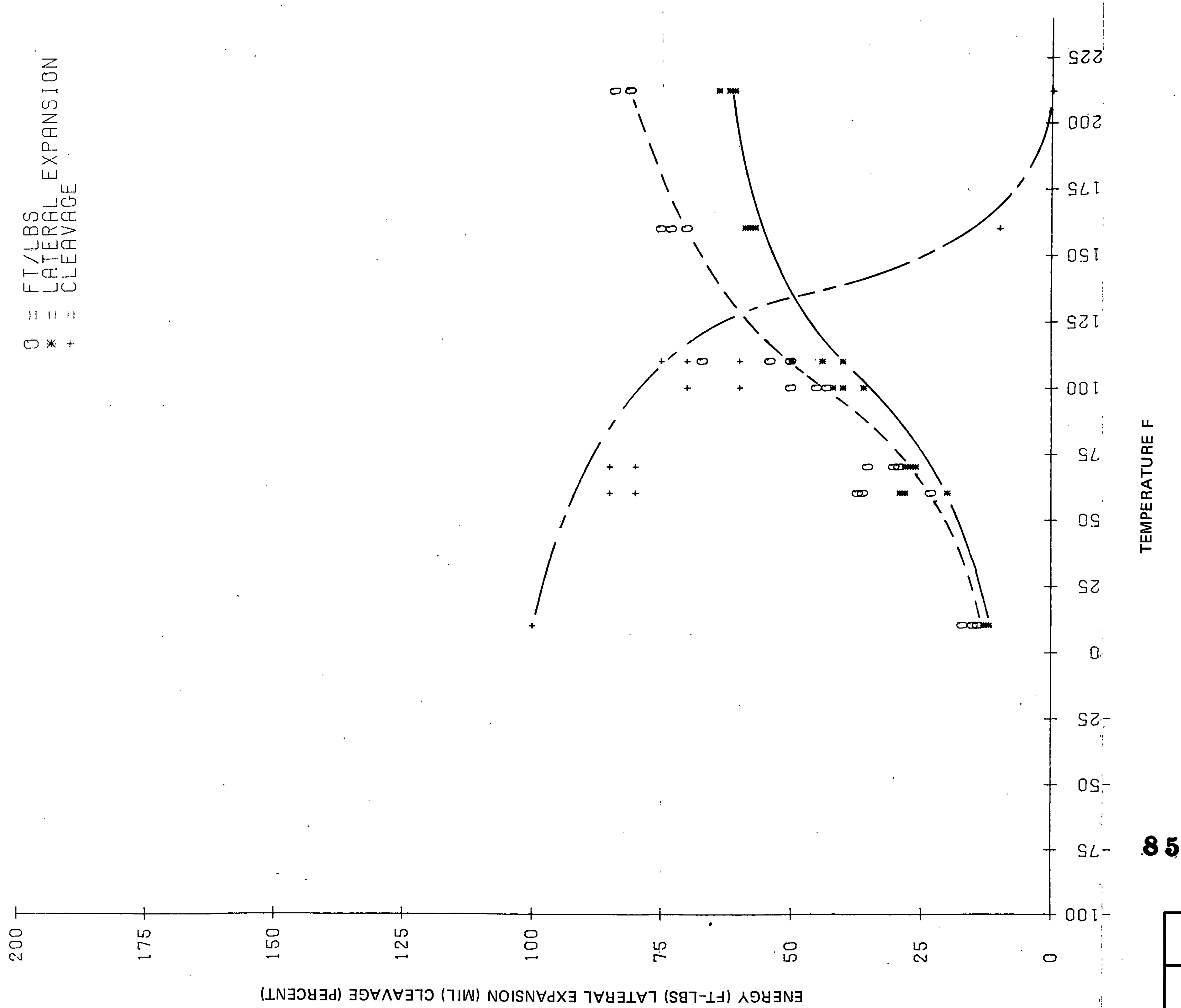
TI APERTURE CARD

8504150391 -11

Also Available On Aperture Card

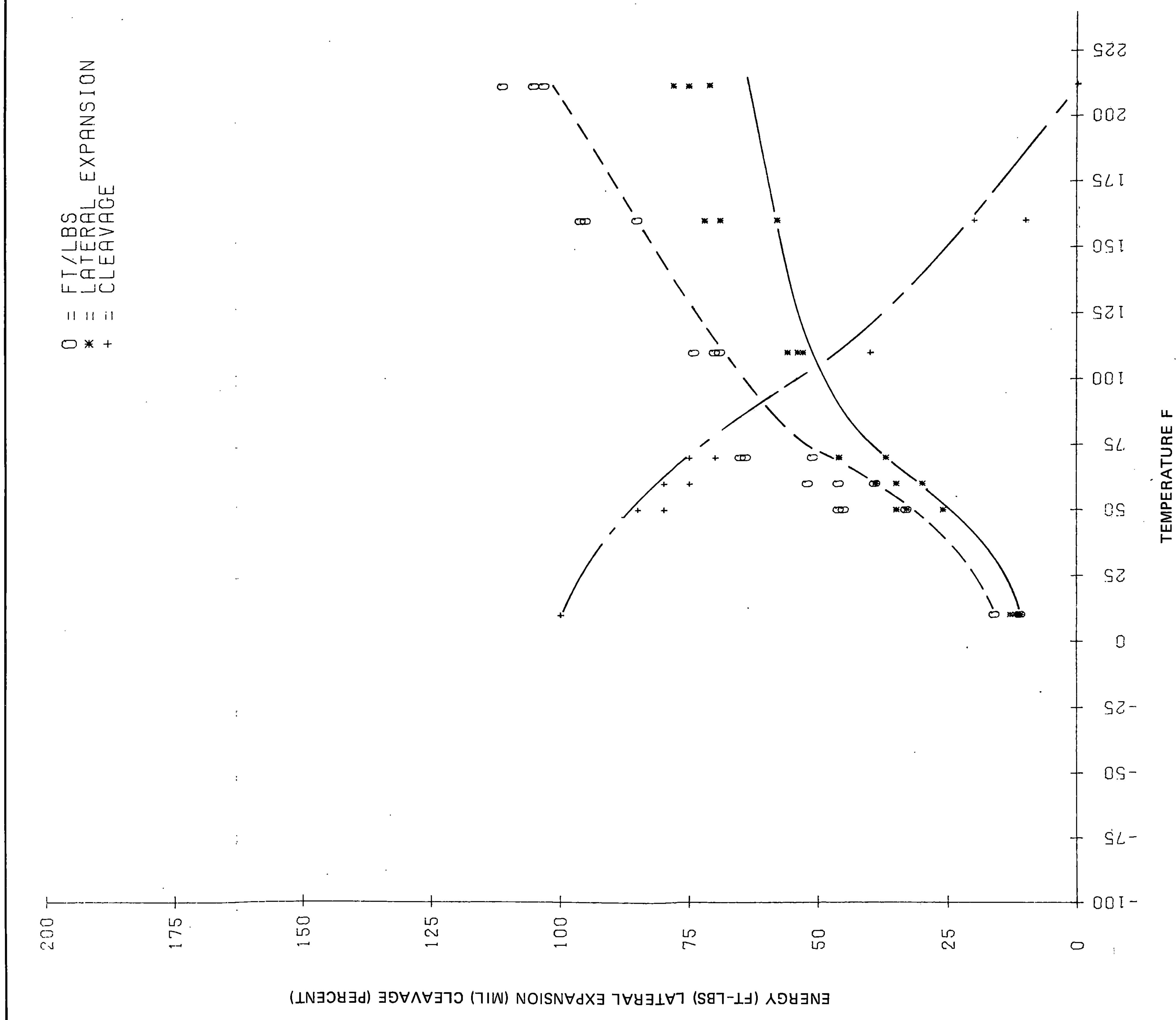
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
UPPER SHELL PLATE
M-604-1
FIGURE 5.2-2



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

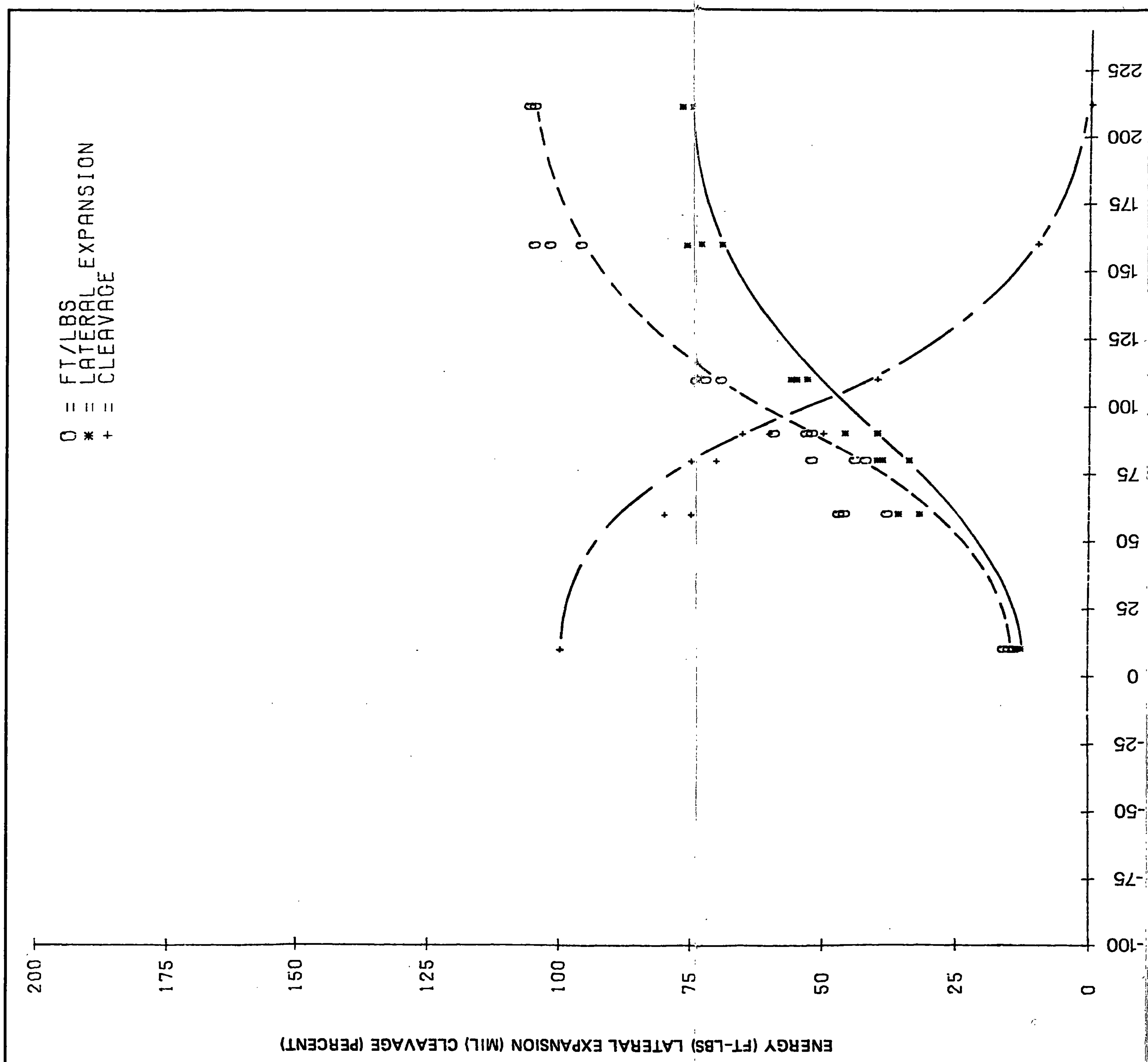
CHARPY TEST RESULTS
UPPER SHELL PLATE
M-604-2
FIGURE 5.2-3



8504150391 -13

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
UPPER SHELL PLATE
M-604-3
FIGURE 5.2-4



ANSTEC APERTURE CARD

Also Available on Aperture Card

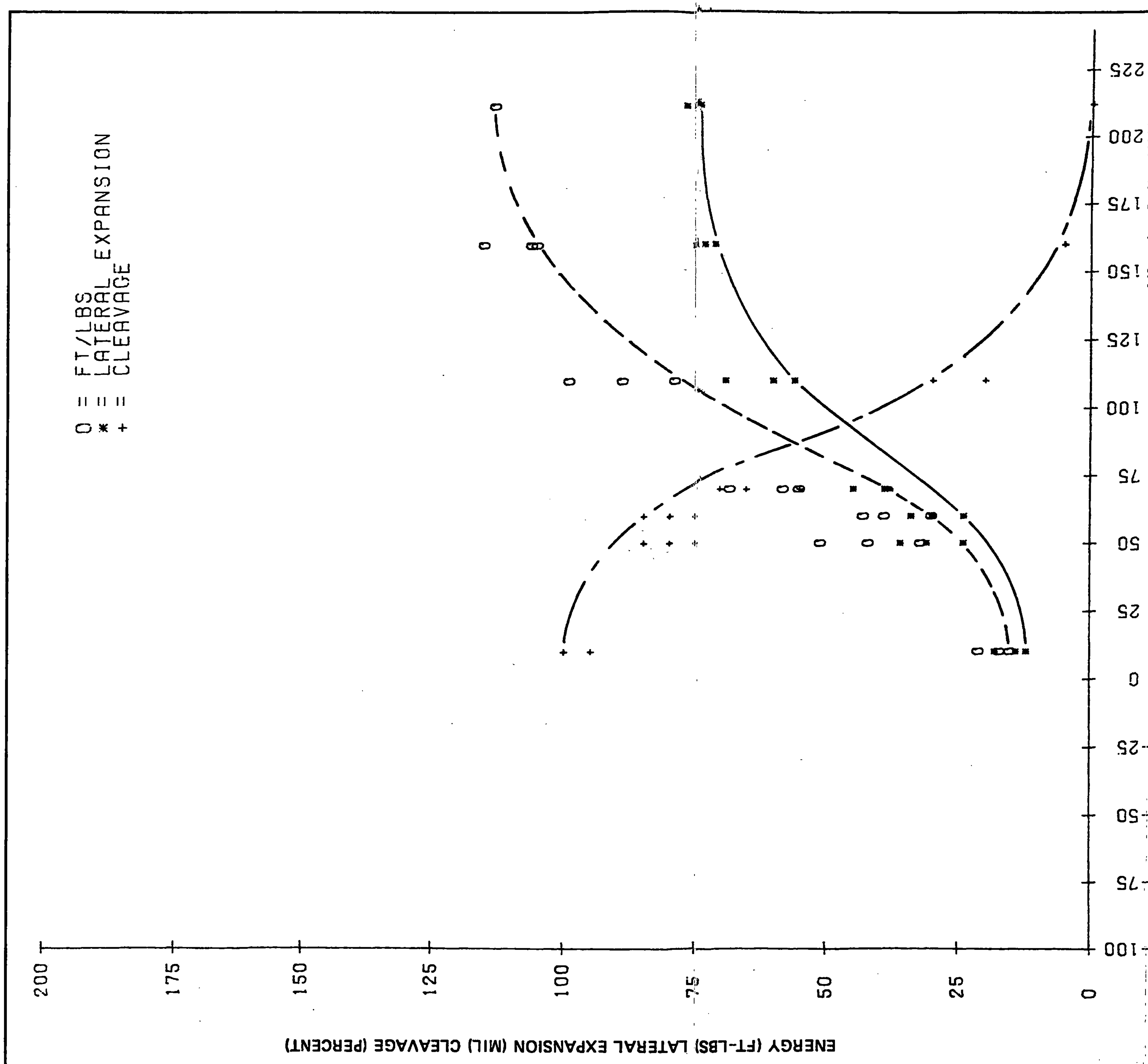
9410270047 - 17

AMENDMENT NO. 9 (10/94)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT - UNIT 2

TRANSVERSE CHARPY TEST RESULTS
INTERMEDIATE SHELL PLATE
M-605-1

FIGURE 5.2-5



ANSTEC APERTURE CARD

Also Available on Aperture Card

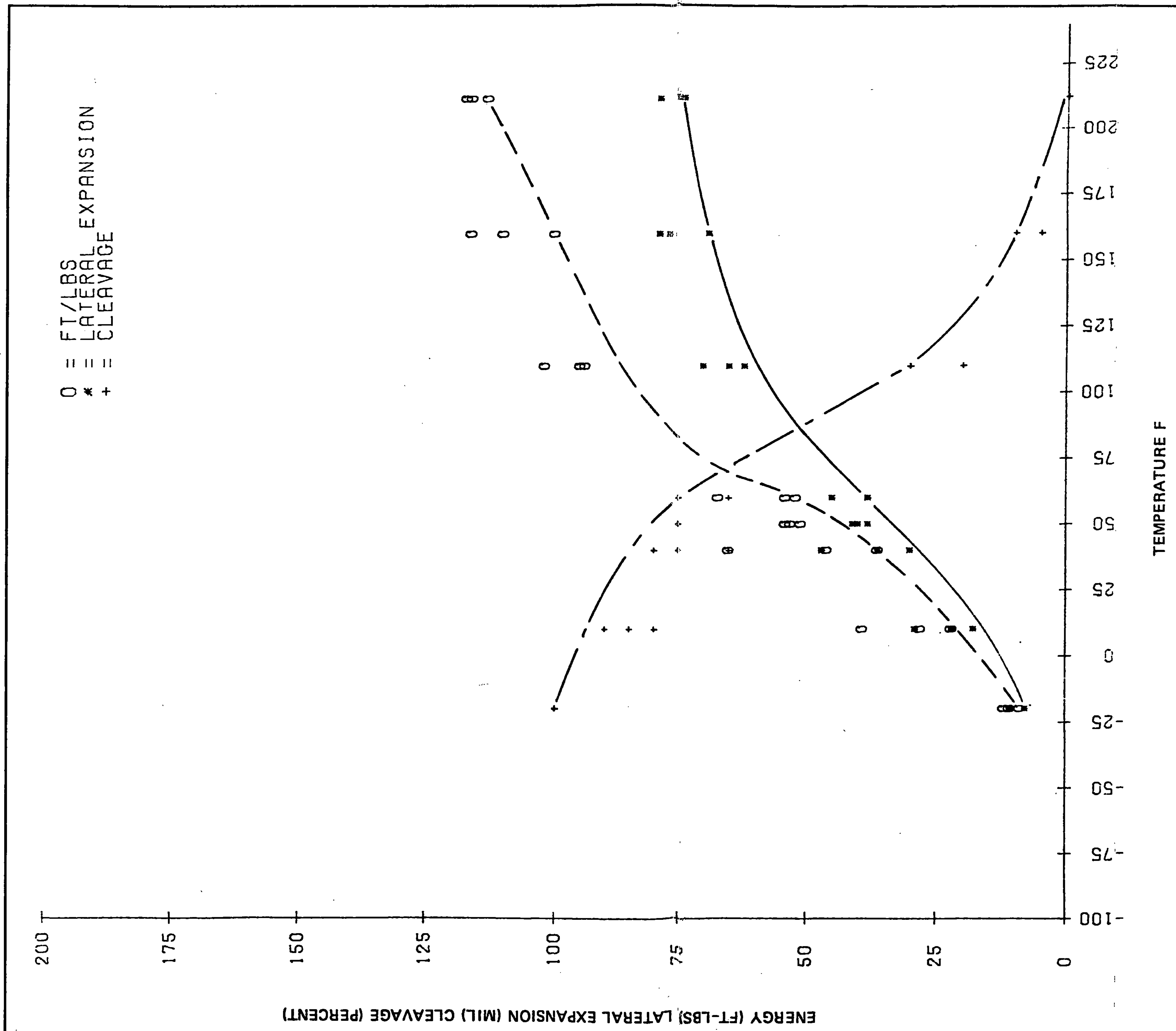
9410270047-18

AMENDMENT NO. 9 (10/94)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT - UNIT 2

TRANSVERSE CHARPY TEST RESULTS
INTERMEDIATE SHELL PLATE
M-605-2

FIGURE 5.2-6



ANSTEC APERTURE CARD

Also Available on Aperture Card

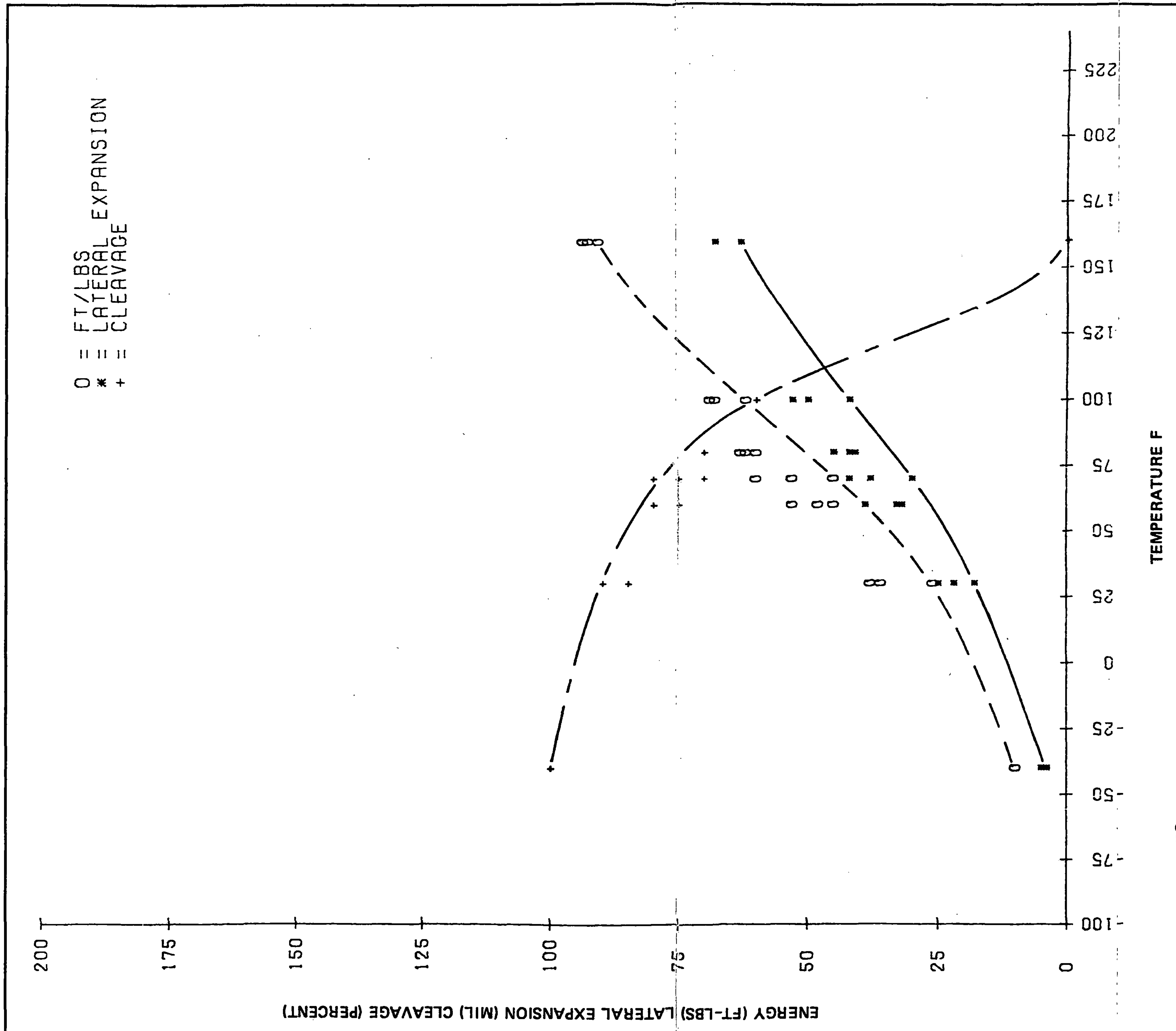
9410270047 - 19

AMENDMENT NO. 9 (10/94)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT - UNIT 2

TRANSVERSE CHARPY TEST RESULTS
INTERMEDIATE SHELL PLATE
M-605-3

FIGURE 5.2-7



ANSTEC APERTURE CARD

Also Available on Aperture Card

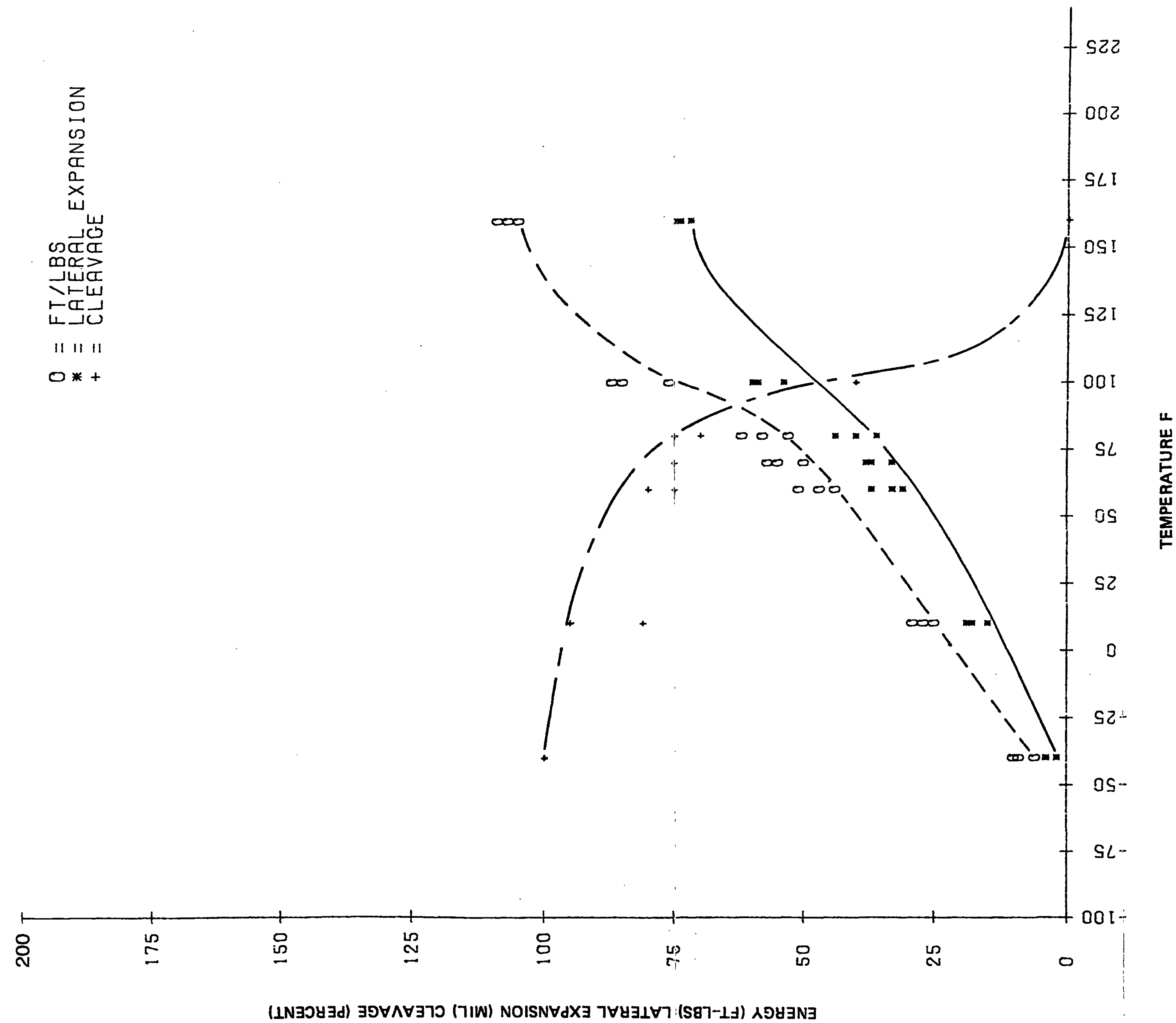
9410270047 - 20

AMENDMENT NO. 9 (10/94)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT - UNIT 2

TRANSVERSE CHARPY TEST RESULTS
LOWER SHELL PLATE
M-4116-1

FIGURE 5.2-8



ANSTEC
APERTURE
CARD

Also Available on
Aperture Card

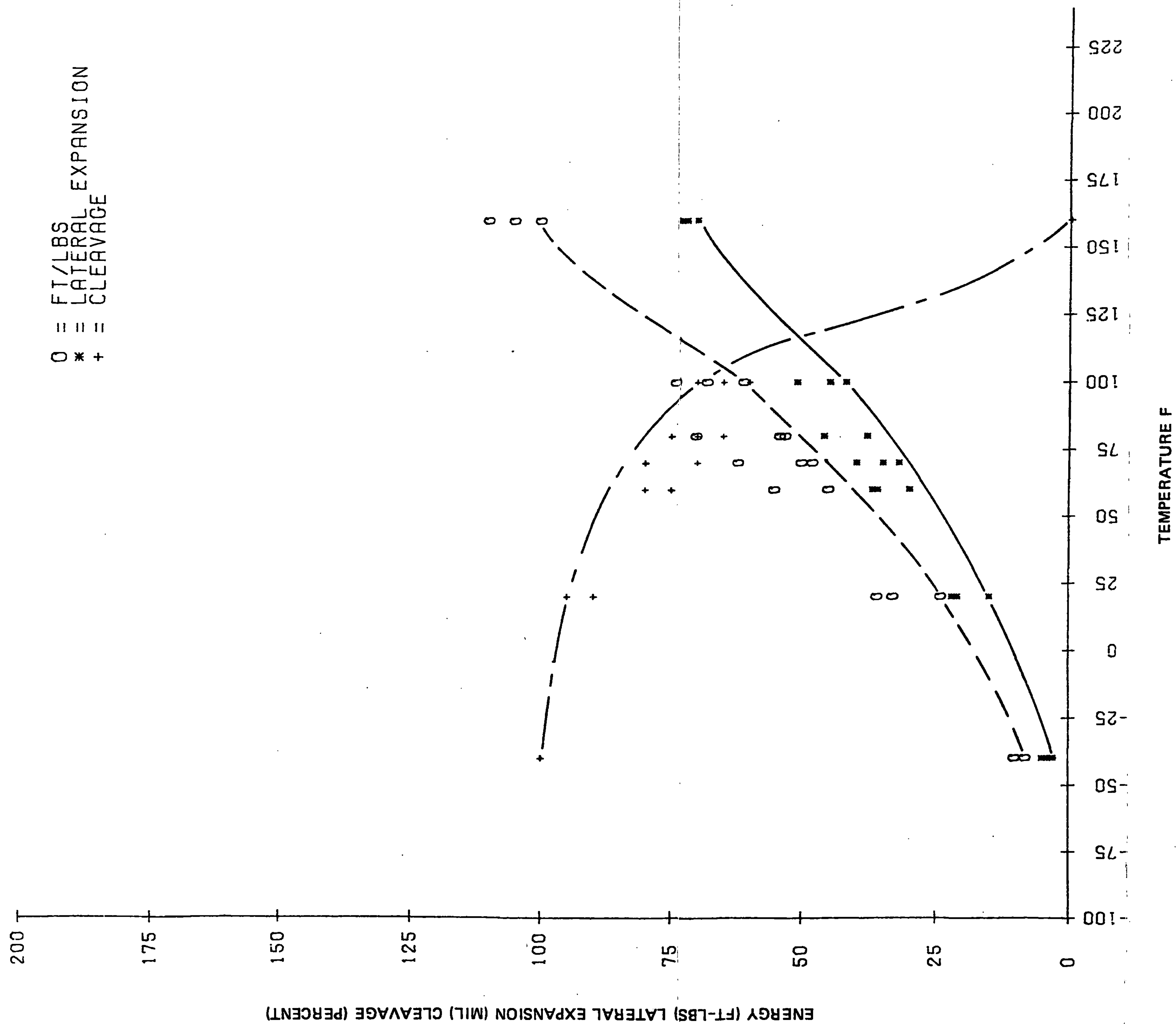
9410270047 - 21

AMENDMENT NO. 9 (10/94)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT - UNIT 2

TRANSVERSE CHARPY TEST RESULTS
LOWER SHELL PLATE
M-4116-2

FIGURE 5.2-9



ANSTEC
APERTURE
CARD

Also Available on
Aperture Card

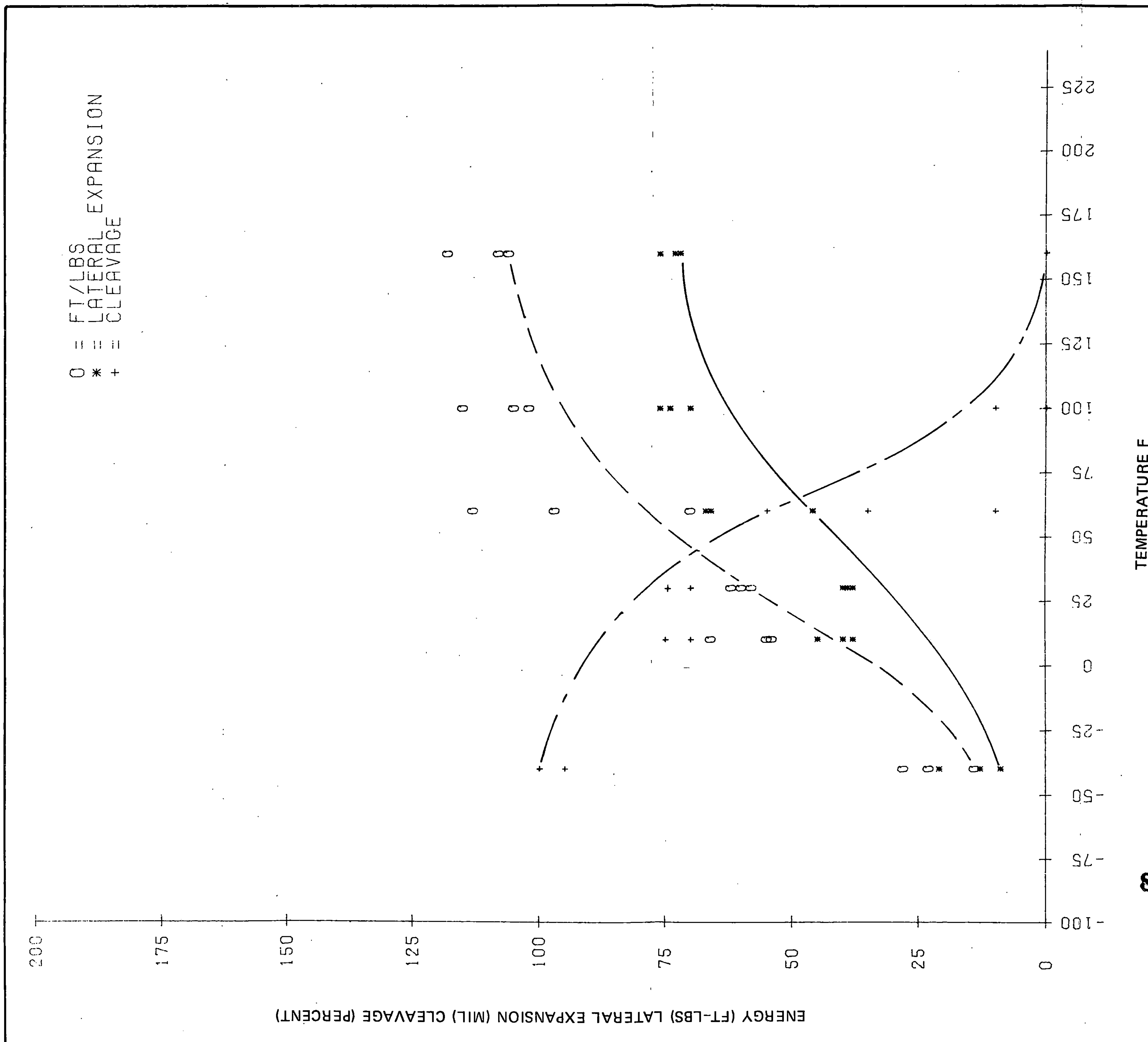
9410270047 - 22

AMENDMENT NO. 9 (10/94)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT - UNIT 2

TRANSVERSE CHARPY TEST RESULTS
LOWER SHELL PLATE
M-4116-3

FIGURE 5.2-10



Also Available On Aperture Card

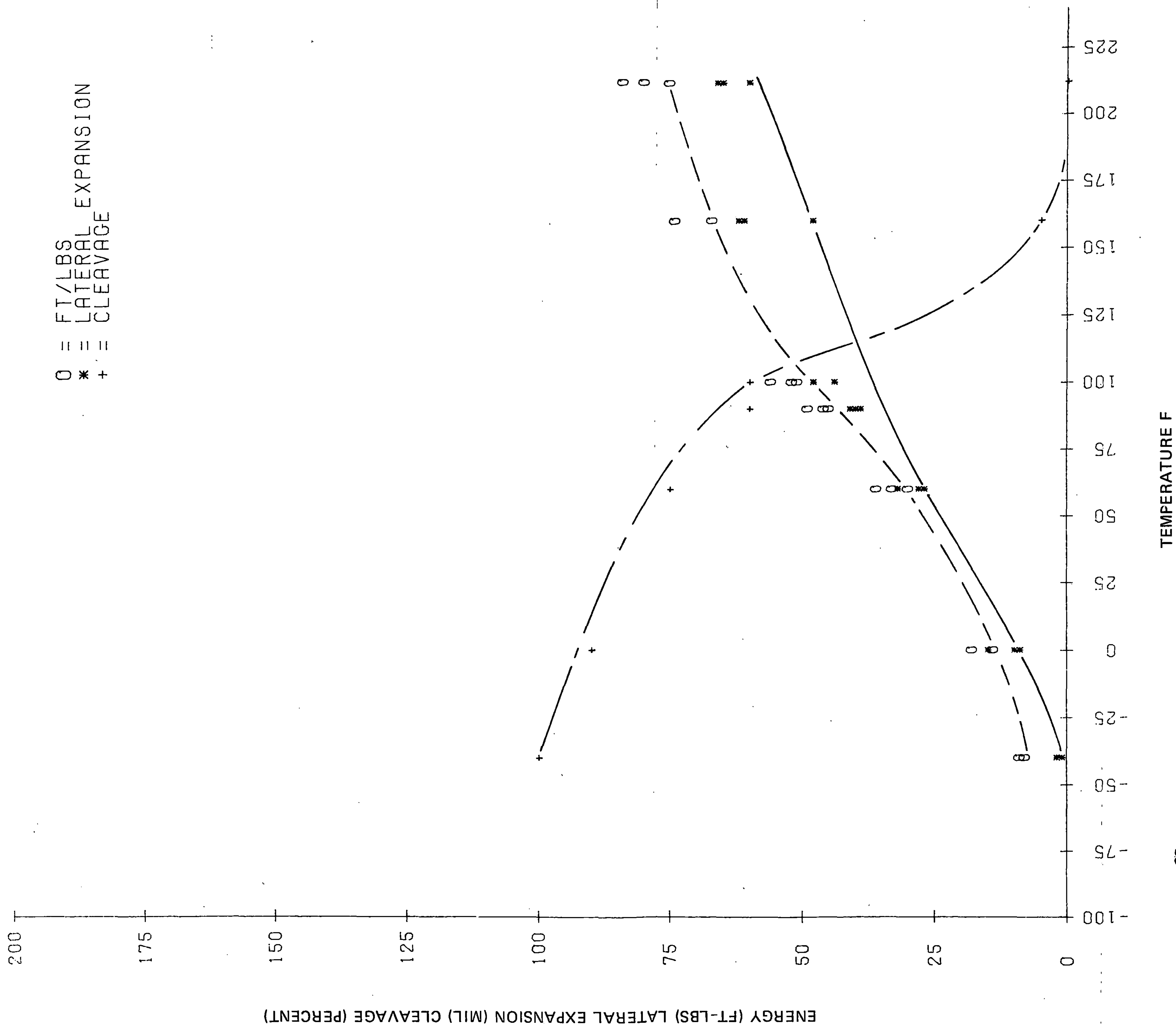
TI APERTURE CARD

8504150391-20

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
BOTTOM HEAD DOME
M-4112-1

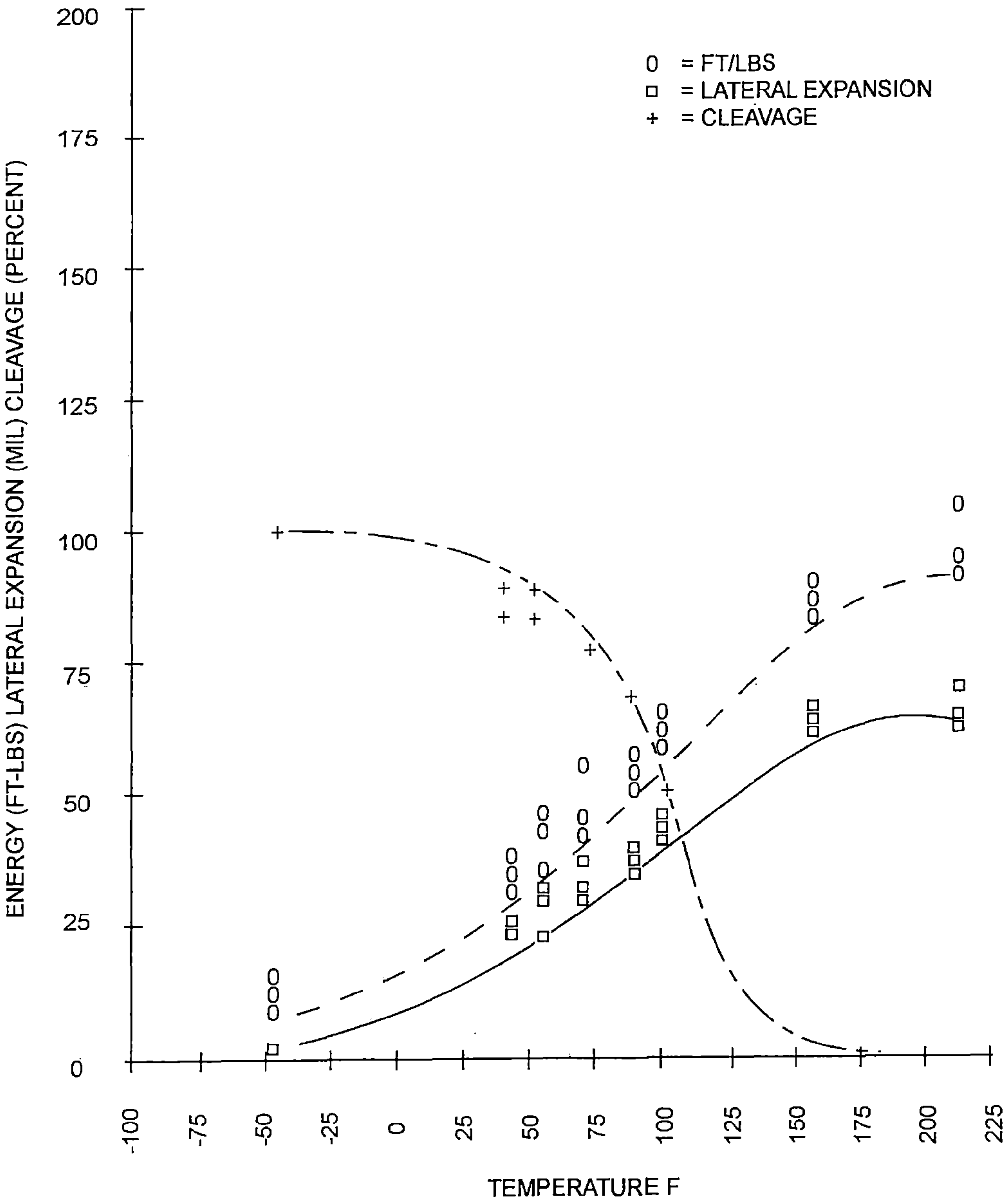
FIGURE 5.2-11



8504150391 -21

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
BOTTOM HEAD TORUS
M-4111-1
FIGURE 5.2-12

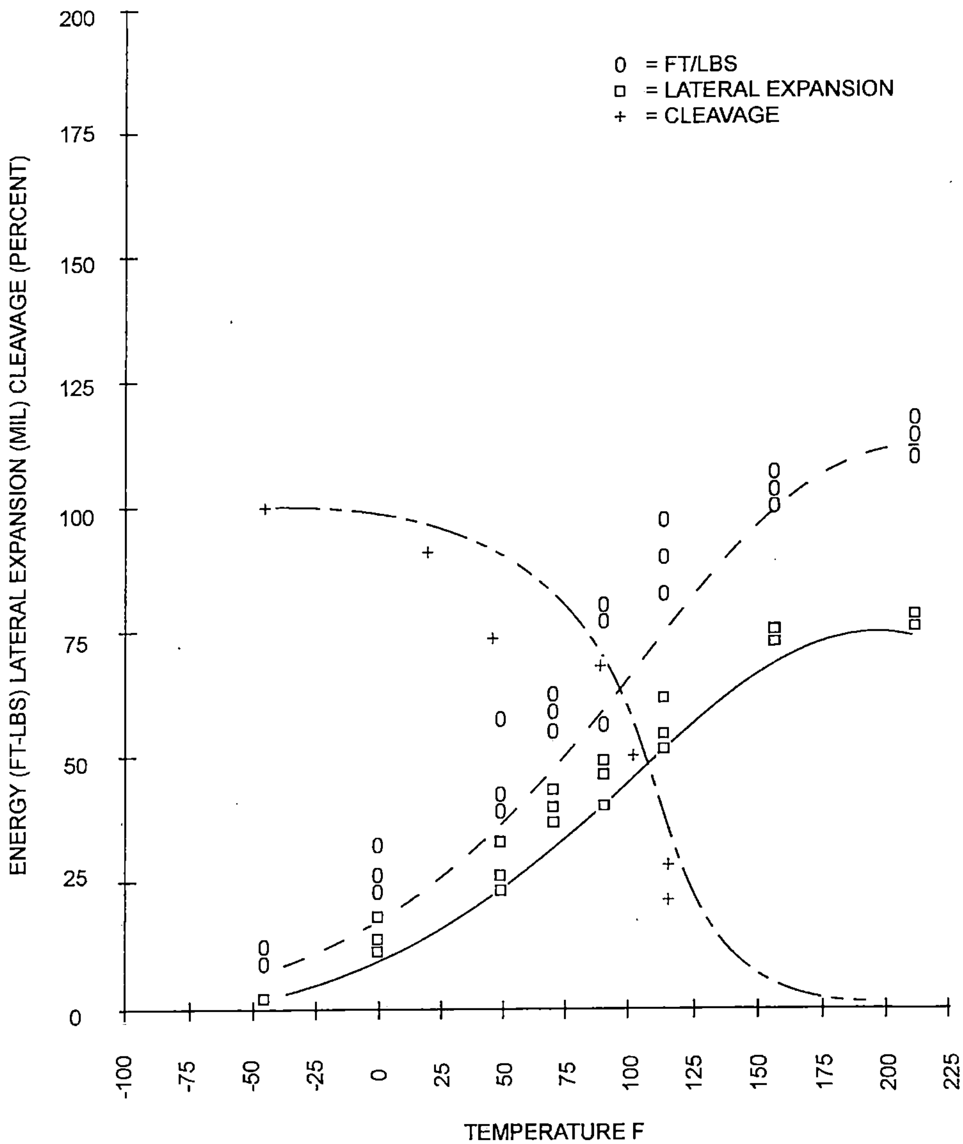


Note:
Historical informational only.
The RVCH has been replaced.

Amendment No. 18 (01/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
CLOSURE HEAD DOME
M-4110-1
FIGURE 5.2-13

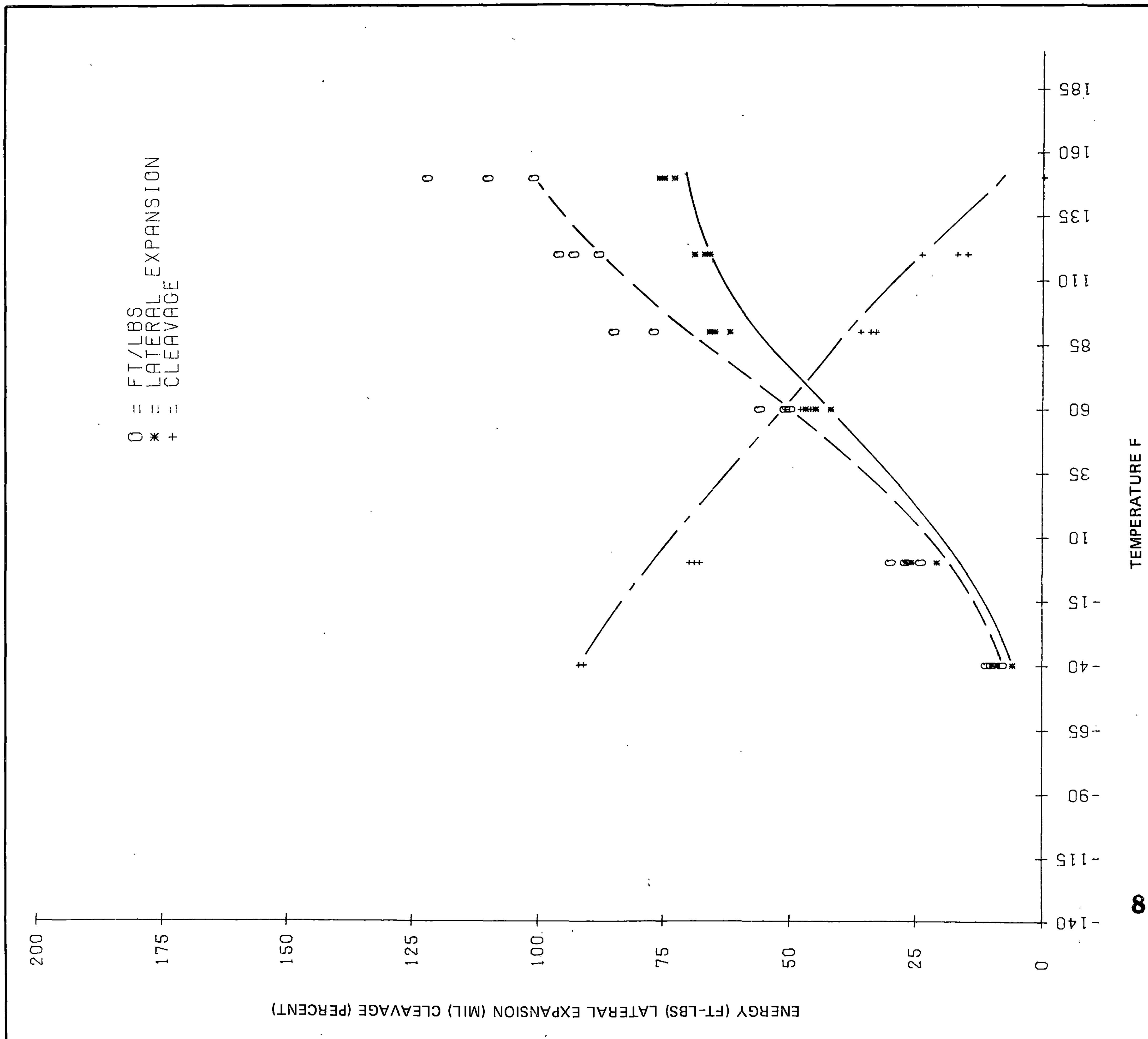


Note:
Historical informational only.
The RVCH has been replaced.

Amendment No. 18 (01/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
CLOSURE HEAD TORUS
M-4109-1
FIGURE 5.2-14



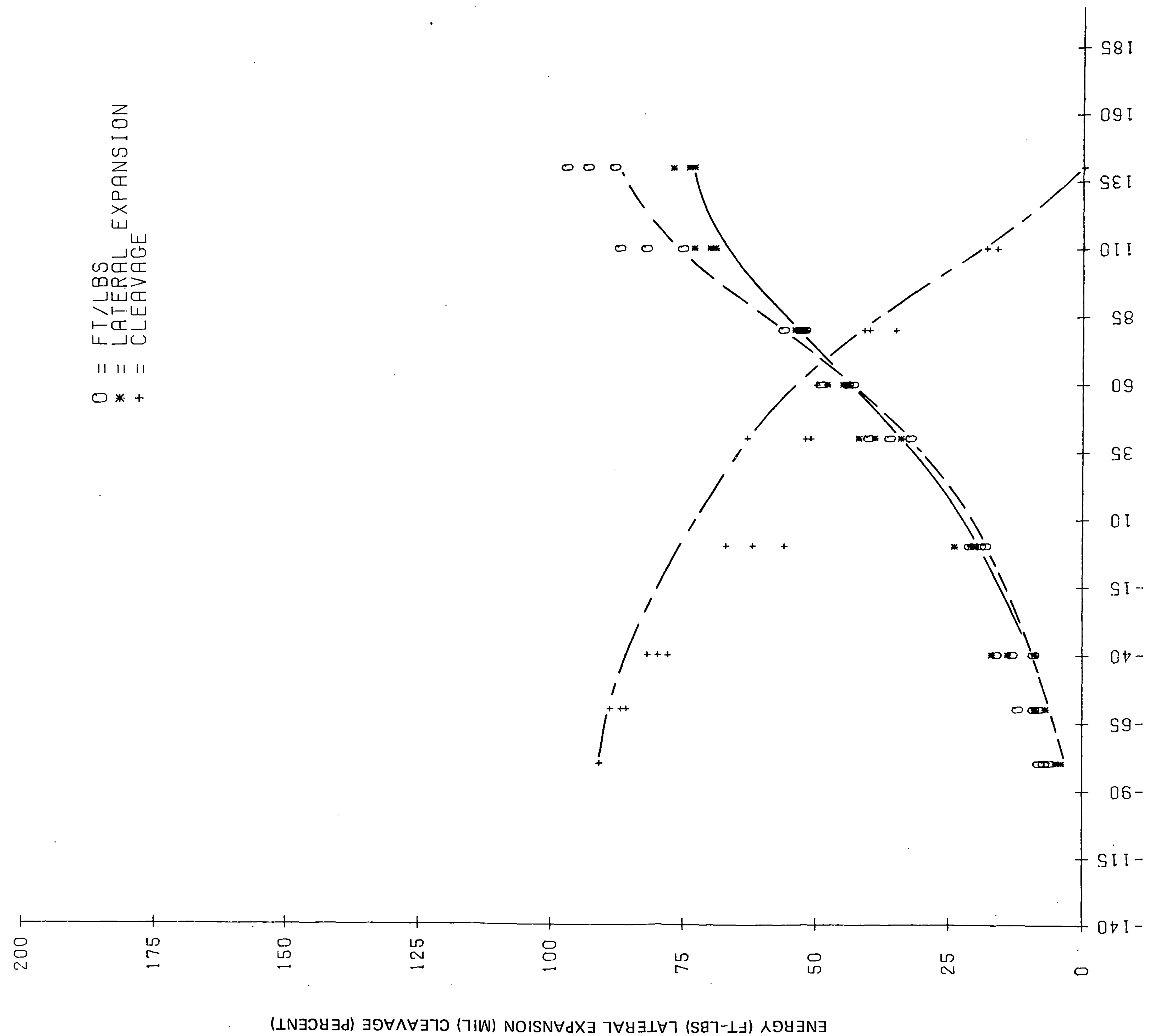
Also Available On Aperture Card

TI APERTURE CARD

8504150391 -24

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
OUTLET NOZZLE SAFE ENDS
M-4105-1 & 2
FIGURE 5.2-15



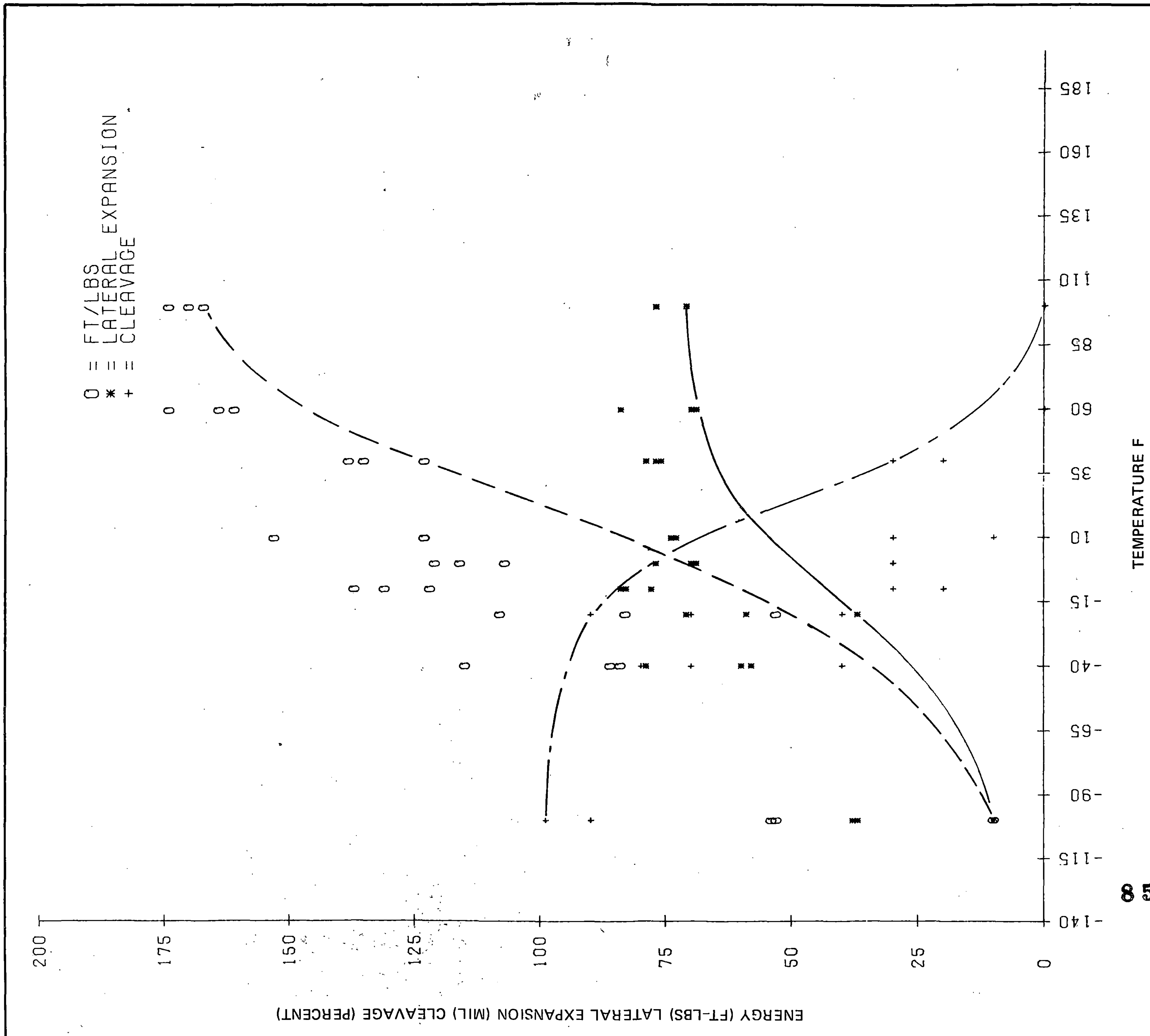
Also Available On Aperture Card

TI APERTURE CARD

8504150391 -25

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
INLET NOZZLE SAFE ENDS
M-4104-1 THRU 4
FIGURE 5.2-16



8504150391 -26

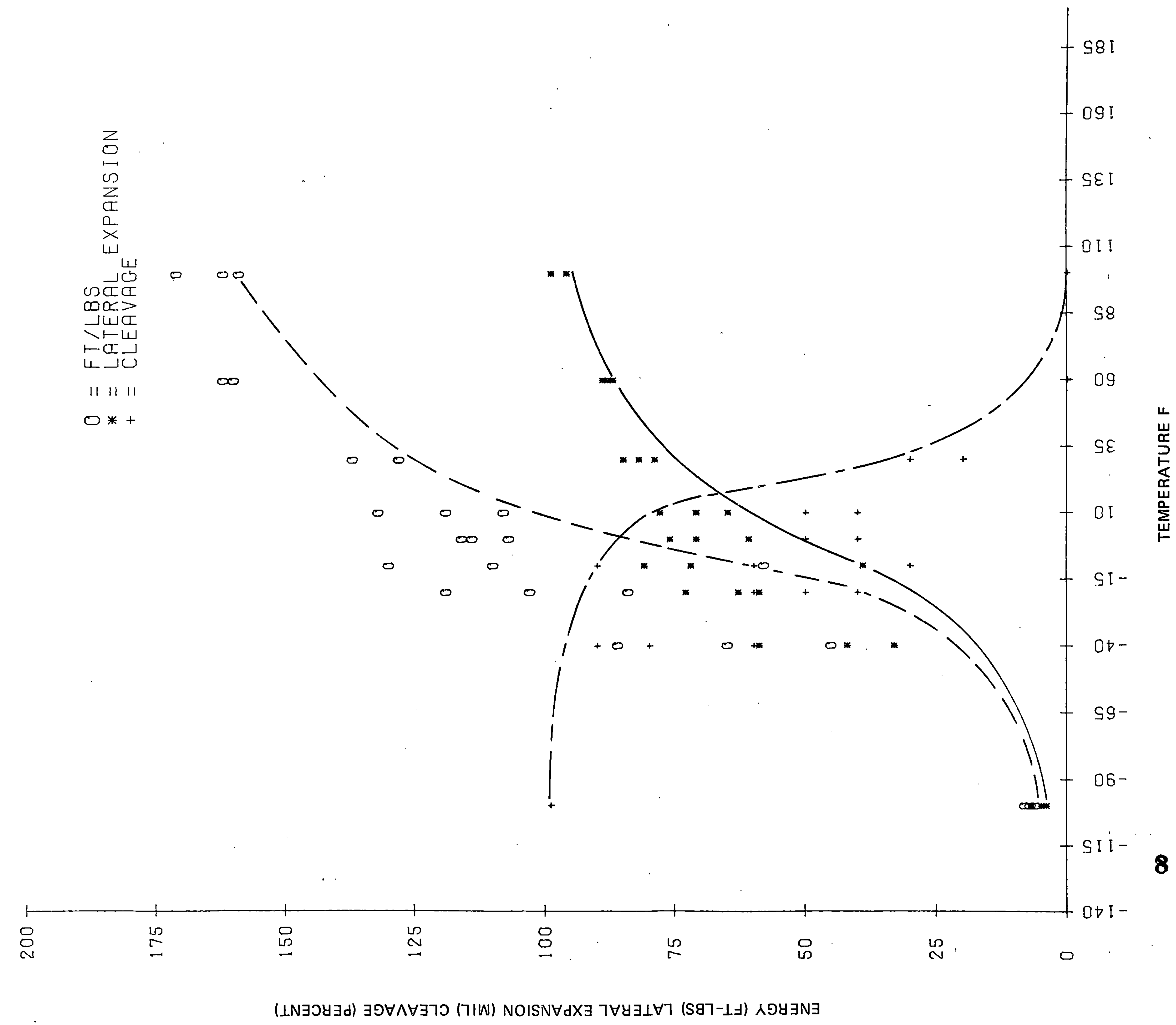
Also Available On Aperture Card

TI APERTURE CARD

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
OUTLET NOZZLE
M-4103-1

FIGURE 5.2-17



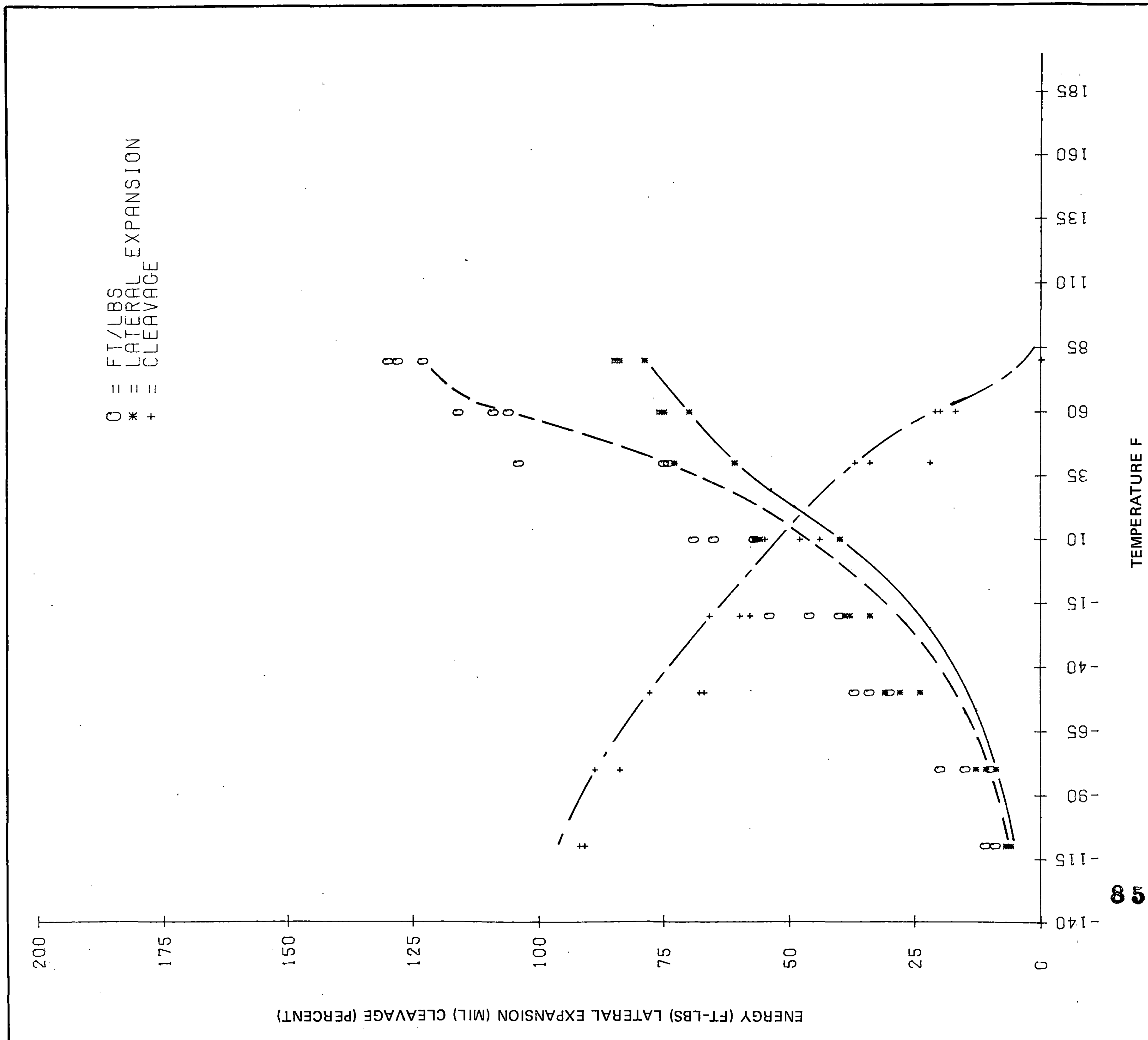
Also Available On Aperture Card

TI APERTURE CARD

8504150391-27

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

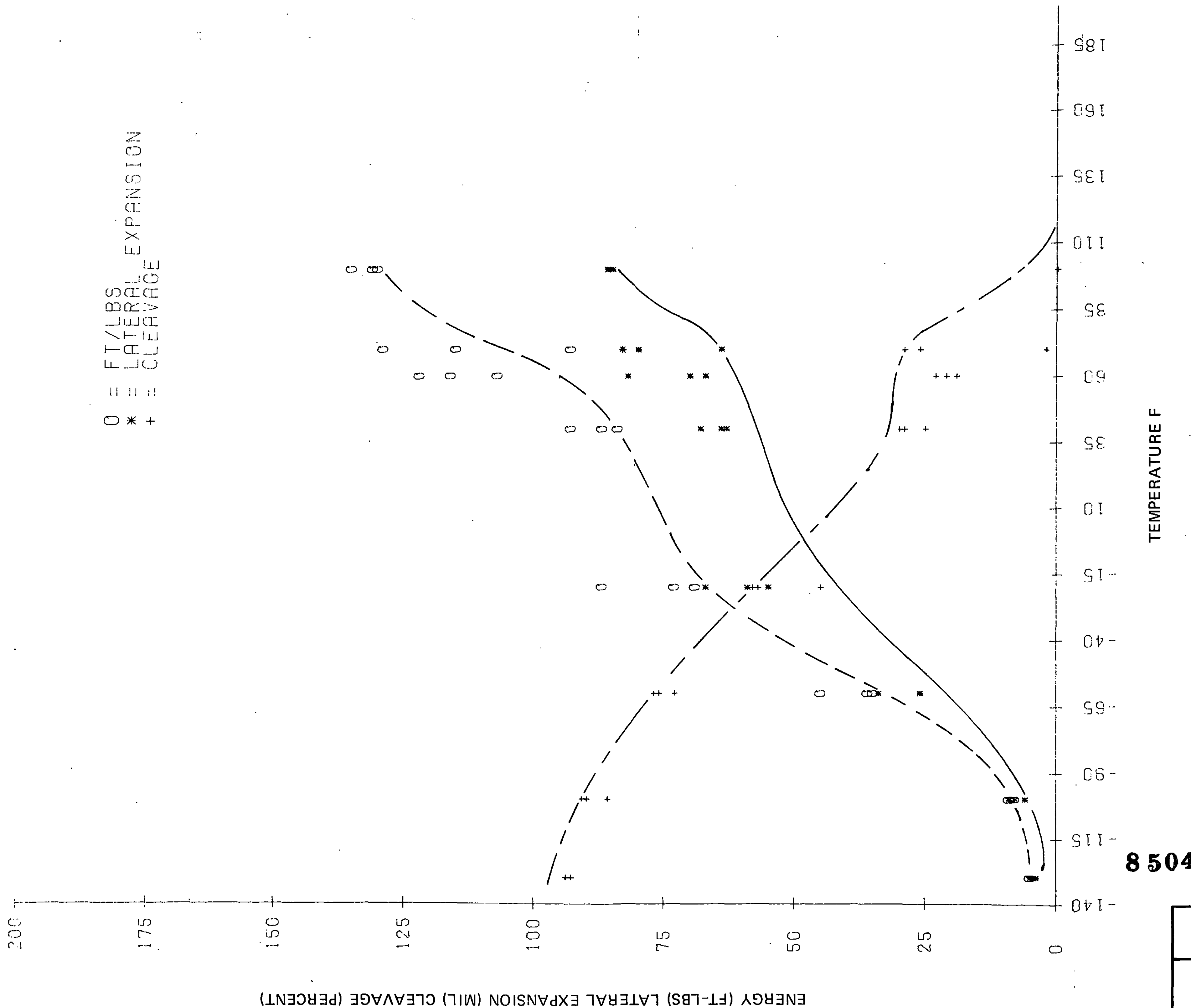
CHARPY TEST RESULTS
OUTLET NOZZLE
M-4103-2
FIGURE 5.2-18



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
INLET NOZZLE
M-4102-1

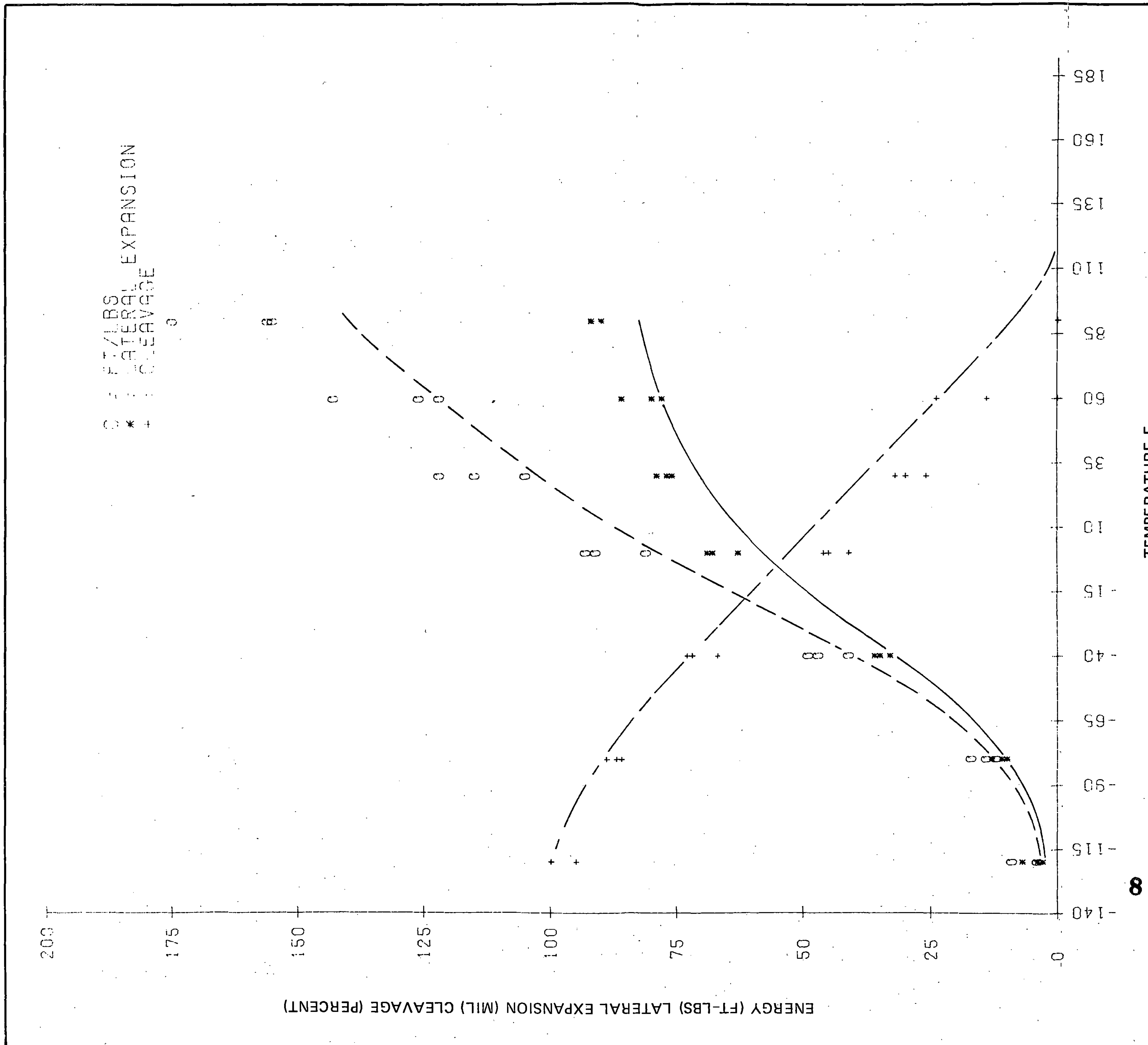
FIGURE 5.2-19



8504150391 -29

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
INLET NOZZLE
M-4102-2
FIGURE 5.2-20



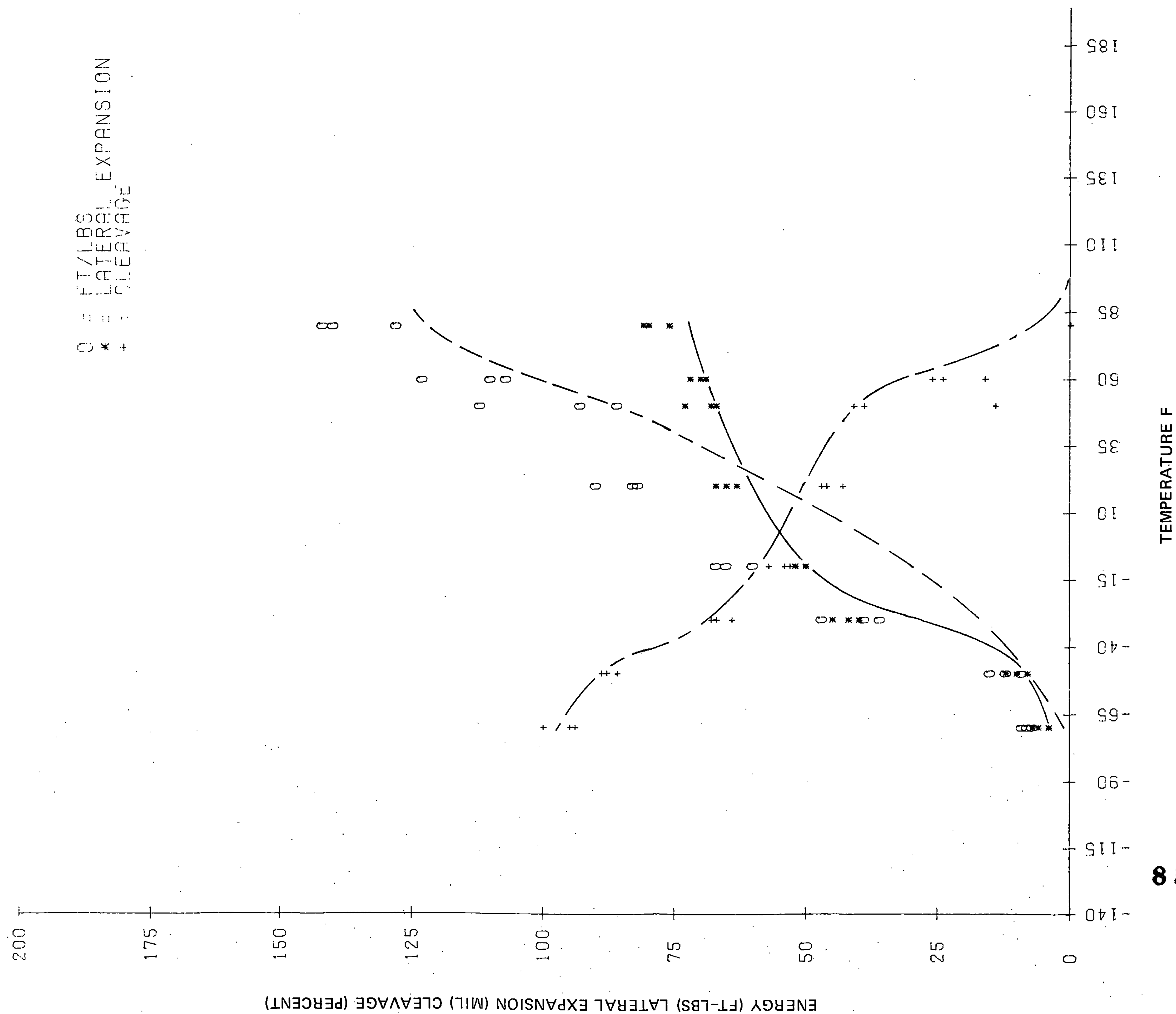
Also Available On
Aperture Card

TI
APERTURE
CARD

8504150391 -30

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
INLET NOZZLE
M-4102-3
FIGURE 5.2-21



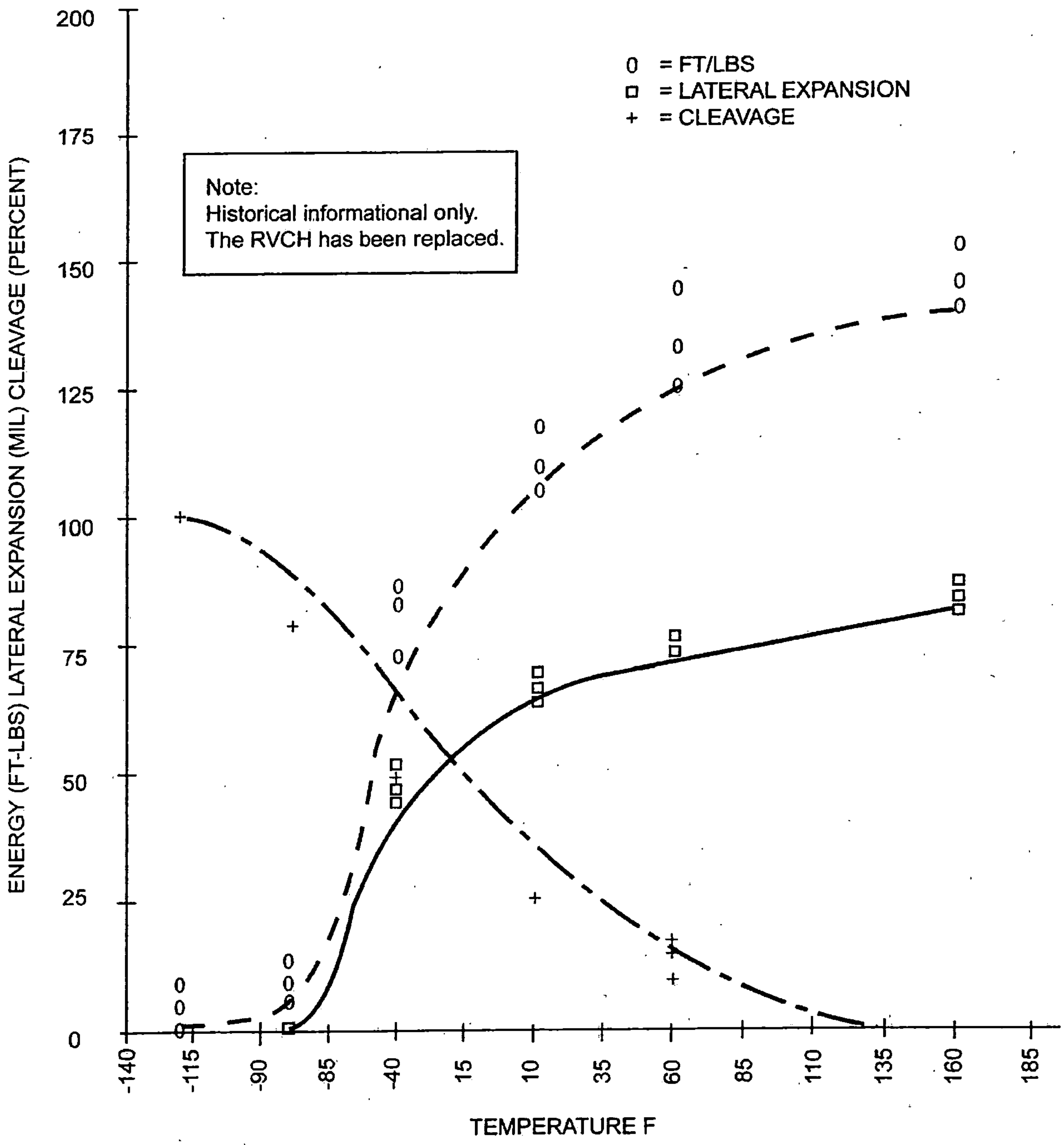
Also Available On Aperture Card

TI APERTURE CARD

8504150391 -31

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

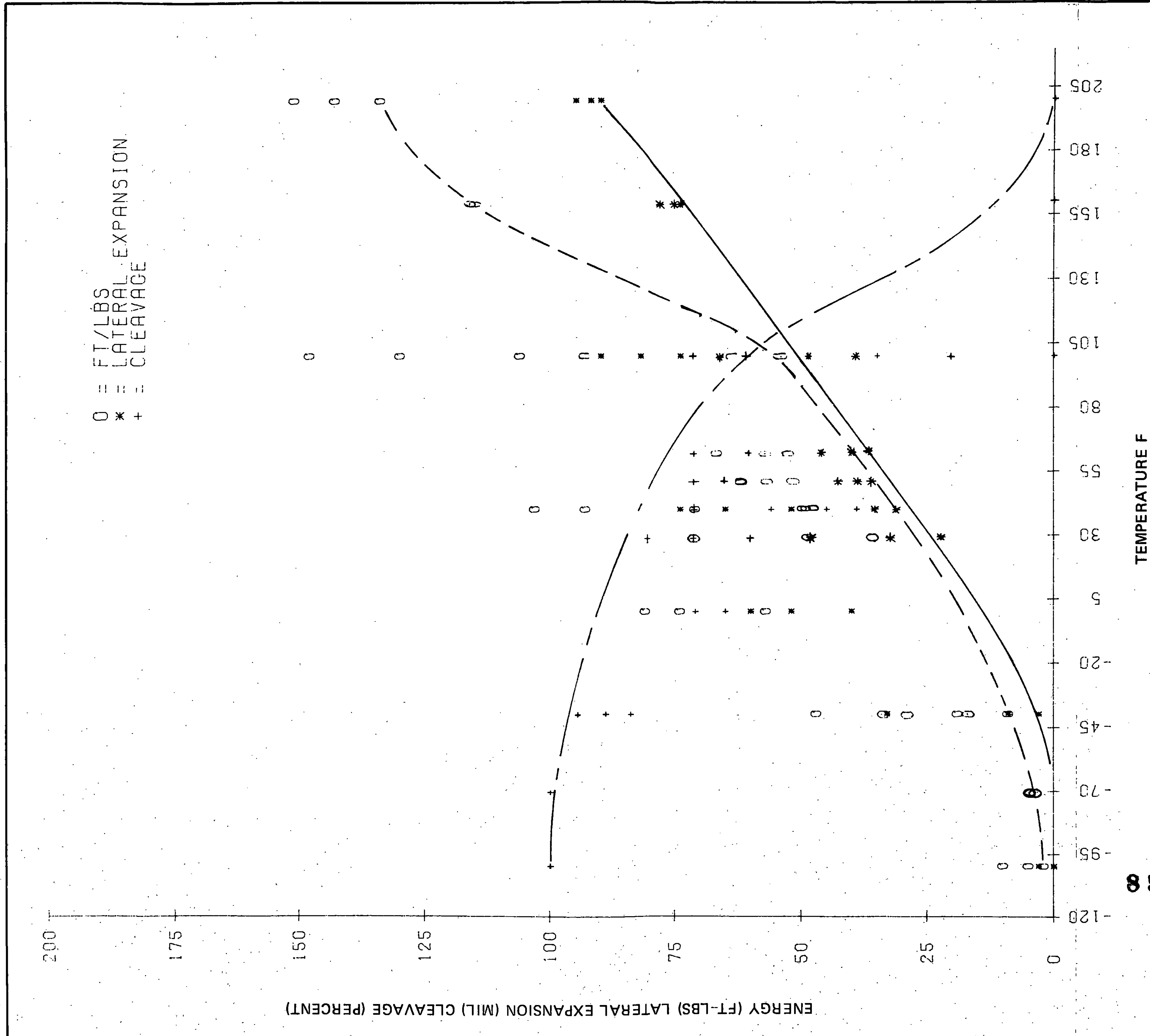
CHARPY TEST RESULTS
INLET NOZZLE
M-4102-4
FIGURE 5.2-22



Amendment No. 18 (01/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
CLOSURE HEAD FLANGE
M-4101-1
FIGURE 5.2-23



8504150391 -33

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

CHARPY TEST RESULTS
VESSEL FLANGE
M-602-1

FIGURE 5.2-23a

DELETED

Amendment No. 19 (06/09)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

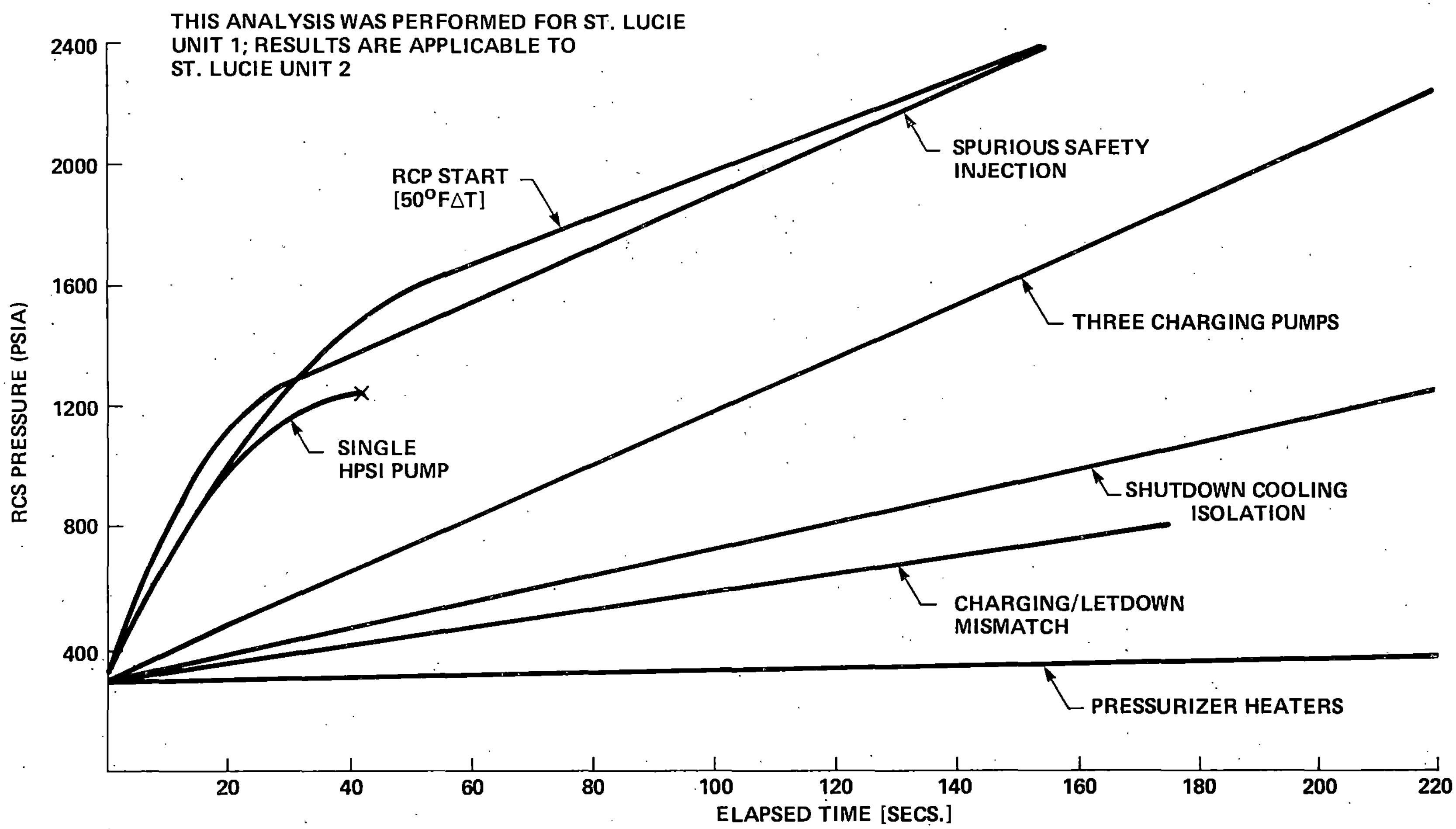
FIGURE 5.2-24

DELETED

Amendment No. 19 (06/09)

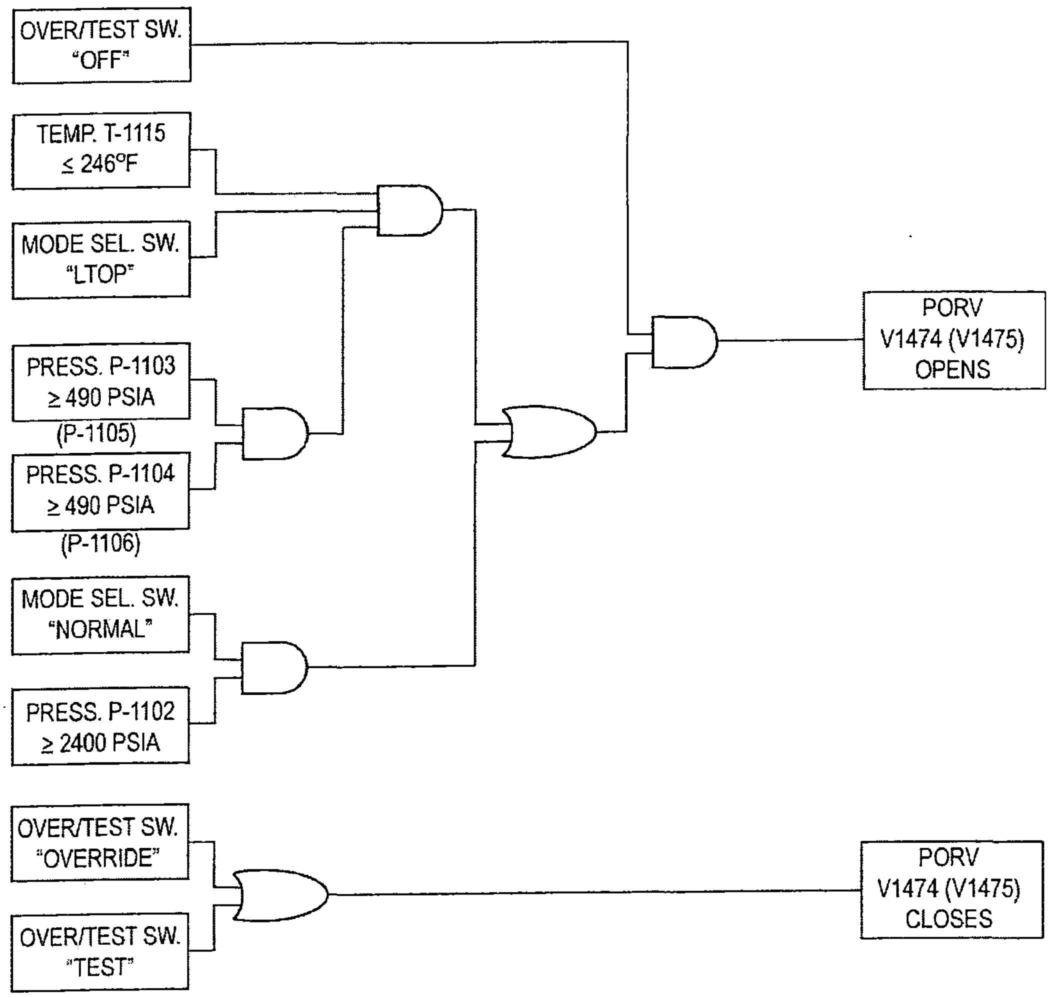
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FIGURE 5.2-25

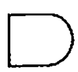


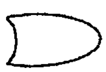
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

WATER-SOLID RCS OVERPRESSURIZATION TRANSIENTS
FIGURE 5.2–26



LEGEND:

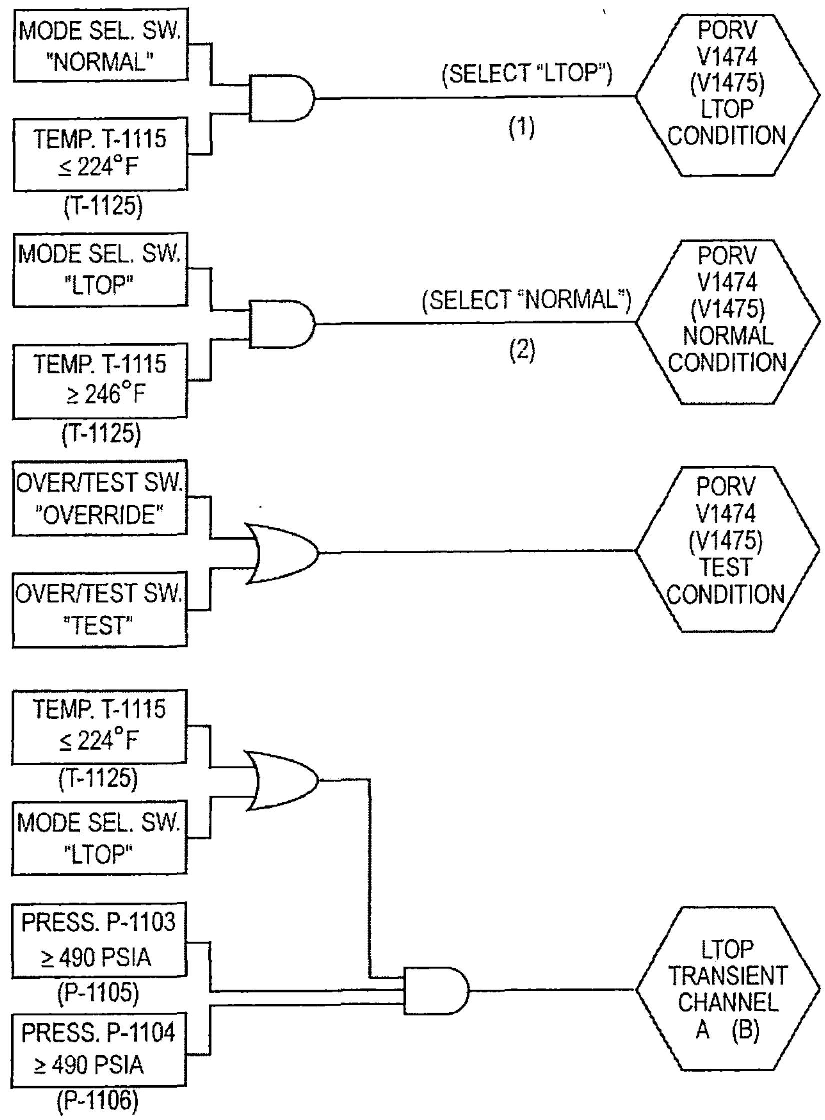
 "AND" GATE - ALL INPUTS ARE REQUIRED TO SATISFY GATE FOR LOGIC TO CONTINUE.

 "OR" GATE - EITHER INPUT IS REQUIRED TO SATISFY GATE FOR LOGIC TO CONTINUE.

AMENDMENT NO. 19 (06/09)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

PORV V1474 (V1475)
ACTUATION LOGIC
FIGURE 5.2-27



NOTE : PORVS V1475/1474 and channel A/B will have their own independent alarms.

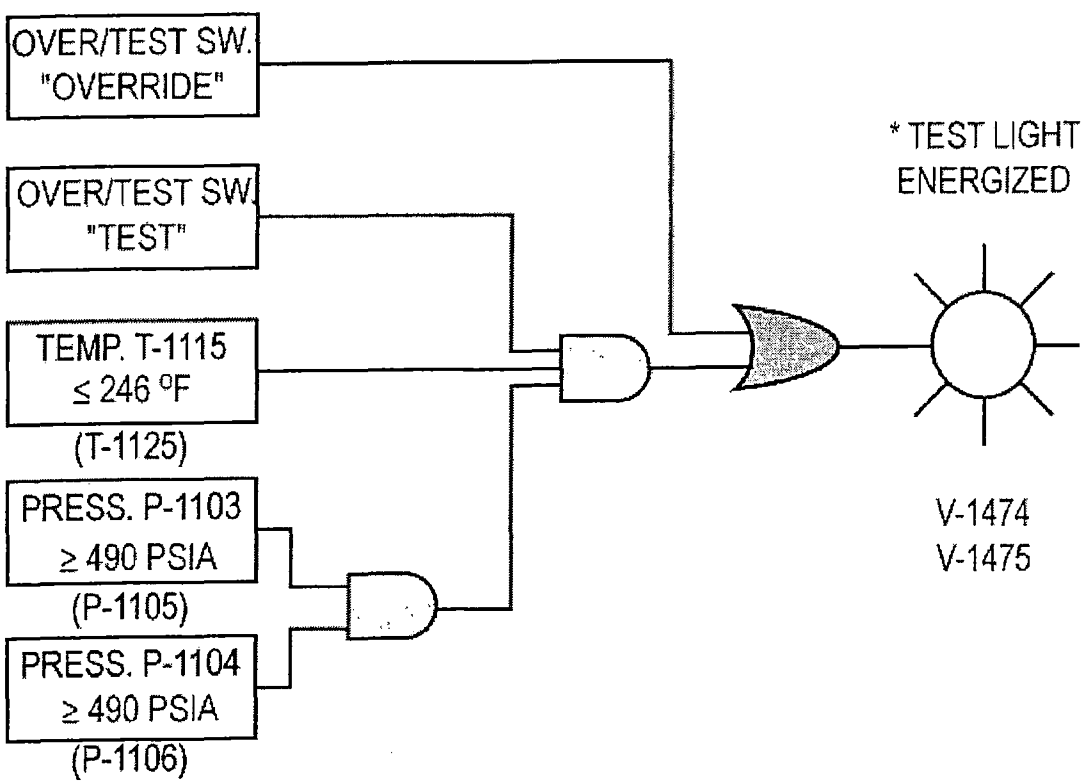
(1) This alarm will not clear unless the PORV mode select switch is in the "LTOP" position and the motor operated isolation valve of the associated PORV is "Open."

(2) This alarm will not reset unless the PORV mode select switch is in the "NORMAL" position and its associated isolation valve is "OPEN". Therefore one alarm will be lit during normal operation as long as one PORV remains isolated.

AMENDMENT NO. 19 (06/09)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

PORV ANNUNCIATORS
FIGURE 5.2-28



* LTOP OR NORMAL PORV CONTROL CIRCUITRY HAS TESTED SATISFACTORY. PORV UNDER TEST DOES NOT PHYSICALLY OPEN. EACH PORV WILL HAVE ITS OWN TEST LIGHT.

AMENDMENT NO. 19 (06/09)

FLORIDA POWER & LIGHT COMPANY **ST. LUCIE PLANT UNIT 2**
PORV CIRCUITRY TEST LOGIC **FIGURE 5.2-29**

APPENDIX 5.2A

OVERPRESSURE PROTECTION FOR ST. LUCIE UNIT 2 - PRESSURIZED WATER REACTOR

ABSTRACT

This Appendix documents the original sizing procedure and the EPU overpressure protection for the St. Lucie Unit 2 Reactor Coolant System and steam generators. The values presented for original sizing represent Cycle 1 conditions.

Overpressurization of the Reactor Coolant System and steam generators is precluded by means of pressurizer safety valves, main steam safety valves and the Reactor Protection System. Pressure relief capacity for the steam generators and Reactor Coolant System is conservatively sized to satisfy the overpressure requirements of the ASME Code, Section III. The safety valves in conjunction with the Reactor Protection System, are designed to provide overpressure protection for a loss-of-load incident with a delayed reactor trip.

This appendix documents the standard sizing procedure for both primary and secondary safety valves during the design phase. The ability for this equipment to provide overpressure protection from EPU operating conditions is also documented in this appendix. The analyses demonstrating the ability of these valves to provide adequate overpressure protection is documented in Section 15.2.

Because of the procurement lead times, the valves must be sized quite early in the design process since this information is required to prepare interface requirements and valve specifications. The type of analysis required by Section 5.2.2 of the USNRC Standard Review Plan can only be performed once the design has been finalized and the exact operating limits are known. The EPU overpressure protection analyses demonstrate the initial sizing methodology provides sufficient margin for the valves to continue to provide the required overpressure protection.

1. INTRODUCTION

Overpressure protection for the reactor, steam generators, and Reactor Coolant System is in accordance with the requirements set forth in the ASME Code, Section III. Overpressure protection is ensured by means of pressurizer safety valves, main steam safety valves, and the Reactor Protective System. Analysis of all reactor and steam plant transients causing pressure excursions is conducted. The worst case transient, loss-of-load, in conjunction with a delayed reactor trip is the design basis for the pressurizer safety valves. The pressurizer safety valves, main steam safety valves, and Reactor Protective System maintain the Reactor Coolant System below 110 percent of design pressure during worst case transients. The main steam safety valves are sized to pass a steam flow equivalent to a power level of 2750 Mwt, which is greater than the initial licensed power level of 2570 Mwt. Steam generator pressure is limited to less than 110 percent of steam generator design pressure during worst case transients.

2. ANALYSIS

2.1 METHOD

CE performed the original parametric study to determine the design basis incident for sizing the pressurizer safety valves. The design basis incident was a loss-of-load with a delayed reactor trip. The analysis was performed using digital computer codes which have been verified by transient data from operating plants and accurately models the thermal, hydraulic, and nuclear performances of the Reactor Coolant and steam system. The digital codes used in the transient analysis include reactor kinetics, thermal and hydraulic performance of the Reactor Coolant System, and the thermal and hydraulic performance of the steam generators. The computer simulation includes effects of reactor coolant pump performance, elevation heads, inertia of surge line water and friction drop in the surge line. Worst case initial conditions and nuclear parameters were assumed for the parametric analysis. The reactor was assumed to trip at a RCS pressure of 2420 psia, with the pressurizer safety valves assumed to lift at a pressure of 2525 psia, which is 25 psi above the system design pressure. During the analysis, the throat area associated with these valves was increased parametrically until the above design basis incident analysis indicates that a further increase in throat area does not result in a significant decrease in RCS peak pressure.

The EPU overpressure analysis uses the EPU Safety Analysis results in Chapter 15 to demonstrate the overpressure protection requirements set forth in ASME Code Section III are met with the current safety valves.

2.2 ASSUMPTIONS

The following are the assumptions used for the original (pre-EPU) valve designs:

a) At the onset of the loss-of-load transient, the Reactor Coolant and Main Steam Systems are at maximum rated output plus a two percent uncertainty. By choosing the highest possible power output, the heatup rate of the reactor coolant loop is maximized, hence the rate of pressurization is also maximized.

b) Moderator coefficient is 0.0. Analytical studies supported by core data show that the moderator coefficient can vary between 0.0 and -3.5×10^{-4} for various phases of core life. Therefore, a coefficient of 0.0 is chosen to maximize the power/pressure transient.

c) Doppler coefficient of -.8 x 10^{-5} ΔK/K/F is used in the loss-of-load analysis. By choosing a relatively small Doppler coefficient, the reduction in reactivity with increasing fuel temperature is minimized thereby, maximizing the rate of power rise. Actual operating coefficients can be expected to range from -1.4 x 10^{-5} at zero power to -1. x 10^{-5} ΔK/K/F at full power.

d) No credit is taken for letdown, pressurizer spray, secondary bypass, or feedwater addition after turbine trip in the loss-of-load analysis. Letdown and pressurizer spray both act to reduce reactor coolant system pressure. By not taking credit for these systems, the rate of pressurization and peak pressure are increased. By not taking credit for the addition of feedwater, the steam generator secondary inventory is depleted at a faster rate. This in turn reduces the capability of the steam generator to remove heat from the reactor coolant loop, thereby maximizing the rate of Reactor Coolant System pressurization.

e) The analysis reflects consideration of plant instrumentation error and safety valve setpoint errors. For example, all safety valves are assumed to open at their maximum popping pressure. This extends the period of time before energy can be removed from the system. The reactor trip setpoint errors are always assumed to act in such a manner that they delay reactor trip, again resulting in maximum pressurization.

f) Pressurizer pressure at the onset of the incident is 2200 psi. By using the lower limit of the normal plant operating pressure, the time required to trip the plant on high pressure is increased.

The EPU analysis maintains the same assumptions with the following adjustments to account for EPU conditions:

1) The power uncertainty has been revised from 2% to 0.3%. The Doppler coefficient has been revised from -.8 x 10^{-5} ΔK/K/F to -.9 x 10^{-5} ΔK/K/F.

2) The initial pressurizer pressure is 2180 psia. The secondary side peak pressure analysis conservatively accounts for pressurizer spray to delay the time to the high RCS pressure reactor trip.

3) The Loss of Condenser Vacuum case analyzed is consistent with SRP Section 5.2.2 in that the trip signal generated is the second reactor trip signal (first safety grade trip). An additional case was performed that considers a reactor trip on the third reactor trip signal (second safety grade trip).

2.2.1 MAIN STEAM SAFETY VALVE SIZING

The discharge piping serving the main steam safety valves is designed to accommodate rated relief capacity without imposing unacceptable backpressure on the main steam safety valves.

The main steam safety valves were sized to pass a steam flow equivalent to a power level of 2750 Mwt. This limited steam generator pressure to less than 110 percent of steam generator design pressure during worst case transients. The EPU secondary side peak pressure Safety Analysis demonstrates the steam generator pressure will remain less than 110 percent of the steam generator design pressure with an initial power level of 3030 MWt (3020 MWt plus 0.3%). The main steam safety valves consist of two banks of valves with staggered set pressures. The valves are spring loaded type safety valves procured in accordance with ASME Code, Section III.

Figure 5.2A-1 depicts the pre-EPU steam generator pressure transient for this worst case loss-of-load incident. As can be seen on Figure 5.2A-1, the steam generator pressure remained below 110 percent of design pressure during the incident. Figure 5.2A-5 is the steam generator pressure transient for the EPU conditions. Figure 5.2A-5 shows the pressure remains below 110% of the steam generator design pressure.

2.2.2 PRESSURIZER SAFETY VALVE SIZING

The quench tank and pressurizer safety valve discharge piping are sized to preclude unacceptable pressure drops and backpressure which would adversely affect valve operations. The discharge piping pressure drops and backpressure have been evaluated for EPU conditions and found to be acceptable.

Pressurizer safety valve backpressure is limited by the design pressure of the valve bellows. These bellows prevent any accumulated backpressure from being imposed on the valve spring, thus allowing valve operation at its design setpoint rather than at its setpoint plus backpressure.

The design basis incident for sizing the pressurizer safety valves is a loss of turbine generator load during which the reactor trip on turbine trip is not operable. The reactor can be tripped on one of the following:

a) High pressurizer pressure trip;

b) Steam generator low level trip;

c) High reactor power trip;

d) Manual trip

The first of these to occur is the high pressurizer trip.

If the high pressure trip were to become inoperative, other reactor trips would proceed to shut the reactor down as their setpoints are exceeded.

The original pressurizer valve sizing was based on a series of loss-of-load studies run with various sizes of pressurizer safety valves. As can be seen on Figure 5.2A-2 after the pressurizer safety valve capacity increases to a certain size, additional increase in capacity has negligible effect in reducing the maximum system pressure experiences during the loss-of-load transient. The pressurizer safety valves are chosen so as to minimize the maximum pressure experienced during the loss-of-load transient. The minimum specified pressurizer safety valve capacity is identified on Figure 5.2A-2.

Figures 5.2A-1, 5.2A-3, and 5.2A-4 present curves of steam generator pressure, maximum Reactor Coolant System pressure and core power versus time for the worst case loss of turbine generator load used to evaluate the pre-EPU plant response. As can be seen on Figures 5.2A-1 and 5.2A-3 the maximum steam generator pressure and reactor coolant loop pressures remain below 110 percent of design during this worst case transient. Figures 5.2A-5 and 5.2A-6 present the EPU peak pressure analyses for the loss of load cases. Figures 5.2A-5 and 5.2A-6 show that the maximum steam generator pressure and the RCS maximum pressure remain below 110% of design pressure.

Figure 5.2A-1 shows the first and second banks of main steam safety valves open at approximately 5.6 and 8.8 seconds, respectively. The main steam safety valves remove energy from the Reactor Coolant System and thus mitigate the pressure surge. The pressurizer safety valves are conservatively assumed to open at one percent above the normal Reactor Coolant System design pressure. Figure 5.2A-3 shows that the pressurizer safety valves open 7.9 seconds after the initiation of the upset condition.

2.2.3 ACCEPTABILITY OF SAFETY VALVE BLOWDOWN

An evaluation[1] of the EPRI test results for the St. Lucie Unit 2 pressurizer safety valve showed that the valve ring adjustments which assure stable valve operation resulted in blowdowns of approximately 10 percent. An analysis was performed, as outlined below, to demonstrate that the extended blowdown would not adversely affect overpressure protection or plant operation, per ASME Code requirements.

An extended blowdown of the safety valves could result in swelling of the pressurizer liquid level due to flashing and possible liquid carryover through the safety valves. Since the safety valve design specification specifies dry saturated steam flow conditions, it is desirable to show that these conditions are maintained during the extended blowdown. It is also desirable to verify that the RCS remains in a subcooled condition in order that steam bubble formation in the RCS is precluded.

A computer analysis was performed of the loss-of-load event with delayed reactor trip, similar to that used in safety valve sizing, except that a conservative 20 percent safety valve blowdown and initial conditions biased to maximize pressurizer liquid level were assumed. The purpose of this analysis was to determine the pressurizer liquid level response and the RCS subcooling under these conservative conditions. In order to introduce additional conservatism, into the results, an additive adjustment was made to the computer-calculated pressurizer levels on the basis of a very conservative pressurizer model. This model assumed that the initial saturated pressurizer liquid did not mix with the cooler insurge liquid, that the initial liquid remained in equilibrium with the pressurizer steam space, and that the steam which flashed during blowdown remained dispersed in the liquid phase and caused the liquid level to swell. The adjusted pressurizer water level vs. time curve showed a maximum of 95.5 percent (expressed as the percentage of the distance from the lower level nozzle to the upper level nozzle; corresponds to 1395 ft^3), below the safety valve nozzle elevation of 107 percent, so that dry saturated steam flow to the safety valves is assured throughout the blowdown. The computer analysis also showed that adequate subcooling was maintained in the RCS during the blowdown, so that steam bubble formation is precluded.

In addition, the licensing safety analyses of the St. Lucie Unit 2 Chapter 15 pressurization events were re-evaluated to determine the impact of assuming a 15 percent blowdown for the pressurizer safety valves in lieu of the five percent generally assumed. The evaluation indicated that, for the FWLB event analysis, which produces the greatest increase in pressurizer level, the increased blowdown would not result in the pressurizer liquid level reaching the safety valve nozzle elevation and thus normal safety valve operation would be assured. It is therefore concluded that a 15 percent safety valve blowdown does not adversely affect the conclusions of the St. Lucie Unit 2 Chapter 15 safety analyses.

In summary, the evaluation of the 10 percent blowdown of the St. Lucie Unit 2 pressurizer safety valves shows that plant overpressure protection is not adversely impacted and that the conclusions of the St. Lucie Unit 2 Chapter 15 safety analyses are not changed.

PCM 96139M replaced the relief valves with a more rigid forged steel valve body design which has an actual blowdown of 4%. This 10% blowdown evaluation is not relevant to EPU operations.

3. CONCLUSIONS

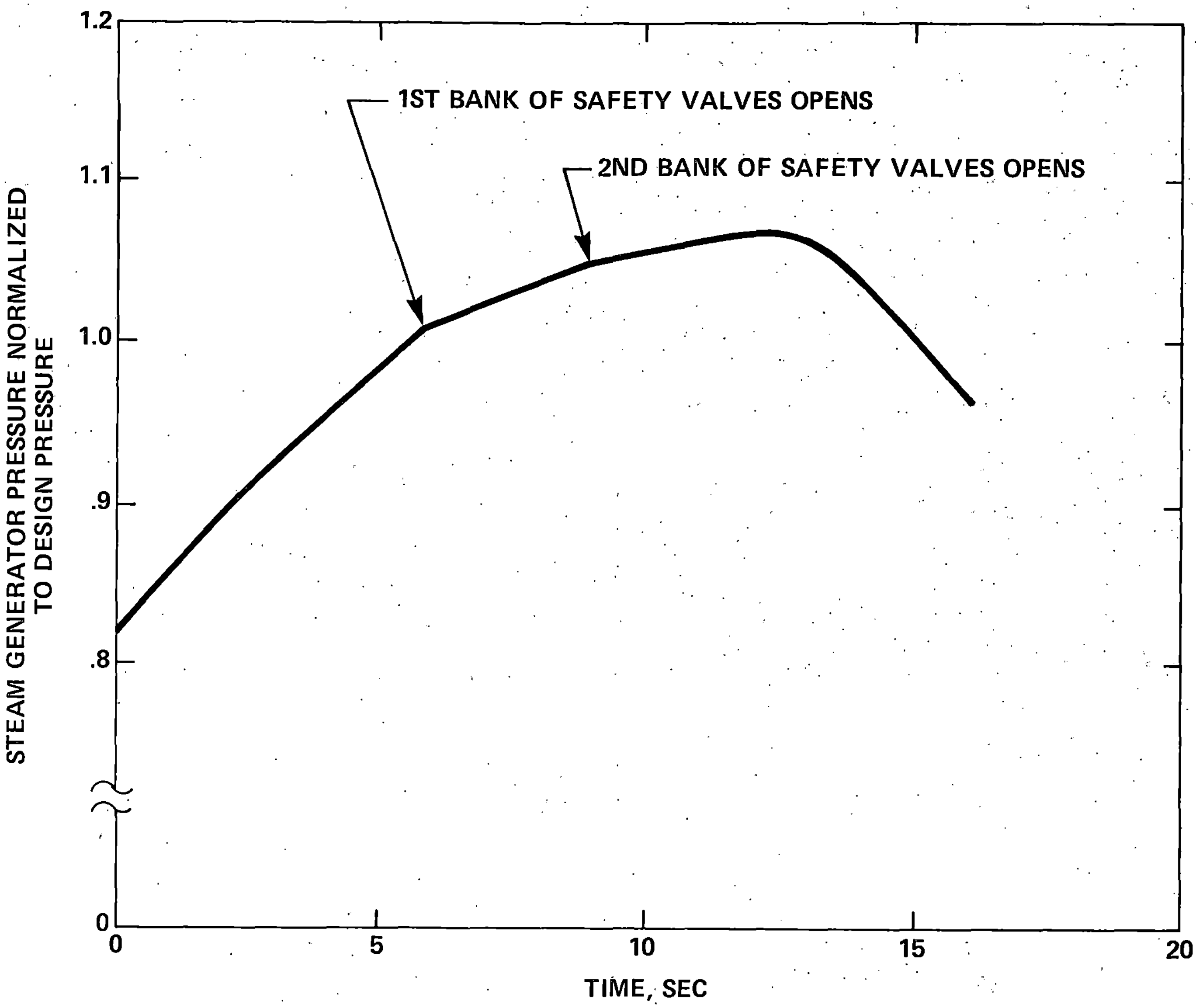
The steam generators and RCS are protected from overpressurization in accordance with the guidelines set forth in the ASME Boiler and Pressure Vessel Code, Section III. The maximum RCS pressure is shown as a function of time on Figure 5.2A-6 for the loss of turbine-generator load. As can be seen on Figure 5.2A-6, the maximum pressure remains below 110 percent of design pressure during this worst case transient. Figure 5.2A-5 depicts the steam generator pressure transient for this worst case loss-of-load accident. As can be seen on Figure 5.2A-5, the steam generator pressure also remains below 110 percent of design pressure during the incident. Both primary and secondary safety valves satisfy the requirements per Subsection NB7421 of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, regarding the required number and capacity of pressure relief devices. Also, in accordance with code requirements, the 10 percent blowdown of the St. Lucie 2 pressurizer safety valves was shown to be acceptable.

Furthermore, the analysis of a complete loss-of-load incident is described in Section 15.2. In the event that a complete loss-of-load occurs without a simultaneous reactor trip, this analysis reveals that the protection provided by the high pressurizer pressure trip, pressurizer safety valves and main steam safety valves is sufficient to assure that the integrity of the RCS and Main Steam System is maintained.

The analysis in the above section was repeated without taking credit for reactor trip on high pressurizer pressure (the first safety grade trip signal). The resulting primary-side and secondary-side pressures are within the acceptance criteria of Standard Review Plan Section 5.2.2.

REFERENCE: SECTION 5.2A

(1) Summary Report on the Operability of Pressurizer Safety Valves in CE Designed Plants, CEN-227, Combustion Engineering, Inc., December 1982.

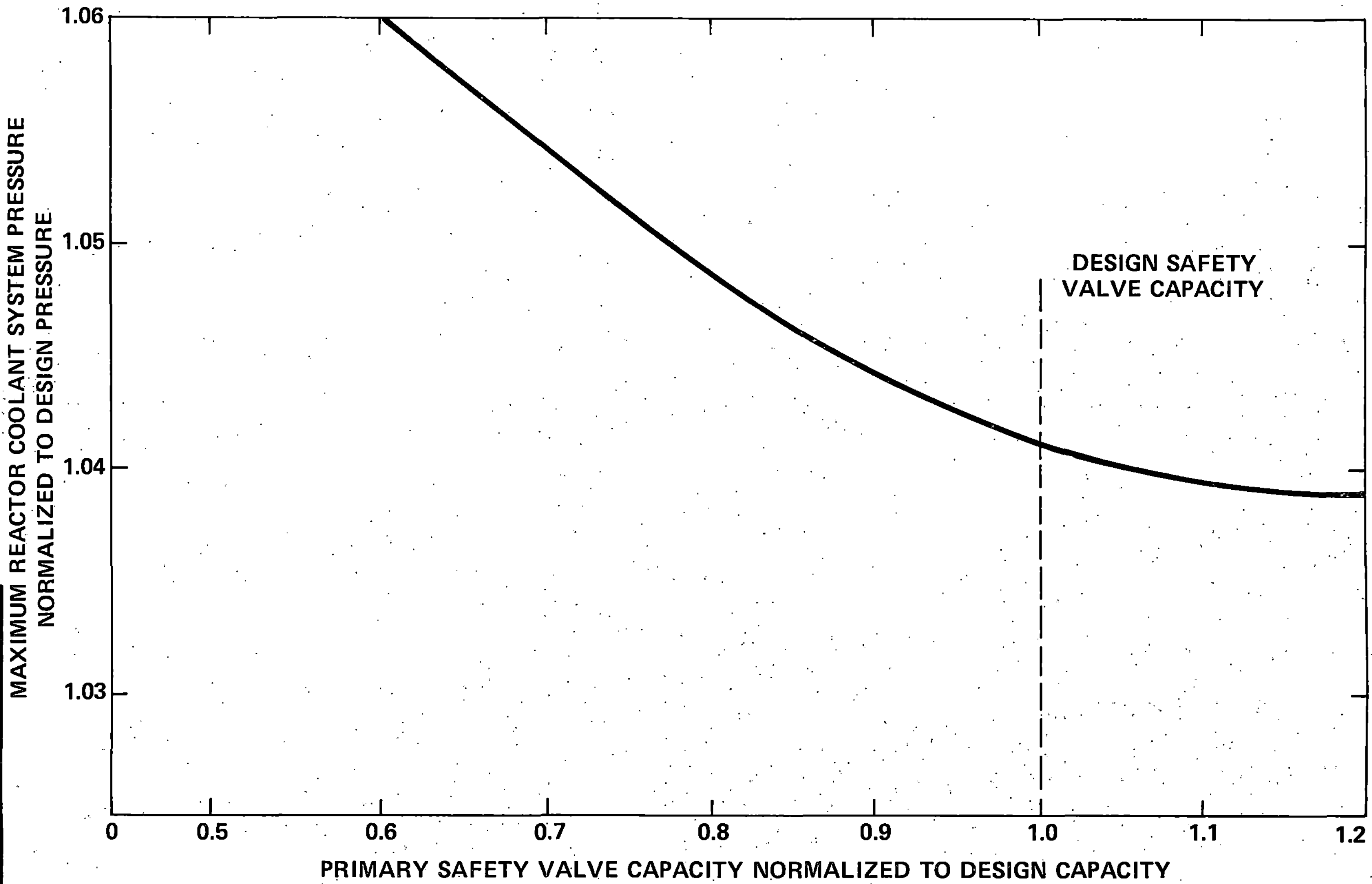


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 2

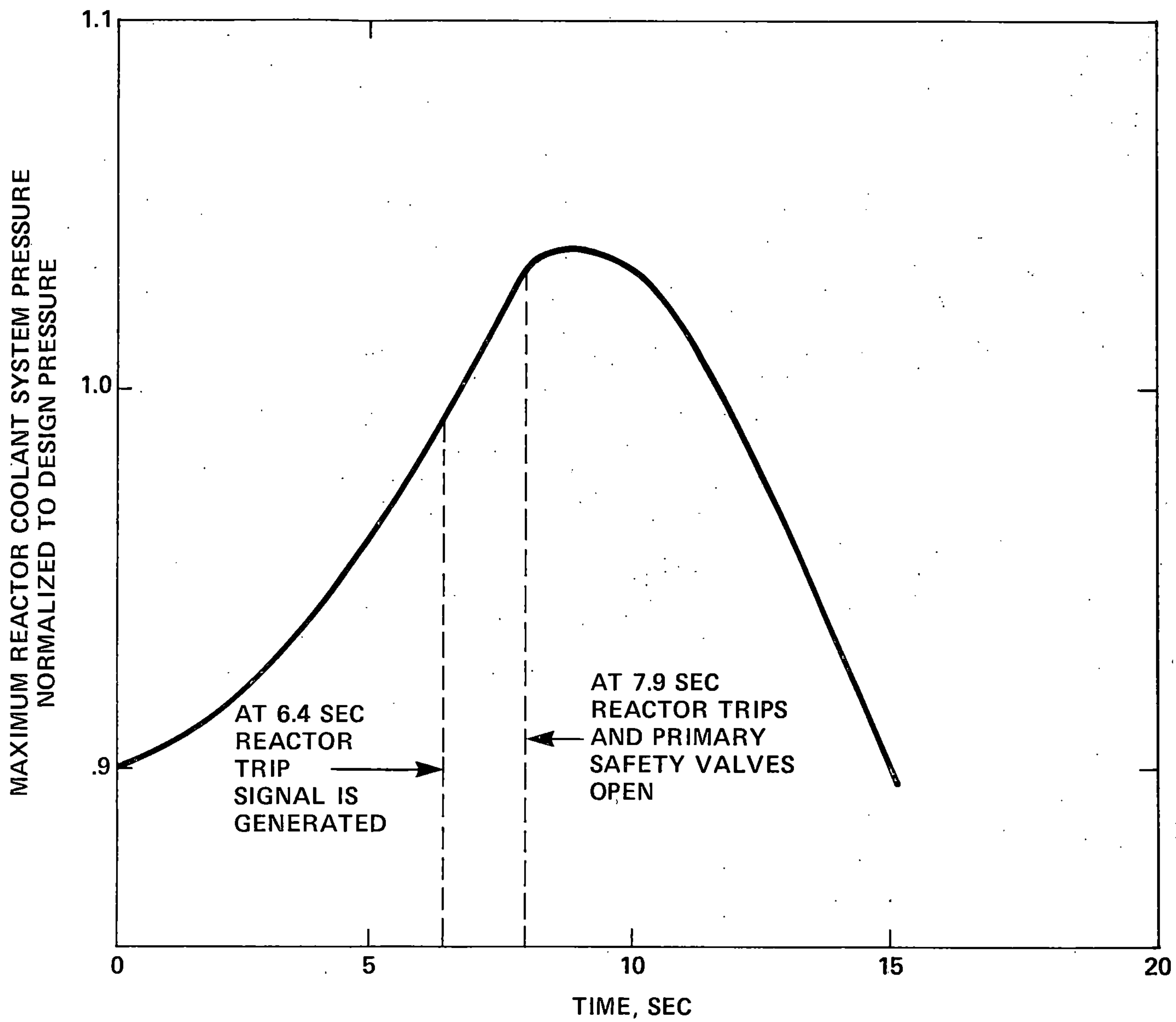
STEAM GEN. PRESSURE COMPLETE LOSS OF TURBINE - GENERATOR LOAD WITH DELAYED REACTOR TRIP

FIGURE 5.2A–1



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

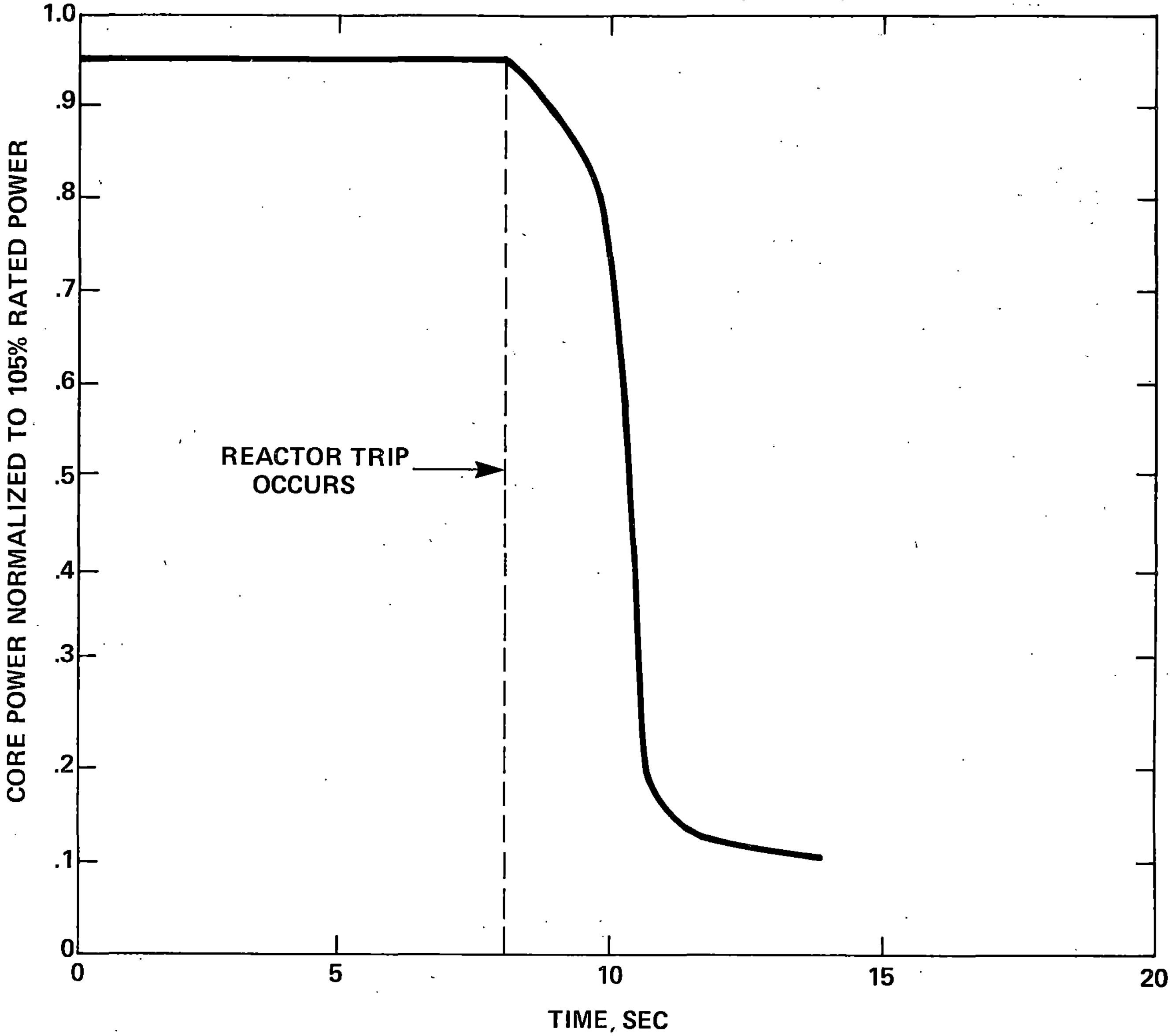
OPTIMIZED SAFETY VALVE SIZING
FIGURE 5.2A–2



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MAX. REACTOR COOLANT SYST. PRESSURE VS. TIME FOR WORST CASE LOSS OF LOAD INCIDENT WITH DELAYED REACTOR TRIP

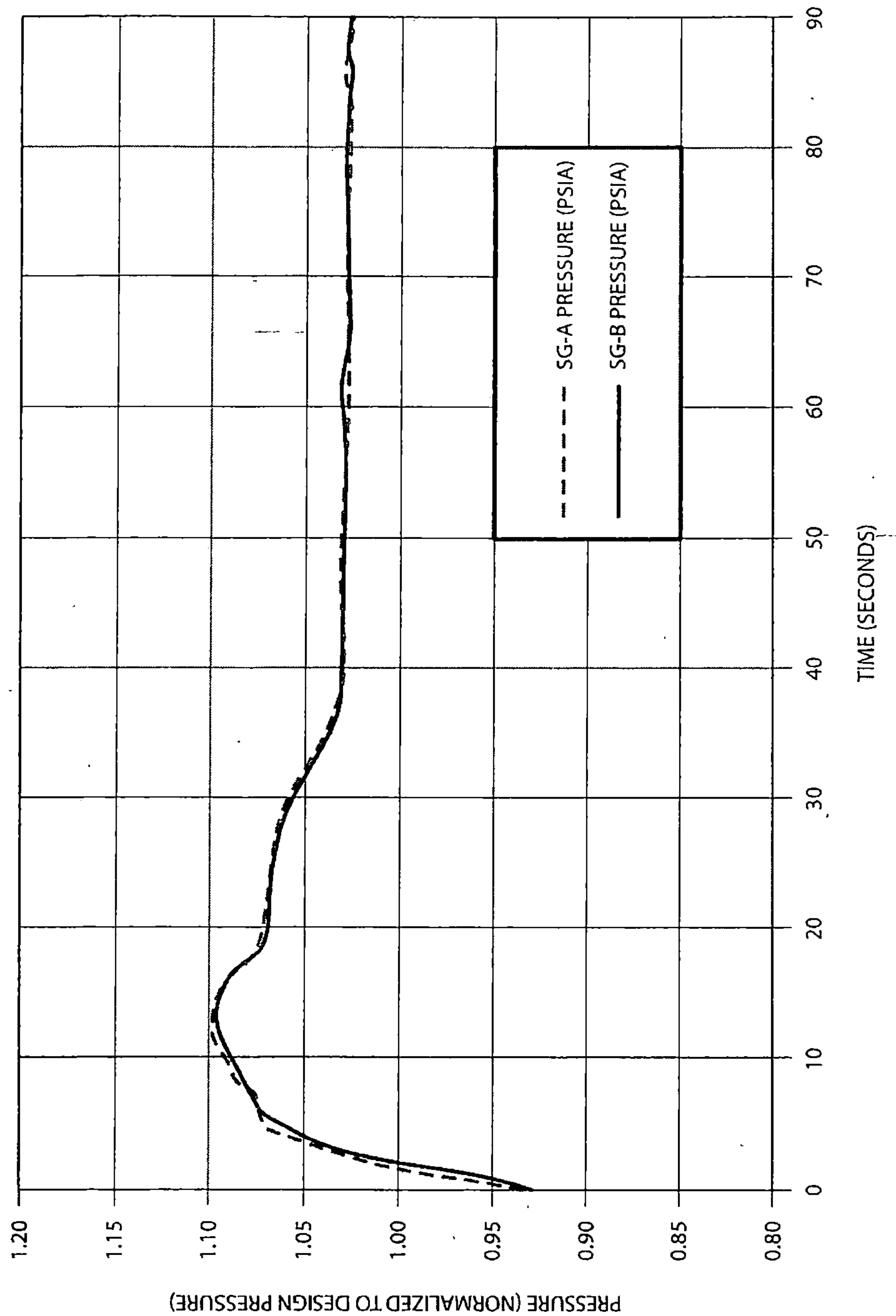
FIGURE 5.2A–3



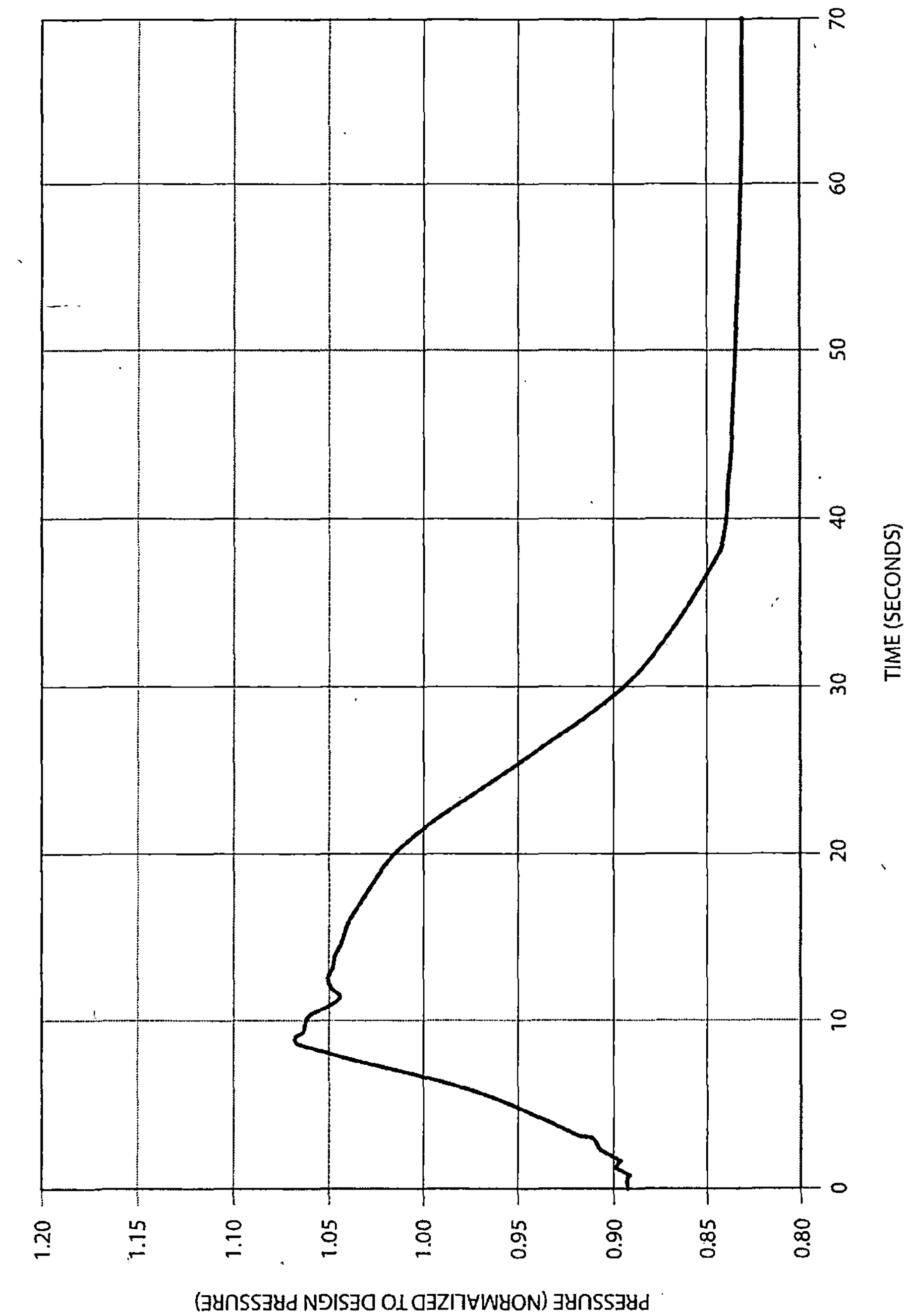
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MAX. REACTOR POWER VS. TIME FOR WORST CASE LOSS OF LOAD INCIDENT

FIGURE 5.2A–4



FLORIDA POWER & LIGHT COMPANY **ST. LUCIE PLANT UNIT 2**
STEAM GENERATOR PRESSURE LOSS OF CONDENSER VACUUM (LOSS OF LOAD WITH DELAYED REACTOR TRIP) **FIGURE 5.2A-5**



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MAXIMUM REACTOR COOLANT SYST. PRESSURE VS. TIME FOR LOSS OF CONDENSER VACUUM (LOSS OF LOAD WITH DELAYED REACTOR TRIP)
FIGURE 5.2A-6

APPENDIX 5.2B

ANALYSIS OF ST. LUCIE UNIT 1 NATURAL CIRCULATION COOLDOWN

WITHOUT UPPER HEAD VOIDING

AND ST. LUCIE UNIT 2 CONDENSATE

STORAGE TANK REQUIREMENTS

INTRODUCTION

An analytical evaluation of natural circulation cooldown to shutdown cooling system entry conditions without formation of voids was performed for St. Lucie Unit 1 in 1980. The analytical evaluation was performed with a detailed thermal hydraulic model utilizing the RETRAN computer code. This code uses for inputs the RCS fluid volumes, flow junctions and heat conductors.

The 1980 St. Lucie Unit 1 analysis is applicable to St Lucie Unit 2. St. Lucie Unit 2 "is essentially the same design as St. Lucie Unit 1" (Subsection 1.3.1). Because of this similarity, the fluid volumes, flow junction and heat conductors for Unit 2 are virtually identical to the St. Lucie Unit 1. Therefore, if the analysis was to be performed on Unit 2 the numerical values of the nodal points, would be very similar to the ones used for Unit 1. The differences in the numerical values would be mainly caused by the 16 x 16 fuel geometry used in Unit 2. However, these differences would not appreciably change the end result.

Additionally, the selection of auxiliary equipment, associated with this analysis, have caused some physical arrangements to differ between the two Units. These physical differences, however, have not caused any significant changes in the flow conditions.

The reactor coolant system pressure must be reduced to 275 psia for shutdown cooling initiation. Consequently to prevent the formation of voids the upper 1-lead fluid must be cooled to a value less than the corresponding saturation temperature of 409.5°F. After that time de-pressurization to shutdown cooling system entry conditions can occur without void formation in the reactor coolant. Hot Leg temperature cooldown rates of 30°F/hr and 50°F/hr to 325°F were investigated to determine the cooldown time required for the fluid temperature in the reactor vessel upper head to reach shutdown cooling entry conditions without void formation.

THERMAL-HYDRAULIC MODEL

The analysis was performed with a detailed thermal-hydraulic model utilizing the RETRAN (Reference 1) computer code. Figure 5.2B-1 presents a schematic drawing of the fluid volume, flow junctions and heat conductors used in the model. Table 5.2B-1 provides a description of these volumes junctions, and conductors. Specific features of the model include: a detailed nodalization of the upper portion of the reactor vessel including a representation of the reactor vessel walls and internals; a number of automatic control systems including those for charging pumps, the letdown flow control valve and the pressurizer heaters; and a non-equilibrium thermal-hydraulic model for the pressurizer.

PRE-EPU ANALYSIS RESULTS

The St. Lucie Unit 2 SGs have been replaced. The natural circulation capability of the RSGs has been verified (Reference 4). Therefore, the original responses, as they applied to the OSGs, remain appropriate following the SG replacement.

An analysis of a St. Lucie Unit 1 natural circulation cooldown from full power for a hot leg temperature cooldown rate of about 30°F/hr to 325°F demonstrates that the reactor vessel upper head fluid cools to 409.5°F (shutdown cooling entry conditions) in 16.1 hours. The results are presented in Figure 5.2B-2. The condensate supply required for this cooldown is 218,500 gallons.

The analysis for a hot leg temperature cooldown rate of about 50°F/hr to 325°F demonstrates that the reactor vessel upper head fluid cools to 409.5°F (shutdown cooling entry conditions) in 14.2 hours. The results are presented in Figure 5.2 B-3. The condensate supply required for this cooldown is 193,000 gallons.

The analysis for a hot leg temperature cooldown rate of about 50°F/hr to 325°F was repeated using very conservative assumptions regarding fluid mixing in the upper reactor vessel in order to determine a bounding cooldown time for operating guidelines. The results demonstrate that the reactor vessel upper head fluid cools to 409.5°F (shutdown cooling entry conditions) in 25.7 hours. The condensate supply required for this cooldown is 270,500 gallons.

In addition, FP&L has investigated the amount of water required to bring St. Lucie Unit 2 through four hours of hot standby followed by a Natural Circulation Cooldown without upper head voiding. Table 5.2B-2 indicates the amount of condensate water required to meet a variety of different cooldown rates. The maximum conservative value of 330,900 gallons is composed of 60,400 gallons required for hot standby coupled with 270,500 gallons needed for natural circulation cooldown. The condensate requirement of 330,900 gallons will be met via revising FP&L procedures to require the operator to follow a hot standby Natural Circulation Cooldown procedure to establish makeup to the condensate storage tank and to maintain the volume of the tank continuously above the technical specification Limit. The makeup supply of condensate water can be supplied from the two 500,000 gallon city water storage tanks located on the St. Lucie Unit 1 site. Pumping capabilities can be provided by the St. Lucie Unit 1 diesel generators.

We have concluded that this approach adequately accomplishes the goal of bringing the plant to shutdown conditions in a safe manner.

RECOMMENDATION

The above results show that for a hot leg temperature cooldown rate of 50°F/hr to 325°F, the upper head fluid can be cooled to shutdown cooling system entry conditions without void formation in approximately 14.2 hours.

In order to provide additional conservatism, it is recommended that for natural circulation cooldown to shutdown cooling system entry conditions without void formation, the hot leg temperature cooldown rate be about 50°F/hr to 325°F followed by a soak at 325°F for 20.4 hours for a total cooldown time of approximately 25.7 hours from cooldown initiation. Figure 5.2B-4 shows the recommended plant cooldown rate. The condensate supply required for this cooldown is 270,500 gallons.

OTHER ANALYSIS

The NRC staff indicated that the St. Lucie Unit 1 natural circulation cooldown event would not, by itself, provide sufficient basis to satisfy the boron mixing and natural circulation cooldown test requirements associated with Reactor System Branch, Branch Technical Position 5-1. Therefore, St. Lucie

Unit 2 was required to: 1) provide a report on the San Onofre Unit 2 Natural Circulation Cooldown and Mixing Test, and 2) justify that the test data is applicable to St. Lucie Unit 2. The information from the San Onofre report was used on a comparison basis for NRC approval and provides the similarities of the results and the applicability to St. Lucie Unit 2. (2)

Based on San Onofre Nuclear Generating Station Report CEN-259, the estimated cooling water requirement for St. Lucie Unit 2 is 276,000 gallons. Therefore, St. Lucie Unit 2 has a sufficient amount of water in the CST to meet the requirements of BTP RSB 5-1[3].

EFFECT OF UPRATE TO 3020 MWT

To evaluate the natural circulation capability of St. Lucie Unit 2 at an EPU power level of 3020 MWt, the CENTS computer Code (Reference 5) was used to simulate the plant response to a loss of offsite power followed by a natural circulation cooldown from hot standby conditions to shutdown cooling system entry conditions. The CENTS code is an NRC approved code that is acceptable for referencing in licensing applications for Combustion Engineering designed pressurized water reactors.

The previously analyzed scenarios were simulated. The resulting total cooldown time and condensate volume required can be compared to the 1980 St. Lucie Unit 1 analysis values.

At an EPU power level of 3020 MWt, the simulations demonstrate that the plant can be cooled down to SDC entry conditions using the same equipment the plant has prior to EPU operation while maintaining pressure control (no voids in the RCS) for a loss of offsite power event.

The 3020 MVW uprate at St. Lucie Unit 2 will not adversely impact the natural circulation cooldown capability of the plant for the following reasons:

- The maximum core ΔT during the 30°F/hour and the 50°F/hour cooldown Is lower than the normal full power ΔT of 55°F.
- The reactor coolant system pressure is reduced to 275 psia for SDC initiation and the upper head fluid is cooled to a value less than the corresponding saturation temperature of 409.5°F.
- The condensate volume requirement established in the historical evaluation is bounding for the 3020 MWt uprate.

The EPU analysis also shows:

- Acceptable results were determined for natural circulation cooling during the hot standby period for expected residual heat rates immediately following reactor shutdown from the 3020 MWT EPU conditions.
- The ADVs are adequate to achieve cooldown to the SDC entry point in a reasonable time period. SDC entry conditions can be achieved In less than 10 hours at a cooldown rate of 30°F/hour and less than 15 hours at a cooldown rate of 50°F/hour which Includes four hours In hot standby.

The results of the CENTS simulation for the 30°F/hour cooldown show that it will take about 9.3 hours to bring the RCS to shutdown cooling entry conditions (325°F and a pressure of 275 psia). The condensate supply required for this cooldown is about 117,000 gallons.
The results of the CENTS simulation for the 50°F/hour cooldown show that it will take about 14.5 hours to bring the RCS to shutdown cooling entry conditions. The condensate supply required for this cooldown is abut 164,200 gallons.

The most limiting cooldown simulation required cooling down at a 50°F/hour rate to 325°F and then maintaining the hot leg temperature at 325°F for 5.1 hours in order to allow the SDC pressure to be reached without flashing of the upper head fluid. The condensate supply required for this cooldown is about 281,000 gallons. The total cooldown time is 23.6 hours.

REFERENCES:

(1) RETRAN-A Program For One-Dimensional Transient Thermal-Hydraulic Anlysis of Complex Fluid Flow Systems, Volumes 1, 2, 3 and 4, EPRI CCM-5, December 1978.

(2) FPL Letter L-84-68 from J W Williams to D G Eisenhut dated March 13, 1984.

(3) NRC Letter from E G Tourigny to W F Conway dated April 12, 1988.

(4) AREVA NP Inc. Document 77-5069878-001, "Replacement Steam Generator Report for Florida Power and Light St. Lucie Unit 2."

(5) Westinghouse Owners Group Topical Report WCAP-15996-P-A, Rev. 1, "Technical Description Manual for the CENTS Code," dated March 2005.

TABLE 5.2B-1

MODEL GEOMETRY DESCRIPTION

FLUID VOLUME	DESCRIPTION
1	Combined hot leg volume.
2	Combined steam generator inlet plenum volume.
3	Combined steam generator tube volume from tube sheet to top of tube bundle.
4	Combined steam generator tube volume from top of tube bundle to tube sheet.
5	Combined steam generator outlet plenum volume.
6	Combined cold leg volume upstream of reactor coolant pump.
7	Combined reactor coolant pump volume.
8	Combined cold leg volume downstream of reactor coolant pump.
9	Reactor vessel downcomer volume.
10	Reactor vessel inlet plenum volume.
11	Core volume.
16	Volume from top of active core to fuel alignment plate.
12	Outlet plenum volume from fuel alignment plate to upper guide structure support plate.
13	CEA shroud volume.
17	Upper head volume from upper guide structure support plate to top of CEA shroud.
14	Upper head volume above top of CEA shroud.
32	Surge line volume.
34	Pressurizer volume.
51	Steam generator shell side volume.

TABLE 5.2B-1 (Cont'd)

FLOW JUNCTION	DESCRIPTION
1	Flow from Volume 12 to Volume 1.
2	Flow from Volume 1 to Volume 2.
3	Flow from Volume 2 to Volume 3.
4	Flow from Volume 3 to Volume 4.
5	Flow from Volume 4 to Volume 5.
6	Flow from Volume 5 to Volume 6.
7	Flow from Volume 6 to Volume 7.
8	Flow from Volume 7 to Volume 8.
9	Flow from Volume 8 to Volume 9.
10	Flow from Volume 9 to Volume 10.
11	Flow from Volume 10 to Volume 11.
17	Flow from Volume 11 to Volume 16.
12	Flow from Volume 16 to Volume 12.
13	Flow from Volume 16 to Volume 13.
14	Flow from Volume 13 to Volume 17.
15	Flow from Volume 17 to Volume 12.
18	Flow from Volume 14 to Volume 17.
35	Flow from Volume 32 to Volume 1.
36	Flow from Volume 34 to Volume 32.
37	Spray flow to Volume 34.
38	Charging flow to Volume 8.
39	Letdown flow from Volume 6.
81	Feedwater flow to Volume 51.
82	Atmospheric relief valve flow from Volume 51.
83	Steam bypass valve flow to condenser from Volume 51.
84	Steam dump valve flow to condenser from Volume 51.
91	Steam flow to turbine from Volume 51.

TABLE 5.2B-1 (Cont'd)

HEAT CONDUCTOR	DESCRIPTION
1	Fuel conductor connecting fuel to Volume 11.
2	Steam generator tubes connecting Volume 3 and Volume 51.
3	Steam generator tubes connecting Volume 4 and Volume 51.
4	Metal in reactor vessel walls adjacent to Volume 14.
5	Metal associated with upperhead drive shafts to Volume 14.
6	Metal associated with CEA shrouds connecting Volume 13 and Volume 17.
7	Metal associated with CEA shrouds connecting Volume 13 and Volume 12.
8	Metal associated with CEA shrouds connecting Volume 13 and Volume 12.
9	Upper guide structure support plate connecting Volume 17 and Volume 12.
10	Metal associated with upper guide structure adjacent to Volume 12.
11	Metal in reactor vessel wall adjacent to Volume 17.
12	An effective conductor to allow axial heat conduction between Volume 14 and Volume 17.

TABLE 5.2B-2

NATURAL CIRCULATION COOLDOWN

CONDENSATE STORAGE TANK REQUIREMENTS FOR PRE-EPU OPERATION

Condition	Length of Condition (hours)	C.S.T. Req't (gal)	Reference
1. Case 1			
Hot standby	4.0	60,400	Ref Table 10.4-2
Cooldown (@30°F/HR)	16.1	218,500	Ref App. 5.2B
Total	20.1	278,900	
2. Case 2			
Hot standby	4.0	60,400	Ref Table 10.4-2
Cooldown (@50°F/HR)	14.2	193,000	Ref App. 5.2B
Total	18.2	253,400	
3. Case 3*			
Hot standby	4.0	60,400	Ref Table 10.4-2
Cooldown (@50°F/HR)	25.7	270,500	Ref App. 5.2B
Total	29.7	330,900	

* Case 3 - Extreme conservative assumptions regarding fluid mixing in the upper reactor vessel

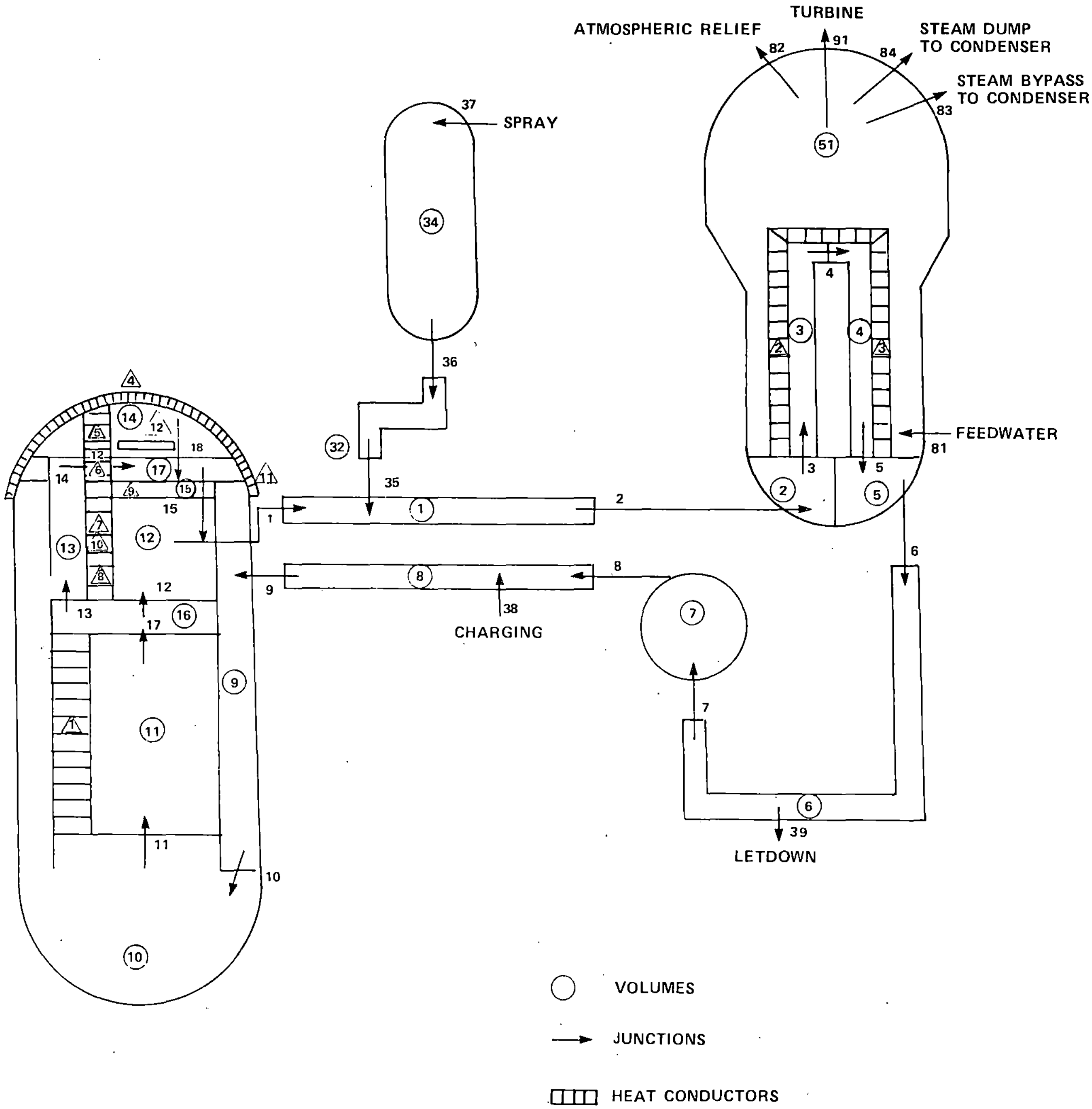
TABLE 5.2B-3

NATURAL CIRCULATION COOLDOWN

CONDENSATE STORAGE TANK REQUIREMENTS FOR EPU OPERATION

Condition	Length of Condition (hours)	C.S.T. Req't (gal)	Reference
1. Case 1			
Hot standby	4.0	74,000	Ref Table 10.4-2
Cooldown (@30°F/hr)	9.3	117,000	Ref App. 5.2B
Soak to SDC Entry Pressure	9.5	74,000	
Total	22.8	265,000	
2. Case 2			
Hot standby	4.0	76,300	Ref Table 10.4-2
Cooldown (@50°F/HR)	14.5	164,200	Ref App. 5.2B
Soak to SDC Entry Pressure	5.1	40,500	
Total	23.6	281,000	

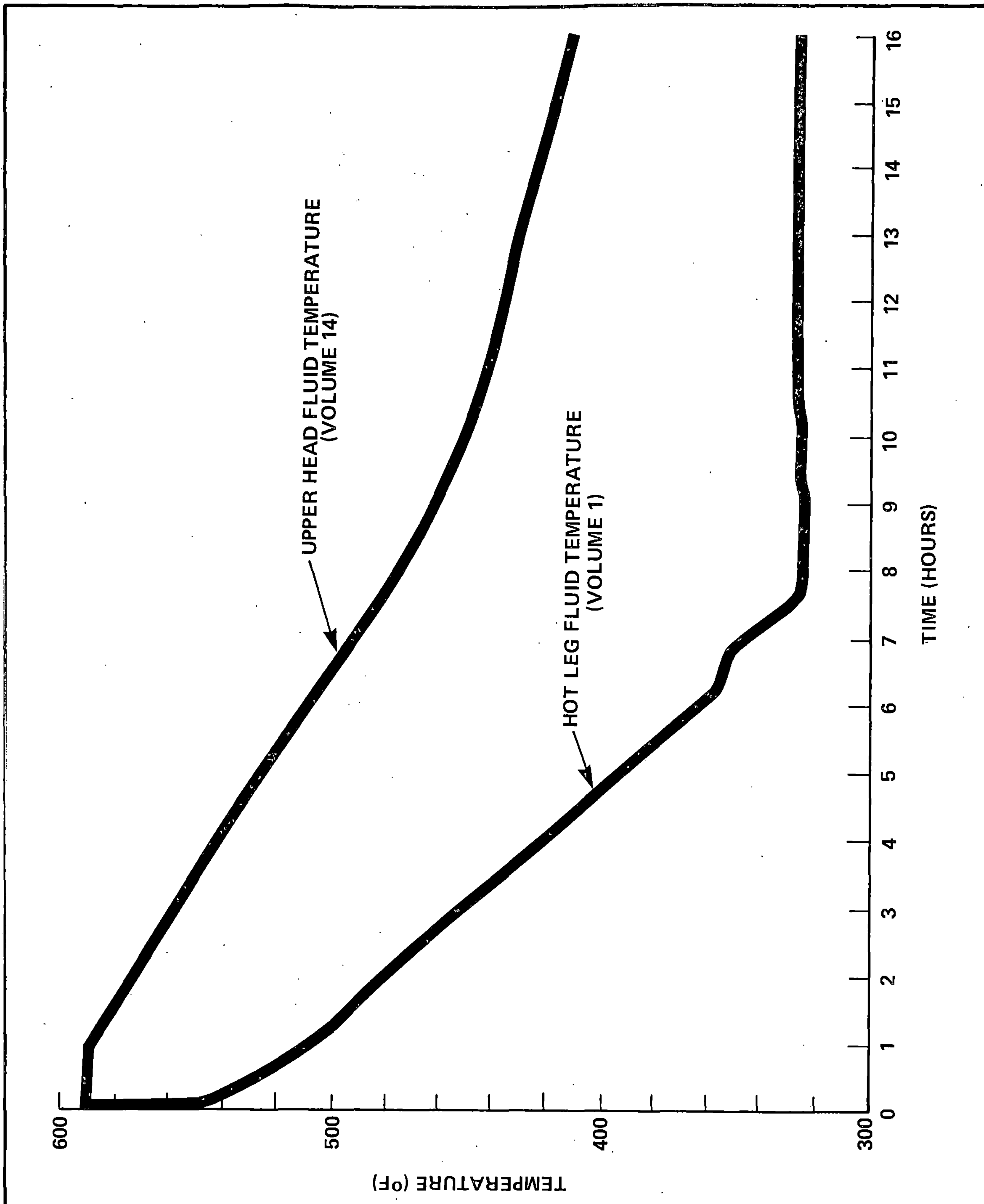
* Case 2 - Extreme conservative assumptions regarding fluid mixing in the upper reactor vessel head



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MODEL VOLUME, JUNCTION, AND CONDUCTOR GEOMETRY

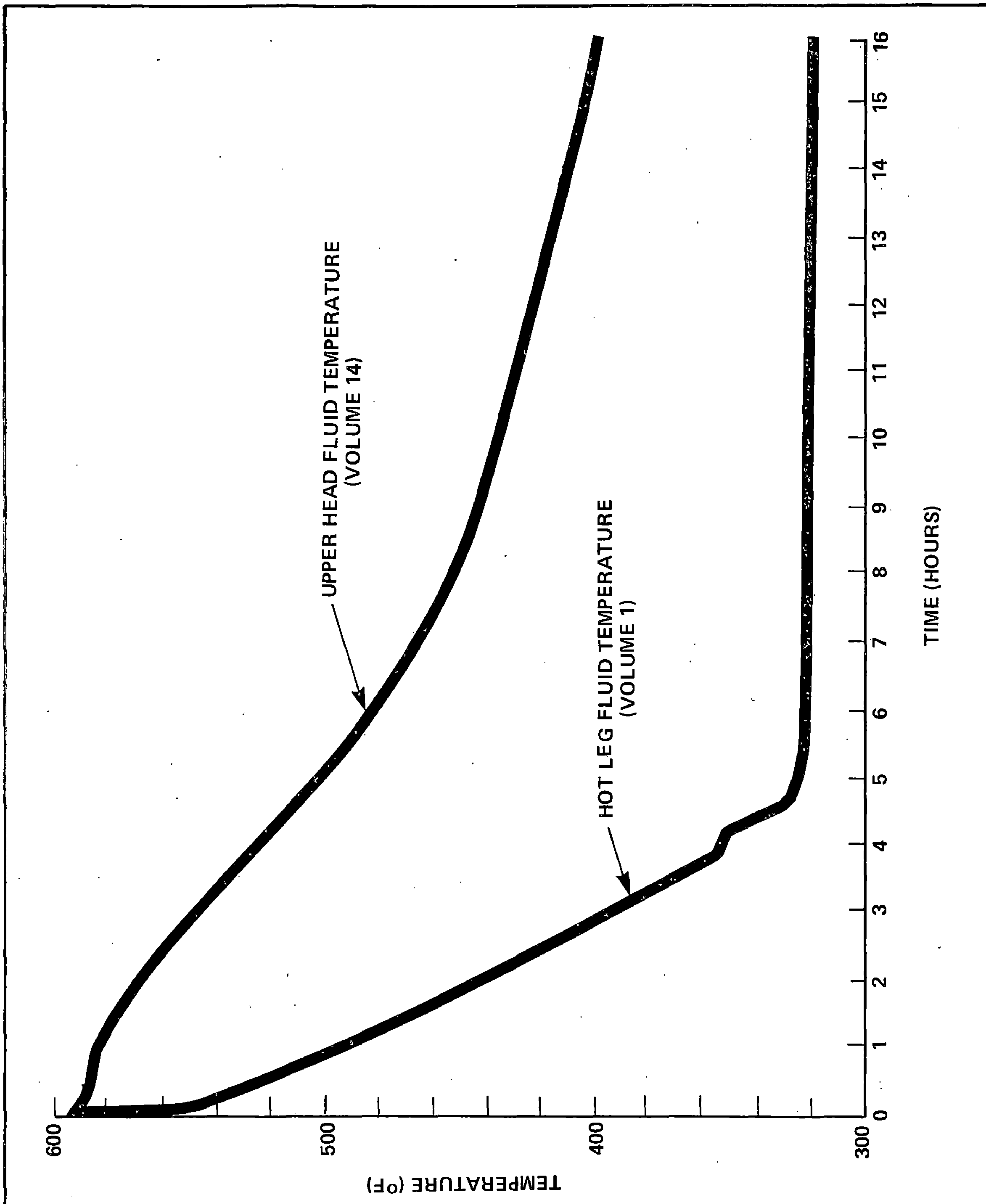
FIGURE 5.2B-1



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR COOLANT TEMPERATURE
VS TIME COOLDOWN AT
30° F/HR TO 325° F

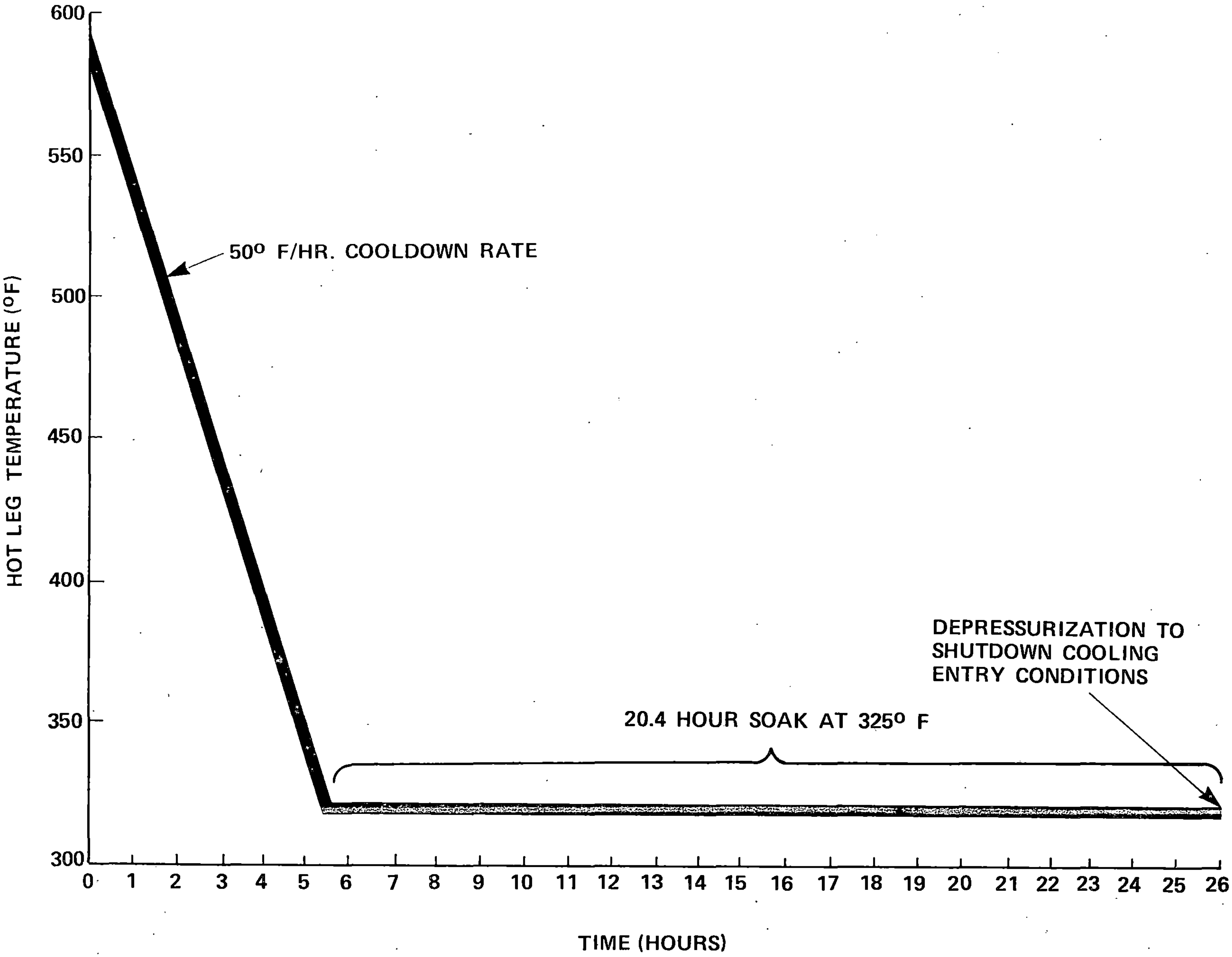
FIGURE 5.2B-2



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR COOLANT TEMPERATURE
VS TIME COOLDOWN AT
50° F/HR TO 325° F

FIGURE 5.2B-3



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

RECOMMENDED COOLDOWN GUIDELINE

FIGURE 5.2B-4

APPENDIX 5.2C

ST. LUCIE UNIT 1

NATURAL CIRCULATION COOLDOWN

APPENDIX 5.2C

Natural Circulation Cooldown

A combined thermal/stress analysis was performed to determine the range of stresses produced in the reactor vessel head due to voiding of the upper head region. In performing the thermal analysis, "worst case" assumptions were made in defining the fluid temperature and initial vessel temperatures. The assumptions which define the thermal transient are as follows and illustrated on Figure 5.2C-1.

a) The upper head is drained down to the upper guide support structure plate in 20 minutes with both fluid and metal temperatures remaining at 600F.

b) The water level is held at this height for 40 minutes with a fluid temperature of 300F.

c) The head is refilled over a 20 minute period with 300F water.

d) The upper head remains filled with 300F water for a period of time.

e) The heat transfer coefficient for the water is large (H=500 BTU/ft^2-hr-F).

f) The heat transfer coefficient for steam is very small (H=0. BTU/ft^2-hr-F).

Temperatures calculated for this transient were applied to a stress analysis model. The results of this analysis indicate that the highest stresses occur in the "knuckle" region of the head near the inside radius. The magnitude of stresses produced for this transient were found to be no more severe than the stresses occurring during a normal cooldown of 100F/hr.

In addition the results of this analysis demonstrate that:

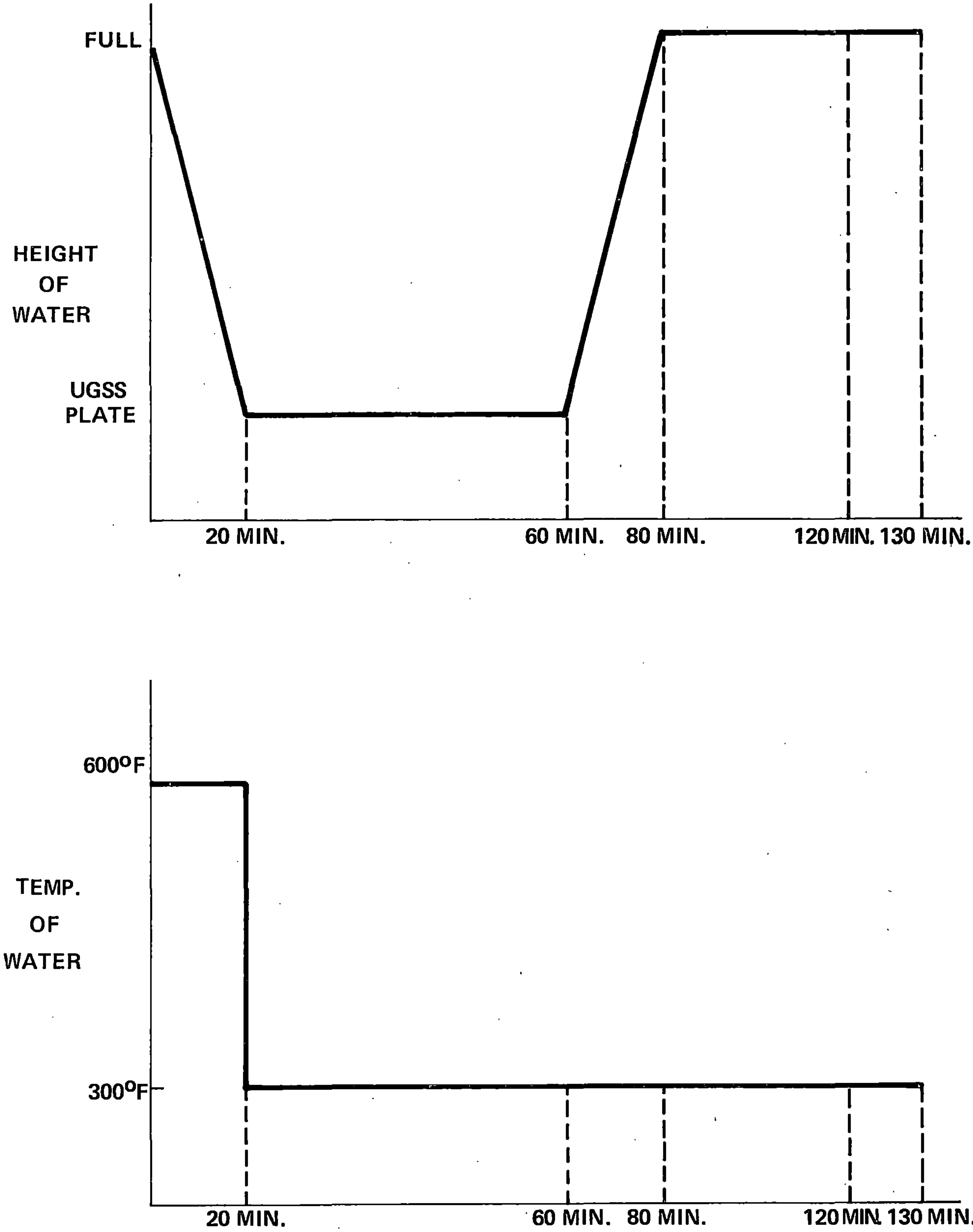
a) A more rapid refill of the head does not cause higher stresses since the thermal conductivity through the reactor vessel wall is the limiting heat transfer mechanism.

b) The water level holddown time does have an effect on the stresses in the head. Longer holddown times decrease the stress in the "knuckle" region because of axial heat flow which removes heat from the head.

c) The thermally-induced stresses in the nozzle region of the reactor vessel are small in comparison to the stresses due to pressure loading only.

d) The deformations and rotations in the control element drive mechanism nozzles are negligible due to the thermal transient.

e) No separation occurs at the O-ring seal region of the flange, hence, no leakage occurs.



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FORCING FUNCTION USED FOR THERMAL TRANSIENT
FIGURE 5.2C-1

5.3 REACTOR VESSEL

5.3.1 REACTOR VESSEL MATERIALS

5.3.1.1 Material Specifications

The principal ferritic materials used in the reactor vessel are listed in Table 5.2-3. These materials are specified to be in accordance with ASME Code, Section III, 1971 Edition including Summer 1972 Addenda.

5.3.1.2 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 reactor vessel head flange is drilled to match the reactor vessel flange stud bolt locations. The stud bolts are fitted with spherical washers located between the closure nuts and the reactor vessel head flange. These washers maintain stud alignment during boltup when flexing of the reactor vessel head must be accommodated. The lower surface of the reactor vessel head flange is machined to provide a mating surface for the reactor vessel closure seals.

The reactor 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 reactor vessel flange is drilled and tapped to receive the closure studs and is machined to provide a mating surface for the reactor vessel closure seals. A tapered transition section connects the reactor vessel flange to the cylindrical shell.

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

The inlet and outlet nozzles on the reactor vessel are located on a common plane just below the reactor vessel flange. Ample thickness in this reactor vessel course, provides most of the reinforcement required for the nozzles. Additional reinforcement is provided by the individual nozzle attachments. A boss located around the outlet nozzles on the inside diameter of the reactor vessel wall provides a mating surface for the core support barrel and guides the outlet reactor 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. The lower region of the nozzles shell segment has an externally tapered transition and joins the intermediate and lower shell assemblies.

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

5.3.1.3 Special Methods of Nondestructive Examination

Prior to, during, and after fabrication of the reactor vessel, nondestructive tests based upon ASME Code Section III are performed on all welds, forgings, plates and bolting listed below:

a) All full-penetration, pressure-containing welds are 100 percent radiographed to the standards of ASME Code, Section III. Weld preparation areas, back-chip areas, and final weld surfaces are 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, are inspected by liquid-penetrant tests of the root pass, the lesser of one-third of the thickness or each 1/2 inch of weld deposit, and the final surface. Additionally, the base metal weld preparation area is magnetic-particle examined prior to overlay with nickel-chromium-iron weld metal.

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

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

Any cracks and linear indications.

Rounded indications in a line separated by less than 3/16 inch

Four or more rounded indications in a line separated by less than 1/16 inch edge to edge.

Ten or more rounded indications in any six in.^{2} in the most unfavorable locations.

c) Plates are subjected to ultrasonic examination using straight beam techniques. Ultrasonic examination is performed in both the as received condition and after forming. Rejection is 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.

Any defect that shows a total loss of back reflection that cannot be contained within a circle whose diameter is the greater of three inches or one half the plate thickness is unacceptable. Two or more defects smaller than described above, which cause a complete loss of back reflection, are 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 are magnetic particle examined after accelerated cooling to the magnetic-particle criteria cited above.

d) All vessel bolting material receives ultrasonic and magnetic-particle examination during the manufacturing process.

The bolting material receives a straight-beam, radial-scan, ultrasonic examination with a search unit whose search area does not exceed one square-inch. The standard for rejection is 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 whose area does not exceed one square inch. A reference specimen of the same composition and thickness containing a notch (located on the inside surface) one inch in length and a depth of three percent of nominal section thickness, or 3/8 inch, whichever is less, is 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 are used.

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

Non destructive testing of a vessel is performed throughout fabrication. The non-destructive examination requirements including calibration methods, instrumentation sensitivity, and reproducibility of data, are in accordance with requirements of ASME Code, Section III (see Table 5.2-1). Strict quality control is maintained in critical areas such as calibration of test instruments.

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

After hydrostatic testing, a set of ultrasonic examination is performed to insure that the welds meet the acceptance criteria of ASME Code, Section XI.

<u>Replacement Reactor Vessel Closure Head (RVCH) Non-Destructive Inspection (NDE)</u>

Table 5.3-11 summarizes the quality assurance program inspections for the replacement RVCH. Within this table are identified all of the non-destructive test and inspections required by the RVCH Certified Design Specification as well as all test and inspections required by the applicable Codes. The applicable Codes for the RVCH are ASME Section III, Div. 1, for Class 1 Components, 1989 Edition No Addenda and ASME Section XI, 1998 Edition with Addenda through 2000. As required by the Certified Design Specification for the RVCH and summarized in Table 5.3-11, the NDE performed on the PSL2 RVCH as well as the acceptance criteria is more stringent than that required by the applicable ASME Codes.

In addition to the NDE summarized in Table 5.3-11, there are those inspections which the equipment supplier performs to confirm the adequacy of materials received, and those performed by the producer of the materials in the production of the basic materials. Procedures for performing all of the examinations are consistent with the requirements of the applicable ASME Codes and are reviewed by qualified FPL and Owner's Agent representatives. These procedures have been developed to provide the highest assurance of quality in the materials and fabrication. They consider not only flaw size, but equally as important; how the material is fabricated, the orientation and type of possible flaws, and the areas of most severe service conditions. The volumetric inspections (Ultrasonic Testing technique) of the forging were done using both the straight beam and angle beam techniques. In addition, the surfaces most subject to damage as a result of forging, heat treating, forming, fabricating, and hydrostatic testing received 100% surface inspections by Magnetic Particle or Liquid Penetrant test at various stages during the processes and again after final completion of the hydrostatic pressure testing of the RVCH.

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

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

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

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

Regulatory Guide 1.31, "Control of Stainless Steel Welding," June 1973 (R1) is addressed in Subsection 5.2.3.4.

Regulatory Guide 1.34, "Control of Electroslag Weld Properties," December 1972 (R0) is addressed in Subsection 5.2.3.3.

Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel (Components," May 1973 (R0) is addressed in Subsection 5.2.3.3.

Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," May 1973 (R0) is addressed in Subsection 5.2.3.4.

Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel," May 1973 (R0) is addressed in Subsection 5.2.3.3.

Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility," December 1973 (R0) is addressed in Subsection 5.2.3.3.

Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," April 1977 (R1) is addressed in Subsection 5.3.1.6.7.

5.3.1.5 Fracture Toughness

The reactor vessel beltline plate material were Charpy impact tested in the weak direction to establish RT_{NDT} as required by 10 CFR 50, Appendix G. Vessel weld heat-affected-zone materials for all plates were not tested as these materials were not available during fabrication nor were they required to be tested by the applicable code. Weld heat-affected zone materials for the applicable reactor vessel beltline most limiting plate are tested as part of the reactor vessel irradiation surveillance program Described in Subsection 5.3.1.6.

Impact test data for the reactor vessel beltline weld materials were obtained from weld metal certification tests performed in accordance with the ASME (Code, Section III, Subarticle NB-2400. The highest RT_{NDT} for the beltline region weld materials was -40°F.

The highest reported RT_{NDT} for reactor vessel beltline plate materials was +30°F (intermediate shell plate M-605-1). For other reactor vessel plate materials the highest RT_{NDT} value reported was +50°F (upper shell plates M-604-1 and 2). Predictions of end of life adjusted reference temperature (adjusted RT_{NDT}) corresponding to the shift at the 30 ft.-lbs. value of Charpy V-Notch fracture energy are made for the beltline region plates and welds. These predictions are calculated using the initial reference temperature, a shift to account for irradiation embrittlement and a margin term to account for uncertainty in the initial measurement and the prediction method. Current predictions are maintained in the NRC docket under 10 CFR 50.61. The predictions indicate that the end of life transition temperatures for the beltline materials are safely below the 10 CFR 50.61 screening limit. Predictions are also used as a basis for heatup and cooldown curves as noted in Section 5.3.2. The predictions are periodically benchmarked by the reactor vessel surveillance program (Section 5.3.1.6).

The lowest Charpy upper shelf energy (weak direction) for the reactor vessel beltline material is 91 ft-lb, well above the 75 ft-lb minimum required by 10 CFR 50, Appendix G.

RT_{NDT} values for each weld adjacent to a nozzle, a flange, and a shell region near a geometric discontinuity in the reactor vessel were evaluated. IE Bulletin No. 78-12, dated September 29, 1978, required the review and submission of all reactor vessel welding materials test data to the NRC(4). Review of the data in the IE Bulletin response shows that no welding materials used in CE vessels has an RT_{NDT} higher than -10°F. Therefore, the weldments in the RV have RT_{NDT} $\leq$ -10°F. This is less than the RT_{NDT} of the adjacent base materials, and therefore these weldments are not limiting for operation.

Conformance to Regulatory Guide 1.2, "Thermal Shock to Reactor Pressure Vessels," November 1970 (RO) is described below:

A reactor vessel materials irradiation surveillance program and test samples are provided with each reactor vessel. The recommended surveillance program is in accordance with 10 CFR 50, Appendix H and ASTM E185, "Recommended practice for Surveillance Tests for Nuclear Reactor Vessels." This program provides for periodic measurement of the radiation induced changes in reactor vessel materials properties and provides assurance that these materials retain adequate toughness to preclude brittle failure under postulated accident conditions. (Regulatory Guide 1.2 was withdrawn in 1991 and was superceded by 10 CFR 50.61.)

As required by 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock" (PTS Rule), an analysis was conducted which projected values of RT_{PTS} (at the INNER vessel surface) of reactor vessel beltline materials at the expiration date of the operating license.

The controlling beltline material from the standpoint of PTS susceptibility was identified to be the intermediate shell plate M-605-2. The RT_{PTS} for this limiting material at the expiration date of the license is less than 270°F which is the screening criterion for the limiting material at the expiration date. The current actual EOL RT_{PTS} results for all the beltline materials are maintained in the NRC Docket file under 10 CFR 50.61 for the unit.

Experimental irradiation programs have been undertaken to provide information on changes in toughness properties due to neutron irradiation, and recovery of these properties by thermal annealing. The Heavy Section Steel Technology research and development program sponsored by the ASME has and is developing improved basic data on reactor vessel materials properties.

Low residual alloy steel has been utilized for the reactor vessel beltline material which has been shown to be resistant to neutron radiation embrittlement to further assure adequate fracture toughness throughout vessel life.

5.3.1.6 Materials Surveillance Program (1)

The irradiation surveillance program is conducted to assess the radiation induced changes in the strength and toughness properties of the reactor vessel beltline materials. Changes in these properties are evaluated by the comparison of pre- and post irradiation test results. The surveillance test specimens are irradiated under conditions which represent, as closely as practical, the irradiation conditions of the reactor vessel.

This surveillance program is consistent with the objectives of ASTM E185-82 (Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels) and 10 CFR 50, Appendix H; "... to verify the initial predictions of the surveillance material response to the actual radiation environment ..." and "... to determine the conditions under which the vessel can be operated with adequate margins of safety against fracture." Tables 5.3-9 and 5.3-9a summarize the information necessary to comply with 10 CFR Part 50, Appendix H. The tables include information on the withdrawal schedule, the origin of the surveillance specimens, and specimen types, quantities, and chemical compositions (Tables 5.2-5 and 5.2-6 provide complete chemical composition data). The one difference between the plant surveillance program and the requirements presented in Appendix H, is the following:

The surveillance capsules are attached to the cladding on the inside vessel wall in the beltline region. This modification to 10 CFR 50, Appendix H, Section II.C.2 is described in CENPD-155(1), and the procedures described therein are considered acceptable(2).

5.3.1.6.1 Test Materials Selection

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 are manufactured is obtained from the plate in the reactor vessel beltline which would become the limiting plate with respect to reactor operation during its lifetime. This material (intermediate shell plate M-605-1) is selected on the basis of highest initial RT_{NDT}, chemical composition and fluence. The test materials are 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 are prepared is maintained. The results of mill test chemical analyses for the six plates of the beltline region of one vessel are presented in Table 5.2-5.

Specimens are prepared from three metallurgically different materials, including base metal, weld metal and heat-affected zone (HAZ) materials. In addition, material is included from a standard heat of ASTM A533-B.

Class I manganese-molybdenum nickel steel made available by the USNRC sponsored Heavy Section Steel Technology (HSST) program. This standard reference material (SRM) is used as a monitor for Charpy impact tests permitting comparisons among the irradiation data from operating power reactors and experimental reactors. Compilation of data generated from post-irradiation tests of these correlation monitors is carried out by the HSST program.

5.3.1.6.1.1 Base Metal

Base metal test material is manufactured from a section of intermediate shell plate M-605-1 which is 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 section of shell plate used for the base metal test material is adjacent to the test material used for ASME Section III tests and is at a distance of at least one plate thickness from any water-quenched edge. This material is 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 heat affected zone material are produced by welding together sections from the selected base metal plate and another plate from the reactor vessel beltline. The HAZ test material is 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 is followed in the manufacture of the weld metal and HAZ test materials. The welded plates are heat-treated to metallurgical conditions that are representative of the final metallurgical conditions of similar materials in the completed reactor vessel.

5.3.1.6.1.3 Archive Material

Representative stock (archive material) to provide two additional sets of test specimens for each material for encapsulation are provided with full documentation and identification.

5.3.1.6.2 Test Specimens

5.3.1.6.2.1 Type and Quantity

The total quantity of specimens furnished for baseline (unirradiated) and post-irradiation testing is presented in Table 5.3-3. Each of the test materials has been chemically analyzed for approximately 21 elements, including those listed in Subsection 4.1.3 of ASTM E185. The types of specimens are drop weight, Charpy and tension specimens for baseline testing, and Charpy impact and tension specimens for post-irradiation testing.

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 5.3-4. The date from tests of these specimens provide the basis for determining the radiation induced property changes of the reactor vessel materials.

Twelve drop weight test specimens each of base metal (longitudinal and transverse), weld metal, and HAZ material are provided for establishing the Nil-ductility transition temperature (T_{NDT}) of the unirradiated surveillance materials. These data form the basis for RT_{NDT} determination.

Thirty Charpy impact 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 E185-73 for developing the initial Charpy impact energy transition curve and reference temperature RT_{NDT} for these materials.

Eighteen tension 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 E185-73 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

Tension and Charpy impact test specimens are used to measure radiation induced strength and toughness property changes of the surveillance materials. The type and quantity of test specimens provided for monitoring the properties of the irradiated materials over the lifetime of the vessel are presented in Table 5.3-5.

5.3.1.6.3 Surveillance Capsules

The surveillance test specimens are placed within corrosion-resistant capsule assemblies for protection from the reactor coolant during irradiation. The capsules also serve to physically locate the test specimens in selected positions within the reactor vessel and to facilitate the removal of a desired quantity of test specimens when a specified radiation exposure has been attained. Six surveillance capsule assemblies are provided for the reactor vessel.

A typical capsule assembly, illustrated on Figure 5.3-1 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 psi 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 cladding. 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 tension 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.

5.3.1.6.3.1 Charpy Impact Compartments

Each Charpy impact compartment (Figure 5.3-2) 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.

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.

5.3.1.6.3.2 Tensile - Monitor Compartments

Each tensile-monitor compartment (Figure 5.3-3) contains three tension test specimens, a set of flux spectrum monitors and a set of temperature monitors. 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 reactor coolant.

5.3.1.6.4 Neutron Irradiation and Temperature Exposure

The predicted changes in the properties of the reactor vessel materials are based on data from specimens irradiated to various fluence levels and in different neutron energy spectra.

5.3.1.6.4.1 Flux Measurements

Fast neutron flux measurements are obtained by insertion of threshold detectors into each of the six surveillance capsules. Such detectors are particularly suited for the application because their effective threshold energies lie in the range of interest (0.5 to 15 Mev).

These neutron threshold detectors and the thermal neutron detectors, presented in Table 5.3-6, can be used to monitor the thermal and fast neutron spectra incident on the test specimen. These detectors possess reasonably long half-lives and activation cross sections covering the desired neutron energy range. These neutron threshold detectors exceed the number required by ASTM E482.

One set of flux spectrum monitors is included in each tensile-monitor compartment. Each detector is placed inside a sheath which identified 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 housing as shown on Figure 5.3-3 at three axial locations in each capsule assembly (Figure 5.3-1) to provide an axial profile of the fluence level 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 I, 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.

5.3.1.6.4.2 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 specimen temperature as well as that of the reactor vessel. During irradiation, instrumented capsules are not practical for a surveillance program extending over the design life-time 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 5.3-7. 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.

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 on Figure 5.3-3 which are placed at three axial locations in each capsule assembly (Figure 5.3-1) to provide an axial profile of the maximum temperature to which the specimens were exposed.

5.3.1.6.5 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 5.3-8.

The test specimens contained in the capsule assemblies are used for monitoring the radiation-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 seen by the adjacent reactor vessel wall.

The capsule assemblies are placed in capsule holders positioned circumferentially about the core at locations which include the regions of maximum flux. Figure 5.3-4 presents the exposure locations for the capsule assemblies.

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

5.3.1.6.6 Withdrawal Schedule

The capsule assemblies retain within their holders until the test specimens contained therein have been exposed to predetermined levels of fast neutron fluency. At that time, the capsule assembly is removed and the surveillance materials are evaluated. The capsule assembly withdrawal schedule is presented in Table 5.3-9.

The surveillance capsule withdrawal schedule for St. Lucie Unit 2 was established in accordance with 10 CFR 50, Appendix H, Paragraph II.C.3(b).

A significant advantage results from withdrawal of the first surveillance capsule after one cycle (1.1 EFPY), because it provides an early indication of the validity of the reactor vessel fluency and reference temperature shift predictions used to set the vessel operating limits. Actual dosimetry and shift measurements are then available for projecting radiation induced changes in the toughness properties of the vessel beltline materials.

The target exposure levels for the surveillance capsules are determined at the azimuthal locations at the time intervals indicated in the withdrawal schedule in 10 CFR 50, Appendix H, Section II.C.3.b.

Withdrawal schedules may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule. The three standby capsules are provided in the event they are needed to monitor the effect of a major core change or annealing of the vessel.

5.3.1.6.7 Irradiation Effects Prediction Basis

The irradiation effects prediction basis used to originally select the surveillance plate is shown on Figure 5.3-5. (It is no longer used as a prediction basis, but is maintained as historical information.) This figure was developed using transition temperature shift data for SA 533 B Class 1 plate typical of that used in the fabrication of the reactor vessel beltline materials. This method is more conservative than Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," April 1977 (R1), which was in effect at the time, throughout the design lifetime of the reactor vessel.

Regulatory Guide 1.99 was revised (Rev. 2) in 1988 and has been used as the prediction basis since. These prediction bases have been used to establish the surveillance capsule withdrawal schedule (Table 5.3-9) and the pressurized thermal shock submittal for 10 CFR 50.61 (Section 5.3.1.5). It was also used to predict the radiation induced change in reference temperature for the reactor vessel beltline as part of the determination of the heatup and cooldown operating limit curves (see Subsection 5.3.2). Once actual post-irradiation surveillance data becomes available, these data will be incorporated, along with the current applicable prediction basis, to adjust the reactor vessel operating limit curves.

5.3.1.7 Reactor Vessel Fasteners

The stud material for the reactor vessel closure head is SA 540 Grade 324 Class III material. The nuts and washers for the reactor vessel are SA 540 Grade B23 material. These materials meet the requirements of 10 CFR 50, Appendix G and Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," October 1973 (R0). Material tests for the stud, nut and washer materials demonstrate adequate toughness in accordance with ASME Code, Section III, 1971 Edition through Summer 1972 Addenda, and an acceptable level of ultimate tensile strength which is consistent with the recommendations of Regulatory Guide 1.65 (R0) (see Table 5.2-13). Test results (Table 5.2-13) demonstrate that the stud, nut and washer material meets the 25 mil lateral expansion, 45 ft-lb criteria of 10 CFR 50 Appendix G at 10°F.

5.3.2 PRESSURE-TEMPERATURE LIMITS

All components on the Reactor Coolant System are designed to withstand the effects of cyclic loads due to Reactor Coolant System temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trips and startup and shutdown operations.

During unit startup 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.

The maximum allowable Reactor Coolant System pressure at any temperature is based upon the stress limitations to prevent brittle fracture. These limitations are derived by using the rules contained in ASME Section III including Appendix G, Protection Against Nonductile Failure and the rules contained in 10 CFR 50, Appendix G, Fracture Toughness Requirements. Compliance with the criteria in 10 CFR 50, Appendix H is discussed in Subsection 5.3.1.6. The reactor vessel beltline materials primarily establish the RCPB pressure-temperature limitations.

5.3.2.1 Limit Curves

The reactor vessel beltline material consists of six plates. The nil-ductility transition temperatures (T_{NDT}) of each plate is established by drop weight test. Charpy tests are 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 +30°F is established.

For the remaining material in the Reactor Coolant System, a limiting RT_{NDT} of +60°F is established in the transverse direction. This determination is made based on the data presented in Tables 5.2-7 through 5.2-13. No RCPB welds outside the reactor vessel are limiting for operation. The controlling items for the system P-T limit curves are the beltline region of the RV, the reactor vessel flange, and the RV flange to upper shell transition region. The beltline region has added to its initial RT_{NDT}, the predicted shift in RT_{NDT} from plant operation for the period of P-T Limit Curve applicability, and the associated margin for the prediction. This produces a much higher RT_{NDT} than is allowed anywhere else in the RCPB (maximum RT_{NDT} = 60°F).

As a result of fast neutron irradiation in the beltline region, RT_{NDT} increases with operation. The techniques used to analytically and experimentally predict the integrated fast neutron ($E \geq 1$ MeV) fluxes and their effects on the reactor vessel are described in Subsections 5.3.1.6.4 and 5.3.1.6.7.

The measured reference transition temperature shift for a surveillance specimen 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 is obtained from the measured surveillance specimen exposure by application of the calculated azimuthal neutron flux variation.

The actual shift in adjusted RT_{NDT} is established periodically during plant operation by testing of reactor vessel material surveillance capsule specimens, 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 on Figure 5.3-4. The surveillance capsules are removed from their location in the reactor vessel for testing. The data obtained is compared to that used to develop the predicted limitation curves presented in the Technical Specifications. If this information indicates differences to the existing predictions, the curves are reevaluated. 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. The adjusted RT_{NDT} corresponding to the end of the period of pressure temperature limit curve applicability is determined as an input into the calculation of the curves. The technical specification limit lines are based on the methods in the ASME Code, Section XI, Appendix G.

This page intentionally blank.

5.3.2.2 Operating Procedures

Pressure-temperature limitations and additional information are described in the plant Technical Specifications. The pressure-temperature limit curves have been prepared in accordance with Appendix G, ASME Code, Section XI. Maintenance of Reactor Coolant System pressure and temperature within these prescribed limits ensures that the integrity of the reactor coolant pressure boundary is maintained.

5.3.3 REACTOR VESSEL INTEGRITY

CE designed and fabricated the reactor vessel for St. Lucie Unit 2. CE has been involved in reactor vessel design and fabrication since the late 1950's, and this proven expertise is reflected in the St. Lucie Unit 2 reactor vessel and the satisfactory performance of a large number of reactor vessels in the operating plants.

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

5.3.3.1 Design

The applicable design code for the reactor vessel is tabulated in Table 5.2-1. The materials of construction are described below and are listed in Table 5.2-3. A schematic of the reactor vessel is shown on Figure 5.3-8. Additional design information can be found in Subsection 5.3.1.2. The reactor vessel design parameters are given in Table 5.3-9.

5.3.3.2 Materials of Construction

The reactor vessel materials of construction are listed in Table 5.2-3. The reactor vessel shell is fabricated from SA-533, Grade B, Class 1, material. This material has a tensile strength of 80 k/in.^{2} and a yield strength of 50 k/in.^{2}. This shell material responds well to quench and tempered heat treatment, which in combination with fine-grained melting practice produces high quality plate with excellent toughness properties. The nozzles, also having excellent toughness properties, are fabricated from SA-508, Class 2 forgings. The welding materials used include Mil Spec. E-18193, B-4 wire for submerged arc processes and E-8018C-3 material for manual arc processes. The stainless steel cladding utilized is nominal SA 240, Type 304.

5.3.3.3 Fabrication Methods

The reactor vessel construction is basically that of formed plates welded into cylinders and hemispherical heads. Flanges on the closure head, upper shell, and the nozzles are forgings. This typifies construction of the reactor vessel in the preceding introductory material. No special fabrication methods were used in the fabrication of the reactor vessel. The basic design and fabrication of the reactor vessel is 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 504 inches high (overall) by 172 inches inside diameter and is all welded manganese molybdenum nickel steel plate and forging construction.. The internal surfaces that are in contact with the reactor coolant are clad with 1/8 inch minimum Type 304 austenitic stainless steel and have a finish acceptable for inservice inspection of weld joints. The closure head flange and reactor vessel shell flange provide the structural rigidity necessary for bolting the head to the shell. The reactor vessel is supported by three support pads integral with the nozzle forgings.

The reactor vessel fabrication process is begun with an upper vessel assembly which consists basically 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 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.

At the same time, a lower vessel assembly is in fabrication. The lower vessel assembly is basically the lower shell and the bottom head. The lower shell is formed from three 120 degree segment plates 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 upper and lower vessel assemblies are welded together to complete the reactor vessel.

During fabrication, the closure head is considered separately since it is bolted to the reactor vessel and not joined, except for hydrostatic testing, until at the site and ready for operation.

The closure head assembly consists basically of the closure head flange, dome, torus, control element drive mechanism (CEDM) housings, and instrument nozzles. The closure head torus is constructed from four peel segments formed from plate material. The closure head dome, torus, and flange are welded together to form the closure head assembly. Penetrations are then machined in the closure head for 91 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 inch diameter studs which are threaded into the vessel flange and extend through the closure head flange.

The original Reactor Vessel Closure Head (RVCH) supplied by Combustion Engineering is described in the paragraph above. However, the original RVCH has been replaced with a replacement RVCH manufactured by AREVA NP Inc. The replacement RVCH is a uni-block (one-piece) forging in which the head dome and the head flange are integral with no connecting welds. After the head is forged it is machined to design dimensions and the interior surface is clad with stainless steel weld overlay. Penetrations are then machined in the closure head for 91 CEDM nozzles, 10 instrumentation nozzles and one vent nozzle. The RVCH flange mating surface is machined and 54 stud holes are machined to match and accommodate the 54 seven inch diameter studs which are threaded into the Reactor Vessel (RV) and extend through the RVCH to facilitate attachment of the RVCH to the RV. Attachment (welding) of the CEDM, Instrumentation, and vent nozzles complete the RVCH assembly.

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

5.3.3.4 Inspection Requirements

Inspection requirements of ASME Code, Section III, 1971 Edition including Summer 1972 Addenda are discussed in Subsection 5.3.1.3.

5.3.3.5 Shipment and Installation

The reactor vessel was shipped by barge to the site, while mounted on the shipping skid used for installation. The reactor vessel was protected by closing all openings (including the top of the vessel) with wooden shipping covers.

The closure head was shipped with separate skids and metal covers. Reactor vessel surfaces and covers were 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.

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

The reactor vessel surveillance program is described in detail in Subsection 5.3.1.6. The program is designed on the basis of 10 CFR 50, Appendix H and ASTM E185-73. When combined with the use of highly radiation resistant materials in the beltline of the reactor vessel, this surveillance program provides maximum assurance, consistent with commercial requirements, of the integrity of the reactor pressure vessel in terms of strength and fracture resistance.

5.3.3.7.2 In-service Inspection

In-service inspection requirements of ASME Code, Section XI, as governed by 10 CFR 50.55a(g), are as indicated in Subsection 5.2.4.

5.3.3.8 Reactor Vessel Thermal Shock Analysis

In response to the NUREG-0737 Item II.K.2.13 requirement to perform an analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater, FP&L participated in CE Owners Group activities which resulted in submittal to the NRC staff of review report CEN-189[3]. This report provided the required analysis, describing the analytical methods used in all analytical evaluations and reporting plant-specific analysis results in separate appendices. Appendix (F) of CEN-189 contains results specifically for the St. Lucie Unit 2 reactor vessel.

The results contained in report CEN-189 and in Appendix (F) demonstrate that the St. Lucie Unit 2 reactor vessel can safely withstand a small break loss-of-coolant accident with extended loss of feedwater for the full design life without crack initiation.

SECTION 5.3: REFERENCES

1. "C-E Procedure for Design, Fabrication, Installation and Inspection of Surveillance Specimen Holder Assemblies," Combustion Engineering Topical Report, CENPD-155P, September 1974.

2. Letter from O.D. Parr (NRC) to F.M. Stern (CE), May 15, 1975.

3. "Evaluation of Pressurized Thermal Shock Effects Due to Small Break LOCA's with Loss of Feedwater for Combustion Engineering NSSS," CEN-189, December, 1981.

4. "Information Requested by IE Bulletin 78-12. Atypical Material in Reactor Pressure Vessel Welds," A.E. Scherer to H.D. Thornburg, LD-79-036, June 8, 1979.

5. "Analysis of Capsule W-83, Florida Power and Light Company, St. Lucie Plant Unit No. 2, Reactor Vessel Material Surveillance Program," Babcock & Wilcox Report # BAW-1880, September 1985.

6. "Analysis of Capsule 263° From the Florida Power and Light Company St. Lucie Unit 2 Reactor Vessel Material Surveillance Program," Westinghouse Report # WCAP-15040, February 2010.

TABLE 5.3-1

This table has been deleted.

TABLE 5.3-2

This table has been deleted.

TABLE 5.3-2a

ST LUCIE UNIT 2

REACTOR VESSEL BELTLINE MATERIAL PROPERTIES

Location	Code Number	Residual Cu%	Chemistry Ni%	RT(NDT)
Lower Shell Plate	M-4116-1	0.06	0.57	+20
Lower Shell Plate	M-4116-2	0.07	0.60	+20
Lower Shell Plate	M-4116-3	0.07	0.60	+20
Intermediate Shell Plate	M-605-1	0.11	0.61	+30
Intermediate Shell Plate	M-605-2	0.13	0.62	+10
Intermediate Shell Plate	M-605-3	0.11	0.61	0
Girth Weld	101-171	0.07	0.08	-70
Lower Shell Long Seam Welds	101-142A	0.04	0.10	-50
	101-142B	0.05	0.09	-50
	101-142C	0.04	0.09	-50
Intermediate Shell Long Seam Weld	101-124A	0.04	0.06	-50
	101-124B	0.03	0.06	-50
	101-124C	0.04	0.07	-50

TABLE 5.3-2b

FLUENCE AND RT(PTS)

1986 and 2023

ST. LUCIE UNIT 2

Location	Code Number	Fluence n/cm^2 Jan 1986	RTPTS OF Jan 1986	Fluence n/cm^2 Apr 2023	RTPTS °F Apr 2023
Intermediate Shell	M-605-1	3.06×10^{18}	125	4.79×10^{19}	177
Intermediate Shell	M-605-2	3.06×10^{18}	116	4.79×10^{19}	179
Intermediate Shell	M-605-3	3.06×10^{18}	95	4.79×10^{19}	147
Lower Shell	M-4116-1	3.06×10^{18}	90	4.79×10^{19}	114
Lower Shell	M-4116-2	3.06×10^{18}	95	4.79×10^{19}	125
Lower Shell	M-4116-2	3.06×10^{18}	95	4.79×10^{19}	125
Intermediate Shell Longitudinal Seam	101-124A	3.06×10^{18}	5	4.79×10^{19}	13
Intermediate Shell Longitudinal Seam	101-124B	3.06×10^{18}	1	4.79×10^{19}	5
Intermediate Shell Longitudinal Seam	101-124C	3.06×10^{18}	5	4.79×10^{19}	13
Lower Shell Longitudinal Seam	101-142A	3.06×10^{18}	5	4.79×10^{19}	14
Lower Shell Longitudinal Seam	101-142B	3.06×10^{18}	9	4.79×10^{19}	21
Lower Shell Longitudinal Seam	101-142C	3.06×10^{18}	5	4.79×10^{19}	13
Intermediate to Lower Girth Seam	101-171	3.06×10^{18}	-4	4.79×10^{19}	16

TABLE 5.3-3

TOTAL QUANTITY OF SPECIMENS

Quantity of Specimens

Type of Specimen	Orientation	Base Metal	Weld Metal	HAZ Metal	SRM[a]	Totals	
Drop Weight	Longitudinal	12	----	----	----	12	48
	Transverse	12	12	12	----	36	
Charpy Impact	Longitudinal	78	----	----	39	117	423
	Transverse	102	102	102	----	306	
Tensile	Longitudinal	18	----	----	----	18	126
	Transverse	36	36	36	----	108	
Totals		258	150	150	39		597

(a) Standard Reference Material

TABLE 5.3-4

TYPE AND QUANTITY OF SPECIMENS FOR UNIRRADIATED TESTS

Type of Specimen	Orientation	Base Metal	Weld Metal	HAZ Metal	SRM[a]	Totals	
		Quantity of Specimens					
Drop Weight	Longitudinal	12	----	----	----	12	48
	Transverse	12	12	12	----	36	
Charpy Impact	Longitudinal	30	----	----	15	45	135
	Transverse	30	30	30	----	90	
Tensile	Longitudinal	18	----	----	----	18	72
	Transverse	18	18	18	----	54	
Totals		120	60	60	15		255

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

TABLE 5.3-5

TYPE AND QUANTITY OF SPECIMENS FOR IRRADIATION EXPOSURE AND IRRADIATED TESTS

Type of Specimen	Orientation	Quantity of Specimens: Base Metal	Weld Metal	HAZ Metal	SRM[a]	Totals	
Charpy Impact	Longitudinal	48	----	----	24	72	288
	Transverse	72	72	72	----	216	
Tensile	Longitudinal	----	----	----	----	----	54
	Transverse	18	18	18	----	54	
Totals		138	90	90	24		342

(a) Standard Reference Material

TABLE 5.3-6

MATERIALS FOR NEUTRON THRESHOLD DETECTORS

Material	Reaction	Threshold Energy (MeV)	Half-Life
Uranium	U^{238} (n, f) Cs^{137}	0.7	30.2 Years
Sulphur	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, α) Co^{60}	7.0	5.3 years
Titanium	Ti^{46} (n, p) Sc^{46}	8.0	84 days
Cobalt	Co^{59} (n, γ) Co^{60}	Thermal	5.3 years

TABLE 5.3-7

COMPOSITION AND MELTING POINTS OF 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

TABLE 5.3-8

TYPE AND QUANTITY OF SPECIMENS CONTAINED IN EACH IRRADIATION CAPSULE ASSEMBLY

Capsule Location	Base Metal			Weld Metal		HAZ Material		SRM[d]	Total Specimens	
	Impact L[b]	Impact T[c]	Tensile	Impact	Tensile	Impact	Tensile	Impact	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

a. Deleted
b. L - Longitudinal
c. T - Transverse
d. Standard Reference Material - NUREG/CR4947, Analysis of the A302B and A533B Standard Reference Materials in Surveillance Capsules of Commercial Power Reactors

TABLE 5.3-9

CAPSULE ASSEMBLY REMOVAL SCHEDULE[e]

Location on Vessel Wall	Approximate Removal Schedule (EFPY)	Predicted Fluence (n/cm^2)	Lead Factor[d]
83°	1.1[a]	1.47×10^{18}	1.33
97°	26	2.3×10^{19}	1.33
104°	Standby[c]	---	0.99
263°	11[b]	1.05×10^{19}	1.34
277°	44/(EOL)[c]	4.5×10^{19}	1.33
284°	Standby[c]	---	0.99

a. Actual removal time (Reference 6).

b. Actual removal time (Reference 6).

c. As required by ASTM E185, one standby capsule will be removed at the end of license fluence and available for testing.

d. Lead Factor is defined as the capsule fluence/RV base metal peak fluence (Reference 6).

e. Capsule removal schedule changes required NRC approval per 10 CFR 50, Appendix H.

TABLE 5.3-9a

ST. LUCIE UNIT 2 SURVEILLANCE PROGRAM

Capsule No.	Azimuthal Location	Withdrawal Schedule EFPY	Lead Factor	Surveillance Materials	Specimen Type No. & Orientation		Chemical Composition	
1	83°	See Table 5.3-9	<1.5	1. Plate M-605-1	12	CVN-L	0.11	Cu
					12	CVN-T	.008	P
					3	Tensile		
				2. Weld Metal* Linde 124 Lot No. 0951 Mil B-4 Wire Heat Heat No. 83637	12	CVN	0.07	Cu
					3	Tensile	.009	P
				3. HAZ material Plate M-605-1	12	CVN-T	0.11	Cu
					3	Tensile	.008	
2	97°	See Table 5.3-9	≤1.5	1. Plate M-605-1	12	CVN-L	0.11	Cu
					12	CVN-T	.008	P
					3	Tensile		
				2. Weld Metal Linde 124 Flux Lot No. 0951 Mil B-4 Weld Wire Heat No. 83637	12	CVN	0.07	Cu
					3	tensile	.009	P*
				3. HAZ material Plate M-605-1	12	CVN-T	0.11	Cu
					3	ensile	.009	P
3	104°	See Table 5.3-9	≤1.5	1. Plate M-605-1	12	CVN	0.11	Cu
					3	Tensile	.008	P
				2. Weld Metal* 124 Flux, Lot 0951 Mil B-4 Weld Wire Heat No. 83637	12	CVN	0.07	Cu
					3	Tensile	.009	P
				3. HAZ material Plate M-605-1	12	CVN	0.11	Cu
					3	Tensile	.008	P
				4. SRM Material, HSST Plate 01	12	CVN-L	Ref.	C**

*Weld Metal specimens are fabricated from the same heat of wire and lot of flux as was used in the weld seam No. 101-171 (closing girth seam weld)
** Ref. C: "Heavy Section Steel Technology Program, Semi-Annual Progress Report for period ending 8/31/81" ORNL - 4377.

TABLE 5.3-9a (Con't)

Capsule No.	Azimuthal Location	Withdrawal Schedule EFPY	Lead Factor	Surveillance Materials	Specimen Type No. & Orientation	Chemical Composition
5	277°	See Table 5.3-9	≤1.5	1. Plate M-605-1	12 CVN-L 12 CVN-T 3 Tensile	0.11 Cu .008 P
				2. Weld Metal* Linde 124 Flux Lot No. 0951 Mil. B-4 Wire Heat No. 83637	12 CVN 3 Tensile	0.07 Cu .009 P
				3. HAZ Material Plate M-605-1	12 CVN 3 Tensile	0.11 Cu .008 P
4.	263°	See Table 5.3-9	≤1.5	1. Plate M-605-1	12 CVN-T 3 Tensile	0.11 Cu .008 P
				2. Weld Metal* Linde 124 Lot 0951 Mil. B-4 Wire Heat No. 83637	12 CVN 3 Tensile	0.11 Cu .008 P
				3. HAZ Material Plate M-605-1	12 CVN 3 Tensile	0.11 Cu .008 P
				4. SRM HSST Plate 01	12 CVN	Ref. C**
6.	284°	See Table 5.3-9	≤1.5	1. Plate M-605-1	12 CVN-L 12 CVN-T 3 Tensile	0.11 Cu .008 P
				2. Weld Material* Linde 124 Lot 0951 Mil B-4 Heat 83637	12 CVN 3 Tensile	0.07 Cu .009 P
				3. HAZ Material Plate M-605-1	12 CVN 3 Tensile	0.11 Cu .008 P

* Weld Metal specimens are fabricated from the same heat of wire and lot of flux as was used in the weld seam No. 101-171 (closing girth) seam weld)

** Ref. C: "Heavy Section Steel Technology Program, Semi-Annual Progress Report for period ending 8/31/81" ORNL - 4377.

TABLE 5.3-10

REACTOR VESSEL DESIGN PARAMETERS

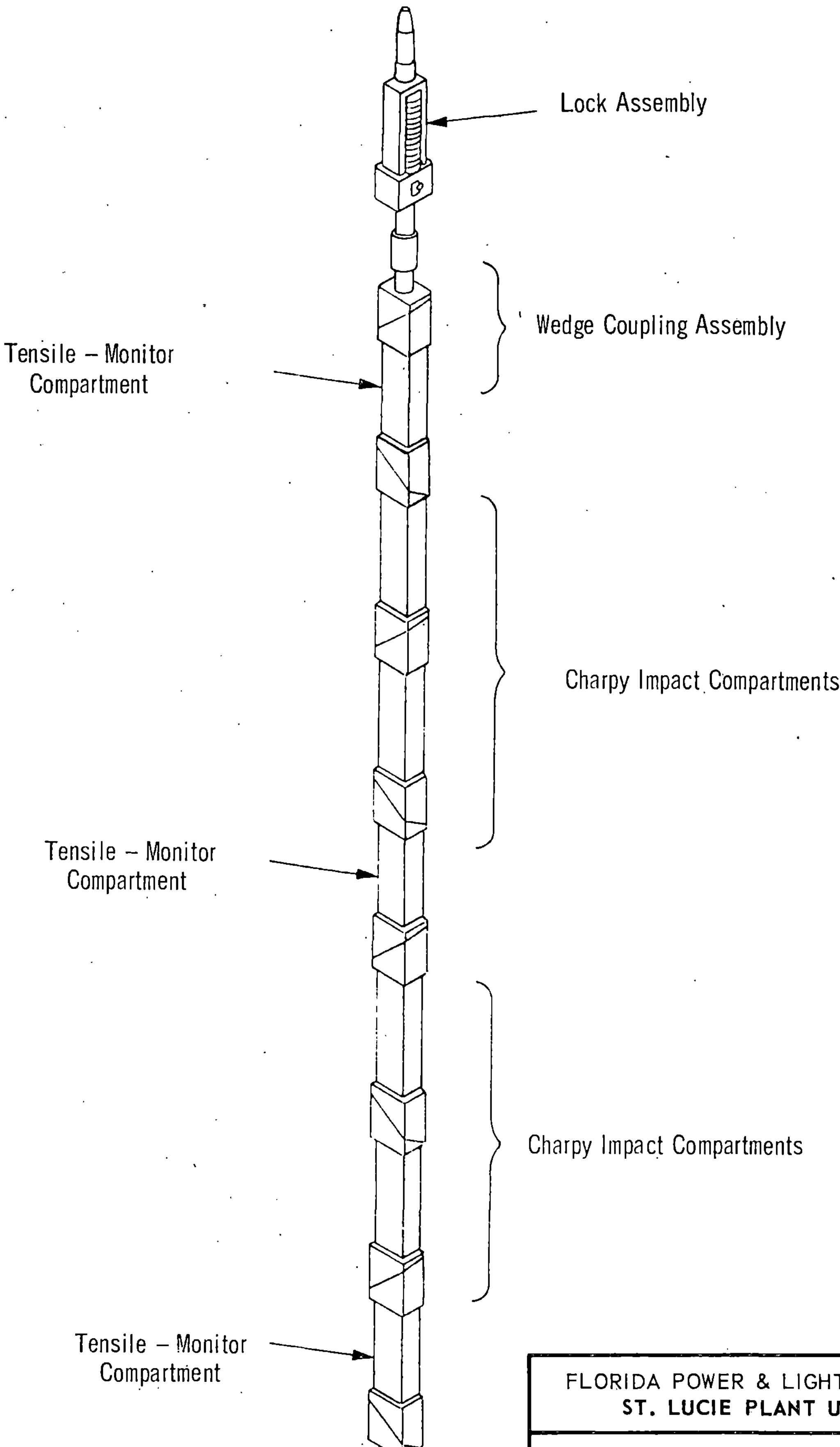
Design Pressure, psia	2500
Design Temperature, F	650
Nozzles	
Inlet (4 ea), ID, in.	30 nominal
Outlet (2 ea), ID, in.	42 nominal
CEDM (91) ID, in.	2.728
Instrumentation (10), ID, in.	4.625
Vent (1), nominal, in.	3/4
Dimensions	
Inside Diameter, nominal, in.	172
Overall Height, Including CEDM Nozzles, in.	502 3/8
Height, Vessel without Head, in.	408 9/16
Wall Thickness, minimum, in.	8 5/8
Upper Head Thickness, minimum, in.	7 1/2
Lower Head Thickness, minimum, in.	4 3/8
Cladding Thickness, minimum, in.	1/8
Cladding Thickness, nominal, in.	7/32
Dry Weights	
Head, lb	151,299
Vessel, without flow skirt, lb.	654,890
Studs, Nuts, and Washers, lb.	37,838
Flow skirt, lb.	3,509
Total	847,536

TABLE 5.3-11

REPLACEMENT REACTOR VESSEL CLOSURE HEAD NON-DESTRUCTIVE TEST

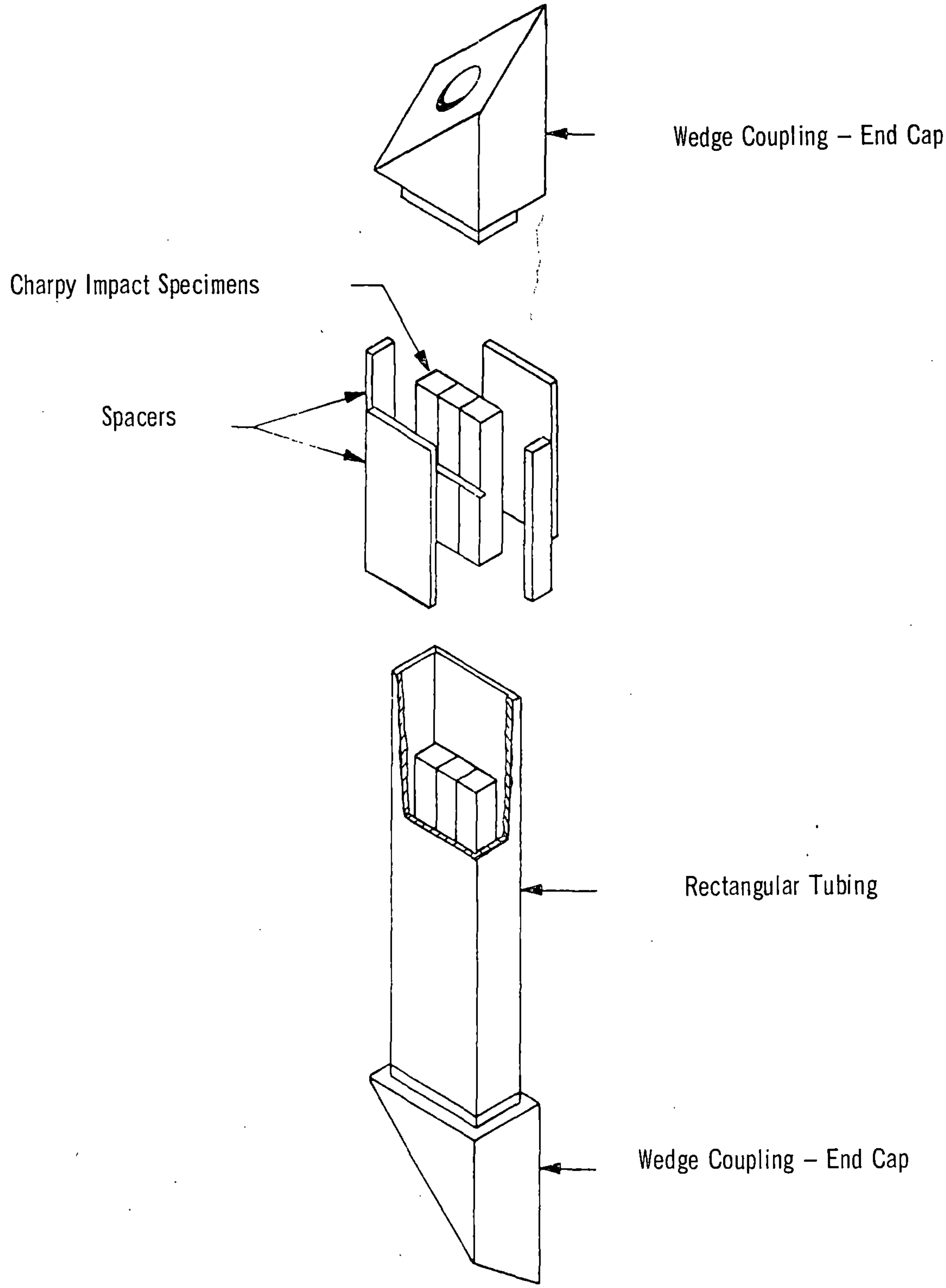
	RT	UT	PT	MT
1. RVCH Mono-Block Forging				
1.1 After Rough Machining		YES		YES
1.2 After Final Machining		YES		
1.3 Machined Surfaces To Be Clad			YES(4)	
1.4 External Unclad Surfaces				YES(1)
1.5 All Clad Surfaces		YES(2&3)	YES(5)	
1.6 Final Machined O-Ring Groove			YES(6)	
2. SB-167 UNS 6690 CEDM and ICI Tubing		YES(10)	YES(7)	
3. SB-166 UNS 6690 CEDM Nozzles		YES(10)	YES(7)	
4. SA-479 Type 304 ICI Nozzle Adapter		YES(10)	YES(7)	
5. Weldment				
5.1 All Weld Prep Areas			YES(8)	
5.2 Root Pass Of All Welds			YES	
5.3 Final Surface Of All Welds			YES(9)	
5.4 Nozzle To Forging Area		YES(11)		
5.5 CEDM and ICI Nozzle to Adapter Butt Weld	YES			
5.6 Vent Pipe Nozzle To Vent Pipe Butt Weld	YES			

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



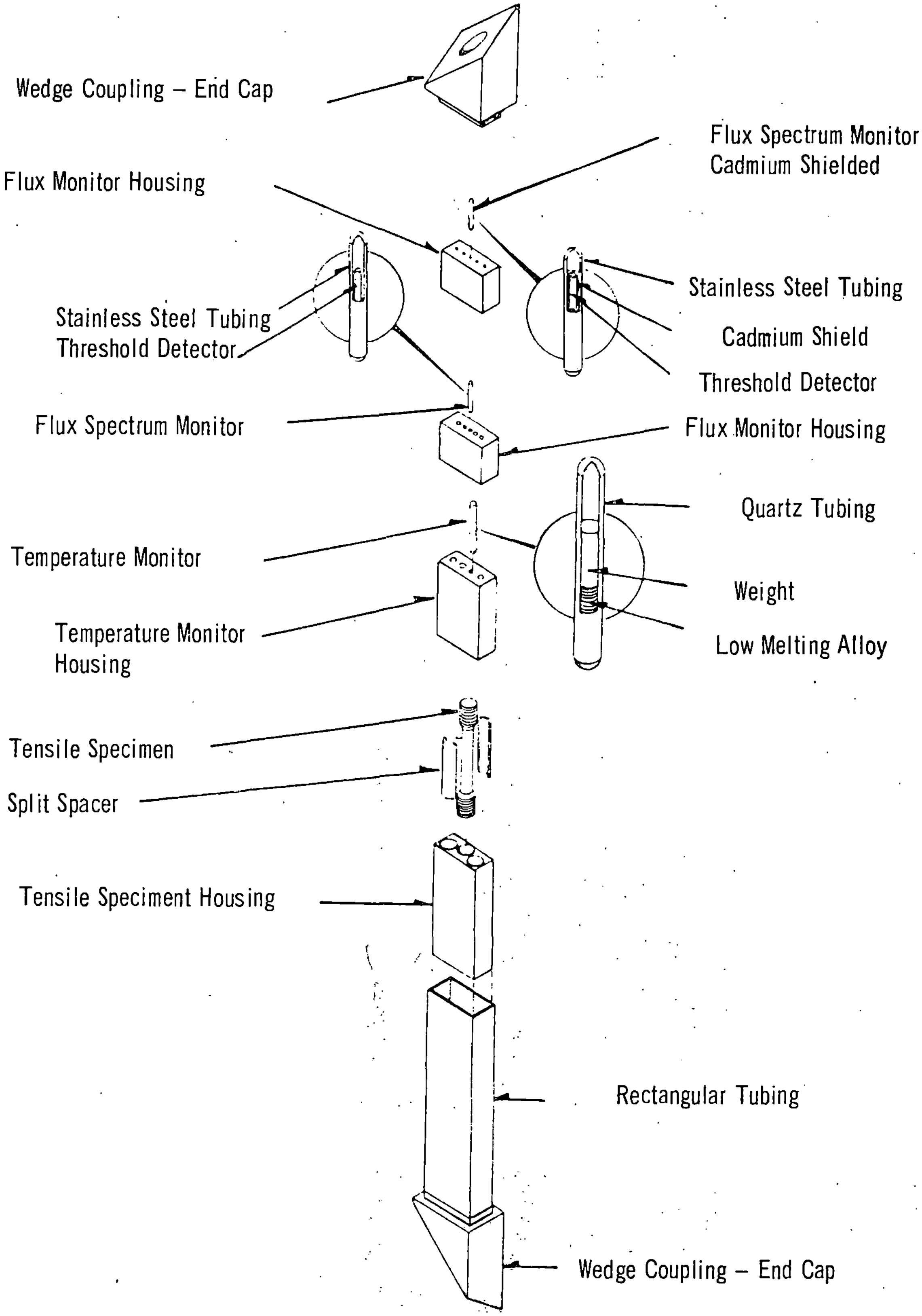
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

TYPICAL SURVEILLANCE CAPSULE ASSEMBLY
FIGURE 5.3-1



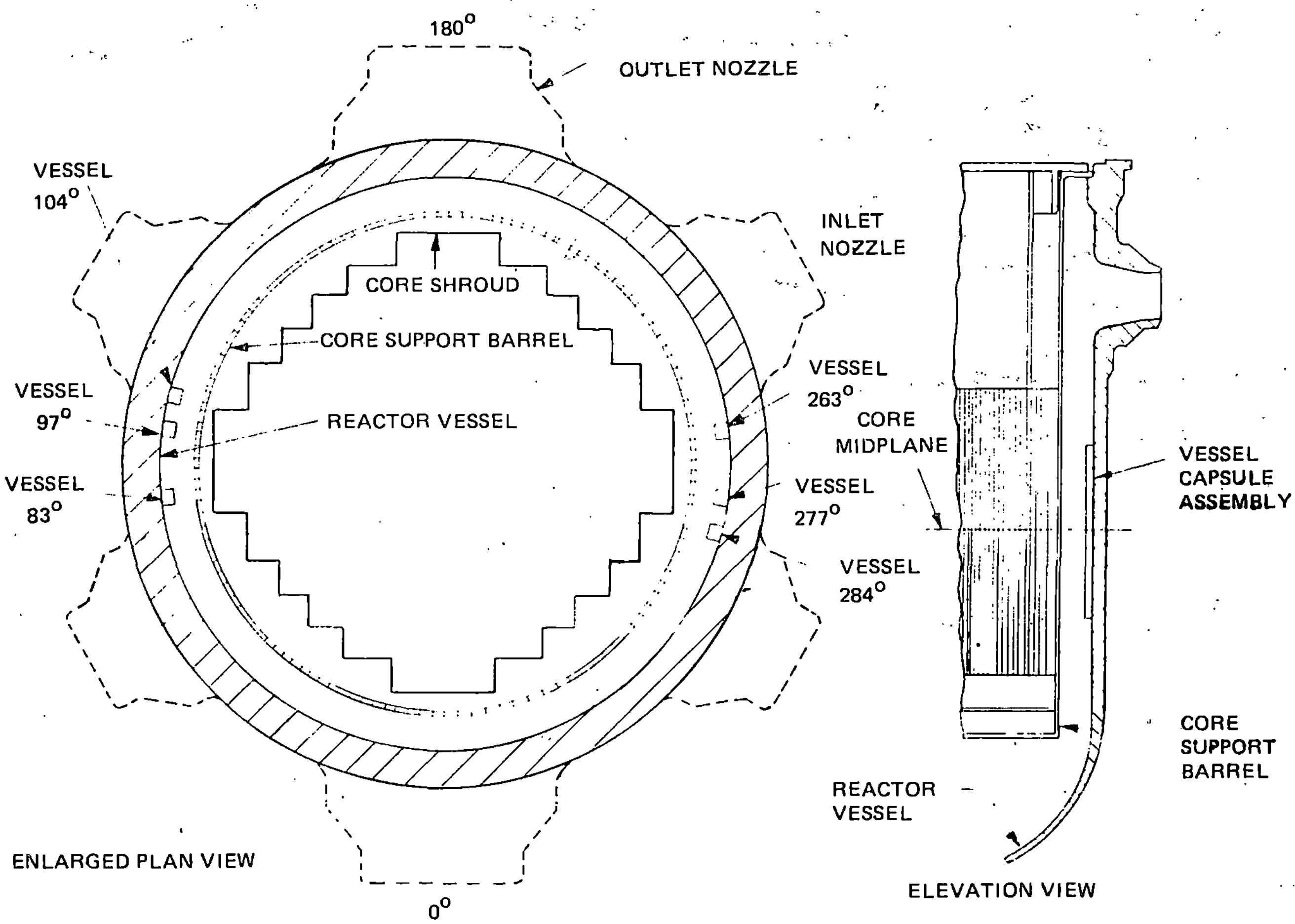
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

TYPICAL CHARPY IMPACT COMPARTMENT ASSEMBLY
FIGURE 5.3-2



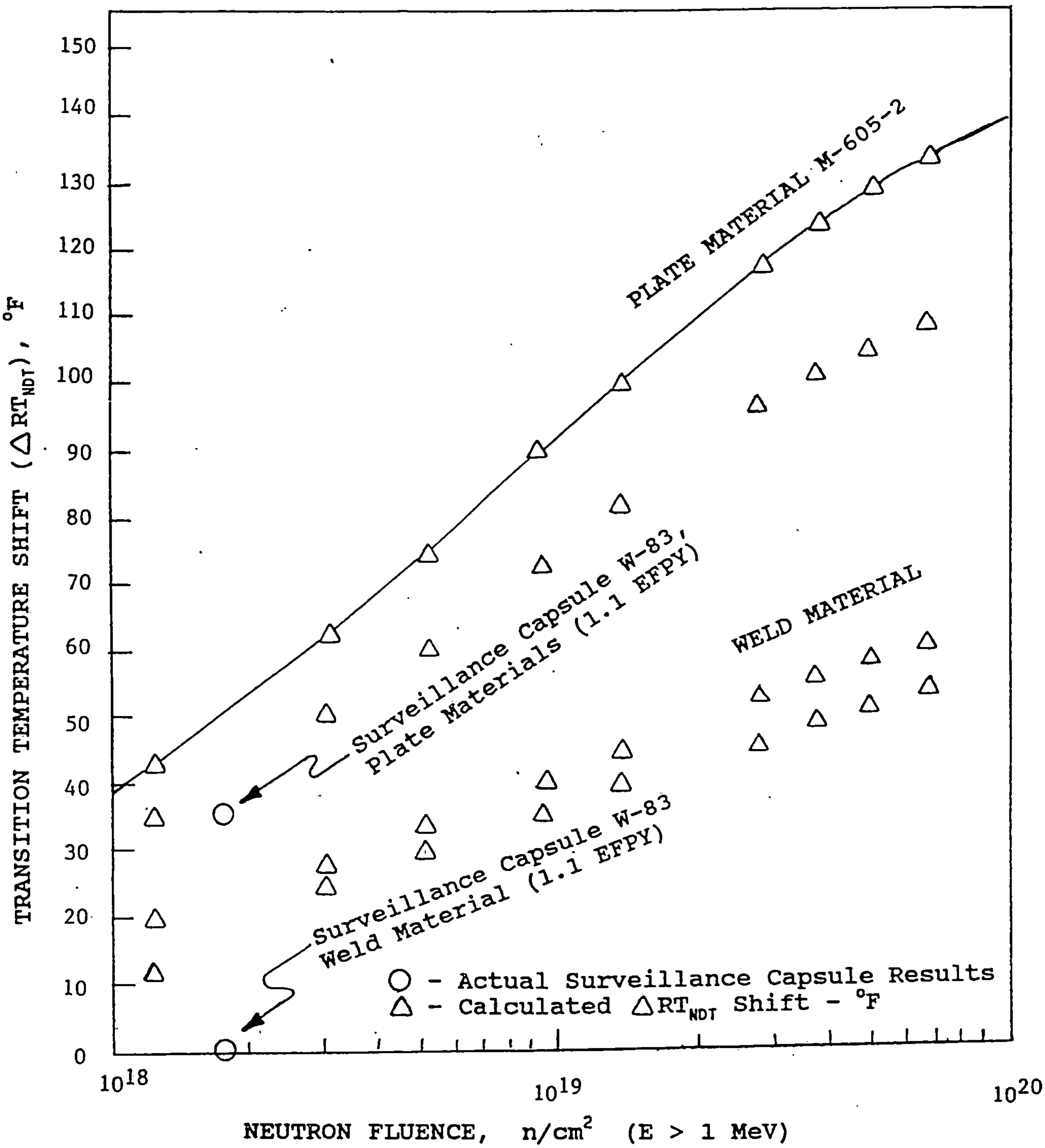
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

TYPICAL TENSILE -- MONITOR COMPARTMENT DETAIL
FIGURE 5.3-3



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

LOCATIONS OF SURVEILLANCE CAPSULE ASSEMBLIES
FIGURE 5.3-4



Note: This data is historical and no longer used. Refer to Section 5.3.1.6.7 for current prediction basis.

Amendment No. 12 (12/98)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

TRANSITION TEMPERATURE SHIFT
FOR ST. LUCIE UNIT 2
PLATE AND WELD METAL

FIGURE 5.3-5

DELETED

AMENDMENT NO. 19 (06/09)

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 2

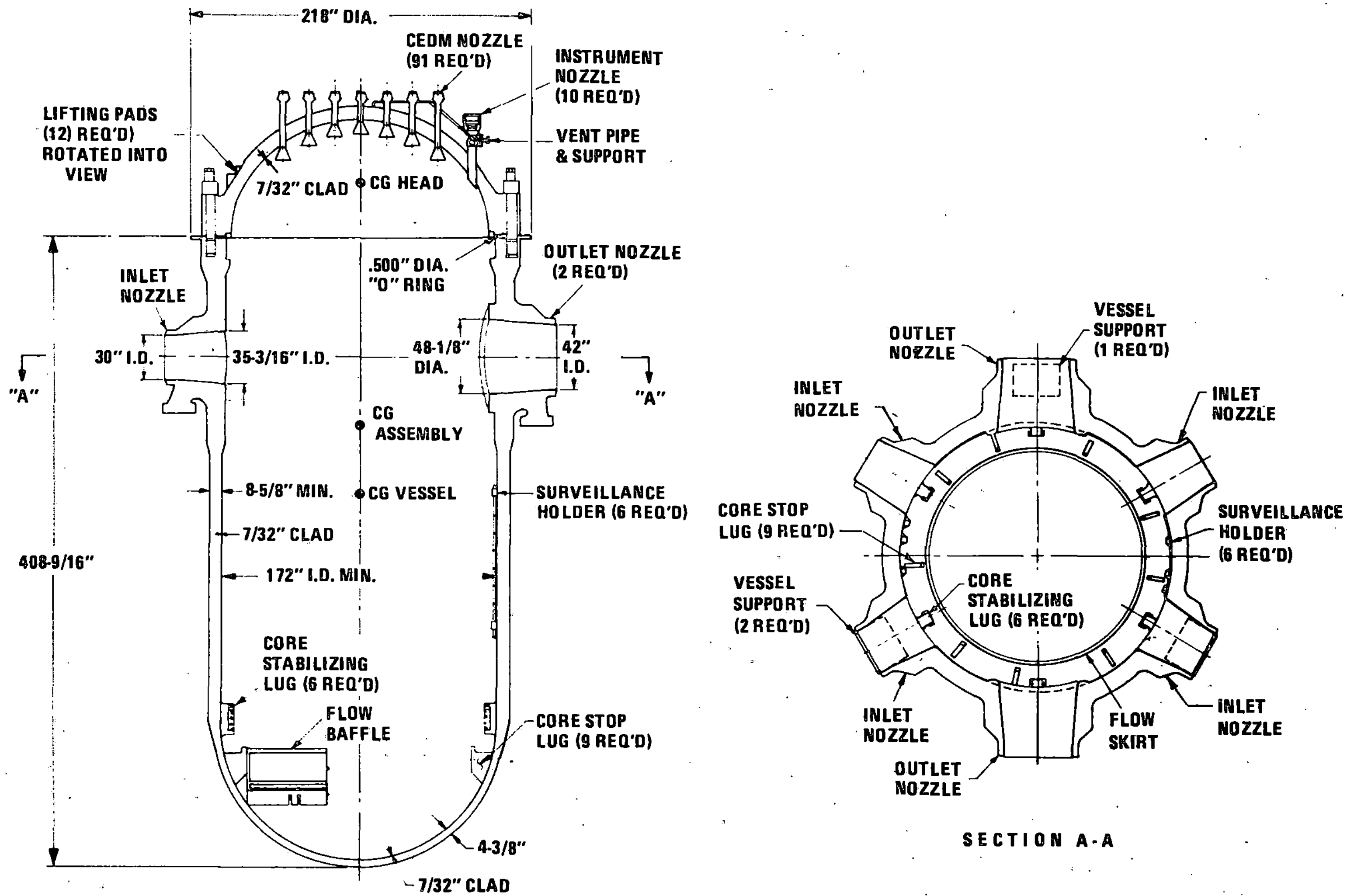
FIGURE 5.3-6

DELETED

AMENDMENT NO. 19 (06/09)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FIGURE 5.3-7



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR VESSEL

FIGURE 5.3-8

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR COOLANT PUMPS

5.4.1.1 Design Bases

The reactor coolant pumps provide sufficient forced circulation flow through the Reactor Coolant System to assure adequate heat removal from the reactor core during power operation. A low limit on the reactor coolant pump flowrate is established to assure that specified fuel design limits are not exceeded. Design flow is derived on the basis of the thermal-hydraulic considerations presented in Section 5.1.

The reactor coolant pump and motor assembly, in conjunction with the fly-wheel, provide sufficient coast down flow following loss of power to the pumps to assure adequate core cooling.

The reactor coolant pump pressure boundary is designed for the transients given in Subsection 3.9.1.1 so that the ASME Code, Section III allowable stress limits are not exceeded for the specified number of cycles.

The reactor coolant pump parameters and design requirements are listed in Table 5.4-1.

5.4.1.2 Description

The reactor coolant is circulated by four vertical, single bottom suction, horizontal discharge, centrifugal, motor driven pumps as shown in Figure 5.4-1. The design parameters for the pumps are given in Table 5.4-1. The piping and instrumentation diagram for the reactor coolant pump is shown in Figure 5.1-6. The pump performance curve is shown in Figure 5.4-2.

5.4.1.2.1 Reactor Coolant Pump Assembly

The reactor coolant pump assembly consists of the following:

a) Pump Case and Motor Mount

b) Rotating Assembly
(Containing the impeller with a welded impeller locknut)

c) Shaft Seal 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 cartridge replacement. This assembly 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 impeller, water lubricated radial hydrostatic bearing, rotor, seal coolant recirculating impeller and rotating elements of the seal cartridge assembly. The radial bearing, one of three used for pump motor shaft support, is located just above the pump impeller. The upper radial bearing and the axial thrust bearing are located on the motor shaft. The seal cartridge and recirculating impeller are located above the thermal barrier formed by the close clearance between the pump shaft and the pump case cover.

5.4.1.2.1.3 Shaft Seal Assembly

The seal cartridge consists of four-multiface type mechanical seals; three full pressure seals mounted in tandem and a fourth low pressure vapor seal designed to withstand system design pressure when the pumps are 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 controlled bleedoff flow is collected in the volume control tank of the Chemical and Volume Control System. Leakage past the vapor seal is collected in the Waste Management System through drainage via floor drains to the containment sump.

The seal cartridge assembly is cooled by circulating the controlled leakage through a coiled tube heat exchanger cooled by component cooling water and integral with the pump case cover. The seal coolant recirculation is done by the recirculating impeller located directly below the seal cartridge. The seal cartridge concept reduces the time required for seal maintenance thereby lowering personnel radiation exposure time. The seal cartridge can be removed without draining the pump case. Details of the seal cartridge are shown in Figure 5.4-3.

The seal design life is at least two years. Each seal is designed to accept the full operating pressure of the Reactor Coolant System. The first three seals of the cartridge assembly normally operate with a pressure differential equal to one-third of the operating pressure and with only a slight pressure differential across the vapor seal. The seal rotors are titanium carbide operating against a hard carbon faced stator.

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

The heat exchanger cooling water is supplied from the Component Cooling Water System. Two 10 hp ac oil lift pumps are used to support the pumpmotor shaft assembly during startup and shutdown of the reactor coolant pumps. The motor-pump hearing support system includes a Kingsbury double acting thrust bearing, upper and lower radial bearings in the motor and a radial hydrostatic bearing located above the pump impeller. The piping and instrumentation diagram for the lube oil and cooling system of the pumps is shown on Figure 5.1-6. The flywheel and motor-pump rotating assembly has a minimum total moment of inertia of 100,000 lb-ft^2 to improve pump coastdown characteristics in order to meet system requirements during a loss of pump power condition.

Each pump-motor assembly is equipped with an anti-rotation device shown in Figure 5.4-4 to preclude reverse rotation caused by backflow through the impeller. The device stops the pump when it decelerates from normal speed (900 rpm) to zero speed while the remaining reactor coolant pumps continue to operate. The anti-reverse device consists of a rotating disc keyed to the motor shaft, and a stationary disc which is bolted to the motor frame. The stationary disc contains several detents each with ramped sides and flats on top of the detents and in the troughs between them. The rotating elements contain several holes in which the retaining pins are located. When reactor coolant pump rotation stops, each pin drops to the flat between detents, and reverse rotation is prevented by the pin which bears against the vertical side of a detent. When motor rotation is started in the normal direction, the pins ride up the ramped sides of the detents and are locked against the sides of the holes in the rotating disc by centrifugal force. No parts are in contact when the motor is operating at rated speed and no lubrication is required for the device. One pin is capable of holding the pump stationary against the torque produced by reverse flow or by the application of 100 percent voltage in reversed phase rotation.

The reactor coolant pump motor is sized for continuous operation at the flows resulting from four-pump operation or partial pump operation with 0.74 specific gravity water. The motor service factor is sufficient to allow 500 heatup cycles. The motors are designed to start and accelerate to speed under full load when 80 percent or more of normal voltage is applied. The motors are contained within NEMA Standard 1-1.20 drip-proof enclosures and are equipped with electrical insulation suitable for a zero to 100 percent humidity and radiation environment of 30R/hr of gamma. The motor cross section is shown in Figure 5.4-5.

5.4.1.2.1.5 Reactor Coolant Pump Lube Oil Collection System

A Reactor Coolant Pump Lube Oil Collection System is provided to meet the following criteria:

a) Capable of collecting lube oil from all potential pressurized and unpressurized leakage sites in the reactor coolant pumps' lube oil systems,

b) Capable of draining lube oil from the collection system at the pumps to a safe location at a rate in excess of the largest anticipated leak in the lube oil systems,

c) Seismically analyzed to insure the system will remain on the reactor coolant pump motors during design basis earthquake conditions, and

d) Capable of collecting 225 gallons of lube oil. This quantity is in excess of the quantity which would require pump shutdown (approximately 15 gallons for each pump). Also, the capacity is in excess of the entire volume of the lube oil contained in one reactor coolant pump (190 gallons).

5.4.1.3 Evaluation

The reactor coolant pumps are sized to deliver flow that equals or exceeds the design flowrate 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 are performed to evaluate reactor coolant pump performance during the post core load hot functional testing to verify adequate flow.

Leakage from the reactor coolant pump past the pump shaft is controlled by the shaft seal assembly. Reactor coolant entering the seal chambers is cooled and collected in closed systems so that reactor coolant leakage to containment is essentially zero. In the event of a seal malfunction, instrumentation in the form of pressure transmitters, a flow meter, and a temperature detector is provided to alert the operator to a potential problem.

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 doses comparable to the guidelines established for design basis accidents. Low component cooling water flow to each pump is indicated and alarmed in the control room. The component cooling water flow from the reactor coolant pumps is sensed by four separate redundant transmitters and low flow is indicated and alarmed in the control room. If the component cooling water flow from the reactor coolant pumps is not restored in 10 minutes, the system automatically trips the reactor and the reactor coolant pumps could be tripped by the operator manually, allowing the system to be cooled down by natural circulation flow.

The reactor coolant pumps, by design and field experience, are not susceptible to seal failure resulting from loss of seal cooling water. The reactor coolant pumps are equipped with four series-arranged face seals, all of which are designed for 2500 psid. The 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 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. This 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, the seal leakage should not exceed the normal operating seal leakage.

The seals have been specified and tested for 10 minutes of RCP operation without cooling water to the RCP seals without incurring seal damages. See Subsection 9.2.2.3.1

In the event of an actuation of containment isolation and subsequent isolation of CCW, the reactor coolant pumps are tripped, as discussed above, resulting in no requirement for component cooling water.

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 following design conditions and material specifications for the flywheels are consistent with the recommendations of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," October 1971 (R0). Technical Specifications require a program for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

5.4.1.4.1 Flywheel Material Specification

The material used to manufacture the flywheel was produced by a process that minimizes flaws by a commercially acceptable process such as the vacuum melt and degassing process which provides adequate fracture toughness properties. The acceptance criteria for flywheel design is compatible with the safety philosophy of the reactor coolant pressure boundary criteria as appropriate considering the inherent design and functional requirement differences between the pressure boundary and the flywheel.

a) The nil-ductility transition temperature (NDTT) of the material, as obtained from the dropweight tests (DWT) performed in accordance with the Specification ASTM E208-66T was no greater than 10°F.

b) The Charpy V-Notch (Cv) upper shelf energy level, in the "weak" (WR) direction, as obtained per ASTM A370 was no less than 50 ft.-lbs. A minimum of three Cv specimens were tested from each plate or forging.

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) of at least 100 Ksi $\sqrt{in}$. Compliance was demonstrated by either of the following:

   1) Testing of the actual material of the flywheel to establish the K_{Ic} (dynamic) value at the normal operating temperature.

2) Use of a lowerbound fracture toughness curve obtained from tests on the same type of material. The curve was translated along the temperature coordinate until the K_{Ic} (dynamic) value of 45 $Ksi\sqrt{in.}$ is indicated at the NDTT of the material, as obtained from dropweight tests.

d) Each finished flywheel was subjected to a 100 percent volumetric ultrasonic inspection from the flat surface per ASME Code, Section III. This inspection was performed on the flywheel after final machining and overspeed test.

e) The flywheel is flame cut; at least 1/2 in. of stock was left on the outer and bore radii for machining to final dimensions.

f) The flywheel was subjected to a magnetic particle of liquid-penetrant examination per ASME Code, Section III before final assembly. The inspection was performed on finished machine bores, key ways, and on both flat surfaces to a radial distance of eight in. minimum beyond the final largest machined bore diameter but not including small drilled holes. There are no stress concentrations such as stamp marks, center punch marks, or drilled or tapped holes within eight in. of the edge of the largest flywheel bore.

5.4.1.4.1.2 Flywheel Design Criteria

The flywheel is designed to withstand normal operating conditions, anticipated transients, and the design basis loss of coolant accident loadings combined with the safe shutdown earthquake loadings.

The following criteria are satisfied:

a) The combined stresses, both centrifugal and interference, at normal operating speed do not exceed 1/3 of the minimum specified yield strength for the material selected in the direction of maximum stress.

b) The design speed of the flywheel is 125 percent of normal operating speed.

c) The combined centrifugal and interference stresses at design speed are limited to 2/3 of the minimum specified yield strength where design overspeed is 125 percent of normal operating speed.

d) The motor and pump shaft and bearings can withstand any combination of normal operating loads, anticipated transients, and the design basis loss of coolant accident combined with the safe shutdown earthquake.

e) Each flywheel was tested at design speed, 125 percent of normal operating speed, as defined in 2.b above.

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

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 in the control room. During initial reactor coolant pump testing, the highest reading thermocouple is selected for this temperature measurement channel. High temperature is detrimental to motor winding insulation life, and may be caused by high ambient temperature, reduction in the cooling air flow to the stator, or inadequate time delay between successive starts of the motor.

5.4.1.5.1.2 Motor Bearing Temperatures

Temperature indication for the following oil lubricated bearings is provided in the control room.

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 thrust bearings are also provided with high temperature alarms.

5.4.1.5.1.3 Pump Controlled Bleedoff Temperature

The temperature of the controlled bleedoff flow is provided. 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 Seal Water Heat Exchanger Outlet Temperature

Temperature sensors (TE-1151, 1161, 1171, 1181) are provided at the seal water outlet. A high temperature condition is an indication that the cooler has developed a leak or that the component cooling water flow has decreased. High temperatures are alarmed.

5.4.1.5.2 Pressure

5.4.1.5.2.1 Reactor Coolant Pump Seal Pressures

The middle seal, upper seal and controlled bleedoff cavities in each reactor coolant pump are provided with pressure transmitters that generate a signal proportional to the pressure within the cavity. High and low pressure alarms for the upper seal cavity and the controlled bleedoff cavity are provided. Pressure indication of all three is also provided.

5.4.1.5.2.2 High Pressure Oil Lift Pump Discharge Pressures

Pressure switches at each high pressure oil lift pump discharge actuate indicating lights and alarms on low pressure in the 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.

5.4.1.5.2.3 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 control room. The pump performance curve (Figure 5.4-2) relates pump differential pressure to pump flow.

5.4.1.5.2.4 Casing Main Closure Gasket Leakage Pressure

A pressure indicator and pressure switch are provided on each reactor coolant pump to monitor the pressure between the double casing main closure gaskets. High pressure in the cavity between the gaskets indicates leakage of the inboard gaskets and is alarmed in the control room.

5.4.1.5.3 Flow

5.4.1.5.3.1 Reverse Rotation Indicator Switch

A flow switch in the lube oil system mounted near the main thrust bearing bracket provides an indication that the reactor coolant pump motor is turning in the reverse direction. This switch causes an alarm in the control room. This feature has been removed from the RCP motors.

5.4.1.5.3.2 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 flowrate and annunciates high and low flow alarms.

5.4.1.5.3.3 Motor Circulating Oil System Flow

This feature has been removed from the RCP motors.

5.4.1.5.4 Level

Each RCP upper lower oil reservoir has a bubbler type level measurement system. A level transmitter is used to measure the air backpressure from each bubbler which is equivalent to the oil level in the reservoir. The excess flow valve in the bubbler limits the excess air flow to the oil reservoir. The transmitter transmits a signal for level indication and high and low alarms in the control room.

5.4.1.5.5 Vibration

Reactor coolant pump motor vibration and axial shaft position is sensed by proximity probes arranged around the motor and pump shaft. These probes are noncontacting. The probes are used to determine axial shaft position relative to the motor case (one probe), provide phase angle information (one probe) and to determine X-Y vibration at the upper motor bearing and the top of the pump mechanical seal (four probes). Pumps 2A1, 2BA2 and 2A2B1 are provided with an additional set of X-Y probes which are installed at the lower motor bearing. These are available for maintenance and troubleshooting and are not normally indicated in the control room.

Excessive X-Y vibration at the upper motor bearing and mechanical seal is alarmed in the control room.

The vendor supplied cables are routed such that they are separated from other plant cables.

5.4.1.6 Testing and Inspection

The reactor coolant pressure boundary is nondestructively inspected as required by ASME Code, Section III for Code Class 1 components. The reactor coolant pump casing inspections include complete radiography and liquid penetrant or ultrasonic testing. The reactor coolant pump receives a hydrostatic pressure test in the vendor's shop and with the Reactor Coolant System. In-service inspection of the reactor coolant pump pressure boundary is performed during plant life in accordance with ASME Code, Section XI.

The reactor coolant pump assembly is performance tested in the vendor's shop over at least the normal operating range in accordance with the Standards of the Hydraulic Institute. Tests also demonstrate the ability of the reactor coolant pump to function under various operating conditions specified. Tests commonly performed are hot and cold performance and start-stop cycling. Vibrations are monitored at several places on the reactor coolant pump during shop testing.

The reactor coolant pump motors undergo a "routine" test in accordance with NEMA MG-1. This test also confirms that the motors are within their vibration limits. Each motor is tested further by being used as the driver for the reactor coolant pump assemblies during the pump manufacturer's shop testing.

To the greatest extent practicable, all conditions of normal operation of the reactor coolant pumps are duplicated during testing.

The reactor coolant pump flywheel inspections and testing are described in Subsection 5.4.1.4.

5.4.2 STEAM GENERATORS

5.4.2.1 Design Bases

The two steam generators are designed to transfer 2710 MWt from the Reactor Coolant System to the Main Steam System, producing approximately 11.8×10^6 lb/h of 896.9 psia saturated steam, when provided with 435°F feedwater. With implementation of the extended power uprate (EPU), it has been verified that the two steam generators have the capability to transfer 3034 MWt, producing approximately 13.3×10^6 lb/h of 886.2 psia saturated steam, when provided with 436°F feedwater. Moisture separators and steam driers in the shell side of the steam generator limit the moisture content of the steam to 0.10 wt% 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 Reactor Coolant System 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: (Note: differences exist between the cycles and transients assumed in the design of Unit 1 and those assumed in the design of Unit 2. Further, there may also be unit differences with respect to those cycles and transients required by plant procedure to be tracked).

a) Ten secondary side hydrostatic tests with secondary side pressurized to 1250 psia with the primary side at atmospheric pressure. The minimum shell side temperature for this test is 70°F.

b) Two hundred secondary side leak tests with the secondary side pressurized from 820 psia to design pressure, with the primary side pressurized so that the tube differential pressure (secondary to primary) does not exceed 820 psi (test condition). The secondary side temperature shall be 70-200°F.

c) Fifteen thousand cycles of adding 70°F feedwater at 600 gpm to each of the steam generators through the main feedwater nozzle when at hot standby conditions (normal condition). The basis is nominal operating conditions assuming intermittent feeding of the steam generators.

d) Eight cycles of adding 32°F feedwater at 650 gpm to each of the steam generators after a loss of normal feedwater. This feedwater flow may be introduced while the secondary side is dry: at 610°F and atmospheric pressure.

e) Four thousand pressure transients of 85 psi across the primary divider plate in either direction caused by starting and stopping reactor coolant pumps (normal condition).

5.4.2.1.1 Steam Generator Materials

The pressure boundary materials used in the construction of the steam generator are listed in Table 5.2-3. These materials are in accordance with the ASME Code, Section III plus code case interpretations as specified in Subsection 5.2.1.

The Code Class 1 components of the steam generator meet the fracture toughness requirements of the ASME Code and 10 CFR 50, Appendix G as discussed in

Subsection 5.2.3.3.1. Fracture toughness data for the steam generator materials is presented in Tables 5.2-8, 5.2-10, and 5.2-11.

During final assembly and shipment, the steam generator primary and secondary sides are brought to a state of cleanliness consistent with the rest of the fluid system in interfaces with as described in Subsection 5.2.3. In addition, the interior of the steam generator is protected by pressurizing with an inert gas during shipment and interim storage. Cleanliness during construction is discussed in Subsection 5.2.3.4.1.2.1.

The chemistry control and corrosion control effectiveness of the secondary side water is discussed in Subsection 10.3.5.

5.4.2.1.2 Steam Generator Description

The nuclear steam supply system utilizes two steam generators (Figure 5.4-6) to transfer the heat generated in the Reactor Coolant System to the secondary system. The design parameters for the steam generators are given in Table 5.4-2.

The steam generator is a vertical U-tube heat exchanger with the reactor coolant on the tube side and the secondary fluid on the shell side.

Reactor coolant enters the steam generator through the 42 inch ID inlet nozzle, flows through 3/4 inch OD 0.0429 inch wall U-tubes, and leaves through two 30 inch ID outlet nozzles. Divider plates in the lower head separate the inlet and outlet plenums. The plenums are carbon steel with stainless steel clad. The reactor coolant side of the tube sheet is Ni-Cr-Fe clad. The U-tubes are Inconel 690 composition.

The steam generator contains 8999 U-tubes for heat transfer for primary to secondary water. Each tube is expanded into the tube sheet so that there is no voids or crevices occurring along the entire length of the tube sheet interface. The tubes are also welded to the Ni-Cr-Fe alloy clad on the reactor coolant surface of the tubesheet. The tube to tubesheet welding conforms with the requirements of the ASME Code, Sections III and IX. Support for the tube bundles are by stainless steel broached tube support plates. Additional support is provided by stainless steel anti-vibration bars to prevent excessive flow-induced vibration.

Feedwater enters the steam generator through the feedwater nozzle where it distributed via a feedwater distribution ring. The feedwater ring is constructed with discharge nozzles which are configured in the form of a "J". These nozzles are welded to the top of the ring and direct the feedwater flow away from the shell. This construction greatly reduces the rate at which the ring drains, helping to provide assurance that the feedwater ring remains full of water as long as there is feedwater flow when the level in the steam generator drops below the feedwater ring (refer to Figures 5.4-6, 16, and 17).

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.

Upon leaving the vertical U-tube heat transfer surface, the steam-water mixture enters the cyclone-type separators. These impart a centrifugal motion in the mixture and separate the water particles from the steam. The water exists 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 double pocket, chevron-type dryers.

The steam generators are mounted on bearing plates which allow controlled lateral motion due to thermal expansion of the reactor coolant piping. Key stops embedded in the concrete base limit this motion in case of a reactor coolant pipe rupture. The top of each unit is restrained from sudden lateral movement by keys and hydraulic snubbers mounted rigidly in the concrete structure.

The steam generators are located at a higher elevation than the reactor vessel. The elevation difference created natural circulation capability sufficient to remove core decay heat following coast down of all reactor coolant pumps.

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 16 flanged spring loaded ASME Code safety valves which discharge to atmosphere. Overpressure protection is discussed in Subsection 5.2.2.

5.4.2.1.3 Steam Generator Tubes

The steam generator are tubed with 0.750 inch OD by .0429 wall tubes. The tubes are fabricated from Inconel 690 to insure compatibility with both the primary and secondary waters. The design incorporates a general corrosion allowance that provided for reliable operation over the plant design lifetime.

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 feedwater 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 or beavering has been eliminated by removing phosphates from the chemistry control system.

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

a) Tube Degradation Mechanisms

The design steady state and transient conditions specified in the design of the steam generator tubes are discussed in Subsection 3.9.1.1.

Alloy 690 tubes are less susceptible to various forms of the degradation.

NRC Information Notice 90-49, "Stress Corrosion Cracking of PWR Steam Generator Tubes," identified conditions of similar steam generators, where circumferential cracks were observed near the tube expansion transition at or near the top of the tubesheet.

Should a circumferential crack be detected the tube may have a stabilizer and tube plug installed. If the plugged tube severs, the stabilizer is designed to reduce the possibility of tube-to-tube contact. The stabilizers the plugs and their installation are designed to function under all operating, transient or test conditions of the steam generator. This installation takes into consideration maintaining integrity under vibrating loads and material compatibility with tube material subject to both reactor coolant and feedwater system environments.

A number of operating plants have experienced a corrosion phenomenon known as "denting".

Denting is caused by the uncontrolled corrosion of carbon steel support structure surfaces surrounding a tube. As the uncontrolled corrosion of carbon steel takes place, the original base metal (iron) is converted to nonprotective magnetite (Fe_3O_4) resulting in a doubling of volume (i.e., twice the volume of the original base metal is occupied by the metal oxide). Because the magnetite is nonprotective, the base metal continues to corrode, producing large localized concentrations of metal oxide. The expanded metal oxide exerts pressure on the steam generator tube and the support. Then pressure in the tube/tube support annulus becomes sufficient to produce yielding in the tube wall, denting results.

Experience from operating steam generators and laboratory testing has demonstrated that two conditions are required to initiate denting:

1) The original clearance between the tube and the support must have become blocked with a porous deposit in which bulk water can enter and be concentrated.

2) The bulk water being concentrated must have impurities that produce chloric acid solutions, which in corroding the carbon steel of the support result in the formation of a nonprotective form of magnetite.

The potential for tube denting has been reduced in the St. Lucie Unit 2 steam generators by the installation of tube support plates and antivibration bars system that are stainless steel with a high chromium content that forms a tight adherent oxide layer. This combination eliminates the potential for denting.

c) Potential Effects of Tube Rupture

The steam generator tube rupture incident 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 Dump and Bypass System. Noncondensible radioactive gases in the condenser are removed by the Main Condenser Evacuation System and discharged to the plant vent. Analysis of a steam generator tube rupture incident, assuming complete severance of a tube, is presented in Section 15.6.

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.

d) Composition of Secondary Fluid and Radiological Considerations

Radioactivity concentration in the secondary side of the steam generator is dependent upon the activity level of the Reactor Coolant System, the primary to secondary leak rate, and the operation of the Steam Generator Blowdown System. An evaluation of shell side radioactivity concentration is given in Section 11.1.

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

Materials used in fabrication of the steam generator are not affected by the radiation levels and doses resulting from operation. Although radiation levels are significant for any internal maintenance operations, procedures and equipment have been developed to minimize individual personnel exposure during these operations by allowing rapid completion of individual maintenance operations.

5.4.2.2 Steam Generator In-service Inspection

a) Preservice and periodic inservice inspections of the steam generators will be performed. These programs are developed to comply with the ASME Code, Section XI requirements as appropriate, to permit examinations of the steam generator Code Class 1 and 2 component parts, including the steam generator tubes (refer to Subsection 5.2.4 and Section 6.6).

b) The parameters of the preservice and inservice inspection programs comply with the guidelines recommended in Regulatory Guide 1.83, "In-service Inspection of Pressurized Water Reactor Steam Generator Tubes," July 1975 (R1)

c) The preservice and inspection program examination method, equipment and reporting requirements comply to Appendix IV of the ASME Code, Section XI. The program parameters governing the criteria used for tube inspection, tube sample sizes, inspection intervals, and acceptance criteria (including plugging limits) are included in Technical Specifications.

5.4.3 REACTOR COOLANT PIPING

5.4.3.1 Design Basis

The reactor coolant piping is designed and analyzed for normal operation and all transients discussed in Subsection 3.9.1. Loading combinations and stress criteria associated with faulted conditions are presented in Subsection 3.9.1. In addition, certain nozzles are subjected to local transients that are included in the design and analysis of the areas affected. Thermal sleeves are installed in the surge nozzles, safety injection nozzles, and charging nozzles to accommodate these additional transients. Principal parameters are listed in Table 5.4-3. The ASME Code and Addenda the piping is designed to is specified in Subsection 5.2.1. The safety injection nozzles have been re-evaluated for the stated transients, and thermal sleeves are not required.

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 14 to 15 Hz and 70 to 75 Hz. The frequency ranges account for mechanical vibratory excitation of the reactor coolant pump and impeller vane passing pressure variations.

5.4.3.2 Description

Each of the two heat transfer loops contains five sections of pipe; one 42 inch internal diameter pipe between the reactor vessel outlet nozzle and steam generator inlet nozzle, two 30 inch internal diameter pipes from the steam generator's two outlet nozzles to the two reactor coolant pump suction nozzles, and two 30 inch internal diameter pipes from the reactor coolant pump discharge nozzles to the reactor vessel inlet nozzles. These pipes are referred to as the hot leg, the suction legs, and the cold legs, respectively. The other major section of reactor coolant piping is the surge line, a 12 inch schedule 160 pipe between the pressurizer and the hot leg in Loop 2B, and the spray line, a 4 inch Schedule 160 pipe at the pressurizer reduced to two 3 inch schedule 160 pipes between the 4 inch pipe and each cold leg in Loops 2B1 and 2B2. Arrangement of this piping is further described in Subsection 5.1.1.

To minimize the possibility of stress corrosion cracking, the reactor coolant piping is fabricated from SA-516 GR 70 base material mill clad with SA-240, Type 304L stainless steel. The surge line, spray lines, and other small lines are totally made of stainless steel. Nozzles are shop fabricated with safe ends to preclude dissimilar field welds.

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. In addition, to ascertain the integrity of the piping during plant life, necessary in-service inspections required by Section XI of the ASME Code are performed where required on the reactor coolant piping. To facilitate such inspections, longitudinal weld seems have been oriented at the 90 degrees and 270 degrees locations where feasible.

The 42 inch and 30 inch pipe diameters are selected to obtain reactor 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 pressure differential between the pressure and the heat transfer loops is no more than five percent of the Reactor Coolant System design pressure. The spray line sizing is discussed in Subsection 5.4.10.

To reduce the amount of field welding during plant fabrication, the 42 inch and 30 inch 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.

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 during the life of the plant. In addition, the fully assembled RCS is subjected to the required hydrostatic tests and post hydrostatic nondestructive testing. Fracture toughness of the reactor coolant piping is discussed in Subsection 5.2.3.

During the design, driving frequencies are accommodated by proper location of piping spring characteristics. The dynamic effects of system operation are also considered and piping restraints sized accordingly.

Further assurance of the continued structural integrity of the system during plant life is obtained from the in-service inspections performed in accordance with ASME Code, Section XI, and described in detail in Subsection 5.2.4.

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. Subsections 5.2.3.3.3 and 5.2.3.4.3 discuss the nondestructive examination program used during fabrication and construction. Inservice inspections of the Reactor Coolant System piping are discussed in Subsection 5.2.4. Inservice inspection requirements are specified in the Technical Specifications.

5.4.4 MAIN STEAM LINE FLOW RESTRICTIONS

The main steam line, venturi type, flow elements located in each main steam line and within the containment are components of the feedwater control system and also act as flow restrictors to impede the discharge of steam into the containment in the event of a steam line break accident downstream of the flow element. Further information is found in Section 10.3 and Table 10.3-1.

5.4.5 MAIN STEAM ISOLATION SYSTEM

The Main Steam Isolation System is discussed in Section 10.3.

5.4.6 REACTOR CORE ISOLATION COOLING SYSTEM

This subsection is not applicable to St. Lucie Unit 2.

5.4.7 RESIDUAL HEAT REMOVAL SYSTEM

5.4.7.1 Design Bases

5.4.7.1.1 Summary Description

The Shutdown Cooling System (SDCS) is used in conjunction with the Main Steam and Main or Auxiliary Feedwater Systems to reduce the temperature of the Reactor Coolant System (RCS) in post shutdown periods from normal operating temperature to the refueling temperature. The initial phase of the cooldown is accomplished by heat rejection from the steam generators to the condenser or atmosphere. For a normal cooldown the reactor coolant hot leg temperature is reduced to 325°F and the pressurizer pressure is reduced to 276 psia*. Then, the SDCS can be put into operation to reduce the reactor coolant temperature to the refueling temperature of 140°F, or less, and to maintain this temperature during refueling. During refueling, a portion of the shutdown cooling flow is directed to the CVCS and is processed by the CVCS ion exchangers and filters for purification of the RCS. This purified flow is then returned to the RCS.

The SDCS is used in addition to the atmospheric dump valves and the Auxiliary Feedwater System to cooldown the Reactor Coolant System following a small break LOCA (see Section 6.3). Operating plants with similar designs (with some generic differences due to necessary requirements) are St. Lucie Unit I and Millstone 2.

Assuming a loss of offsite power (LOOP), the plant cooldown process to reach cold shutdown conditions is comprised of two phases of heat removal. The initial phase is accomplished by controlling heat rejection to the atmosphere through the secondary atmospheric dump valves. The Reactor Coolant System (RCS) is depressurized by the main or auxiliary spray system or manual control of letdown and/or charging pumps throughout the RCS cooldown while maintaining RCS pressure temperature requirements in accordance with the St. Lucie Unit 2 Technical Specifications and subcooled margin limitations. Safety Injection tanks may be isolated when depressurizing to reach shutdown cooling entry pressure condition. With offsite power available, forced RCS flow is maintained during a plant cooldown. However, for a LOOP, natural circulation flow is used during the plant cooldown until the reactor coolant pumps are restarted after offsite power is restored.

Once the RCS has been adequately cooled down and depressurized, the Shutdown Cooling System (SDCS) is aligned and the cooldown proceeds by rejecting heat to the SDCS heat exchangers. Assuming loss of offsite power, the most limiting postulated single failure is a failure of one emergency power train. A loss of one DC emergency train would effectively prevent AC power off one diesel generator, even if operating, from being transferred to the onsite electrical system. The single failure disables one train of components associated with the atmospheric dump valves, chemical and volume control systems, auxiliary feedwater system, and shutdown cooling systems. However, the plant can be cooled down to cold shutdown conditions with one emergency power train and one train of system components within 36 hours as shown in Subsection 5.4.7.5.

5.4.7.1.2 Functional Design Bases

The following functional design bases apply to the Shutdown Cooling System:

* The operating limit for SDC entry has been conservatively established at 275 psia.

a) No single active failure prevents at least one complete train of the SDCS from being brought on line, whether this is during normal plant cooldown or following a design basis event.

b) The post LOCA functional requirements defined in Subsection 5.4.7.1.1 are met assuming the failure of a single active component during the short-term immediately after a LOCA, or a single active or limited leakage passive failure of a component during the long-term cooling phase following a LOCA. Additional analysis of passive failures in the SDCS can be found in Appendix 3.6F.

c) No single failure allows the SDCS to be overpressurized by the Reactor Coolant System. Shutdown Cooling System components are provided with overpressure protection by use of interlocks, valve arrangement, and relief valves (see Subsections 5.4.7.2.2 and 5.4.7.2.3).

d) The SDCS lowers the Reactor Coolant System temperature as follows:

   1) two train emergency cooldown (based on Reactor Coolant System cooldown to 350 F, hot leg temperature, in 3.5 hours after shutdown.)

      350F - 200F - approximately 5.5 hours after shutdown

2) One train cooldown (Emergency Condition)

350° F - 200° F: approximately 10.5 hours after activation of the SDCS

Emergency cooldown curves are shown on Figures 5.4-7 and 5.4-8.

e) The components of the Shutdown Cooling System are designed in accordance with Subsection 5.4.7.2.4.

f) Materials are selected to preclude system performance degradation due to the effects of short and long term corrosion.

g) In the event of a single active failure and to assure availability of the system when required, redundant components are provided. Instrumentation to assure proper system operation is described in Subsection 5.4.7.2.2, item (e).

h) The system design provides for inspection and testing of components to ensure availability and proper operation.

i) Pressure relief valves in the SDCS suction lines protect the system from overpressurization during system operation when the suction valves are open and the system is not isolated from the Reactor Coolant System. The valves are sized considering transients due to inadvertent operation of charging pumps, HPSI pumps, and pressurizer heaters. Additional pressure relief valves are provided to protect isolated sections of piping from thermal overpressure (see Subsections 5.4.7.2.3 and 6.3.2.2.6).

j) The SDCS is designed against the effects of internally and externally generated missiles as described in Section 3.5, and designed against the dynamic effects of pipe rupture as outlined in Section 3.6.

k) The motor operators for valves V3480, V3481, V3545, V3651, and V3652 are located above the maximum water level expected during the recirculation phase of Safety Injection System operations.

l) The design location, arrangement, and installation of the system and its components are such that it withstands the effects of earthquakes and other natural phenomena, without loss of capability of performing its safety function as specified in General Design Criterion 2. Operability requirements and analysis of this system and components relative to Regulatory Guides 1.29, "Seismic Design Classification," February 1976 (R2) and 1.48 "Design Limits and Loading Combinations for Seismic Category I Fluid System Components," May 1973 (R0) are described in Section 3.2 and Section 3.9.

m) No components of the SDCS for St. Lucie Unit 2 are shared by St. Lucie Unit 1, thereby satisfying GDC 5.

n) The SDC operation is a closely monitored mode of operation. In the event of a closure of a SDC suction line isolation valve, the operator sees indications on the instruments in the control room and shuts the LPSI pump off avoiding overheating and damage. Adequate cooling of the core is provided by the redundant SDC train. The cooling curve for a one train cooldown is shown on Figure 5.4-8.

5.4.7.2 System Design

5.4.7.2.1 System Schematic

The SDCS piping and instrumentation diagram, shutdown cooling mode diagram, and flow point data are shown on Figures 6.3-1a, b, c and 6.3-2d, and Table 6.3-4d, respectively. For a two train cooldown, the pressurizer pressure and hot leg temperature of the reactor coolant system vary from 276 psia* and 325°F at initiation of shutdown cooling to atmospheric pressure and less than 140°F at refueling conditions.

The SDCS contains two shutdown cooling heat exchangers and employs one or both low pressure safety injection pumps throughout shutdown cooling. During initial shutdown cooling, a portion of the reactor coolant flows out the shutdown cooling nozzles located on the reactor vessel outlet (hot leg) pipes and is circulated through the shutdown cooling heat exchangers by the LPSI pumps. The return to the reactor coolant system is through the safety injection nozzles in reactor coolant pump discharge (cold) legs.

The SDCS suction line isolation valves are interlocked to prevent overpressurization of the SDCS by the reactor coolant system. These interlocks are described in Subsection 5.4.7.2.3 and Section 7.6.

Shutdown cooling and LPSI flow are measured by orifice meters FT-3301 and 3306 installed in each LPSI header or by flow instruments FT-3312, 3322, 3332 and 3342 installed in the individual injection headers. The information provided by these flow elements is used to manually maintain total flow control during shutdown cooling operation.

The cooldown rate is controlled by adjusting the flow rate through the heat exchanger(s) with the throttle valves (HCV-3512, HCV-3657) on the discharge of the heat exchangers in conjunction with valves FCV-3301 and FCV-3306. The injection header motor-operated isolation valves, HCV-3615, 3625, 3635 and 3645 are manually controlled to maintain the required total shutdown cooling flow rate to the core.

During initial cooldown the temperature differential for heat transfer is large, thus only a portion of the total shutdown flow is diverted through the heat exchangers. As cooldown proceeds, the temperature differentials become less and the flow rate through the shutdown cooling heat exchangers is increased to the maximum achievable.

* The operating limit for SDC entry has been conservatively established at 275 psia.

At this time, full shutdown cooling flow (with the exception of any bypass valve leakage and purification flow) is through one or both of the shutdown cooling heat exchangers. A portion of the shutdown cooling flow from either of the loops through connections on the discharge side of the Low Pressure Safety Injection (LPSI) pumps is diverted to purify Reactor Coolant inventory during periods of shutdown. The flow is directed to the Chemical and Volume Control System (CVCS) just upstream of flow element FE-2202 and is processed by the CVCS ion exchanger and filters. The flow is then returned to the suction of the LPSI pumps via the shutdown cooling lines (Figure 6.3-1a,b and 9.3-5a).

5.4.7.2.2 Component Description

a) Low-Pressure Safety Injection Pumps

The Low Pressure Safety Injection (LPSI) pumps are used jointly as part of the SDCS and SIS. During all periods of plant operation, when SIS operability is required, the LPSI pumps are manually aligned for emergency core cooling operation.

During shutdown cooling. LPSI pump 2A takes flow from the Reactor Coolant System hot leg 2A and discharges through the shutdown cooling heat exchanger 2A. Train A shutdown cooling flow is returned to the Reactor Coolant System through the LPSI header A to the Reactor Coolant System cold legs 2A1 and/or 2A2. Likewise, LPSI pump 2B takes flow from the Reactor Coolant System hot leg 2B and discharges through the shutdown cooling heat exchanger 2B. Train B shutdown cooling flow is returned to the Reactor Coolant System cold legs 2B1 and/or 2B2. Thus, two completely independent shutdown cooling trains are provided.

The LPSI pump design flowrate of 3000 gpm was based on maintaining a core ΔT (approximately equal to the full power ΔT) at shutdown cooling 3 1/2 hours after shutdown for the pre-stretch core decay heat. The LPSI pump characteristics are further discussed in Section 6.3.2.2.2. Note that the SDC flow required to maintain full power ΔT at 3 ½ hours after shutdown with post-stretch core decay heat is approximately 4000 gpm which is beyond the capacity of a single SDC train. However, this ΔT would only be approached during a natural circulation cooldown without RCPs available. For a natural circulation cooldown, a soak period of at least 20 hours is maintained to preclude voiding in the reactor head prior to initiating SDC. At this time, the SDC flow required to maintain a full power core $\Delta T$is less than 2300 gpm.

b) Shutdown Cooling Heat Exchangers (SDCHX)

The shutdown cooling heat exchanger removes core decay heat, sensible heat, and LPSI and reactor coolant pump heat during plant cooldown and cold shutdown. The shutdown cooling heat exchangers are sized to lower the reactor coolant to the refueling water temperature (140°F) 27 1/2 hours after shutdown, assuming two shutdown cooling heat exchangers and two LPSI pumps are operating.

A conservative fouling factor is assumed, resulting in an additional area margin for the shutdown cooling heat exchangers. The shutdown cooling heat exchanger characteristics for the shutdown cooling mode are given in Table 5.4-4.

The design pressure of the shutdown cooling heat exchanger is based on the maximum operating pressure of the LPSI pump suction piping plus the shutoff head of the LPSI pump. The design temperature of the shutdown cooling heat exchanger is based on the SDC initiation temperature.

c) Piping

All SDCS piping is austenitic stainless steel. All piping joints and connections are welded except for a minimum number of flanged connections that are used to facilitate equipment maintenance or accommodate component design.

d) Valves

1) Motor Operated Valves

Design pressures and temperatures for the SDCS motor operated valves are provided in Table 6.3-5.

Throttle valves HCV-3512, HCV-3657, FCV-3301 and FCV-3306 are provided for remote control of the shutdown cooling heat exchanger tube side and bypass flow.

The two SDCS suction lines are equipped with three remotely controlled isolation valves in each line. A remotely controlled valve (V3545) serves as a crossover valve between the two SDCS suction lines. Five of the suction line isolation valves are located inside containment, and two valves are located outside containment. The shutdown cooling suction valves power and control diagram is shown on Figure 7.6-1.

All motor operated valves in the ECCS are equipped with Limitorque motor operators. These operators allow disengagement of the motor for handwheel operation, but are automatically re-engaged when the motor is activated. As a result, it is not possible for the motor operated valves to be disabled from automatic operation due to a prior manual handwheel operation.

The following SDCS motor operated valves are subject to the requirements of NRC Generic Letter 89-10: FCV-3301, FCV-3306, V3456, V3457, V3480, V3481, HCV-3512, V3517, V3536, V3539, V3651, V3652, HCV-3657, V3658, V3664, V3665.

As a result of evaluations performed to address NRC Generic Letter 95-07 "Pressure Locking and Thermal Binding of Safety Related Power-Operated Gate Valves", valve V3545 is maintained "Locked Open" during normal operation to eliminate susceptibility to pressure locking, V3651, V3652, V3480 and V3481 have a 3/16" hole drilled on the RCS side of the disk to prevent pressure locking.

The LPSI pump suction isolation valves (V3432 and V3444) are motor operated, to facilitate SDCS alignment from the control room.

2) Manual

Manual isolation valves are provided to isolate equipment for maintenance.

Manual valves exist in the flow paths of the SDCS trains which, if improperly aligned, could prevent flow from that train. These valves are V3206 and V3207, which are just downstream of the LPSI pumps. These are "locked open" and administrative procedures are used to assure their proper position. There is no single manual valve which, if misaligned, would disable both SDCS trains.

3) Relief Valves

Protection against overpressure of components within the shutdown cooling system is provided by relief valves. A description of relief valves, their set pressures and capacities is provided in Subsection 6.3.2.2.6.

4) Check Valves

Check Valves prevent reverse flow through the system during shutdown cooling.

5) Power Supplies

In addition to the preferred sources of offsite power, independent onsite standby electrical power supplies are provided for the SIS equipment by the diesel generators. A detailed description of the onsite standby power sources is given in Section 8.3. Diesel generator load sequencing is provided in Section 8.3.

e) Instrumentation

Operation of the SDCS is controlled and monitored through the use of installed instrumentation. The instrumentation provides the capability to determine heat removal, cooldown rate, shutdown cooling flow, and the capability to detect degradation in the flow or heat removal capacity. The instrumentation provided for the SDCS consists of:

1) Temperature measurements - Shutdown cooling heat exchanger inlet and the temperature of the shutdown cooling flow to the low pressure header. All temperatures are indicated in the control room. The shutdown cooling heat exchangers inlet temperature, and the low pressure header temperatures are recorded to facilitate control of the Reactor Coolant System cooldown rate.

2) Pressure Measurements - LPSI header pressure and shutdown cooling heat exchanger inlet pressure. These pressures are indicated in the control room, and, when used with the low-pressure pump performance curves, provide an alternate means of measuring system flow rate.

3) Flow Measurements - Total shutdown cooling flow rate is measured by flow indicators FI-3301 and 3306 or alternatively by FI-3312, 3322, 3332, and 3342.

The instrumentation is discussed further in Sections 7.4 and 7.6.

5.4.7.2.3 Overpressure Control

Protection against overpressure of the SDCS is provided by relief valves and interlocks. A description of relief valves is provided below and in Subsection 6.3.2.2.6.1.

There are six relief valves in the SDCS suction lines. These valves are sized to protect the components and piping from overpressure due to thermal expansion of the fluid. Valves V3482 and V3469 have a set pressure of 2485 psig and a capacity of five gpm. Relief fluid from these valves is collected in the quench tank. These valves are designed to 1974 ASME, Section NB, Quality Group A (see Figure 6.3-1b).

Valves V3483 and V3468 have a set pressure of 335 psig and a capacity of 155 gpm. Relief fluid from these valves is collected in a holdup tank in the Waste Management System. These valves are designed to 1974 ASME, Section NC, Quality Group B (see Figure 6.3-1b).

In addition to protecting the components and piping from overpressure due to the thermal expansion of the fluid, valves V3666 and V3667 are sized to protect the components and piping from overpressure due to inadvertent starting of the charging pumps, HPSI pumps, and pressurizer heaters. These valves have a set pressure of 335 psig and a capacity of 2300 gpm. Relief fluid collected from these valves is collected in the containment sump. These valves are designed to 1974 ASME, Section NC (to 1975 summer addenda) Quality Group B (see Figure 6.3-1b). When calculating the capacity of valves V3666 and V3667, the capacity of each valve was taken to be greater than the combined flow rate of two HPSI pumps, three charging pumps and the fluid forced out of the pressurizer when the backup heaters are actuated. This is a very conservative sizing, as the inadvertent actuation of the HPSI pumps, the charging pumps, and simultaneous energization of the pressurizer heaters is an unlikely coincidence. The largest flow rate which would be expected is from two safety injection pumps delivering approximately 1100 gpm at the Reactor Coolant System at 335 psig.

For the Shutdown Cooling mode, the LPSI pump suction is aligned to the hot leg of the RCS. The flow from the discharge side of the pump goes to the cold leg of the RCS. Due to this arrangement, the LPSI pump would not be dead headed by an RCS pressure surge. Relief valves in the shutdown cooling suction lines prevent isolation of the line from any anticipated overpressure events.

Valves V3666 and V3667 are located inside containment in close proximity of the containment sump. The discharge fluid from these relief valves is routed directly to the containment sump in order to prevent the accidental spread of Reactor Coolant System inventory outside the containment. The discharged liquid, which is collected and contained in the containment sump, will then be transported to the Liquid Waste Management System for processing. The containment sump is a large collecting reservoir provided to supply water to the Containment Spray and Safety Injection System for long term recirculation. The design of the containment sump is described in Subsection 6.2.2.2.3.

The use of the containment sump as an atmospheric collection tank minimizes the possibility of introducing any backpressure on the relief valve discharge piping. The relatively short piping runs in the discharge piping reduces the potential for any downstream pressure buildup or pressure fluctuations. This design assures maximum relief valve capacity at design conditions.

Since the SDCS is not designed to accommodate full Reactor Coolant System pressure, isolation of the system suction lines is assured by interlocks on the four suction line isolation valves inside containment. An independent interlock, utilizing pressurizer pressure is provided for each of the valves. For each SDCS train, power is supplied to the interlocks on these valves by two independent power supplies (one power supply for each interlock). This ensures isolation of the SDCS from the Reactor Coolant System should a spurious signal inadvertently open one of these valves. Each interlock is designed to prevent opening of its associated valve whenever pressurizer pressure is greater than 276 psia. An audible alarm sounds in the control room whenever any of the valves are off the full closed position. The interlock also provides automatic closure of the valves prior to an actual or simulated pressurizer pressure signal exceeding 515 psia. When power to an interlock is removed, the interlock fails as is. This precludes the loss of LPSI pump suction flow and, thereby, damage to the pump is avoided. As there are two isolation valves with interlocks in each LPSI pump suction train it is still possible to close off the LPSI pump line, as required, by both automatic and manual means. It should be noted that the basic purpose of the interlock is to provide double barrier protection between the Reactor Coolant System and the SDCS before the Reactor Coolant System returns to normal operating temperature and pressure. Each of the seven suction line isolation valves is equipped with open/close position indication in the control room.

Valve control circuitry is covered in Section 7.6. Interlocks are discussed further in Section 7.6. Interlocks are provided on the safety injection tank isolation valves for overpressure protection of the SDCS. The SIT interlocks are discussed in Section 7.6 and Subsection 6.3.2.2.1.

5.4.7.2.4 Applicable Codes and Classifications

Table 5.4-4 and Section 6.3 provides the applicable codes for system components. Refer to Section 3.2 for safety and seismic classifications.

5.4.7.2.5 System Reliability

The SDCS is designed to perform its design function assuming a single failure, as described in Subsection 5.4.7.1.2.

To assure availability of the SDCS when required, redundant components and power supplies are utilized. The Reactor Coolant System can be brought to refueling temperature utilizing one of two LPSI pumps and one of two shutdown cooling heat exchangers. However, with the design heat load, the cooldown would be considerably longer than the specified 27 1/2 hour time period.

Inadvertent overpressurization of the SDCS is precluded by the use of pressure relief valves and interlocks installed on the shutdown cooling suction line isolation valves and safety injection tanks isolation valves (see Subsections 5.4.7.2.3, Sections 6.3, and 7.6).

The instrumentation, control, and electric equipment pertaining to the SDCS are designed to applicable portions of IEEE Standards. See Section 7.5 for further details.

In addition to normal offsite power sources, physically and electrically separated and redundant onsite emergency power supply systems are provided to power safety related components. See Chapter 8 for further details.

Since the SDCS is essential for a safe shutdown of the reactor, it is a seismic Category I system and designed to remain functional in the event of a safe shutdown earthquake (see Section 3.10).

For a long-term performance of the SDCS without degradation due to corrosion, only materials compatible with the pumped fluid are used.

Environmental conditions are specified for system components to ensure acceptable performance in normal and applicable accident environments (see Section 3.11).

5.4.7.2.6 Manual Actions

Plant shutdown is the series of operations which bring the reactor from a hot standby condition to cold shutdown. The Shutdown Cooling System is used in conjunction with the Main Steam, and Main or Auxiliary Feedwater Systems to reduce the temperature of the Reactor Coolant System to the refueling temperature. The initial phase of the cooldown is accomplished by heat rejection from the Steam Generators to the condenser. In the case of no condenser vacuum, the atmospheric dump valves may be used.

For a normal cooldown, the reactor coolant hot leg temperature is reduced to 325F and the pressurizer pressure is reduced to 276 psia*. The Shutdown

* The operating limit for SDC entry has been conservatively established at 275 psia.

Cooling System is then put into operation to reduce the reactor coolant temperature to the refueling temperature.

The SDCS is a manually aligned and actuated system. Alignment to the shutdown cooling mode is accomplished via remote (from the control room or from the local control station outside the reactor containment) operated valves with the exception of the minimum flow isolation valves which are manually aligned in the plant. These valves also have a handwheel for local operation. Once system alignment to the shutdown cooling mode is accomplished, the SDCS system, and hence plant cooldown, can be controlled remotely from the control room during normal plant conditions.

Required manual actions for normal two train SDCS alignment and operation are listed below (Functionally, these actions allow for SDCS alignment and are not intended to provide appropriate sequencing of steps). All actions are performed from the control room, except as noted:

a) The containment spray isolation valves for the shutdown cooling heat exchangers (MV-07-03, MV-07-04) are closed.

b) The safety injection tank isolation valves (V3614*, V3624*, V3634, V3644) may be closed or the tanks depressurized. Note: In order to close these valves, power must be restored to the valves at the motor control center.

c) The shutdown cooling heat exchanger (SDC HX) flow control valves (HCV-3657*, HCV-3512) are operated as required to control cooldown and the SDC HX inlet valves (V3517*, V3658) are opened.

d) The SDC HX bypass valves (FCV-3306*, FCV-3301) and the SDC warm-up valves (V3536*, V3539) are operated as required to initiate and control SDC operation.

e) The SDC HX outlet valves (V3456*, V3457) are opened.

f) The LPSI pumps are operated as required.

g) The LPSI Recirculation valves (V3767*, V3205) are locked closed locally at the valves.

h) The LPSI pump suction valves (V3444*, V3432) from the RWT and the containment sump are locked closed.

i) The hot leg suction valves (V3480*, V3481*, V3664*, V3651, V3652, V3665) are opened (the normal position for the hot leg suction cross-tie V3545 is locked open). Note: In order to operate valves V3480, V3481, V3545, V3651 and V3652, power must be restored to the valves at the motor control center outside the control room.

j) CCW is established to the SDC HXs by opening valves HCV-14-3A and HCV-14-3B. The LPSI header isolation valves (HCV-3615*, HCV-3625*, HCV-3635, HCV-3645) are opened as required.

k) The shutdown cooling throttle valves (HCV-3657*, HCV-3512) are adjusted as necessary to maintain the RCS cooldown rate at 75°F/hr. or less. The total SDCS flow is maintained greater than or equal to 3000 gpm through one or both shutdown cooling trains when required by Technical Specifications. At other times, SDC flow is maintained as required to meet the heat removal requirements. The SDCS bypass flow control valves (FCV-3306, FCV-3301) are manually controlled by the operator.

A graph of Reactor Coolant System temperature vs. time after shutdown for a typical cooldown is presented on Figure 5.4-7.

Shutdown cooling is continued throughout the entire period of plant shutdown to maintain a refueling water temperature of 140°F or less. When RCS pressure is reduced to atmospheric and shutdown cooling is in operation, shutdown purification flow may be initiated to purify the circulating reactor coolant in the chemical and volume control system by using the connection between the SDCS and the chemical and volume control system.

The shutdown cooling function is used during the early stages of plant start up to control reactor coolant temperature. Heat generated by the reactor coolant pumps and by core decay is removed as required by the shutdown cooling system. Prior to continuing plant heatup above SDC design conditions, the interconnections to the reactor coolant system are isolated and the safety injection system is aligned for emergency operation. The six isolation valves (V3480*, V3481*, V3664*, V3651, V3652 and V3665) in the shutdown cooling suction lines are closed, the cross connect, V3545 is opened. After reactor coolant pump operation is allowed (reactor coolant system pressures ≥ 265 psia) and prior to reaching 325°F (reactor coolant system hot leg temperature) or 276 psia**, the LPSI pumps are stopped and the shutdown cooling heat exchangers are aligned for their containment spray function.

Initiation of shutdown cooling with the most limiting single failure (loss of one shutdown cooling train) can be accomplished using the procedures under plant cooldown for the operable train (i.e., operating the valves with (*) for train A, or the valves without (*) for train B). The SDC isolation valves are unique in that one of the two valves inside containment on each physical train is powered from separate electrical trains (i.e. V3480 and V3651 are powered by the B bus, and V3481 and V3652 are powered from the A bus).

In the event of a loss of a power (emergency or normal) supply, the SDCS suction line crossover valve (V3545) may be utilized to provide at least one complete shutdown cooling train. The operator selects the system flow path with an active available power supply (emergency or normal), and since the SDCS suction line crossover valve (V3545) is a normally Locked Open valve, the SDC function can continue. At times of reduced RCS inventory, the cross-tie valve can be closed to guard against the simultaneous loss of both trains resulting from an inadvertent loss of RCS level. Closure of the cross-tie valve is procedurally controlled to ensure that it is opened prior to pressurizing the RCS.

Following certain accidents, (feedwater line break, small break LOCA, steamline break, or loss of offsite power), it may become necessary to initiate shutdown cooling with reactor coolant system hot leg conditions which exceed the normal shutdown cooling initiation temperature (325F

* Indicates valves on shutdown cooling train 'A'.
** The operating limit for SDC entry has been conservatively established at 275 psia.

Reactor Coolant System hot leg temperature). However, shutdown cooling is not initiated at conditions which exceed the design temperature of the SDCS components (350F Reactor Coolant System hot leg temperature).

5.4.7.3 Performance Evaluation

The design point of the SDCS is taken at 27 1/2 hours after plant shutdown. At this point, the design heat load is to maintain a refueling temperature of 140F with 100F component cooling water. Two shutdown cooling heat exchangers and two LPSI pumps are assumed to be in operation at the design flow of 3000 gpm per shutdown cooling heat exchanger train. The SDCHX size is determined at this point, since it demands the greatest heat transfer area due to the relatively small ΔT between reactor coolant and component cooling water. The cooldown rate is limited to a maximum of 75° F/hr throughout the cooldown. The emergency cooldown curve is shown on Figure 5.4-7.

For the most limiting single active failure (see Table 5.4-5) in the SDCS, pressurizer pressure and Reactor Coolant System hot leg temperature are reduced to 276 psia and 325° F by other heat rejection means (atmospheric vent valves, steam bypass through the condenser). From these initial conditions, Reactor Coolant System temperature can be brought to 200° F in approximately 20.8 hours following shutdown cooling initiation using one LPSI pump and one shutdown cooling heat exchanger. The cooldown curve for an emergency one train cooldown is shown on Figure 5.4-8.

A failure modes and effects analysis (FMEA) for the SDCS is presented in Table 5.4-5. The analysis demonstrates that the SDCS can withstand any single active failure and still perform its design function. Analysis of passive failures in the SDCS is summarized in Appendix 3.6F.

On December 5, 1980, the Davis-Besse Nuclear Power Station experienced an inadvertent actuation of the emergency core cooling system (ECCS) in the recirculation Mode. At the time, the system was in the Hot Shutdown Mode, as repairs were being made to the electrical system. An electrical short circuit or ground caused the Safety Injection System to be actuated. As the Reactor Coolant System (RCS) was at a higher pressure than the Safety Injection pumps' shut off head (2100 vs. 1600 psig) the water from the Refueling Water Storage Pool did not enter the RCS. The electrical failure also caused a change to the Recirculation Mode. During the mode changeover period (1-1/2 minutes) the piping was aligned such that water from the RWSP drained into their SIS Recirculation Sump (about 15,000 gallons drained). After this switchover time, the High Pressure and Low Pressure Pumps (HPSI and LPSI Pumps) were aligned to a dry sump (save for the 15,000 gallons mentioned above). The major concern here is that the Safety Injection Pumps could become air bound, which could result in significant damage to the pumps, causing them to become inoperable.

If an electrical failure of the type that occurred at Davis-Besse could take place at St. Lucie Unit 2, the following fluid system response would occur. The LPSI and HPSI pumps are aligned with the RWT during normal plant operation. For this event no flow would be delivered to the RCS because the RCS pressure exceeds the shut off head of the LPSI and HPSI pumps. After the SIAS, but before the RAS, the only flow through the pumps would be mini-flow back to the RWT. This is done to prevent damage to the pumps when no flow can be delivered to the RCS.

Upon initiation of the RAS, the LPSI pumps are automatically shut off and the mini-flow valves are closed. This stops all flow through the HPSI pumps. The operator must take manual action to stop the HPSI pumps. Any degradation to the HPSI system as a result of this would have no effect on the decay heat removal abilities of the unit.

At this point Davis-Besse's piping layout allowed flow from the RWSP to the containment sump during the valve transition period. The St. Lucie Unit 2 Design prevents this flow with a pair of check valves (one in each train) just downstream from the containment sump isolation valves. Because the outlet nozzle on the RWT is approximately 12 feet above these check valves, a pressure of about 5 PSI holds them closed.

The April 19, 1980 event which occurred at Davis Besse inadvertently aligned the decay heat removal (DHR) pumps to the recirculation mode (and a dry containment sump) while the RCS was in a quasi-refueling mode. This resulted in air in the suction line of the system. As a result, the DHR system was down for approximately 2 1/2 hours while the lines were vented. During this delay the temperature limit for the refueling mode was exceeded.

Should this unlikely combination of events occur at the St. Lucie Unit 2 plant, no damage would occur to the DHR system. After the SIAS and before the RAS, the LPSI pumps would be aligned to their suction lines, which is the condition they were in prior to the event. After the RAS, the LPSI pumps would not be connected to the dry containment sump due to two conditions: the suction lines are isolated from both the RWT and the sump by valves V3444 and V3432 which do not receive an RAS; also the RAS automatically turns off the LPSI pumps. However, the HPSI pumps will be connected to the dry sump and could become damaged by becoming airbound if the operator does not shut them off soon enough. This damage will not affect the decay heat removal capabilities of the DHR system.

The preceding discussions compared the vulnerability of the St. Lucie Unit 2 fluid system to two similar events that occurred at Davis-Besse. It is concluded that the results for St. Lucie Unit 2 would be much less severe and could be handled by operator action.

5.4.7.4 Preoperational and In-Service Testing

Preoperational tests were conducted to verify proper operation of the SDCS. The preoperational tests included testing of the automatic flow control,(automatic control was subsequently removed in favor of manual control), verification of adequate shutdown cooling flow, and verification of the operability of all associated valves. In addition, a preoperational hot functional performance test was made on the installed shutdown cooling heat exchangers.

For availability of the SDCS, components of the system are periodically tested as part of the Safety Injection System testing, as described in the Technical Specification. As discussed in NRC Generic Letter 2008-01 and INPO SER 2-05, the presence of unanticipated gas voids within fluid systems can challenge the ability of systems to perform their design functions due to issues such as gas binding, water hammer, injection delay times, etc. Requirements for maintaining system operability with respect to gas intrusion are contained within Technical Specifications and Gas Accumulation Management Program procedures. The heat exchanger bypass valves leakage is tested on a frequency determined by historical test data. The system and component tests are sufficient to demonstrate the continued operability of the SDCS.

In addition to flow tests, the SDCS also undergoes a series of preoperational and in-service hydrostatic tests. Preoperational hydrostatic tests were conducted in accordance with Section III of the ASME Code while in-service hydrostatic tests are carried out as required by Section XI of the ASME Code.

ASME Boiler and Pressure Vessel Code Case N-416, "Alternative Rules for Hydrostatic Testing or Replacement of Class 2 Piping," may be used within the limitations stated in the "Inquiry" and "Reply" sections of the Code Case to satisfy the inservice pressure testing requirements of the ASME Code, Section XI.

St. Lucie Unit 2 had RSGs installed in Fall 2007. The RSGs have been evaluated with respect to natural circulation (see AREVA Doc. 77-5069878-01), and have been shown to have natural circulation capabilities equal to or better than the OSGs. Additionally, the RSGs impact on the Condensate Storage Tank, Shutdown Cooling System, and Auxiliary Feedwater was evaluated. It was considered that the OSGs requirements bound the RSGs, and no modifications to these systems is required. (Reference 8)

5.4.7.5 Compliance With Branch Technical Position RSB 5-1

(See Appendix 5.2B for additional analysis related to natural circulation cooldown.)

This section contains information which is historical in nature in that it documents FPL's response to the Branch Technical Position; however, compliance with RSB 5-1 is still required and update of this section should occur, as appropriate.

BRANCH POSITION

A. Functional Requirements

The system(s) which can be used to take the reactor from normal operating conditions to cold shutdown shall satisfy the functional requirements listed below.

1. The design shall be such that the reactor can be taken from normal operating conditions to cold shutdown using only safety-grade systems. These systems shall satisfy General Design Criteria 1 through 5.

RESPONSE:

Cooldown to cold shutdown conditions employ the auxiliary feedwater system, the main steam system, the chemical and volume control system, the component cooling water system, and the shutdown cooling system. The initial plant cooldown is accomplished by heat rejection to the atmosphere by the steam generator atmospheric dump valves. Four safety grade atmospheric dump valves, two per steam generator, are provided at St. Lucie Unit 2. The atmospheric dump valves, valve operators, and power supplies are all built in accordance with seismic Category I, Quality Group B requirements. Two atmospheric dump valves are supplied from vital dc bus SA; the other two atmospheric dump valves are supplied from vital dc bus SB. The valves are also supplied with handwheels to allow them to be operated manually. Should a single failure occur making two atmospheric dump valves inoperable from the control room, the other two valves are sufficient for plant cooldown.

During loss of offsite power the reactor coolant system is depressurized using auxiliary spray. The auxiliary spray is safety grade and has vital power supplied by emergency onsite power (4160 volt ac bus, diesel generator). Redundant auxiliary spray valves are provided.

Boration is accomplished using the chemical and volume control system. This system incorporates redundant charging pumps, boric acid makeup tanks and charging pump suction and delivery paths. This system satisfies the single failure criterion and can function without offsite power.

When the plant reaches the appropriate temperature and pressure, the shutdown cooling system is aligned, and the cooldown proceeds by rejecting heat to the shutdown cooling system heat exchangers.

BRANCH POSITION

A.2 The system(s) shall have suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) the system function can be accomplished, assuming a single failure.

RESPONSE:

Assuming loss of offsite power, the most limiting single failure associated with the thirty-six hour criterion is the failure of a dc bus and associated diesel generator. This failure disables the auxiliary spray valve and one train of components associated with the chemical and volume control system, the auxiliary feedwater system, the component cooling water system, and the shutdown cooling system. The sequence of operator actions described below is required to cool down to the shutdown cooling system entry conditions.

1. Secondary Steam Removal

The steam generated for decay heat removal and for system cooldown after a loss of offsite power is discharged through the atmospheric dump valves (ADVs). The two steam generator dump valves, operators, and power supplies are safety grade. There are two dc operated ADVs located on each of the two main steam lines. The ADVs are capable of automatic modulating service using ac control power and are capable of open/close service from the control room using dc power only. Each ADV is sized to pass 50 percent of the steam flow required to hold the Reactor Coolant System at hot standby for two hours followed by a 75°F/hr cooldown to the shutdown cooling system entry temperature of 350°F during natural circulation conditions, assuming that only 125,000 gallons of condensate are available from the condensate storage tank (see Appendix 5.2B for analysis of natural circulation cooldown at lower cooldown rates). The ADVs are ASME Code Class 2, seismic Category I with Class 1E qualified electrical components and a single valve capacity of 54,000 lb/hr at 40 psig. Should a single failure of an emergency power train occur making one ADV on each steam generator inoperable, the remaining two ADVs are capable of releasing the design basis required steam flow.

In addition to the ADVs, the following means are also available for dumping secondary steam.

- Steam may be released from either or both of the steam generators through the auxiliary feedwater pump turbine steam supply lines which are also safety grade (seismic Category 1, Quality Group B).

- Steam may be bypassed to the condenser if it is available.

- Steam traps may be bypassed, if necessary, to relieve steam.
- Vent valves and drain valves may be used, if necessary, to relieve steam.

In summary, four 50 percent capacity safety grade (seismic Category I, Quality Group B) atmospheric steam dump valves with multiple, diverse backup means of relieving steam are provided. See Section 10.3 for more information.

2. Primary Depressurization

All or a portion of the charging flow may be used for auxiliary spray flow to cool and depressurize the pressurizer in the event that reactor coolant pumps and thus main spray flow are not available. Auxiliary spray flow can be provided by operating one or more of the three safety grade charging pumps (Seismic Category I, ASME Class 2) after opening one or both of the two safety grade auxiliary spray isolation valves, SE-02-3 and SE-02-4 (seismic Category I, ASME Class 1), and after closing both safety grade charging line isolation valves, SE-02-1 and SE-02-2 (seismic Category I, ASME Class 1). The charging pumps are powered from the emergency onsite power sources while the auxiliary spray valves are dc powered. Only one charging pump and one auxiliary spray isolation valve need be operable in order to provide adequate auxiliary spray system operation. However, both charging line isolation valves must be closed to ensure that adequate charging flow is diverted through the auxiliary spray line. Three alternatives are available for plant depressurization if both offsite and one emergency power train fails. Plant shutdown without letdown and without auxiliary spray is addressed in Subsection 9.3.4.3.1.3.4. The second alternative involves manual action inside containment to close the fail-open charging line isolation valve. With the failed charging line isolation valve closed, adequate auxiliary spray and normal charging flow can be obtained throughout a plant cooldown with one charging pump, one auxiliary spray valve and one charging line isolation valve operable from the available emergency power train. (Note: The charging line isolation valves would not require manual action inside the containment since the current plant design uses 125V DC valves powered from a battery backed electrical bus; therefore, the valves would be available for use during this scenario). The third alternative involves operation of the safety grade pressurizer power operated relief valves (PORVs) to cool and depressurize the pressurizer (within the 36 hour cooldown criterion). The PORVs provide a redundant diverse means of pressurizer depressurization in addition to the use of the pressurizer auxiliary spray system. See Subsection 9.3.4 for more information.

As the RCS pressure is being reduced to the shutdown cooling entry pressure, the safety injection tanks (SITs) have their pressure reduced in accordance with operating procedures. An interlock with pressurizer pressure prevents the SIT isolation valves from being closed until RCS pressure drops to shutdown cooling entry pressure. Isolation of the tanks requires restoring power to the valve operators at the motor control center, in order to close the isolation valves from the control room. However, the tanks can be depressurized from the control room by using the vent valves. Two vent valves are provided on each tank. Each valve on a tank is powered from a separate emergency power source.

To initiate shutdown cooling, the SDC isolation valve breakers (for V3480, V3481, V3545, V3651 and V3652) must be manually closed at the Motor Control Center to allow for remote operation from the control room.

3. Primary Boration and Inventory Makeup

The St. Lucie Unit 2 design incorporates three safety grade charging pumps (seismic Category I, ASME Code Class 2), redundant safety grade charging pump borated water sources (two seismic Category I, ASME Code Class 2 boric acid make-up tanks and one seismic Category I, ASME Code Class 2 refueling water tank), redundant charging pump suction paths, and redundant charging pump delivery paths. The charging pumps and all related automatic control valves are connected to vital power if the normal power supply system should fail. During the plant cooldown, the Chemical Volume and Control System (CVCS) borates the RCS to cold shutdown boron concentration and accommodates the reactor coolant shrinkage, taking suction from the boric acid makeup and refueling water tanks. The minimum amount of stored boric acid solution that is maintained in either boric acid makeup tank is sufficient to bring the RCS to the required cold shutdown boron concentration.

Since the original response, and according to the implemented Boric Acid Concentration Reduction Program (Licensing Amendment #40), sufficient boric acid is added from the boric acid makeup tanks to the RCS to achieve a condition where the cooldown can be concluded using inventory and boric acid concentration from the refueling water tank.

Should offsite power be unavailable, letdown flow will be isolated and the normally closed motor-operated valve, V2504, will be operated manually to provide suction for the charging pumps from the refueling water tank. This valve is powered from vital MCC 2B5, but no credit is taken for its remote operation because its operator is not Class 1E. The capability of the CVCS to borate and to makeup is not compromised as a result of this event. A single failure of one emergency power train would leave at least one charging pump and one borated water source operable for RCS injection off the available emergency dc bus. One charging pump will allow the RCS to be borated to a cold shutdown boron concentration within the period that the plant is initially held at hot standby. Borated makeup water from the refueling water tank becomes available upon the opening of valve V2504 manually via the handwheel at the valve location within the Reactor Auxiliary Building. The procedure to bring the plant from hot standby to cold shutdown conditions is identical to that described in Subsection 9.3.4.3.1.3.4 for a shutdown without letdown and without auxiliary spray.

As a result of the Boric Acid Concentration Reduction Program, the boron concentration needed to maintain the required shutdown margin was calculated for each temperature during cooldown. Throughout the cooldown, the RCS boron concentration is maintained above the concentration needed to maintain the required shutdown margin.

4. Secondary Makeup

During cooldown, the auxiliary feedwater system (AFWS) and atmospheric dump valves provide a means to bring the reactor coolant system temperature down to the shutdown cooling system entry temperature. The auxiliary feedwater pump(s) are started and the safety grade auxiliary feed isolation valves are opened by an auxiliary feedwater actuation signal (AFAS). The auxiliary feedwater system is designed such that no single active failure coupled with a loss of offsite power prevents plant cooldown. A failure modes and effects analysis is provided in the St. Lucie Unit 2 Table 10.4-3.

The AFWS contains one steam driven and two motor driven auxiliary feedwater pumps. Each pump by itself is capable of maintaining the RCS in hot standby. Also, the steam driven or either of the motor driven auxiliary feedwater pumps is capable of delivering enough feedwater condensate for a plant cooldown. The AFWS is designed to safety grade requirements (seismic Category I, Safety Class 3, ASME Code Section III). Each of the motor driven auxiliary feedwater pumps utilize a Class 1E ac safety related power supply and the turbine driven pump train relies strictly on dc power supply. If the motor driven auxiliary feedwater pumps were operating due to an AFAS when offsite power was lost, the pumps would automatically restart using diesel generator power. If one diesel generator were to fail, the remaining motor driven pump or the turbine driven pump can be used to provide full capacity auxiliary feedwater flow. Auxiliary feedwater system valves required for cold shutdown have control switches in the control room and locally as well.

The 400,000 gallon condensate storage tank (CST) provides the water supply for the AFWS. When no tornado warnings are in effect, 297,800 gallons are available for St. Lucie Unit 2. The remaining usable portion of the condensate storage volume is provided for the St. Lucie Unit 2 secondary system makeup during normal plant operations. The CST is designed as seismic Category I, ASME Code Class 3, and is enclosed in a concrete protective barrier in order to withstand any tornado and/or missile effects.

Each ADV is sized to pass 50 percent of the flow required to bring the Reactor Coolant System to the SDCS entry temperature with reactor coolant pumps off, assuming that only 125,000 gallons of condensate is available from the CST. The quantity of water needed for St. Lucie Unit 2 cooldown is determined assuming a case wherein the unit is brought to hot standby and held there for two hours and then cooled down at the maximum rate of 75°F/hr until the shutdown cooling window of 350 F is reached. This requires 129,000 gallons, as indicated in Table 10.4-2. During emergency conditions (except the hypothetical tornado missile which drains the St. Lucie Unit 1 CST) there is sufficient water in the CST to allow hot standby operation for 23 hours and a subsequent cooldown to 350F over four hours. See UFSAR Subsections 9.2.6 and 10.4.9 for further information. Also see UFSAR Appendix 5.2B for additional analysis of natural circulation cooldown and CST requirements.

For EPU operation, two cases were evaluated. For one case the unit is held at hot standby for two hours and then cooled down to shutdown cooling entry conditions. The other case is held at hot standby for four hours and then cooled down to shutdown cooling entry conditions. Both cases are evaluated at a cooldown rate of 75°F/hr. If the plant is held at hot standby for two hours before cooldown, the quantity of water needed is 139,000 gallons. If the plant is held at hot standby for four hours before cooldown, the quantity of water needed is 154,000 gallons.

BRANCH POSITION

A.3 The system(s) shall be capable of being operated from the control room with either only onsite or only offsite power available. In demonstrating that the system can perform its function assuming a single failure, limited operator action outside of the control room would be considered acceptable if suitably justified.

RESPONSE:

All systems used for cooldown to hot shutdown are capable of being operated from the control room. Operator action outside the control room resulting from single failure assumptions are identified in the response to position A.2 above.

Cooldown to cold shutdown using SDCS can be accomplished with limited operator action from outside the control room as described in Subsection 5.4.7.2.6.

BRANCH POSITION

A.4 The system(s) shall be capable of bringing the reactor to a cold shutdown condition, with only offsite or onsite power available within a reasonable period of time following shutdown, assuming the most limiting single failure.

RESPONSE:

The initial phase of cooldown is accomplished by heat rejection to the atmosphere from the atmospheric dump valves (see Position A.2). Once the Shutdown Cooling System (SDCS) is aligned, the cooldown proceeds by rejecting heat to the SDCS heat exchangers. Assuming loss of offsite power, the most limiting postulated single failure is that of one vital dc bus and associated diesel generator. This single failure disables one train of components associated with the ADVs, CVCS, Auxiliary Feedwater System and Shutdown Cooling System. However, the plant can be cooled down to cold shutdown conditions with one diesel generator and train of system components within 36 hours.

BRANCH POSITION

B. RHR System Isolation Requirements

The RHR system shall satisfy the isolation requirements listed below.

1. The following shall be provided in the suction side of the RHR system to isolate it from the RCS.

   (a) Isolation shall be provided by at least two power-operated valves in series. The valve positions shall be indicated in the control room.

   (b) The valves shall have independent diverse interlocks to protect against one or both valves being opened during an RCS increase above the design pressure of the RHR system.

RESPONSE:

The SDCS is designed to provide adequate isolation between the SDCS and the safety injection tanks or the RCS when the RCS is above the design pressure of the SDCS (350 psig) as follows:

There are two parallel paths with two isolation valves per path inside containment on the suction line to the SDCS pumps. Each valve has a separate, independent power source and each valve is interlocked with a separate and independent pressurizer signal. Valve opening is prevented until the RCS pressure falls to a value of 276 psia. The setpoint for automatic closure of the SDCS suction line isolation valves is 500 psia to preclude premature isolation of the SDCS*.

Each isolation valve is equipped with open/close position indication in the control room.

It should also be noted that there is a cross connect between the two LPSI pump trains. This cross connect is situated between the two isolation valves on each train. This cross connect piping is opened and closed via valve V3545 (key locked opened). Power is supplied to the valve from either electrical Train A or B via the swing Bus. This cross connect valve can be operated from the control room and also from another control station. Open and closed positions are shown by corresponding lights. These lights are powered from electrical Bus A. In addition, there is 0-100 percent indication for this valve.

Thus, if one of the valves failed or if one of the diesel generators failed, there would still be one functioning LPSI pump train. For example, if valve V3481 failed closed, the piping line to LPSI pump 2A would no longer be

* The requirement is prior to an actual or simulated pressurizer pressure signal exceeding 515 psia.

available; however, the line to LPSI pump 2B would still be open and operable. As a second example; electrical Train A failed - the diesel generator - failed to start - these valves V3652, V3481 and V3664 would not open. Electrical Train B is functional and is used to open the SDC isolation valves (V3480 and V3651). Water flows through the LPSI pump 2A train (up to the cross connect point), flows through the cross connect piping and into the LPSI pump 2B piping train, thus supplying LPSI pump 2B with the necessary flow. Plant procedures ensure that the cross connect valve (V3545) will be opened when both SDC trains are operating.

At times when the SDC system is being operated at reduced levels in the RCS and the suction pressure valve interlock is defeated with the pressurizer manway removed, a pressurizer safety valve removed or the reactor head removed, the suction cross-tie valve, V3545, may be closed to gain additional heat removal performance from the SDC system. In addition, at times when a rapid loss of RCS level could cause air ingestion into the SDC system, such as operation at hot leg mid-loop level, the closed cross-tie valve could preclude the failure of both trains and allow continuation of, or expeditious recovery of, the SDC function.

BRANCH POSITION

B.2. One of the following shall be provided on the discharge side of the RHR system to isolate it from the RCS:

(a) The valves, position indicators, and interlocks described in item 1 (a)-(c).

(b) One or more check valves in series with a normally closed power operated valve. The power-operated valve position shall be indicated in the control room. If the RHR system discharge line is used for an ECCS function, the power-operated valve is to be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure.

(c) Three check valves in series, or

(d) Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leak tightness and the testing is performed at least annually.

RESPONSE:

Two check valves are within the containment barrier.

Design features permit leak testing of each check valve separately during plant operation to fulfill staff requirements for high/low pressure isolation.

In addition, normally closed motor operated valves are provided outside the containment. These valves will automatically open on SIAS.

NOTE:

Subsequent to initial plant operation, the technical specifications have been revised to allow testing at intervals greater than annual.

BRANCH POSITION

C. Pressure Relief Requirements

The RHR system shall satisfy the pressure relief requirements listed below.

1. To protect the RHR system against accidental overpressurization when it is in operation (not isolated from the RCS), pressure relief in the RHR system shall be provided with relieving capacity in accordance with the ASME Boiler and Pressure Vessel Code. The most limiting pressure transient during the plant operating condition when the RHR system is not isolated from the RCS shall be considered when selecting the pressure relieving capacity of the RHR system. For example, during shutdown cooling in the PWR with no steam bubble in the pressurizer, inadvertent operation of an additional charging pump or inadvertent opening of an ECCS accumulator valve should be considered in selection of the design bases.

2. Fluid discharged through the RHR system pressure relief valves must be collected and contained such that a stuck open relief valve will not:

   a. Result in flooding of any safety-related equipment.

   b. Reduce the capability of the ECCS below that needed to mitigate the consequences of a postulated LOCA.

   c. Result in a non-isolatable situation in which the water provided to the RCS to maintain the core in a safe condition is discharged outside of the containment.

3. If interlocks are provided to automatically close the isolation valves when the RCS pressure exceeds the RHR system design pressure, adequate relief capacity shall be provided during the time period while the valves are closing.

RESPONSE:

Overpressure protection of the SDCS is provided by relief valves in the suction lines and valves in the LPSI pump discharge header. There are six relief valves in the SDCS suction lines. These valves are sized to protect the components and piping from overpressure due to thermal expansion of the fluid. Valves V3482 and V3469 have a set pressure of 2485 psig and a capacity of five gpm. Relief fluid from these valves is collected in the quench tank and are designed to 1974 ASME, Section NB, Quality Group A (see Figure 6.3-1b).

Valves V3483 and V3468 have a set pressure of 335 psig and a capacity of 155 gpm. Relief fluid from these valves is collected in a holdup tank in the Waste Management System. These valves are designed to 1974 ASME, Section NC, Quality Group B (see Figure 6.3-1b). Further protection is provided by relief valves V3666 & V3667 between the inside and outside containment isolation valves. The relief valves protect the SDCS from inadvertent RCS pressurization during SDCS operation. The valves are sized and designed to provide protection against water-solid overpressure transients. The setpoint and valve capacity for V3666 and V3667 are 335 psig and 2300 gpm, respectively. The valves are capable of passing full safety injection flow. The relief valves at discharge of the LPSI pumps protect the header from pressure developed by temperature changes to the trapped water. The setpoint for the relief valve is 500 psig with a capacity of five gallons per minute (Reference Subsection 6.3.2.2.6.1). See Subsection 5.4.7.2.3 for more information.

BRANCH POSITION

D. Pump Protection Requirements

The design and operating procedures of any RHR system shall have provisions to prevent damage to the RHR system pumps due to overheating, cavitation or loss of adequate pump suction fluid.

RESPONSE:

The shutdown cooling system mode is a closely monitored manual mode of operation. Control room indication of the RCS pressure (PI-1103), the pressure in each LPSI header (PI-3304 (Train B) and PI-3307 (Train A)), each LPSI pumps flow rate (F-3312 (Train A), F-3322 (Train A), F-3332 (Train B), and F-3342 (Train B), and the position of each remotely actuated suction line isolation valve (V3480 (Train A), V3481 (Train A), V3651 (Train B), and V3652 (Train B) permits the operator to take the necessary steps to prevent damage to the pumps resulting from overheating, cavitation, or a loss of adequate suction fluid. If evidence of pump cavitation appears, manual control of LPSI flow can be used to maintain adequate available NPSH. Should the operator determine it necessary to secure one shutdown cooling train, adequate cooling of the core can be provided by the redundant train. Also, the existing design permits a single failure of the A or the B bus to close a suction valve in each train of the SDCS. To avoid a loss of suction to the LPSI pump, plant procedures ensure that the cross connect valve (V3545) will be opened when both SDC trains are operating or that the closure interlock is defeated whenever the cross connect valve is closed.

Low flow alarms aid the operator in evaluating pump performance. The *ultrasonic flow sensors are located in the miniflow lines (FE-03-1-1 (Train A) and FE-03-21 (Train B)) and the discharge lines (FE-03-1 (Train A) and FE-03-2 (Train B)) for the LPSI pumps, with alarm annunciators in the Control Room.

*Note: The ultrasonic flow sensors and alarms were subsequently deleted by PC/M 104-294, due to sensor obsolescence and procedural improvements.

BRANCH POSITION

E. Test Requirements

The preoperational and initial startup test program shall be in conformance with Regulatory Guide 1.68. The program for PWRs shall include tests with supporting analysis to (a) confirm that adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions and permit estimation of the times required to achieve such mixing, and (b) confirm that the cooldown under natural circulation conditions can be achieved within the limits specified in the emergency operating procedures. Comparison with performance of previously tested plants of similar design may be substituted for these tests.

RESPONSE:

The preoperational and startup test programs conform to Regulatory Guide 1.6 (R2). Boron mixing under natural circulation conditions, has been demonstrated in a prototypical test at the San Onofre Nuclear Generation Station (SONGS). The results of this SONGS test have been reported to the NRC under separate correspondence.[1]

A natural circulation test and simulator training will be performed for St. Lucie Unit 2 to demonstrate the capability of cooling the plant. A detailed plan is included in Subsection 14.2.12.42 in a separate correspondence. St. Lucie Unit 1 submittals to the NRC concerning Natural Circulation cooldown have been reviewed with respect to their applicability to St. Lucie Unit 2. It is FPL's opinion that the submittals provided to the staff for Unit 1 are directly applicable to Unit 2. In summary, Unit 2 has revised emergency operating procedures which reflect a more stringent cooldown rate than the existing 75°F/hr rate. However, it is our position that the more stringent cooldown rate is not required to preclude safe cooldown or plant shutdown. All evaluations completed by our NSSS Vendor (CE) concur with our position. The St. Lucie Unit 2 response is covered by the St. Lucie Unit 1 analysis which is addressed in FP&L letters L-80-343 dated October 17, 1980 and L-80-431 dated December 30, 1980. See also Appendices 5.2B and 5.2C which contain the St. Lucie Unit 1 analysis. Appendix 5.2B has also been updated with the St. Lucie Unit 2 analyses for operation at EPU conditions.

(1) FPL letter L-84-68 dated March 13, 1984

BRANCH POSITION

F. Operational Procedures

The operational procedures for bringing the plant from normal operating power to cold shutdown shall be in conformance with Regulatory Guide 1.33. For pressurized water reactors, the operational procedures shall include specific procedures and information required for cooldown under natural circulation conditions.

RESPONSE:

The operational procedures will be in conformance with Regulatory Guide 1.33. Generic guidance for cooldown under natural circulation conditions developed in conjunction with the CE Owner's Group will be included in the procedures.

BRANCH POSITION

G. Auxiliary Feedwater Supply

The seismic Category I water supply for the auxiliary feedwater system for a PWR shall have sufficient inventory to permit operation at hot shutdown for at least 4 hours, followed by cooldown to the conditions permitting operation of the RHR system. The inventory needed for cooldown shall be based on the longest cooldown time needed with either only onsite or only offsite power available with an assumed single failure.

RESPONSE:

Subsection 10.4.9 of the St. Lucie Unit 2 UFSAR states that the quantity of water stored in the condensate storage tank for St. Lucie Unit 2 cooldown is determined assuming a case wherein the unit is brought to hot standby conditions and held there for two hours followed by a cooldown at the maximum rate of 75°F/hr until the shutdown cooling window of 350°F is reached. The total hot standby and cooldown time is 6.5 hours (see Table 10.4-2). This evolution results in about 43,000 gallons of condensate to cool the plant from hot standby to 350°F and about 86,000 gallons of condensate to remove 6.5 hours of decay heat.

A cooldown scenario which involves maintaining hot standby conditions for four hours followed by a cooldown at the maximum rate of 75°F/hr until the shutdown cooling entry conditions are reached is basis for condensate storage per Branch Technical Position RSB 5-1. Pre-EPU calculations show that 49,600 gallons of condensate is required to cool the plant from hot standby to 350°F. However, 100,000 gallons of condensate is now needed to remove 8 hours of decay heat.

Thus 149,600 gallons of condensate storage was required for pre-EPU operation in order to accommodate an initial hold at hot standby for four hours followed by a cooldown. For EPU operation, at a cooldown rate of 75°F/hr to shutdown cooling entry conditions following an initial hold at hot standby for four hours, 154,000 gallons of condensate are needed. The condensate storage tank minimum available volume of water (297,600 gallons) is above the volume needed for a four hour initial hot standby scenario (see Subsection 10.4.9.3).

As indicated in Subsection 10.4.9 (Table 10.4-2, Figure 10.4-10) there is sufficient water in the condensate storage tank during emergency blackout conditions (except when an assumed tornado missile ruptures the Unit 1 storage tank) to remove a total of 27 hours of decay heat plus the sensible heat (stored energy). This volume of water would support a scenario of 23 hours in hot standby followed by a 4 hour cooldown.

See UFSAR Appendix 5.2B for additional analysis of natural circulation cooldown and CST requirements.

5.4.7.6 Generic Letter 88-17 Commitments

Generic Letter 88-17 was issued to address concerns related to the loss of decay heat removal capability during non-power operations based on several industry incidents. The generic letter required the implementation of expeditious actions as well as programmed plant enhancements to address this issue. FPL provided commitments to implement these requirements in letters to NRC (L-88-552 and L-89-38). The commitments made by FPL were in the following areas: 1) personnel training on the events; 2) availability of instrumentation to verify state of RCS and cooling systems performance; this includes the use of core exit thermocouples and SDC supply/return line temperature for core exit conditions and RCS water level indicators for reduced inventory; 3) procedures for normal and off-normal reduced inventory SDC operation and administrative controls for containment closure prior to core uncovery; 4) at least one HPSI pump and charging pumps available to maintain RCS in stable and controlled condition; 5) the performance of time to RCS boiloff, core uncovery and vent area analyses to supplement design information and support procedures/instrumentation; 6) guidelines to ensure perturbation-causing operations are minimized; and 7) controls to avoid RCS pressurization when hot leg nozzle dams are in place simultaneously. In addition, licensees were required to identify and submit appropriate changes for Technical Specifications that restrict or limit the safety benefit of the actions identified in the generic letter.

5.4.8 REACTOR WATER CLEANUP SYSTEM

This section of Regulatory Guide 1.70 (R3) is for boiling water reactors and is not applicable to St. Lucie Unit 2.

5.4.9 MAIN STEAM LINE AND FEEDWATER PIPING

The main steam piping is discussed in Section 10.3. The Feedwater System piping is discussed in Subsection 10.4.7. The main steam and feedwater piping materials are discussed in Subsection 10.3.6.

5.4.10 PRESSURIZER

5.4.10.1 Design Bases

The pressurizer is designed to:

a) Maintain Reactor Coolant System (RCS) operating pressure such that the minimum pressure observed during operation transients is above the setpoint for the safety injection actuation signal (SIAS) and that the maximum pressure is below the high pressure reactor trip.

b) Compensate for changes in reactor coolant volume during the design transients specified in Subsection 3.9.1.1.

c) Provide sufficient water volume to prevent draining the pressurizer as a result of a reactor/turbine trip.

d) Contain a total water volume such that the total mass and energy released to the containment during a LOCA is minimized.

e) Provide sufficient water volume to prevent the pressurizer heaters from being uncovered during a pressurizer outsurge following a 10 percent step decrease or a five percent per minute ramp decrease.

f) Provide sufficient steam volume to allow acceptance of an insurge without the pressurizer water level reaching the primary safety or power operated relief valve nozzles.

g) Provide sufficient steam volume to yield an acceptable pressure response to normal system volume changes during design load change transients.

h) Provide sufficient pressurizer heater capacity to heat up the pressurizer, filled with water at the zero power level, at a rate that ensures a pressurizer temperature (and thus pressure) which maintains an adequate degree of subcooling of the water in the reactor coolant loops as it is heated by core decay heat and/or pump work from the reactor coolant pumps.

5.4.10.2 Description

The pressurizer, as shown on Figure 5.4-10 is a vertically mounted, bottom supported, seismic Category I, cylindrical pressure vessel. Replaceable direct immersion electric heaters are vertically mounted in the bottom head. The pressurizer is furnished with nozzles for spray, surge, safety valves, relief valve and pressure, temperature and water level instrumentation. A manway is provided in the top head. Principal design parameters are listed in Table 5.4-6.

The pressurizer is designed and fabricated in accordance with the ASME Code listed in Table 5.2-1. The interior surface is clad with weld deposited stainless steel.

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 level in the pressurizer is programmed as a function of average reactor coolant temperature (See Figure 4.4-10). High or low water level error signals result in the control actions shown on Figure 5.4-11. The pressurizer surge line is sized to accommodate the flow rates associated with the RCS expansion and contraction due to the transients specified in Subsection 3.9.1.1.

The pressurizer maintains Reactor Coolant System operating pressure and, in conjunction with the Chemical and Volume Control System (CVCS), Subsection 9.3.4, compensates for changes in reactor coolant volume during load changes, heatup, and cooldown. During full power operation, slightly more than one half the pressurizer volume is occupied by saturated water, and the remainder by saturated steam. Reactor Coolant System pressure may be controlled automatically or manually by maintaining the temperature of the pressurizer fluid at the saturation temperature corresponding to the desired system pressure.

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 (See Figure 4.4-10). 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 reactor coolant lowers the pressurizer water level, causing the Reactor Coolant System pressure to decrease. The pressure reduction is partially compensated by flashing of pressurizer water into steam. Pressurizer heaters are automatically energized on low system pressure, generating steam and further limiting pressure decrease. Should the water level in the pressurizer drop below the setpoint, the letdown control valves close to minimum value, and additional charging pumps in the CVCS are automatically started to add coolant to the Reactor Coolant System and restore pressurizer water level. A redundant Class 1E Low-Low Level Heater Cutoff with alarms is installed to assure that all heaters are deenergized on a pressurizer low-low level signal. Therefore, no single failure in the protection system can cause them to be energized while uncovered.

Some of the heaters are connected to proportional controllers, which adjust the heater input to account 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 are normally deenergized but are automatically turned on when needed. The pressure control program is shown on Figure 5.4-12.

When steam demand is increased, the average reactor coolant temperature is raised in accordance with the reactor coolant temperature program. The expanding reactor 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 reactor 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 water level error signal, produced by an in-surge causes the letdown control valves to open, releasing reactor coolant to the CVCS and restoring the pressurizer to the programmed level. Small pressure and reactor 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. During certain transients which result in RCS inventory increases, a safety grade Class 1E Pressurizer high level alarm alerts the operator and allows sufficient time to terminate excess charging. This action functions to keep peak RCS pressure below 2370 psia, thus avoiding a reactor and turbine trip.

The pressurizer spray is supplied from each of the reactor coolant pumps associated with steam generator 2B as shown on Figures 5.1-3 and 5.1-4. 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 valves are sized to use the differential pressure between the reactor coolant pump discharge and the pressurizer to pass the amount of spray required to prevent the pressurizer steam pressure from opening the power operated relief valves during normal load following transients. A small continuous flow is maintained with the pressurizer spray bypass valves through the spray lines at all times to keep the spray lines and the surge line warm to reduce thermal shock during plant transients. An auxiliary spray line is provided from the charging pumps to permit pressurizer spray during plant heatup, or to allow cooling if the reactor coolant pumps are shut down.

In the event of an abnormal transient which causes a sustained increase in pressurizer pressure, at a rate exceeding the control capacity of the spray, a high pressure trip level is reached. This signal trips the reactor and opens the two power operated relief valves. The steam discharged by the power operated relief valves goes to the quench tank where it is condensed.

5.4.10.3 Evaluation

It is shown by analysis, made in accordance with the requirements for ASME Code, Section III, Code Class 1, that the pressurizer is adequate for all normal operating and transient conditions expected during the life of the plant.

Overpressure protection of the Reactor Coolant System is provided by three ASME Code spring-loaded pressurizer safety valves. Refer to Subsection 5.4.13 for more information on safety valves.

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

5.4.10.4 Tests and Inspections

Following shop fabrication, the pressurizer is subjected to the hydrostatic test required by ASME Code, Section III, Code Class 1. Upon completion of field erection of the Reactor Coolant System, the system is hydro-tested to verify the pressure retaining capability of the field piping welds.

Prior to plant startup, the transient performance of the pressurizer is evaluated by determining system normal heat losses, and maximum pressurization and depressurization rates. This data is used to set the pressure controllers and verify the adequacy of the pressurizer water level control system, heaters and spray valves.

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

5.4.10.5 Pressurizer Surge Line Thermal Stratification

Design basis cyclic transients were revisited for the pressurizer surge line and nozzles to address thermal stratification and thermal stripping concerns raised by NRC Bulletin 88-11. This Bulletin required licensees to visually inspect the surge line; to perform analyses to demonstrate that the surge line meets applicable design codes and other UFSAR/regulatory requirements for the design life of the plant; to obtain data on thermal stratification, thermal striping and line deflection; and to perform detailed stress and fatigue analyses on the line to ensure compliance with applicable code requirements. As a result of this analysis which is documented in Reference 6, the NRC concluded (Reference 7) that the results of the analysis performed by the Combustion Engineering Owner's Group (CEOG) for this task may be used for FPL as the basis to update the plant specific code stress reports to demonstrate compliance with applicable Code requirements as requested in NRC Bulletin 88-11. The analysis developed a new set of design basis transients based on the collected data. The new set of design basis transients are provided in Reference 6. The total number of heatup-cooldown cycles remained unchanged at 500. However, the number of stratification cycles that occur during a heatup-cooldown cycle was changed.

The structural integrity of the surge line is maintained for EPU operation and existing pipe support loads remain applicable. The results from the existing pre-EPU analysis of record (Reference 9) are applicable for EPU normal operation and transients.

5.4.11 QUENCH TANK (Pressurizer Relief Discharge System)

5.4.11.1 Design Bases

The quench tank is designed to receive and condense the normal discharges from the pressurizer safety and power operated relief valves and to prevent the discharge from being released to containment.

The quench tank is sized to contain the 567 lbm steam calculated to be discharged by the pressurizer safety valves during the worst case anticipated operating occurrence (loss of load event, see Appendix 5.2A) followed by 575 lbm steam calculated to be discharged by the power operated relief valves during CEA withdrawal accident as the plant returns to power following the loss of load trip.

The quench tank is mounted on a structural steel frame which is designed to seismic Category I requirements. The pressure relief discharge system is designated Quality Group D and non-seismic per ANSI N18.2. Furthermore, the piping system from the pressurizer in the quench tank, although designated non-seismic, has been seismically analyzed to ensure that it withstands safe shutdown earthquake loadings without failure.

A loss of load event immediately followed by an uncontrolled rod withdrawal event is unrealistic given that each result in a reactor trip. Thus, it is sufficient to ensure that the tank volume is such that the tank is capable of accepting and condensing steam released to the quench tank in the limiting event (uncontrolled rod withdrawal). The steam released in an uncontrolled rod withdrawal is 450 lbm for 3020 MWt operation, well below the limit the tank was sized for as stated above. The current quench tank water level and temperature limits can be maintained.

5.4.11.2 System Description

The quench tank, shown on Figure 5.4-13, is an austenitic stainless steel vessel suitable for prolonged contact with borated water. The quench tank is located in the containment at elevation 62 ft which is lower than the pressurizer safety or power operated relief valves to ensure that any leakage or discharge from the valves drains to the quench tank. Nozzles are provided for the safety/relief valve discharge line, vent, drain, instrumentation, makeup water and a manway and rupture disc. The tank prevents the steam released from the pressurizer safety/relief valves from being released to the containment atmosphere. The steam is discharged under water by the sparger and condensed. A 1/2 inch hole in the sparger prevents a vacuum from forming in the safety/relief valve discharge piping following a steam release. Primary water is manually added to cool the tank water after a steam discharge and the pressure is manually relieved to the Waste Management System. Normal operating level can then be reestablished by draining to the reactor drain tank located at elevation 7.58 ft. A relief valve and a rupture disc venting to the containment atmosphere are provided for overpressure protection. The rupture disc is also provided with a stainless steel vacuum support.

The tank normally contains water under a three psig nitrogen overpressure. The quench tank normal water volume of 127 cubic feet covers the sparger and is sufficient to condense the steam released from the pressurizer safety and power operated relief valves as a result of the design basis transient. The water temperature in the quench tank is limited to 281° F based on initial water temperature of 120° F. The tank gas volume is based on limiting the maximum tank pressure to 70 psig after the steam release produced by this sequence of events.

The quench tank design parameters are given in Table 5.4-7. The quench tank and piping connections are shown on the Reactor Coolant System Piping and Instrumentation Diagram, Figures 5.1-3 and 5.1-4. The nitrogen supply piping and the piping from the quench tank to the reactor drain tank is shown on Figures 11.3-1 and 11.2-1, respectively.

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

5.4.11.3 Safety Evaluation

The quench tank is designed and fabricated to the requirements of the ASME Code, Section III, Code Class 3. Although the quench tank is designated non-seismic, seismic Category I supports have been provided. Furthermore, the piping system from the pressurizer to the quench tank, although designated non-seismic, has been seismically analyzed to ensure that it withstands safe shutdown earthquake loadings without failure. Besides these evaluations, the quench tank sparger is analyzed for both seismic forces and blowdown forces. These additional evaluations ensure that the quench tank would withstand these accident related conditions. The quench tank rupture disc minimum capacity is greater than the combined maximum capacities of the pressurizer power operated relief and safety valves at the maximum allowed ASME Code quench tank pressure. Thus the build-up backpressure does not exceed the valve design backpressure for the pressurizer safety valves. Criteria for design against pipe whip are discussed in Section 3.6. Compliance with Regulatory Guide 1.67, "Installation of Overpressure Protection Devices," October, 1973 (R0) is discussed Subsection 3.9.3.

Failure of any of these components in no way compromises the integrity of the reactor coolant pressure boundary or safe shutdown of the plant.

5.4.11.4 Instrumentation Requirements

The quench tank is equipped with water level, pressure and temperature Instrumentation as described below. The location of each of the sensors are indicated on Figure 5.1-4.

5.4.11.4.1 Temperature

The temperature measurement channel consists of a resistance temperature detector (RTD), a temperature transmitter and associated indicator and alarms. Quench tank temperature indication and high-temperature alarm are provided in the control room. A high temperature alarm may be indicative of a safety/relief valve leakage or discharge to the quench tank.

5.4.11.4.2 Water Level

Water level instrumentation consists of a level transmitter and associated indicating and alarming equipment. Quench tank level indication and high and low level alarms are provided in the control room. A low level alarm is indicative of the sparger being uncovered or of insufficient tank fluid volume to quench the design basis steam release. A high level alarm alerts the operator of insufficient gas volume in the quench tank to accept a pressurizer steam release without becoming overpressurized and causing the rupture disc to burst.

5.4.11.4.3 Pressure

The pressure measurement instrument consists of a pressure transmitter and associated indicators and alarms. Quench tank pressure indication and high pressure alarm are provided in the control room. The high pressure alarm alerts the operator to the situation prior to the bursting of the rupture

5.4.11.5 <u>Inspection and Testing Requirements</u>

Inservice inspection of the applicable portions of the pressurizer relief discharge system is performed in accordance with ASME Code, Section XI.

5.4.12 VALVES

5.4.12.1 <u>Design Basis</u>

The safety related valves within the reactor coolant pressure boundary act as pressure retaining vessels and leak tight barriers during normal plant operation, accidents and seismic disturbances. The valves associated with the Reactor Coolant System pressure boundary are shown on Figures 5.1-3 and 5.1-4. These valves are designed in accordance with ASME Code, Section III, Code Class 1 requirements and withstand the effects of the system design transients (see Subsection 3.9.1.1) plus other transients associated with the valve 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 <u>Design Description</u>

All valves in the Reactor Coolant System are constructed primarily of stainless steel. Other materials in contact with the reactor coolant, such as hard facing and packing, are compatible materials. Fasteners, packing, gland assemblies and yoke fasteners are constructed of stainless steel or equal to eliminate corrosion problems. Valves, over 2 inches which are not constructed with diaphragms, bellows, or not normally backseated, are provided with either two independent sets of packing, separated by a lantern ring and leak-off connections piped to the reactor drain tank or provided with the EPRI recommended five ring packing stack-up, with capped leakoff connections, to control valve stem leakage.

5.4.12.3 Design Evaluation

All valves within the reactor coolant pressure boundary are stress analyzed in accordance with ASME Code, Section III, Code Class 1, 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 the ASME Code, Section III as indicated in Table 5.2-1. Valves greater than one inch nominal pipe size are dimensionally checked, including measurements to determine minimum wall thickness.

In-service inspection is performed in accordance with ASME Code, Section XI.

5.4.13 PRESSURIZER SAFETY AND RELIEF VALVES

5.4.13.1 Design Basis

The pressurizer safety and power operated relief valves are designed to the requirements of the ASME Code, Section III, Code Class 1 (refer to Table 5.2-1 for code date). Their location in the RCS, flow paths, and associated instrumentation are shown on Figure 5.1-4.

The design basis for establishing the relieving capacity of the pressurizer safety valves is presented in Appendix 5.2A. The relieving capacity of the pressurizer safety valves are designed to provide overpressure protection for the postulated transients presented in Chapter 15. No credit is taken for power operated relief valves (PORVs) in the Section 15.2 analysis or in the sizing of the safety valves.

The PORVs prevent opening of the pressurizer safety valves during hot standby or power operations based upon the design transient which is a control rod withdrawal accident. This design basis transient assumes the main steam safety valves (described in Subsection 10.3.3) are operational and the reactor is scrammed on high pressure by the Reactor Protective System. The PORVs are also designed to provide the RCS with Low Temperature Overpressure Protection (LTOP) as described in Subsection 5.2.6.

5.4.13.2 Description

The RCS is protected against overpressure by protective and control devices such as the pressurizer spray system, PORVs and the high-pressure reactor trip. In addition to these features, three ASME Code safety valves ensure that RCS piping and components are protected from overpressure in accordance with ASME Code requirements. The pressurizer safety valve is illustrated on Figure 5.4-14. The design parameters are listed in Table 5.4-8. A flanged connection is used to secure these valves to the top of the pressurizer. They are direct acting, spring-loaded safety valves with an enclosed bonnet and a balanced bellows to compensate for backpressure variations.

The RCS has two power operated relief valves connected to the top of the pressurizer as shown on Figures 5.1-4 and 5.1-4a. During certain transient conditions, the RCS pressure will increase at a rate that exceeds the control capacity of the pressurizer spray system. The PORVs are provided to prevent such excursions from unnecessarily lifting the safety valves.

The PORVs are designed to open at the reactor high-pressure trip setpoint (2370 psia). When the plant is at power, one PORV is isolated from the RCS by closure of the upstream motor operated gate valve to avoid an excessive discharge of reactor coolant due to both PORVs lifting.

In the startup and cooldown mode of operation, both PORVs are aligned to the pressurizer. While in the Low Temperature Overpressure Protection (LTOP) mode, the PORV setpoints are at low pressures. The power operated relief valve is illustrated on Figure 5.4-15. The design parameters for the power operated relief valves are given in Table 5.4-9.

Downstream temperature and flow instrumentation actuate their respective alarms in the control room, notifying the operator of PORV leakage.

5.4.13.3 Evaluation

Low temperature overpressure protection by the PORVs is discussed in Subsection 5.2.6. Overpressure protection by the pressurizer safety valves is discussed in Subsection 5.2.2 and the ASME Code Report on safety valve overpressure Protection is included as Appendix 5.2A.

The NRC Staff per GL 90-06 identified actions to enhance the reliability of the PORVs and PORV block valves that perform safety related functions. For St. Lucie Unit 2, the PORVs (V1474 and V1475) perform the safety related function of providing low temperature overpressure protection for the reactor coolant pressure boundary as discussed in Section 5.2.6. The PORVs are not credited with performing any accident mitigation function, nor are they relied upon to cooldown the plant.

5.4.13.4 Tests and Inspections

Both the safety valve and the PORVs are inspected and tested during fabrication in accordance with ASME Code, Section III requirements. The following four paragraphs describe the original safety valves and are maintained as-is for historical purposes.

The inlet and outlet portions of the pressurizer safety valves are hydrostatically tested with water at the appropriate pressures required by the ASME Code. Using a prorated spring to permit valve operation with the available steam supply, each valve is cycled; set pressure and seat leakage tests are performed, and adjustment is made for blowdown. The actual operating spring is then installed and set pressure and seat leakage tests are performed using air at room temperature. Another seat leakage test is then performed using air at 650°F (+25°F, -0°F). Final set pressure and leakage tests are performed in the field at operating temperatures. On the basis of the results of the EPRI PWR Safety and Relief Valve Test Program, which included full scale testing of a St. Lucie Unit 2 model pressurizer safety valve, valve ring adjustments for the St. Lucie Unit 2 pressurizer safety valves were selected to assure stable operation. The resulting valve blowdown is approximately 10 percent (2250 psia blowdown pressure). Note: 10% represents the blowdown of the originally installed safety valves. PCM 96139 installed new valves with a specified blowdown of 4%.

The inlet and outlet portions of the PORVs are hydrostatically tested with water at the appropriate pressures required by the ASME Code. Each valve is cycled at LTOP pressure using room temperature water and at normal steam service pressure using air at 650°F (+25°F, -0°F). Cycling is done by manually energizing and deenergizing the solenoid. During the cycling, the operation of the position indicators on the main valve disc and on the solenoid is checked. Total leakage tests which include main seat leakage and solenoid leakage are performed using room temperature water at LTOP pressure and air at 650°F (+25 °F, -0°F) at normal steam service pressure.

St. Lucie Unit 2, along with other PWR utilities, participated in the EPRI program for the performance testing of PWR safety and relief valves. The EPRI program was conducted in response to NUREG-0578 Section 2.12, "Performance Testing for BWR and PWR Safety and Relief Valve Operability." Models of the St. Lucie Unit 2 pressurizer safety valves and power operated relief valves were tested in the program. The EPRI test program results demonstrated satisfactory operability for the St. Lucie Unit 2 pressurizer safety and power operated relief valves. Results were reported in Combustion Engineering reports CEN-213 (June 1982)[1] for power operated relief valves and CEN-227 (December 1982)[2] for safety valves.

Reference 3 documents NRC acceptance of the Relief and Safety Valve Test Program designed to qualify the operability of prototypical valves and to demonstrate that their operation would not invalidate the integrity of the associated equipment and piping. NRC review of NUREG-0737 items including relief and safety valve testing is discussed in Appendix 1.9A.

Per GL 90-06 (References 4 & 5), the following requirements apply to the PORVs and PORV block valves (V1474, V1475, V1476 and V1477):

1) These valves are Class 1 valves and are to be included in its QA program for maintenance and procurement.
2) Include the PORVs and block valves within the scope of ASME Section XI, Subsection IWV for inservice testing.
3) Include the block valves in the Motor Operated Valve (MOV) test program, GL 89-10, "Safety Related Motor Operated Valve Testing and Surveillance."

In-service testing of PORVs for the LTOP mode is discussed in Subsection 5.2.6.1.3.

5.4.14 COMPONENT SUPPORTS

The structures supporting Code Class 1 components are also discussed in Subsection 3.8.3.

5.4.14.1 Design Bases

The criteria applied in the design of the reactor coolant system supports are that the specific function of the supported equipment be achieved during all normal, earthquake, and LOCA conditions. Specifically, the supports are designed to support and restrain the reactor coolant system components under the combined design basis earthquake and LOCA loadings in accordance with the stress and deflection limits listed in Subsection 3.9.3.1.

5.4.14.2 Description

The Reactor Coolant System support points are illustrated in Figures 3.8-41, 3.8-42, 3.8-43, 3.8-44, and 3.8-52.

a) Reactor Vessel Supports

The reactor vessel is supported by three integral pads at an elevation below the centerline of the vessel nozzles. The integral pads provide vertical vessel support and allow for unrestrained thermal expansion of the vessel.

Keyways located alongside the vertical support pads guide the vessel during thermal expansion and contraction of the Reactor Coolant System and maintain the vessel centerline.

b) Steam Generator Supports

The steam generator is supported at the bottom of 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 Reactor Coolant System. Two keyways within the sliding base guide the movement of the steam generator during the expansion and contraction of the reactor coolant system and, together with a stop and anchor bolts, prevent excessive movement of the bottom of the steam generator during seismic events and following a LOCA.

A system of keys and snubbers located on the upper end guide the top of the steam generator during expansion and contraction of the Reactor Coolant System and provide restraint during seismic events and following a LOCA or a steam line break.

c) Reactor Coolant Pump Supports

Each reactor coolant pump is provided with four vertical spring-type hangers which provide support for normal operation and seismic conditions. In addition each pump has a horizontal hydraulic snubber to dampen torsional oscillation of the pump on the main coolant piping under seismic conditions. The support system is designed to accept the piping and pump movements resulting from the normal and abnormal transient operating conditions described in Subsection 3.9.3.1.

For the case of pipe break in the pump suction line, one structural stop is provided on each suction line to limit the pump horizontal motion.

d) Pressurizer Supports

The pressurizer is supported by a cylindrical skirt welded to 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 LOCA.

5.4.14.3 Evaluation

The Reactor Coolant System supports are designed to the criteria for load combinations and stresses which are presented in Subsection 3.9.3.1. The criteria are used to determine the loads the supports must consider as a result of the effects of pipe rupture and seismic conditions.

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. Sliding supports for the steam generator were 100 percent liquid penetrant inspected and machined sockets were 100 percent magnetic particle inspected at the vendor's shops. The steam generator base plate was 100 percent ultrasonic inspected per ASME III Subsection NF.

The steam generator snubbers were tested in the vendor's shop at the rated load capacity in both tension and compression. The piston creep velocities were measured during these tests for compliance with specification limits. Tests were also conducted for initiation of snubber action on both the tension and compression directions.

SECTION 5.4: REFERENCES

1) "Summary Report on the Operability of Power Operated Relief Valves in CE Designed Plants," CEN-213, Combustion Engineering, Inc., June, 1982.

2) "Summary Report on the Operability of Pressurizer Safety Valves in CE Designed Plants," CEN-227, Combustion Engineering, Inc., December, 1982.

3) Letter, from J.A. Norris (NRC) to C.O. Woody (FPL), "NUREG-0737 Item II.D.1 Performance Testing of Relief and Safety Valves," dated May 11, 1989.

4) Generic Letter 90-06, "Resolution of Generic Issue 70, Power-Operated Relief Valve and Block Valve Reliability, and Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light Water Reactors"

5) NRC Letter, St. Lucie Units 1 & 2 - Response to Generic Letter 90-06 dated August 20, 1993

6) ABB Combustion Engineering Report CEN-387-P, "Pressurizer Surge Line Flow Stratification Evaluation," Rev. 1-P-A, C-E Owners Group, May 1994.

7) NRC Letter to C-E Owners Group, "Safety Evaluation for CEOG Report CEN-387-P, Revision 1, (Bulletin 88-11)," July 14, 1993.

8) AREVA NP Document 77-5069878-01, "Replacement Steam Generator Report for Florida Power and Light St. Lucie Unit 2".

9) CE Design Report F-MECH-DR-006, Rev. 0, "Addendum to the Piping Analytical Reports for FPL St. Lucie Units 1 and 2," January 17, 1994.

TABLE 5.4-1

REACTOR COOLANT PUMP PARAMETERS

Cycle 1

Number	4
Type	Vertical, Limited Leakage, Centrifugal
Shaft Seals	Mechanical (4)
Stationary Face	Carbon CCP-72
Rotating Face Body	ASTM A351 Gr CF8
Rotating Face Ring	Titanium Carbide, Kenna-Metal K 162-B

Note: shaft seals were replaced during the 1998 refueling outage with the N-9000 design, which incorporates the following materials:

Stationary Face	NMCC CNFJ
Rotating Face	Kennametal KZ-801
Design Pressure, psig	2485
Design Temperature, °F	650
Normal Operating Pressure, psig	2235
Normal Operating Temperature, °F	550*
Design Flow, gpm	81,200
Total Dynamic Head, feet	310
Maximum Flow (one-pump operation), gpm	120,000
Dry Weight, pounds (w/o Motor)	75,000
Flooded Weight, pounds (w/o Motor)	82,000
Reactor Coolant Volume, cubic feet	112
Material	
Shaft	ASTM A182 Type F-304
Casing	ASTM A351 Gr CF8M
Casing Wear Ring	ASTM A351 Gr CF8
Hydrostatic Bearing	
Bearing	ASTM A351 Gr CF8
Journal	ASTM A351 Gr CF8
Flywheel	ASTM A543 Cl 1 Type B
Motor	
Voltage, volts	6600
Frequency, hz/phase	60/3
Horsepower/Speed, Hot, hp/rpm	5700/883
Horsepower/Speed, Cold, hp/rpm	7200/883
Horsepower/Speed, Rated	6500/881
Service Factor	1.15
Weight, pounds	99,000 to 105,000

* For EPU operation, normal operating temperature is 551°F.

TABLE 5.4-1 (Cont'd)

Instrumentation		
Seal Temperature Detectors	1	
Pump Casing Pressure Taps	2	
Seal Pressure Detectors	3	
Controlled Bleedoff Flow Detectors	1	
Controlled Bleedoff Temperature Detectors	1	
Motor Oil Level Detectors	2	
Motor Bearing Temperature Detectors	4	
Motor Stator Temperature Detectors	6	
Reverse Rotation Detectors	2	(Removed from the motors)
Vibration Monitors	6	
Oil Lift Pressure Switches	4	
Oil Lift Pressure Gage	1	
Total Seal Assembly Leakage		
Three Pressure Seals Operating, gpm	1.00	
Two Pressure Seals Operating, gpm	1.35	
One Pressure Seal Operating, gpm	1.90	

TABLE 5.4-2

STEAM GENERATOR PARAMETERS

Parameter	Cycle 1[a] Value	RSG[b] Value	EPU[b], [d] Value
Number of units	2	2	
Heat transfer rate, each, Btu/h	4.386×10^9	4.623×10^9	5.179×10^9
Primary side:			
Design pressure/temperature (psia/°F)	2500/650	2500/650	
Coolant inlet temperature, °F	604	595.5	602.4
Coolant outlet temperature, °F	550	549.0	551
Coolant flow rate, each, lb/h	61×10^6	75.8×10^6	
Coolant volume at 68 °F each, ft^3	1629	1900	
Tube size, OD inches	3/4	3/4	
Tube thickness, nominal inches	0.048	0.0429	
Secondary side:			
Design pressure/temperature (psia/°F)	1000/550	1000/550	
Steam pressure, psia	815	896.9[c]	886.2[c]
Steam flowrate each, lb/h	5.6×10^6	5.9×10^6	6.6×10^6
Feedwater temperature at full power, °F	435	435	436
Moisture carryover, weight maximum, %	0.20	0.10	
Reactor Coolant inlet nozzle, No./ID inches	1/42	1/42	
Reactor Coolant outlet nozzle, No./ID inches	2/30	2/30	
Steam Nozzle, No./ID inches	1/34	1/34	
Feedwater nozzles, No./size/schedule	1/16/80	1/18 (16" ID)	
Overall heat transfer coefficient (estimated), Btu/h-Ft^2 - °F	933	1458	1416

(a) Performance parameters are based on full power Cycle 1 operation.

(b) Parameters are based on no fouling, no plugging and continuous blowdown of 1% of total steamflow.

(c) Static steam pressure.

(d) Only the values that change with implementation of EPU are shown.

TABLE 5.4-3

REACTOR COOLANT PIPING PARAMETERS

Parameter	Value
Number of loops (steam generators)	2
Flow per loop, lb/h	See Table 5.1-1
Pipe Size:	
Reactor Outlet: ID/Wall, in	
Elbow	42/4-1/8 W/O Clad
Pipe	42/3-3/4 W/O Clad
Reactor Inlet: ID/Wall, in	
Elbow	30/3 W/O Clad
Pipe	30/2-1/2 W/O Clad
Pump Suction: ID/Wall, in	
Elbow	30/3 W/O Clad
Pipe	30/2-1/2 W/O Clad
Surge Line, Nominal (in.)/Schedule	12/160
Spray Line, Nominal (in.)/Schedule	4/160 & 3/160
Design pressure, psia	2500
Design temperature, F	650
Surge Line Design Temperature F	700
Spray Line Design Temperature F	650

TABLE 5.4-4

SHUTDOWN COOLING HEAT EXCHANGER DATA

Parameter	Value
Quantity	2
Type	Shell and tube horizontal U-Tube
Manufacturer	Tubular Exchangers Manufacturing Association (TEMA)
Service transfer rate (Btu/h-F-ft^2)	277
Heat transfer area per heat exchanger, ft^2	5790
Tube side:	
Fluid	Reactor
Design Pressure, psig	500
Design Temperature, F	400
Material	Austenitic stainless steel
Shell Side:	
Fluid	Component Cooling Water
Design Pressure, psig	150
Design Temperature, F	250
Material	Carbon Steel
At 27-1/2 hours after shutdown:	
Tube Side:	
Flow, million lb/h	1.5
Inlet Temperature, F	135.00
Outlet Temperature, F	116.3
Shell Side:	
Flow, million lb/h	2.41
Inlet temperature, F	100.0
Outlet temperature, F	111.6
Heat load (million Btu-h-HX)	28.0
Code	ASME Section III, 1974 Edition, Class 2 and 3.

TABLE 5.4-5

FAILURE MODES AND EFFECTS ANALYSIS - SHUTDOWN COOLING SYSTEM

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection*	Inherent Compensating Provision	Remarks and Other Effects
1.	SDCS Suction Isolation Valves (Inside containment) V3480, V3481, V3651, or V3652	a) Fails open	Electrical malfunction, Mechanical malfunction	Loss of redundancy for isolating one shutdown cooling train if the RCS pressure should rise above the SDCS isolation valve interlock setpoint during shutdown cooling operation.	Periodic Testing Alarm in control room	The redundant series valve ensures that the SDCS is protected from high RCS pressure.	These valves are inter-locked to prevent in-advertent opening and to automatically close whenever the RCS pressure exceeds the SDCS isolation valve interlock setpoint.
		b) Fails closed	Electrical malfunction, Mechanical malfunction	Partial loss of decay heat-removal capability. Loss of one shutdown cooling train.	Periodic testing, Alarm. ("valve closure initiated") in control Room	Redundant shutdown cooling train ensures adequate cooling al-though the cooling time will be extended.	These valves are FAI and locked closed in the control room during power operation.
2.	LPSI Pump 2A or 2B	Fails to start ,or fails during operation	Electrical malfunction, Bearing failure, seal failure	Loss of flow through one shut-down cooling train to RCS cold leg.	No flow indication from FIC3306 or FIC3301, no dis-charge pressure indication from PI3314 or PI3315, Periodic testing, pump "Run" light	Redundant shutdown cooling train will not be affected.	The reactor coolant system can be brought to refueling temperature using one LPSI pump and one shutdown cool-ing heat exchanger, but the cooldown process would be extended beyond the speci-fied 24 hour time period.
3.	Minimum re-circulation flow line isolation valve V3659, V3660, V3495 or V3496	a)Fails open	Electrical malfunction, Mechanical binding	None during shutdown cooling. Loss of redundancy to prevent pumping of primary coolant to RWT during SDC.	Periodic testing	Redundant series isolation valve provides backup.	These valves are locked open and are required to be closed during shutdown cooling.
		b)Fails closed	Electrical malfunction	Loss of mini-flow protection against operating one LPSI pump dead headed if pump is spuriously act.	Periodic testing	Redundant LPSI train is unaffected.	

TABLE 5.4-5 (Cont'd)

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection*	Inherent Compensating Provision	Remarks and Other Effects
4.	SDC Hx Inlet Isolation Valve V3517, V3658	a) Fails open	Electrical malfunction, Mechanical binding	Inability to isolate shutdown cooling heat exchanger for maintenance.	Valve position indication, periodic testing	Redundant containment spray train.	These valves are locked closed in control room during, power operation and are required to be open during shutdown cooling operation.
		b) Fails closed	Electrical malfunction, Mechanical binding	Inability to align one shutdown cooling heat exchanger for shutdown cooling operation.	Periodic testing, abnormal pressure indication from PI-3303X or PI-3303Y, abnormal temperature indication from TI-3303X or TI-3303Y	Redundant shutdown cooling train.	----
5.	SDC Hx Outlet Isolation Valve V3456 or V3457	a) Fails open	Electrical malfunction, Mechanical binding	Inability to isolate SDC Hx for maintenance.	Valve position indication, periodic testing	None required.	These valves are tucked closed in the control room during power operation and are required to be open during shutdown cooling.
		b) Fails closed	Electrical malfunction, Mechanical binding	Isolating of one shutdown cooling Hx during shutdown cooling operation.	Periodic testing, no ΔT indication from TR-3351 or TR-3352.	Redundant shutdown cooling train will not be affected.	----
6.	LPSI/SDC Hx Bypass Flow Control Valve FCV-3301 or FCV-3306	a) Fails open	Electrical malfunction, Mechanical binding	Excessive shutdown cooling flow will bypass the SDCS Hx. Inability to increase the cooldown cooling rate in affected train.	Valve position indicator, periodic testing, low ΔT indication from TR-3352 or TR-3351	Redundant shutdown cooling train will assure shutdown cooling time will be extended.	These valves are locked open during power operation and power to the valve operators are racked out. The valves are manually adjusted.
		b) Fails closed	Electrical malfunction, Mechanical binding	During early states of shutdown cooling, excessively cooled water would be pumped through one shutdown cooling, train to the reactor core.	Valve position indicator, periodic testing, abnormal ΔT indication from TR-3351 or TR-3352	Operator can manually operate the associated throttle valve. Redundant shutdown cooling train will not be affected.	Operator manually controls these valves during the shutdown cooling mode.
		c) Excessive Seat Leakage	Seat wear or damage	Excessive shutdown cooling flow will bypass the SDCS Hx	Periodic testing, low ΔT indication from TR-3351 or TR-3352	Redundant SDC train will not be affected. Use of injection Header isolation valves for flow control minimizes impact of excessive leakage.	-------

TABLE 5.4-5 (Cont'd)

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection*	Inherent Compensating Provision	Remarks and Other Effects
7.	SDC Hx Flow Control Valve HCV-3512 or HCV-3657	a) Fails open	Electrical malfunction, Mechanical binding	Inability to regulate and maintain cooldown rate. See 6b.	Valve position indicator in control room, Periodic testing, Abnormal flow indication from FR-3301 or FR-3306	Same as 6a.	These valves are locked closed during power operation. These valves are opened by the operator to initiate SDC flow to the Hxs and are manually adjusted by the operator to maintain cooldown.
		b) Fails closed	Electrical malfunction, Mechanical binding	Reduction in core cooling capability. Isolation of one shutdown cooling Hx.	Valve position indicator in control room, periodic testing, no delta t indication from TR-3351 or TR-3352	Redundant shutdown cooling train will not be affected.	During the final stages of shutdown cooling, the SDCS bypass valves are almost completely closed so that a substantial amount of reactor coolant passes through the shutdown cooling heat exchangers.
8.	LPSI Valve HCV-3615, HCV-3625, HCV-3635, or HCV-3645	a) Fails open	Electrical malfunction, Mechanical binding	Inability to gradually warm up the associated shutdown cooling line during the shutdown cooling alignment procedure.	Valve position indicator in control room, periodic testing.	Redundant shutdown cooling trains will not be affected.	The safety injection piping is designed for a limited number of Thermal cycles such as could result from Operating the Shutdown cooling trains without prior warmup.
		b) Fails closed	Electrical malfunction, Mechanical binding	Inability to inject cooled reactor coolant into one of the RCS cold legs.	Valve position indicator in control room, periodic testing, no flow indication from testing, no flow FI-3312, FI-3322, FI-3332, of FI-3342	Redundant LPSI valve - and shutdown cooling train are not affected.	----

TABLE 5.4-5 (Cont'd)

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection*	Inherent Compensating Provision	Remarks and Other Effects
9.	SIT Isolation Valve V3614, V3624, V3634, or V3644	a) Fails open	Electrical malfunction, Mechanical binding	Inability to isolate one safety injection tank from the RCS during shutdown cooling. Possible inadvertent draining of one SIT.	Valve position indicator in control room, periodic testing.	Reduce pressure in affected tank using vent valves.	During shutdown cooling these valves are closed. However, if a LOCA occurs an SIAS will automatically open these valves. Prior to the actuation of shutdown cooling, the SIT pressure is periodically lowered below the RCS pressure to preclude discharging of their contents. Pressure reduction is achieved by bleeding off nitrogen or draining liquid to RWT.
		b) Fails closed	Electrical malfunction, Mechanical binding	None during shutdown cooling.	Valve position indicator in control room, periodic testing	None required.	
10.	LPSI Suction Isolation Valves V3432 , V3444	a) Fails open	Electrical malfunction, Mechanical binding	Inability to double isolate one LPSI pump suction from the RWT during shutdown cooling operation; loss of one shutdown cooling train.	Periodic testing, position indicator in control room.	Redundant LPSI train is unaffected	These valves are normally locked open and are closed just prior to initiation of shutdown cooling. Valves are opened from the control room.
		b) Fails closed	Electrical malfunction, Mechanical malfunction	None during shutdown cooling.	Periodic testing position indicator in control room.	None required	These valves are required to be closed during shutdown cooling operation. Valves are opened from the control room.
11.	SDCS Cross Connect Valve V3545	a) Fails open	Electrical malfunction, Mechanical binding	None during system operation.	Periodic testing, position indicator in control room	None required	This valve is required to be open when both SDC trains are in operation.
		b) Fails closed	Electrical malfunction, Mechanical binding	Inability to establish a crossover flow path between the SDCS trains (inside containment).	Periodic testing, position indicator in control room	The valve is capable of being powered from either diesel generator.	This valve is an FAI and locked open in the control room during power operation. This valve assures operation of at least one SDCS train if one emergency diesel generator fails.

TABLE 5.4-5 (Cont'd)

No.	Name	Failure Mode	Cause	Symptoms and Local Effects Including Dependent Failures	Method of Detection*	Inherent Compensating Provision	Remarks and Other Effects
12.	SDCS Warmup Valve V3536 or V3539	a) Fails open	Electrical malfunction, Mechanical binding	Diversion of some flow from the discharge leg to the suction leg of one SDCS train without passing through the reactor core during shutdown cooling.	Position indicator in control room Periodic testing	The redundant shutdown cooling train will not be affected.	If this valve fails open, there will be a minimal effect on the shutdown cooling time.
		b) Fails closed	Electrical malfunction, Mechanical binding	Inability to gradually warm LPSI pump, SDCHX, and shutdown cooling piping (one train) during the shutdown cooling alignment process.	Periodic testing, position indicator in control room	Redundant shutdown cooling train will not be affected	Gradual warming of the shutdown cooling trains minimize thermal shock at initiation of shutdown cooling. These valves are gradually closed once warm up and flow rate stability have been achieved.
13.	SDCS Suction Isolation Valve (outside containment) V3664 or V3665	a) Fails open	Electrical malfunction, Mechanical binding	No effect on normal or emergency SDCS operation.	Periodic testing	None required	
		b) Fails closed	Electrical malfunction, Mechanical binding	Inability to align one shutdown cooling train for shutdown cooling operation.	Periodic testing	Redundant shutdown cooling train will not affected	These valves are FAI and locked closed in the control room during operation.

* The method of Detection column is used to show that it is possible to detect the failure during or before Shutdown Cooling System Operation.

TABLE 5.4-6

PRESSURIZER PARAMETERS

Property	Parameter
Design pressure, psia	2500
Design temperature, °F	700
Normal operating pressure, psia	2250
Normal operating temperature, °F	653
Internal free volume, minimum, ft^3	1500
Normal (full power) operating water volume, ft^3	800
Minimum (zero to 15% power) operating water volume, ft	450
Spray flow, maximum, gpm	450
Spray flow, continuous, gpm	1.5
Heater type	Immersion
Heater capacity total, kW	1500*
Proportional, kW	300*
Backup, kW	1200*

* Operation with a reduced pressurizer heater capacity of 700 kW is acceptable. This permits up to 16 pressurizer heaters, a total of 800 kW, to be removed from service, as described below:

One proportional pressurizer heater (50 kw), and;

Fifteen backup pressurizer heaters (750 kW). Diesel backed heater banks should be maintained at full capacity to provide margin over the Technical Specification heater requirement of 150 kW.

Heaters available from backup banks (non diesel backed) can be utilized to replace heaters removed from service from the proportional heater banks and diesel backed banks.

TABLE 5.4-7

QUENCH TANK PARAMETERS

Design Pressure (Int./Ext.)	100/15 psig
Design Temperature	350 F
Normal Operating Pressure	3 psig
Normal Operating Temperature	120 F
Internal Volume	209 ft^3
Normal Water Volume	127 ft^3
Normal Gas Volume	82 ft^3
Blanket Gas	Nitrogen
Nozzles	
Pressurizer line inlet, (1)	10 inches, sch 40
Demineralized water inlet, (1)	2 inches
Manway and rupture disc, (1)	18 inches
Vent/relief, (1)	1.5 inches
Drain (1)	2 inches
Temperature instrument, (1)	1 inch
Level instrument (2)	1 inch
Rupture Disc Set Pressure @ 350 F	85 psig @ 350 F
Relief Valve	
Set pressure	70 psig
Accumulation, % of set pressure	10%
Blowdown, % of set pressure	10%
Capacity	500 scfm
Vessel Material	ASME-SA-240-Tp-304
Code	ASME Section III, Class 3 1974 Edition

TABLE 5.4-8

PRESSURIZER SAFETY VALVE PARAMETERS

Property	Parameter
Design Pressure, psia	2500
Design Temperature, °F	700
Fluid	Saturated steam, 2000 ppm H_3BO_3, pH=5.0
Set Pressure, psia	2500 $\pm$ 1% (Note 1)
Minimum Capacity, lb/hr at 3% accumulation	200,000
Orifice Area, square inches	1.838
Accumulation, %	3
Backpressure	
Max buildup/max superimposed, psig	500/300
Approx. Dry Weight, pounds	792
Minimum Blowdown Pressure, psia	2125
Materials	
Body	ASME SA 182 GRF316, ASTM A351, GR. CF8M OR EQUIVALENT
Disc Insert	ASTM A637 GR 718
Nozzle	ASME SA 182 GR F316
Bonnet	ASME SA 105
Spring	ASTM A689
Code	ASME Section III, Class 1, 1974 Edition

Note 1: As -found setpoint is 2500 psia +/-2% for operability. Values are reset to 2500 psia +/-1% during surveillance to allow for drift.

TABLE 5.4-9

POWER OPERATED RELIEF VALVE PARAMETERS

(Valves 1474 and 1475)

I. Operational Requirements	Normal Service	LTOP Service
Fluid	Saturated Steam*	Water*
Operating Pressure, psia	2250	300
Operating Temperature, °F	653	417
Set Pressure, psia	2370	490 (see Note 1)
Backpressure, Buildup, psia	500	150
Minimum Capacity @ Set Pressure	153,000 lb/hr	1420 gpm (see Notes 1 & 2)

*Service: Pressurizer pressure relief of saturated steam and water with 2000 ppm H_3BO_3 pH 5.

II. Design Requirements		
Design Pressure, psia	2500	
Design Temperature, °F	675	
Valve Type	Solenoid Operated Relief Valve	
Seismic Class	I	
Quality Group	A	
Code	ASME Section III	(refer to Table 5.2-1 for code date)
Active	Yes	
Inlet Line Size, Inches	3	
Outlet Line Size, Inches	8	
Body Material	ASME SA 182, Grade F 316	
Failure Position	Fail close on loss of Electrical Power	

Safety Related Function	Open/Close
Solenoid Voltage Requirement	90-140 V D.C.

Notes:

1. Valve capacity corresponds to a nominal setpoint of 465 psia.

2. The valve manufacturer demonstrates via a combination of testing and analysis that the saturated steam and water flow capacity requirements are met.

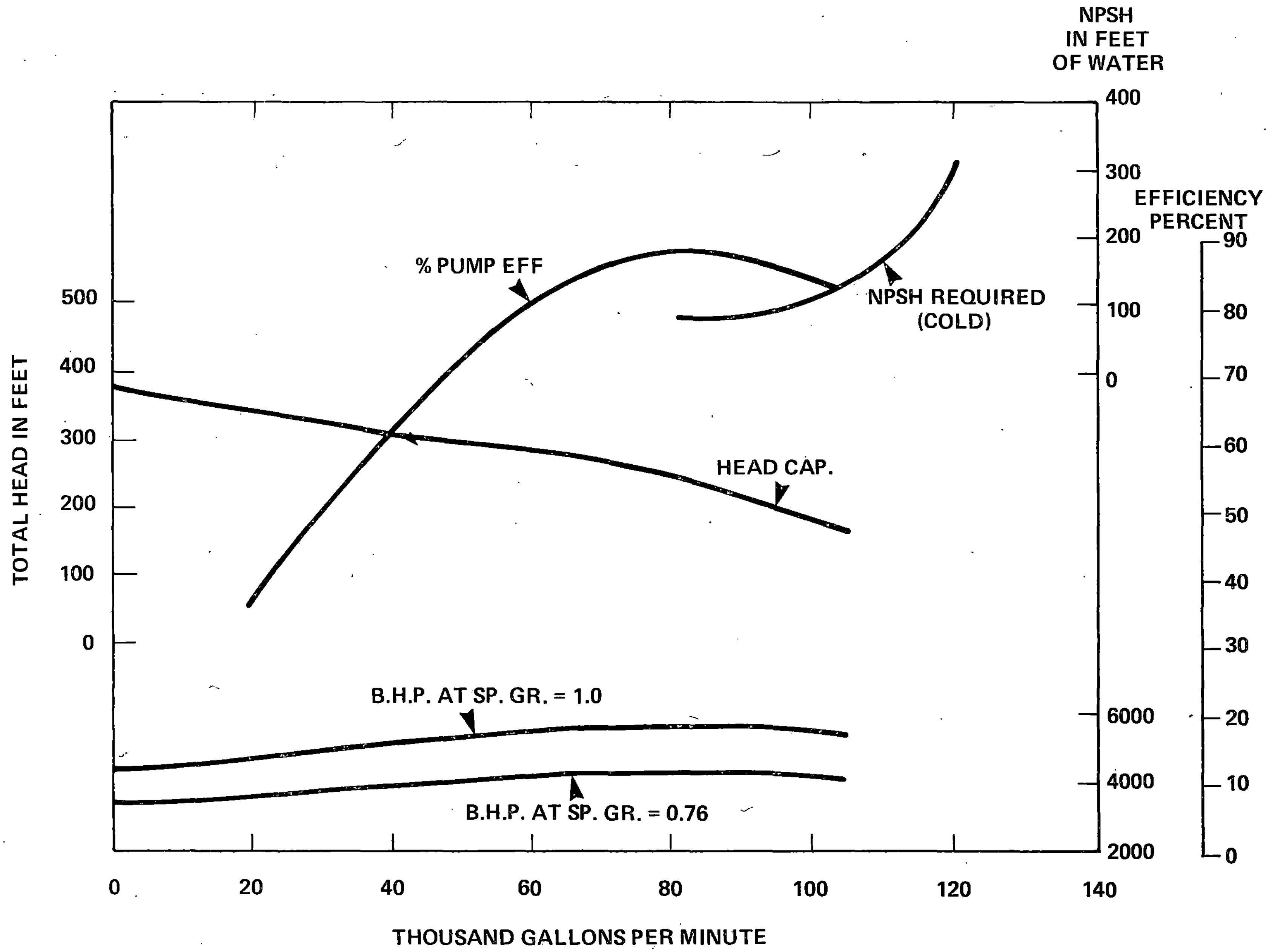
Refer to Drawing
2998-455

Amendment No. 18 (08/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

SECTIONAL ASSY REACTOR PRIMARY COOLANT PUMPS

FIGURE 5.4-1



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR COOLANT PUMP
PERFORMANCE
FIGURE 5.4-2

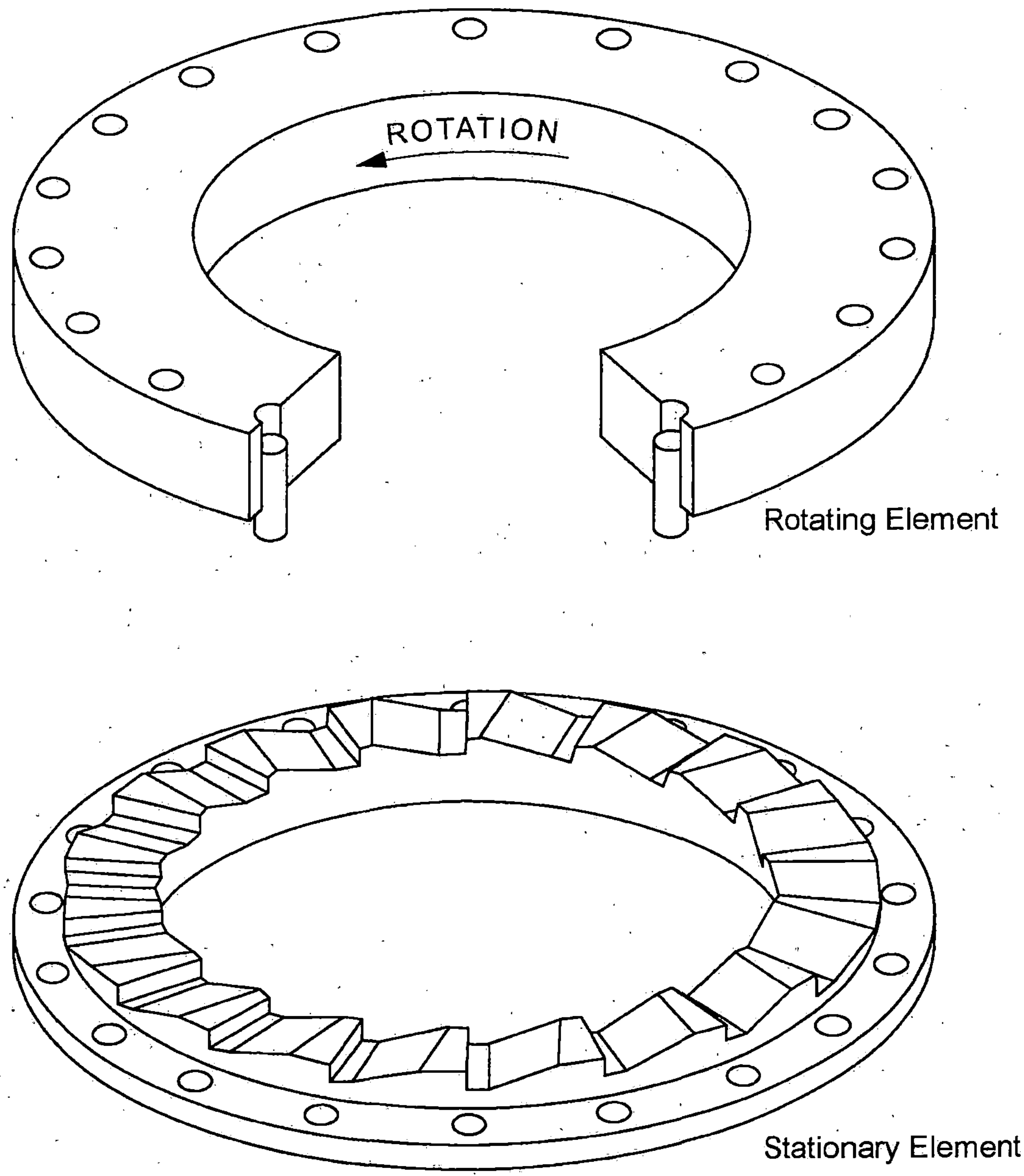
Refer to Drawing
2998-19778

Amendment No. 18 (01/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REACTOR COOLANT PUMP SHAFT
SEAL ARRANGEMENT N-9000 SEAL

FIGURE 5.4-3

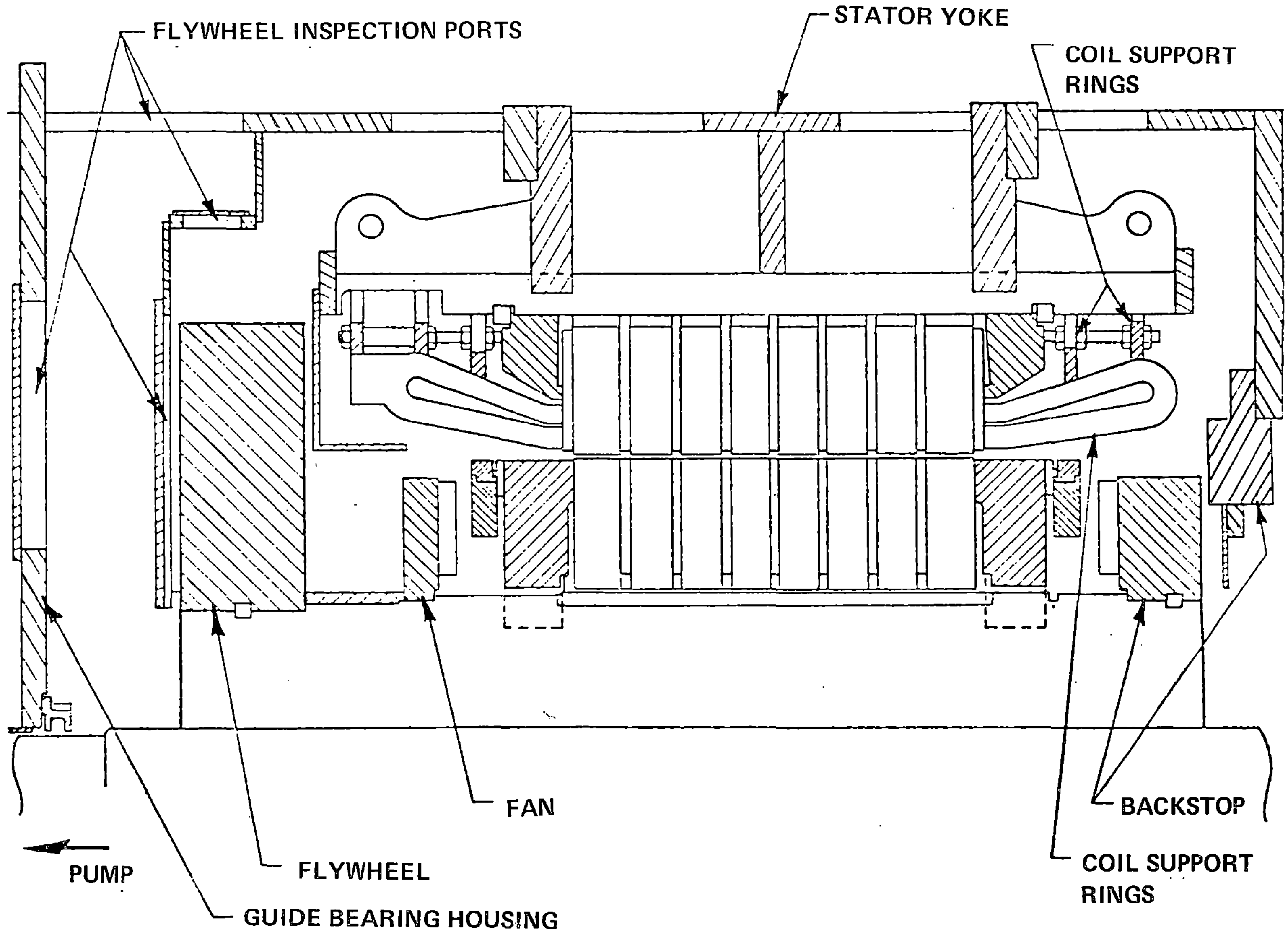


Amendment No. 18 (01/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ANTIROTATIONAL DEVICE

FIGURE 5.4-4



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MOTOR FLYWHEEL ASSEMBLY

FIGURE 5.4-5

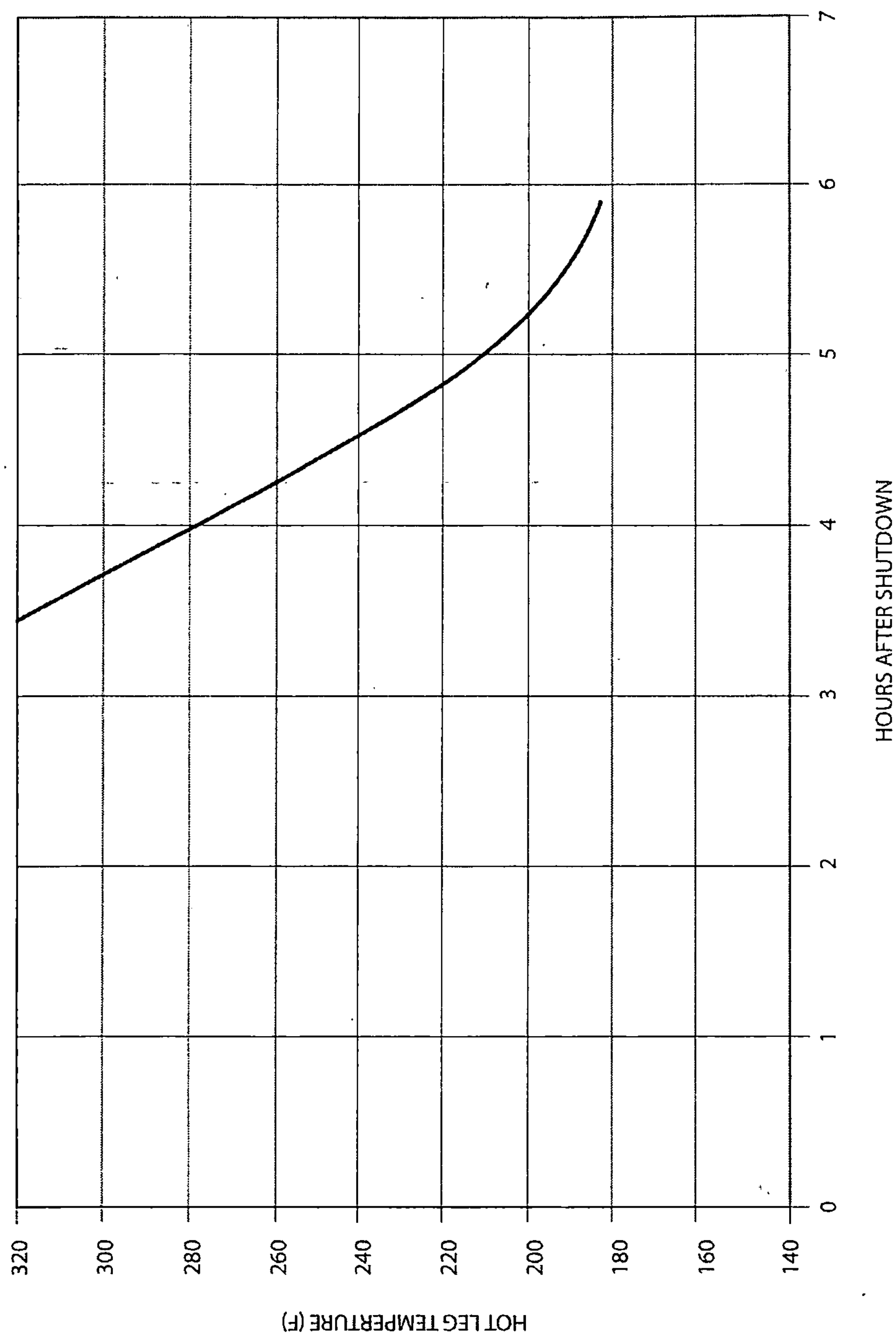
SEE FPL DRAWING 2998-21514

Amendment No. 22 (04/14)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

REPLACEMENT STEAM GENERATOR

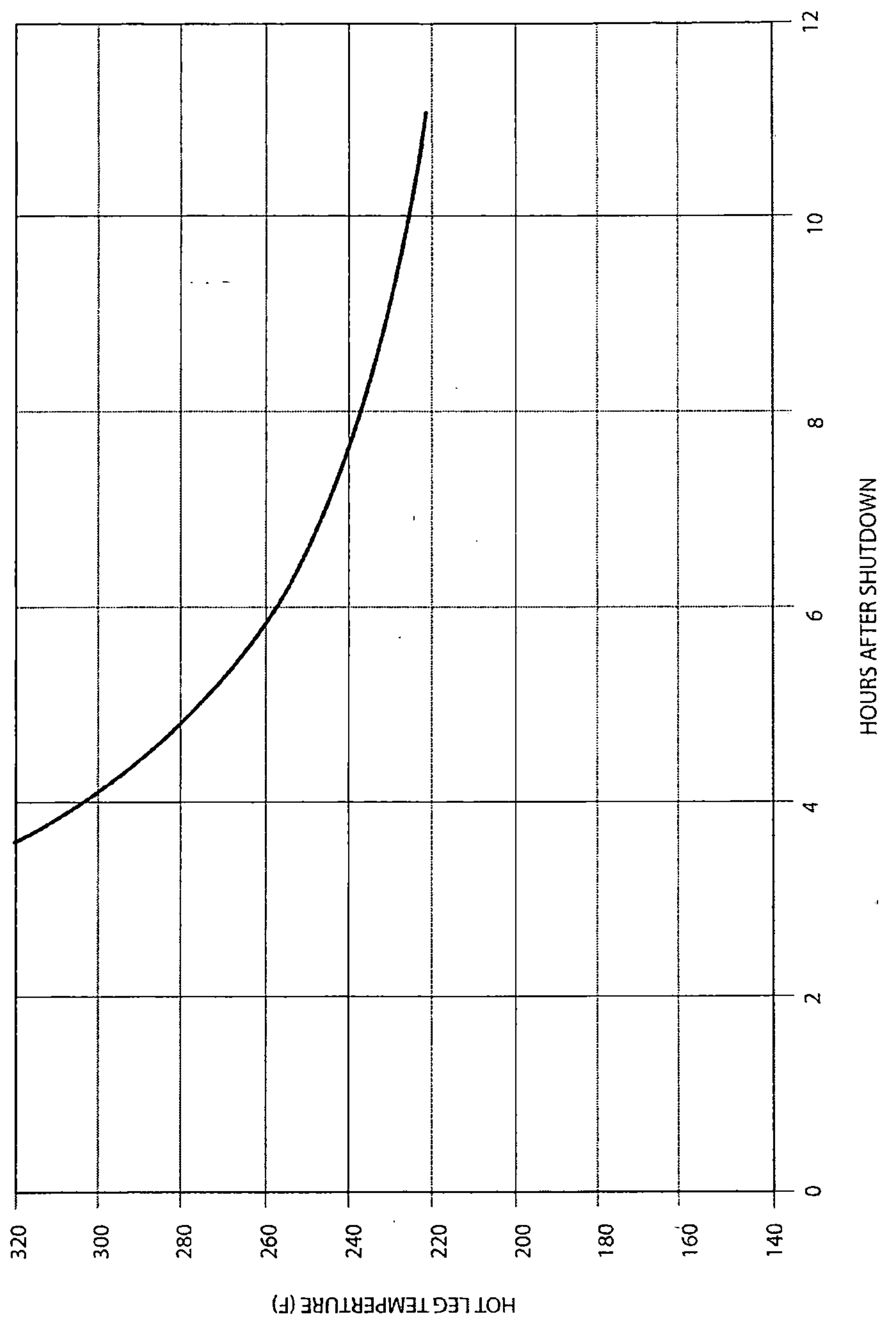
FIGURE 5.4-6



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

TWO TRAIN COOLDOWN
FROM 325 F

FIGURE 5.4-7



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

ONE TRAIN COOLDOWN
FROM 325 F

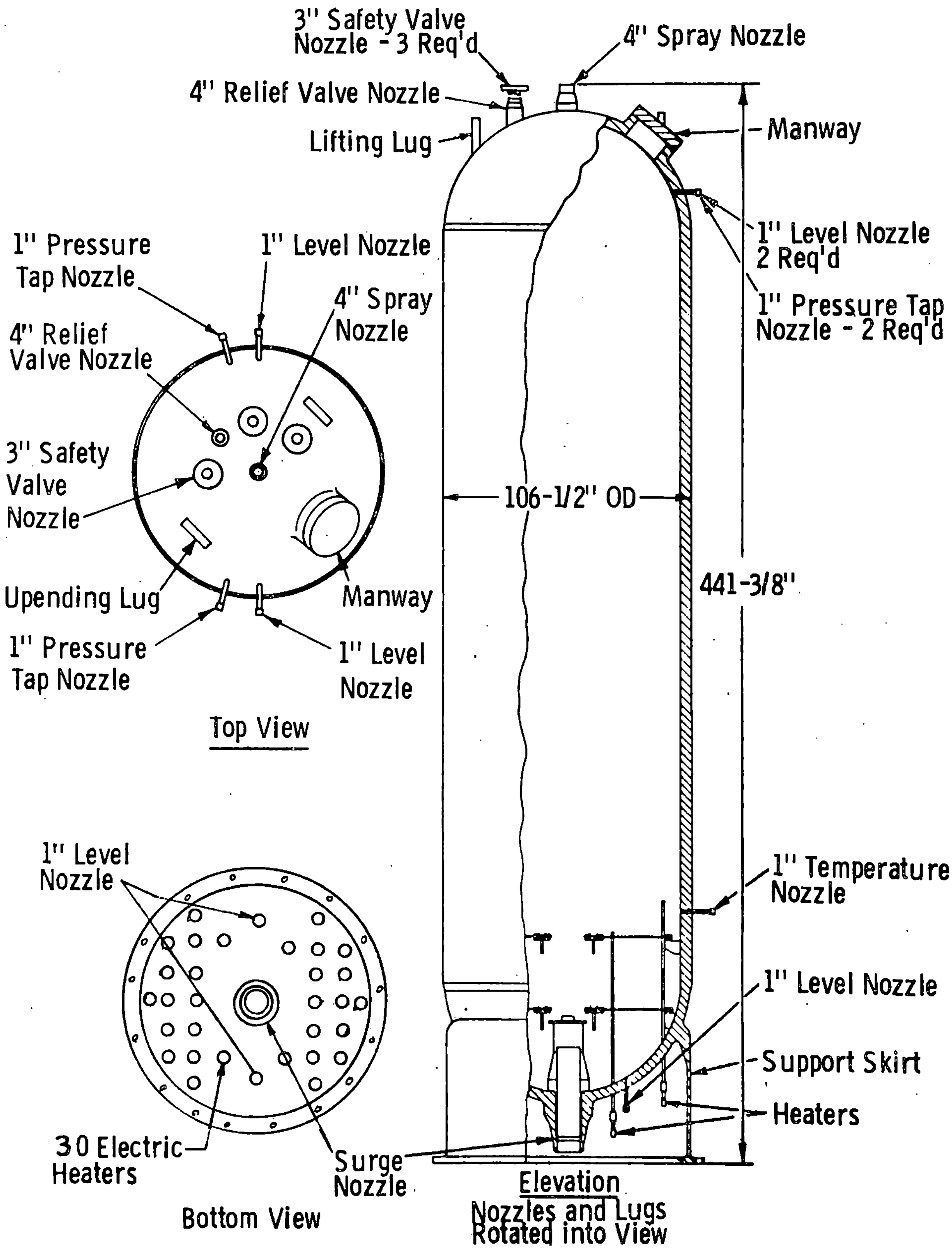
FIGURE 5.4-8

DELETED

Amendment No. 18 (01/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

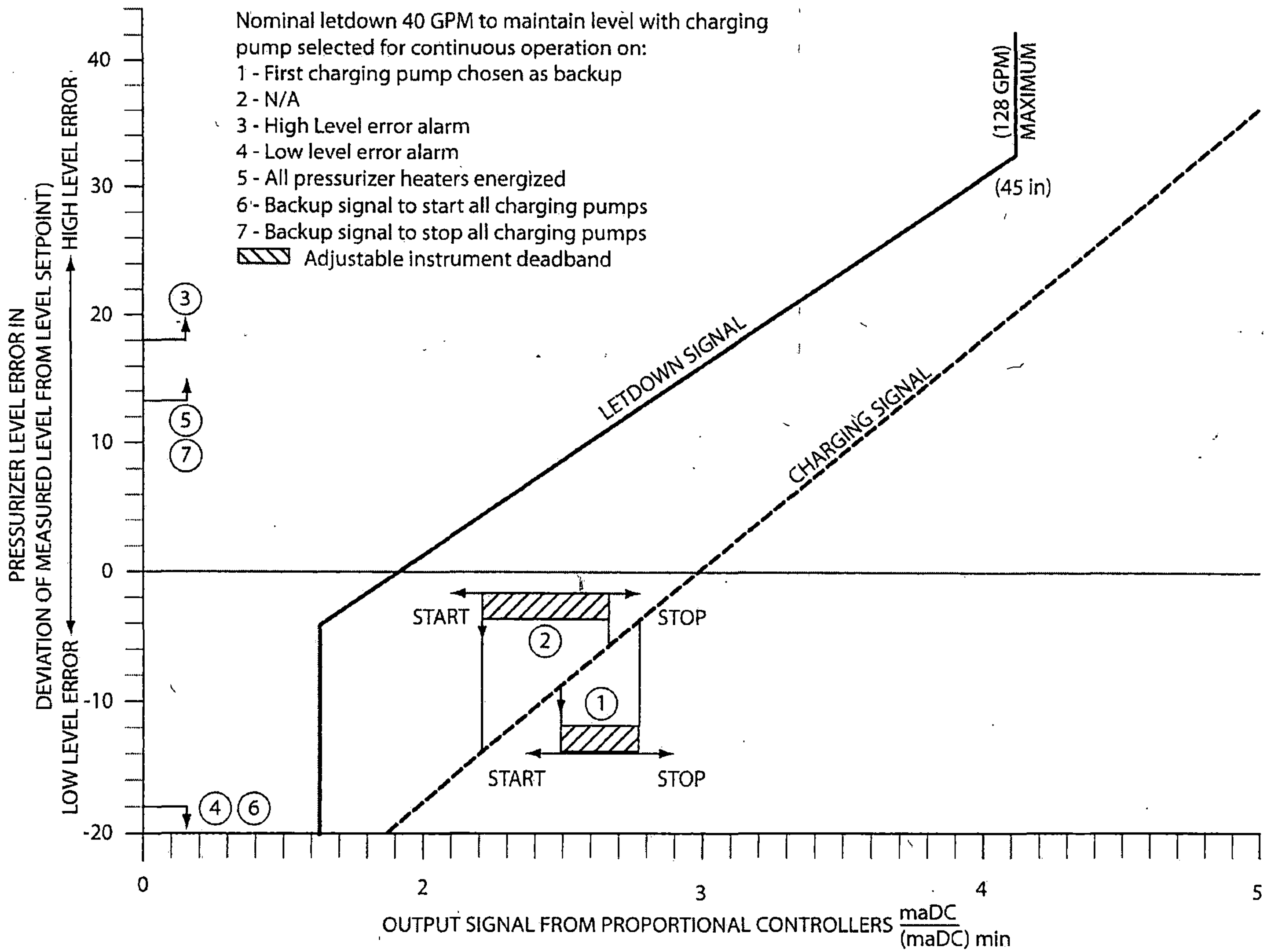
FIGURE 5.4-9



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

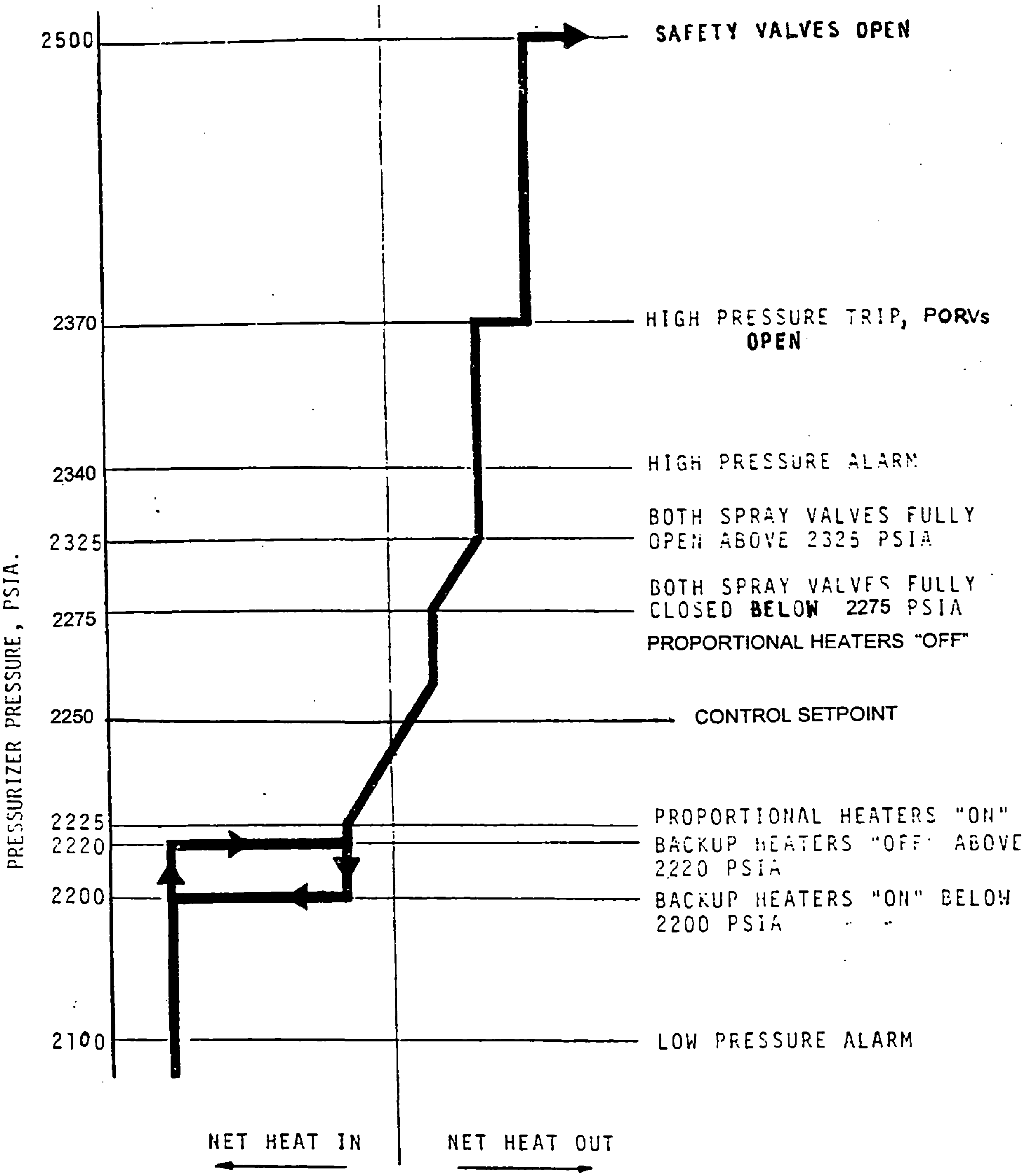
PRESSURIZER

FIGURE 5.4-10



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2
PRESSURIZER LEVEL CONTROL PROGRAM
FIGURE 5.4-11

Amendment No. 21 (11/12)



Amendment No. 13, (05/00)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

PRESSURIZER PRESSURE CONTROL PROGRAM

FIGURE 5.4-12

Refer to Drawing
2998-3858

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

QUENCH TANK

FIGURE 5.4-13

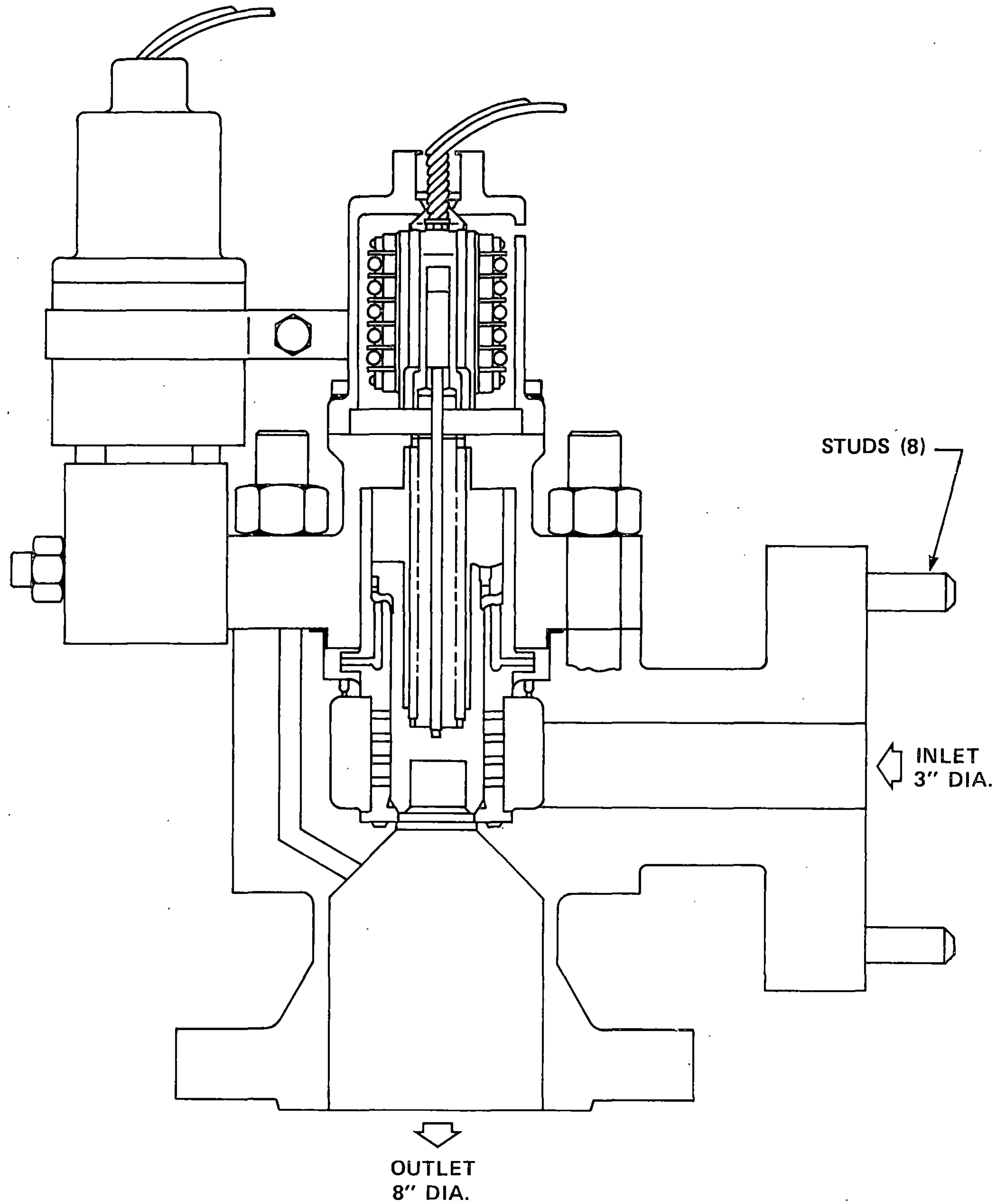
Refer to Drawing
2998-19690

Amendment No. 18 (01/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

PRESSURIZER CODE SAFETY VALVE
REPLACEMENT FORGED BODY DESIGN
SHEET 1 OF 2

FIGURE 5.4-14



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

POWER OPERATED RELIEF VALVE
FIGURE 5.4-15

SEE FPL DRAWING 2998-21367

Amendment No. 22 (04/14)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

STEAM GENERATOR
FEEDWATER RING
SHEET 1
FIGURE 5.4-16

SEE FPL DRAWING 2998-21548

Amendment No. 22 (04/14)

FLORIDA POWER & LIGHT COMPANY **ST. LUCIE PLANT UNIT 2**
STEAM GENERATOR FEEDWATER RING SHEET 2 **FIGURE 5.4-17**

FSAR User Comment Form

FSAR errors or improvement suggestions should be identified below by FSAR Users and forwarded to the appropriate Nuclear Engineering Project Licensing Supervisor.

Originator ____________ Dept ____________ Location ____________ Phone ____________

Plant PTN _____ PSL 1 _____ PSL 2 _____

FSAR Areas Affected

Sections ____________ ____________ Figures ____________ ____________

Comments Attached _______ Below _______

Engineering Review (To be completed by Project Licensing)

Accepted _______ Insufficient Information _______ No Change Required _______

Disposition: ____

Assigned User Comment # ________________ Reviewing Engineer ________________

LIST OF EFFECTIVE PAGES

CHAPTER 6

ENGINEERED SAFETY FEATURES

Page	Amendment	Page	Amendment
6-1	23	6.1-7	-
6-2	23	6.1-8	21
6-3	23	6.1-8a	14
6-4	23	6.1-9	12
6-5	23	6.1-10	19
6-6	23	6.1-10a	-
6-7	23	6.1-11	19
6-8	23	6.1-12	-
6-9	23	6.1-13	12
		6.1-14	12
6-i	21		
6-ii	18	6.2-1	21
6-iia	-	6.2-1a	21
6-iii	21	6.2-2	21
6-iv	21	6.2-2a	21
6-v	13	6.2-3	21
6-vi	13	6.2-4	21
6-vii	21	6.2-5	20
6-viii	23	6.2-6	21
6-viiia	21	6.2-7	21
6-ix	22A	6.2-8	21
6-x	19	6.2-9	21
6-xi	21	6.2-10	21
6-xii	8	6.2-11	21
6-xiii	-	6.2-11a	21
6-xiv	-	6.2-12	21
6-xv	13	6.2-12a	19
6-xvi	21	6.2-12b	21
6-xvii	18	6.2-13	20
6-xviii	-	6.2-14	20
6-xix	-	6.2-14a	20
6-xx	-	6.2-15	-
6-xxi	-	6.2-16	20
6-xxii	-	6.2-16a	1
6-xxiii	-	6.2-17	21
6-xxiv	-	6.2-18	21
6-xxv	-	6.2-19	21
6-xxvi	19	6.2-20	8
		6.2-21	21
6.1-1	18	6.2-22	21
6.1-2	14	6.2-23	21
6.1-3	-	6.2-24	13
6.1-4	18	6.2-25	21
6.1-5	21	6.2-25a	21
6.1-6	-	6.2-26	21
		6.2-27	18
		6.2-28	13

LIST OF EFFECTIVE PAGES

CHAPTER 6

Page	Amendment	Page	Amendment
6.2-29	-	6.2-71	-
6.2-30	23	6.2-72	23
6.2-31	11	6.2-73	18
6.2-32	21	6.2-74	18
6.2-33	13	6.2-75	18
6.2-34	23	6.2-76	12
6.2-34a	19	6.2-77	18
6.2-35	13	6.2-78	-
6.2-36	1	6.2-79	20
6.2-37	19	6.2-80	20
6.2-37a	19	6.2-81	23
6.2-38	19	6.2-82	21
6.2-39	19	6.2-83	21
6.2-40	21	6.2-84	21
6.2-41	18	6.2-85	21
6.2-42	23	6.2-86	21
6.2-42a	20	6.2-87	21
6.2-43	19	6.2-88	21
6.2-44	19	6.2-89	21
6.2-45	18	6.2-90	21
6.2-46	21	6.2-91	21
6.2-47	21	6.2-92	21
6.2-48	21	6.2-93	-
6.2-48a	21	6.2-94	18
6.2-49	21	6.2-95	-
6.2-49a	21	6.2-96	-
6.2-50	21	6.2-97	-
6.2-51	18	6.2-98	-
6.2-52	19	6.2-99	19
6.2-53	18	6.2-100	8
6.2-54	18	6.2-101	21
6.2-55	10	6.2-102	20
6.2-56	18	6.2-103	21
6.2-56a	20	6.2-104	16
6.2-57	18	6.2-105	21
6.2-58	-	6.2-106	21
6.2-59	21	6.2-107	13
6.2-60	18	6.2-108	8
6.2-61	21	6.2-109	8
6.2-61a	1	6.2-110	8
6.2-62	18	6.2-111	8
6.2-63	20	6.2-112	8
6.2-63a	-	6.2-113	1
6.2-64	-	6.2-114	-
6.2-65	18	6.2-115	13
6.2-66	-	6.2-116	8
6.2-67	18	6.2-117	8
6.2-68	18	6.2-118	8
6.2-69	20		
6.2-70	18		
6.2-70a	21		

Page	Amendment	Page	Amendment
6.2-119	8	6.2-165	-
6.2-120	13	6.2-166	1
6.2-121	8	6.2-167	-
6.2-122	8	6.2-168	1
6.2-123	8	6.2-169	-
6.2-124	8	6.2-170	1
6.2-125	8	6.2-171	21
6.2-126	13	6.2-172	21
6.2-127	8	6.2-173	21
6.2-128	8	6.2-174	21
6.2-129	8	6.2-175	1
6.2-129a	8	6.2-176	-
6.2-129b	8	6.2-177	-
6.2-130	13	6.2-178	18
6.2-131	21	6.2-179	13
6.2-131a	21	6.2-180	13
6.2-131b	21	6.2-181	13
6.2-132	21	6.2-182	13
6.2-132a	21	6.2-183	13
6.2-132b	21	6.2-184	21
6.2-132c	21	6.2-185	13
6.2-133	21	6.2-186	8
6.2-134	21	6.2-187	21
6.2-135	21	6.2-188	13
6.2-136	21	6.2-189	13
6.2-137	21	6.2-190	18
6.2-138	21	6.2-191	8
6.2-139	21	6.2-192	17
6.2-140	21	6.2-193	21
6.2-141	21	6.2-194	-
6.2-142	21	6.2-195	-
6.2-143	21	6.2-196	18
6.2-144	21	6.2-197	21
6.2-145	21	6.2-198	-
6.2-146	21	6.2-199	-
6.2-147	21	6.2-200	21
6.2-148	21	6.2-201	12
6.2-149	21	6.2-202	21
6.2-150	21	6.2-203	18
6.2-151	21	6.2-204	-
6.2-152	21	6.2-205	19
6.2-153	21	6.2-206	21
6.2-154	21	6.2-207	21
6.2-155	21	6.2-208	18
6.2-156	21	6.2-209	18
6.2-157	21		
6.2-158	21		
6.2-159	21		
6.2-160	1		
6.2-161	-		
6.2-162	1		
6.2-163	-		
6.2-164	1		

LIST OF EFFECTIVE PAGES

CHAPTER 6

Page	Amendment	Page	Amendment
6.2-210	12	6.2-253	1
6.2-211	14	6.2-254	1
6.2-212	20	6.2-255	1
6.2-213	20	6.2-256	1
6.2-214	20	6.2-257	1
6.2-215	20	6.2-258	1
6.2-215a	20	6.2-259	1
6.2-216	16	6.2-260	1
6.2-217	17	6.2-261	1
6.2-218	14	6.2-262	1
6.2-219	18	6.2-263	1
6.2-220	-	6.2-264	1
6.2-221	14	6.2-265	1
6.2-222	14	6.2-266	1
6.2-223	-	6.2-267	21
6.2-224	2	6.2-268	21
6.2-224a	2	6.2-269	21
6.2-225	-	6.2-270	21
6.2-226	-	6.2-271	21
6.2-227	-	6.2-272	21
6.2-228	13	6.2-273	21
6.2-228a	23	6.2-274	21
6.2-229	12		
6.2-230	14	F6.2-1	8
6.2-231	20	F6.2-2a	21
6.2-232	18	F6.2-2b	21
6.2-233	18	F6.2-3a	21
6.2-234	18	F6.2-4	8
6.2-235	18	F6.2-5	8
6.2-236	18	F6.2-6	8
6.2-237	18	F6.2-7	8
6.2-238	18	F6.2-8	8
6.2-239	18	F6.2-9	21
6.2-240	18	F6.2-9a	21
6.2-241	18	F6.2-10	21
6.2-242	18	F6.2-11	21
6.2-243	18	F6.2-11a	21
6.2-244	21	F6.2-12	21
6.2-244a	21	F6.2-12a	21
6.2-245	13	F6.2-13 (DELETED)	21
6.2-246	8	F6.2-14 (DELETED)	21
6.2-247	8		
6.2-248	21		
6.2-249	1		
6.2-250	1		
6.2-251	1		
6.2-252	1		

LIST OF EFFECTIVE PAGES

CHAPTER 6

Page	Amendment	Page	Amendment
F6.2-15 (DELETED)	21	F6.2-28t	21
F6.2-16	-	F6.2-28u	21
F6.2-17	-	F6.2-28v	21
F6.2-18	-	F6.2-28w	21
F6.2-19	-	F6.2-28x	21
F6.2-20	-	F6.2-28y	21
F6.2-21	-	F6.2-29	-
F6.2-22	8	F6.2-30a	-
F6.2-23	21	F6.2-30b	-
F6.2-24	21	F6.2-30c	-
F6.2-25a	21	F6.2-30d	-
F6.2-25b	21	F6.2-30e	-
F6.2-25c	21	F6.2-30f	-
F6.2-25d	21	F6.2-31	-
F6.2-25e	21	F6.2-32	-
F6.2-25f	21	F6.2-33	21
F6.2-25g	21	F6.2-34	21
F6.2-25h	21	F6.2-35	21
F6.2-25i	21	F6.2-36	8
F6.2-25j	21	F6.2-37	-
F6.2-25k	21	F6.2-38	-
F6.2-25l	21	F6.2-39	-
F6.2-25m	21	F6.2-40	-
F6.2-25n	21	F6.2-41	10
F6.2-26	-	F6.2-42	21
F6.2-27	-	F6.2-43	21
F6.2-28a	21	F6.2-44	19
F6.2-28b	21	F6.2-45	19
F6.2-28c	21	F6.2-46	18
F6.2-28d	21	F6.2-47	18
F6.2-28e	21	F6.2-48	-
F6.2-28f	21	F6.2-49	-
F6.2-28g	21	F6.2-50	-
F6.2-28h	21	F6.2-51	19
F6.2-28i	21	F6.2-52	-
F6.2-28j	21	F6.2-53	21
F6.2-28k	21	F6.2-54	21
F6.2-28l	21	F6.2-55 (DELETED)	21
F6.2-28m	21	F6.2-56	21
F6.2-28n	21	F6.2-56a	21
F6.2-28o	21	F6.2-57	21
F6.2-28p	21	F6.2-58	21
F6.2-28q	21	F6.2-59	21
F6.2-28r	21	F6.2-60	-
F6.2-28s	21	F6.2-61	-
		F6.2-62	18

LIST OF EFFECTIVE PAGES

CHAPTER 6

Page	Amendment
F6.2-63	-
F6.2-64	-
F6.2-65	-
F6.2-66	-
F6.2-67	-
F6.2-68	-
6.2A-1	21
6.2A-2	-
6.2A-3	21
6.2A-4	21
6.2A-5	-
6.2A-6	21
6.2A-7	21
6.2A-8	-
6.2A-9	20
F6.2A-1	-
F6.2A-2	-
6.2B-1	21
6.2B-2	-
6.2B-3	21
6.3-1	18
6.3-2	17
6.3-3	18
6.3-3a	12
6.3-3b	20
6.3-4	15
6.3-5	23
6.3-6	21
6.3-6a	21
6.3-6b	12
6.3-7	18
6.3-8	18
6.3-9	18
6.3-9a	11
6.3-10	19
6.3-11	23
6.3-12	13
6.3-13	12

Page	Amendment
6.3-14	18
6.3-15	16
6.3-16	-
6.3-17	18
6.3-18	13
6.3-18oa	20
6.3-18oa(i)	20
6.3-18oa (ii)	16
6.3-18oa (iii)	-
6.3-18a	18
6.3-18b	21
6.3-18c	16
6.3-18d	16
6.3-18e	23
6.3-18f	18
6.3-19	13
6.3-20	12
6.3-21	11
6.3-22	21
6.3-23	21
6.3-23a	4
6.3-24	-
6.3-25	13
6.3-26	20
6.3-27	20
6.3-28	20
6.3-29	22A
6.3-29a	21
6.3-29b	20
6.3-29c	20
6.3-29d	20
6.3-29e	20
6.3-29f	-
6.3-30	20
6.3-31	20
6.3-32	-
6.3-33	-
6.3-34	-
6.3-35	-
6.3-36	18
6.3-37	18
6.3-37a	18

LIST OF EFFECTIVE PAGES

CHAPTER 6

Page	Amendment
6.3-38	-
6.3-39	12
6.3-40	1
6.3-41	1
6.3-42	1
6.3-43	1
6.3-44	-
6.3-45	1
6.3-46	16
6.3-47	20
6.3-48	-
6.3-49	-
6.3-50	21
6.3-51	-
6.3-52	12
6.3-53	13
6.3-54	-
6.3-55	-
F6.3-1a	18
F6.3-1b	18
F6.3-1c	18
F6.3-2a	10
F6.3-2b	10
F6.3-2c	10
F6.3-2d	1
F6.3-3a	-
F6.3-3b	-
F6.3-4a	-
F6.3-4b	-
F6.3-5a	1
F6.3-5b	1
F6.3-5c	1
F6.3-5d	1
F6.3-5e	1
F6.3-5f	1
F6.3-5g	1
F6.3-5h	1
F6.3-5i	1
F6.3-6a	1
F6.3-6b	1
F6.3-6c	1
F6.3-6d	1
F6.3-6e	1
F6.3-6f	1
F6.3-6g	1
F6.3-6h	1
F6.3-6i	1
F6.3-7a	1
F6.3-7b	1
F6.3-7c	1
F6.3-7d	1
F6.3-7e	1
F6.3-7f	1
F6.3-7g	1
F6.3-7h	1
F6.3-7i	1
F6.3-8a	1
F6.3-8b	1
F6.3-8c	1
F6.3-8d	1
F6.3-8e	1
F6.3-8f	1
F6.3-8g	1
F6.3-8h	1
F6.3-8i	1
F6.3-9a	1
F6.3-9b	1
F6.3-9c	1
F6.3-9d	1
F6.3-9e	1
F6.3-9f	1
F6.3-9g	1
F6.3-9h	1
F6.3-9i	1
F6.3-9j	1
F6.3-9k	1
F6.3-9l	1
F6.3-9m	1
F6.3-9n	1
F6.3-9o	1
F6.3-9p	1
F6.3-9q	1
F6.3-9r	1

Page	Amendment	Page	Amendment
F6.3-9s	1	F6.3-16h	1
F6.3-9t	1	F6.3-17a	1
F6.3-10a	1	F6.3-17b	1
F6.3-10b	1	F6.3-17c	1
F6.3-10c	1	F6.3-17d	1
F6.3-10d	1	F6.3-17e	1
F6.3-10e	1	F6.3-17f	1
F6.3-10f	1	F6.3-17g	1
F6.3-10g	1	F6.3-17h	1
F6.3-10h	1	F6.3-18a	1
F6.3-10i	1	F6.3-18b	1
F6.3-11a	1	F6.3-18c	1
F6.3-11b	1	F6.3-18d	1
F6.3-11c	1	F6.3-18e	1
F6.3-11d	1	F6.3-18f	1
F6.3-11e	1	F6.3-18g	1
F6.3-11f	1	F6.3-18h	1
F6.3-11g	1	F6.3-19	1
F6.3-11h	1	F6.3-19a	1
F6.3-11i	1	F6.3-19b	1
F6.3-12	1	F6.3-19c	1
F6.3-13	1	F6.3-19d	1
F6.3-14a	1	F6.3-19e	1
F6.3-14b	1	F6.3-19f	1
F6.3-14c	1	F6.3-19g	1
F6.3-14d	1	F6.3-19h	1
F6.3-14e	1	F6.3-20	1
F6.3-14f	1	F6.3-21	3
F6.3-14g	1	F6.3-22	1
F6.3-14h	1	F6.3-23	1
F6.3-15a	1	F6.3-24	1
F6.3-15b	1	F6.3-25	-
F6.3-15c	1	F6.3-26	1
F6.3-15d	1	F6.3-27	1
F6.3-15e	1	F6.3-28	1
F6.3-15f	1		
F6.3-15g	1	6.4-1	19
F6.3-15h	1	6.4-2	19
F6.3-16a	1	6.4-3	19
F6.3-16b	1	6.4-4	11
F6.3-16c	1	6.4-5	21
F6.3-16d	1	6.4-6	21
F6.3-16e	1	6.4-7	21
F6.3-16f	1	6.4-8	16
F6.3-16g	1	6.4-9	21

Page	Amendment	Page	Amendment
6.4-10	16		
6.4-11	21		
F6.4-1	19		
F6.4-2	19		
6.5-1	-		
6.5-2	18		
6.5-3	19		
6.5-4	23		
6.5-5	-		
6.5-6	21		
6.5-7	21		
6.5-8	21		
6.5-9	21		
6.5-9a	21		
6.5-10	21		
6.5-10a	21		
6.5-11	-		
6.5-12	-		
6.5-13	19		
6.5-14	19		
6.5-15	19		
6.5-16	18		
6.5-17	15		
6.5-18	21		
6.5-18a	15		
6.5-19	21		
6.5-20	21		
6.5-21	21		
6.5-22	-		
6.5-23	21		
6.5-24	14		
F6.5-1	19		
F6.5-2	19		
F6.5-3	19		
F6.5-4	10		
F6.5-5	10		
F6.5-6	-		
F6.5-7	18		
F6.5-8	21		
F6.5-9	19		
6.6-1	18		
6.6-2	-		

ENGINEERED SAFETY FEATURES

CHAPTER 6

TABLE OF CONTENTS

CHAPTER 6

TABLE OF CONTENTS (Cont'd)

CHAPTER 6

TABLE OF CONTENTS (Cont'd)

ENGINEERED SAFETY FEATURES

CHAPTER 6

LIST OF TABLES

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

CHAPTER 6

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

LIST OF FIGURES

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

CHAPTER 6

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

CHAPTER 6

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

6.0 ENGINEERED SAFETY FEATURES

Engineered safety features (ESF) are designed to mitigate the consequences of postulated design basis accidents (DBA) (notwithstanding the fact that these accidents are highly unlikely). The ESF function to limit, contain, control and terminate an accidental release of radioactive fission products, particularly as a result of accidents that could release large amounts of energy within the containment structure. ESF systems keep exposure levels to the public and plant personnel below applicable limits.

St. Lucie Unit 2 utilizes the following ESF systems:

a) Containment Structure (Subsections 6.2.1 and 6.2.3)

The containment structure is a steel containment vessel surrounded by a reinforced concrete Shield Building. The two structures are separated by an annular air space. The containment vessel is a low leakage cylindrical steel shell with hemispherical dome and ellipsoidal bottom. The vessel is designed to contain the radioactive material that could be released from a loss of integrity of the reactor coolant pressure boundary. The Shield Building is a concrete structure which protects the containment vessel from external missiles, provides biological shielding, and provides a means of controlling radioactive fission products that could leak from the containment vessel if an accident should occur. Detail design information for the containment structure is presented in Subsection 3.8.2.

b) Containment Spray System (Subsections 6.2.2 and 6.5.2)

The Containment Spray System removes heat from the containment by spraying water droplets through the containment atmosphere. The spray system is automatically initiated by the containment spray actuation signal (CSAS) which requires a coincidence of the safety injection actuation signal (SIAS) and the high-high containment pressure signal. The Containment Spray System consists of two redundant subsystems each consisting of a spray pump, a shutdown cooling heat exchanger and a spray header.

When used with the Containment Cooling System to mitigate the consequences of the DBA, one containment spray subsystem along with two containment fan coolers provides 100 percent DBA heat removal capability.

The Containment Spray System operates in conjunction with the Iodine Removal System which adds a hydrazine solution to the spray water in order to remove iodine from the containment post-DBA atmosphere.

c) Containment Cooling System (Subsection 6.2.2)

The Containment Cooling System removes heat from the containment by passing containment air over coils cooled by the Component Cooling Water System.

The system consists of four cooling units each of which consists of cooling coils served by the Component Cooling Water System and an axial flow fan. During normal operation, three of the four units are in service.

On receipt of a SIAS, the fourth unit is automatically started. Two containment fan coolers, along with one containment spray subsystem, provide 100 percent DBA heat removal capability, as noted in part b) above.

d) Shield Building Ventilation System (Subsections 6.2.3 and 6.5.1)

The Shield Building Ventilation System (SBVS) maintains a negative pressure in the Shield Building annulus following a design basis accident, mixes Shield Building in-leakage, with the air in the annulus and with any leakage from the containment, and discharges it through a filter train which includes charcoal adsorbers. Also, the Fuel Pool Ventilation System is connected to the SBVS and maintains a slightly negative pressure in the Fuel Handling Building following a fuel handling accident in that building.

The SBVS consists of two full capacity redundant fan and filter systems located in the Reactor Auxiliary Building which filter air from the annular space and discharge through a vent stack to the atmosphere. On the suction side of each fan, the air is drawn sequentially through an electric heater, a demister, another electric heater, HEPA filter, charcoal absorber, and HEPA afterfilter. A backdraft damper is located on the discharge side of the exhaust fan to preclude backflow through the system. An automatically operated damper is also provided on the discharge side of the fan to control the flow at 6000 cfm when negative pressure is attained in the annulus or Fuel Handling Building.

Both full capacity fan and filter systems are automatically started upon receipt of a containment isolation actuation signal (CIAS) or a high radiation signal from the Fuel Handling Building. One fan-filter system can be manually shutdown and placed in a standby mode, whereby the standby system is automatically restarted if the operating system should fail.

e) Containment Isolation System (Subsection 6.2.4)

The containment isolation system isolates systems that pass through containment penetrations such that any radioactivity that may be released into the containment atmosphere following a postulated design basis accident is contained. The containment isolation system is required to function following, a DBA to isolate non-ESF systems penetrating the containment. Barriers are comprised of seismic Category I piping systems, both inside and outside the containment vessel, and manual and automatic isolation valves. Containment isolation is achieved upon receipt of a CIAS, thus limiting the potential radioactive releases.

f) Combustible Gas Control System (Subsection 6.2.5)

Combustible gas control in the containment following a DBA is accomplished by the use of the Hydrogen Sampling System in conjunction with the redundant, 100 percent capacity hydrogen recombiners.

The Hydrogen Sampling System analyzes the post-DBA containment atmosphere for potential hydrogen buildup via two redundant subsystems that are physically separate and independent of each other. When a hydrogen setpoint is reached, each of two 100 percent capacity redundant hydrogen recombiners are operated which limit the concentration to below the combustible limit. The recombiners process a flow of containment air containing hydrogen at a concentration typical of the average concentration throughout the containment.

The Hydrogen Sampling System and the hydrogen recombiners are operated manually from the control room.

g) Safety Injection System (Section 6.3)

Four safety injection tanks (SITs) provide a passive means of injecting emergency core cooling water into the reactor vessel. Each SIT is connected to one of the four reactor vessel inlet lines. Each SIT contains borated water at refueling concentration and is pressurized with nitrogen. In the event of a LOCA that results in Reactor Coolant System depressurization, the borated water is forced into the Reactor Coolant System (RCS) by the expansion of the nitrogen.

Active safety injection is provided whereby borated water from the refueling water tank is injected into the reactor vessel by two low pressure and two high pressure safety injection (LPSI and HPSI) pumps. The HPSI and LPSI pumps are actuated by a SIAS.

The design capacity from the combined operation of one high pressure pump, one low pressure pump and three safety injection tanks provides required core cooling for any size loss of coolant accident which results in RCS depressurization.

6.1 ENGINEERED SAFETY FEATURES MATERIALS

6.1.1 METALLIC MATERIALS

6.1.1.1 Materials Selection and Fabrication

Metallic materials for ASME Code Class 2 and 3 components are selected for their compatibility with the Reactor Coolant System and containment spray. The materials are described in ASME Code, Section III Articles NC-2160, NC-3120 and ND-2160, ND-3120 respectively and Appendix I to Section III. The mechanical properties of materials specified for use in Class 2 and 3 components are as indicated in ASME Code, Section III, Appendix I or ASME Code, Section II, Parts A, B or C. The pressure retaining materials of the ESF systems are included in Table 6.1-1.

Materials for ESF components listed below which are or may be in contact with core cooling and/or containment spray water are considered compatible with the cooling solutions:

a) Austenitic stainless steels and nickle base alloys are not subject to significant corrosion in borated water or borated water with hydrazine and trisodium phosphate additives.

b) Ferritic components are coated with protective coating systems as discussed in Subsection 6.1.2.

c) Galvanized steels do not represent a significant source of generated hydrogen, as discussed in Subsection 6.2.5.3.

Quality of ESF components is maintained during the stages of component manufacture and plant construction. Specific assurance of integrity is based on conformance with Regulatory Guides 1.31, 1.36, 1.37, and 1.44, as discussed in the following paragraphs.

Non-metallic insulation materials installed on stainless steel piping and equipment of the Engineered Safety Features have a low soluble chloride content to minimize the possibility of chloride induced stress corrosion. Combined leachable chloride and fluoride do not exceed 600 ppm. The insulation of ESF stainless steel piping and equipment conforms completely with the criteria of R.G 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," February 1973 (R0).

System cleanliness during construction, start-up (refer to Section 1.4.2) and operation is in accordance with Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," March 1973 (R0).

ESF system components meet the intent of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," May 1973 (R0). The following requirements are included in the materials specifications for the prevention of sensitization in austenitic stainless steel components.

a) Nonsensitization of austenitic stainless steel material is verified for castings and for wrought materials which are either greater than two inches in wall thickness or are not water quenched after solution heat treatment.

One sample from each production heat of material in a solution heat treatment lot is tested for acceptability under ASTM A262-70 Practice E, "Copper-Copper Sulfate - Sulfuric Acid Test," optionally screened by Practice A, "Oxalic Acid Etch Test."

b) Austenitic stainless steel castings and weld metal, are specified to contain a minimum of five percent delta ferrite.

c) Austenitic stainless steel piping which is subjected to heating above 300°F for bending or any purpose other than welding is resolution heat treated over the full length of pipe.

d) Welding practices for austenitic stainless steel components are controlled in the following manner: 1) the use of high heat input welding process is prohibited; 2) the maximum weld bead size is limited; and 3) the interpass temperature is limited to 350°F maximum.

e) Stainless steel components are cleaned, descaled, pickled, and passivated in accordance with the recommended practices of ASTM A380, "Standard Recommended Practice For Cleaning and Descaling Stainless Steel Parts, Equipment and Systems," to protect against contaminants which are capable of causing stress corrosion cracking.

f) Stainless steel components are cleaned before and after welding to remove any contaminants, which may be capable of causing stress corrosion cracking. Aluminum oxide and/or silicon carbide grinding wheels and stainless steel wire brushes are limited to use on stainless steel only.

g) ESF water systems use deaerated water sources for makeup.

Austenitic stainless steels are provided in the solution annealed condition. Yield strength of the materials are of the order of 40,000 to 50,000 psi. Cold bent piping is resolution annealed after cold bending, except for the following conditions: 1) piping which is bent to a radius of at least 20 times the pipe diameter, in which case the resulting strain in the outer pipe fibers is under two percent, which causes no significant increase in yield strength; and 2) piping having two inch and smaller nominal size, normally operating at a temperature below 212°F and a pressure of below 150 psig, which have been subjected to a cold bending radius of at least five pipe diameters.

The delta ferrite content of welds in austenitic stainless steel ESF components is controlled, as a minimum, in accordance with the essential requirements of MTEB 5-1, "Interim Regulatory Position on Regulatory Guide 1.31," as follows:

a) Austenitic stainless steel filter metals are purchased to the acceptance test requirements of Section III of the ASME Code.

Purchase orders specify an additional requirement: that austenitic stainless steel filler metals, other than SFA-5.4 Type 16-8-2, conform to an analysis demonstrating five to 20 percent ferrite to be present in undiluted weld deposits.

b) Flux bearing filler metals are tested for ferrite yield by chemical analysis using material obtained from weld metal test samples. Bare filler metals, to be deposited by inert gas shielded processes, have ferrite yield predicted from wire analysis. The analyses are applied to the Schaeffler Constitution Diagram for ferrite determination.

c) Weld metal test samples, required for flux bearing filler metals, are produced according to the method described in ASME Specification SFA-5.4. The acceptable ferrite range for filler metals is five to 20 percent. ASME SFA-5.4 Type 16-8-2 is excluded from this requirement.

d) The filler metal supplier makes chemical analysis tests, using material samples as required by the filler metal type. The analytical results are applied to the Schaeffler Constitution Diagram for ferrite determination.

e) The results of the analytical tests required are included in a Certified Materials Test Report, per the requirement of the ASME Code, Section III, NB-2130. The results of the ferrite determination are also included in the certified test report.

f) Production welds, with the exception of fillet welds having a throat dimension of 3/8 in. or under and butt welds in material with nominal thickness of 3/8 in. or under are tested for ferrite using magnetic permeability measuring equipment. The instruments are calibrated from or traceable to those of the National Bureau of Standards, to the procedures described by the AWS Welding Research Document, dated July 1, 1972, "Calibration Procedures for Instruments to Measure Ferrite Content of Austenitic Stainless Steel Weld Metal," as supplemented by AWS Specification A4.2-74. Calibration of the instruments is performed on a weekly basis.

g) Magnetic testing for ferrite is performed about the circumference of pipe welds at 90 degrees or 1/4 weld length intervals on the weld centerline. The minimum required ferrite level, averaged over four readings, is three percent. The four ferrite determinations are not to include any ferrite values below one percent; any location yielding less than one percent ferrite is re-evaluated.

h) Only the minimum ferrite level criterion described above is used for weld acceptance standard.

i) In the event that the acceptable ferrite criterion is not met, a comparison of the weldment service requirements as related to the ASME criteria for fatigue strength, using conservative fatigue data representing weld metal containing fissures when available, is made to enhance the acceptance of low ferrite bearing weld metal.

j) As an alternate, low ferrite welds are examined for fissures. The test sample is a highly restrained flat position welded butt joint 4 x 4 x 1 inch thick, produced with those filler metals and using the parameters employed in the production weldment. The finished weld is sectioned at mid-length and one cut surface examined microscopically at 10X. A single tear or fissure defect less than 1/16 inch or a maximum of three defects, 1/64 to 1/16 inch in length within any 0.2 square inch area of weld metal, is acceptable.

k) Those production welds found unacceptable by the simulated weld tests are removed from the piping system, the joint rewelded and retested.

l) Production welds greater than one inch thick are evaluated for ferrite content according to the procedures and acceptance criteria stated above.

m) Production welds in the material nominal thickness range greater than 3/8 inch, to and including one inch, are tested on a sampling basis, according to the Table II of MTEB 5-1, "Interim Regulatory Position on Regulatory Guide 1.31."

n) Acceptable ferrite in the material thickness range greater than 3/8 inch, to and including one inch, is according to the procedures and acceptance criteria stated above.

o) Production butt welds in material with nominal thickness of 3/8 in. or under are not evaluated for ferrite content since there is no prior industry history of the occurrence of fissuring in welds of under 1/2 in. thickness.

Additionally, in various equipment/component orders and construction activities where it was possible to employ later criteria, the delta ferrite content of stainless steel welds is controlled in accordance with the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," Revisions 2 and 3.

Combustion Engineering (CE) requires vendors to comply with delta ferrite limits given below for components within CE's scope of supply. Delta ferrite in austenitic stainless steel pressure retaining parts shall be controlled to the limits given for the appropriate material as follows.

a) All components except valves

   1) Weld rod or filler metal: five to 20 percent by volume (5Fn to 23Fn), and,

   2) Production Welds: not less than three percent by volume (3Fn).

   3) Deposited weld metal, including overlay cladding, exposed to the temperature range of 800-1500 F subsequent to welding; five to 20 percent by volume (5Fn to 23Fn).

4) Delta ferrite in castings of types CF-3, -3A, -3M, -8M, shall be controlled to the following limits: five to 30 percent volume (5Fn to 35Fn).

Delta ferrite content of all production welds in austenitic stainless steel except for fillet welds with a minimum dimension of 3/8 inch or less and except for welds in material less than 1/4 inch thick are verified as follows:

(a) Measurements are taken along the weld centerline at four equally spaced positions with a calibrated ferrite measuring instrument.

(b) No single measurement shall be less than one percent by volume (1Fn) and the resultant average value will meet the acceptance limits 1) through 4) above.

Delta ferrite content austenitic stainless steel castings is measured and recorded for each casting or a 30 percent minimum random sample of each casting heat if quantity in heat is five or more. A calibrated magnetic measuring instrument is used at random locations on the actual castings.

b) Valves:

Delta ferrite control on valve material consists of requirements that austenitic stainless steel castings and deposited weld metal shall have a chemistry balanced to give five to 25 percent by volume delta ferrite on solidification. This is verified by measurement with a calibrated ferrite measuring instrument.

Sensitization of unstabilized stainless steel parts that are wetted by the process fluid (all 300 types except 321, 347 and 348) is controlled as follows:

1) All components except valves;

(a) All raw material shall be supplied in the solution annealed condition.

(b) Wrought material to be welded shall have a maximum carbon content of 0.065 percent.

(c) Wrought material shall not be exposed to the temperature range of 800-1500 F except due to welding.

(d) All unstabilized, austenitic stainless steel raw material, except castings are tested and pass ASTM Test Method A262-70, Procedure E.

2) Valves:

Valve specifications include the requirement that the valve supplier control sensitization during manufacturing process.

6.1.1.2 Composition, Compatibility and Stability of Core and Containment Spray Coolants

As indicated in Subsection 6.1.1.1, ESF materials are chosen to be compatible with the cooling solutions which consist of the following:

a) The Reactor Coolant System contains borated water with a chemistry as described in Subsection 9.3.4. Should a LOCA occur this water is ejected into the containment, and drains to the containment sump.

b) The refueling water tank (RWT) stores borated water as described in Subsection 6.3.2. Should a LOCA or Main Steam Line Break (MSLB) inside containment occur, a portion of the RWT water is injected into the RCS by the safety injection pumps and a portion is sprayed into containment by the Containment Spray System (reference Subsections 6.2.2 and 6.5.2). The containment spray solution is mixed with the Iodine Removal System water which contains trace amounts of hydrazine for iodine removal (Subsection 6.5.2). The hydrazine does not affect the pH of the spray solution.

c) Should a steam or feedwater break occur inside containment, demineralized feedwater pipe drains to the containment sump. Feedwater chemistry is discussed in Subsection 10.3.5.

The above sources of essentially pH neutral or slightly acidic borated water immerse and dissolve the solid trisodium phosphate dodecahydrate (TSP) emplaced in stainless steel mesh baskets located in the vicinity of the sump. Subsequently, within two hours post-accident, the recirculating water mixture is stabilized at a neutral pH in accordance with Branch Technical Position MTEB 6-1, "pH for Emergency Coolant Water." As a result, the pH level of the post-accident water chemistry reduces the probability of stress corrosion cracking of austenitic stainless steel components. Other ESF materials such as pump and valve seals designed for the cooling solutions remain unaffected by neutral pH water.

Potential hydrogen generation from possible corrosion of aluminum and zinc metals inside containment is discussed in Subsection 6.2.5.

Details on the storage of TSP and hydrazine, the post-accident pH time history, and related information pertaining to the Containment Spray/ Iodine Removal Systems, are provided in Subsections 6.2.2 and 6.5.2. Borated water storage in the RWT is discussed in Subsection 6.5.2.

6.1.2 ORGANIC MATERIALS

Asphalt and wood are not used as materials of construction inside containment. Plastics present in containment are limited to an insignificant amount. The major lubricant present in significant amounts is the lube oil associated with the reactor coolant pumps, but this oil is contained in a closed system and is not exposed to the accident environment.

Insulation on the piping and equipment inside the containment is either reflective metallic type or conventional nonmetallic. Nonmetallic insulation inside the containment is jacketed with type 304 stainless steel. Jacketing covers the entire length of the insulation including valves and fittings, thus there is no exposed nonmetallic insulation to react with the borated water. Insulation is further discussed in Subsections 6.1.1.1 and 6.2.2.

Cable insulations are qualified to withstand the DBA environment. Insignificant amounts of polyvinyl chloride (PVC) insulation are used inside containment. Refer to Section 3.11 for further environmental qualification details concerning electrical equipment.

Protective coatings (paints) are applied inside the primary containment for decontamination purposes and the original plant coatings are listed in Table 6.1-2. In Service Level I areas, the significant coating quantities conform with the intent of ANSI Standards N512, "Protective Coatings (Paints) for the Nuclear Industry," June 1974, and N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," May 1972. Service Level I areas are defined as those areas inside the containment vessel. Significant surface areas of concrete, masonry, and steel receive protective coatings successfully tested under DBA conditions. Nonqualified coatings represent an insignificant quantity. Qualifying tests were conducted at independent laboratories or at the coating manufacturers laboratories. Regulatory Guide 1.54 (RO) requires that the radiolytic and pyrolytic characteristics of the protective coatings are tested, analyzed, and certified, in accordance with the above ANSI Standards, that no decomposition products will be released such as to interfere with the safe operation of any engineered safety feature. The physical-chemical characteristics of the above protective coatings are ensured by tests, conducted per ANSI Standards N512 and N101.2, to show that combustible properties are at or below the acceptable level delineated in those ANSI Standards. Tests deemed necessary to demonstrate satisfactory performance have been conducted.

Application of field applied coatings is done in compliance with the recommendations of each product manufacturer and per approved site-coating procedures. Quality assurance during manufacturing, storage, application and inspection of field applied coatings meets the intent of ANSI Standard N101.4, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities," Nov. 1972, in conjunction with the general QA requirements of ASME NQA-1-1994, and thereby are consistent with the requirements of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants" June 1973.

Protective coatings were originally intended for a 40 year service life. Consistent with St. Lucie's response to NRC Generic Letter 98-04 (see Section 6.3.2.2.2a), visual inspections and condition assessments of Service Level 1 coatings inside the containment building are performed every refueling outage. In the event that recoating is desired, the original coating is removed before application of any new coating. Note that coating parameters (i.e., thermal conductivity, thickness and volumetric heat capacity) are used in the accident analysis of Section 6.2.1.

The total amount of protective coatings and organic materials used inside the containment that do not meet the requirements of ANSI N101.2 (1972) and Regulatory Guide 1.54 is approximately 6 cubic feet. The amount of unqualified protective coatings is about 2.5 cubic feet. The equipment so coated is located in various parts of the containment. Although these coating schemes have not undergone DBA qualification testing, some of these coating schemes are baked on enamel which will not readily peel off under DBA environments. The remaining unqualified organic material is NUKEM 750 caulking material used as expansion joint filler (2.5 cubic feet) and miscellaneous organics (approximately 1 cubic foot). Use of NUKEM 750 was justified and was deemed acceptable by the NRC. [1],[2]

Evaluation of the effect of unqualified materials inside containment on combustible gas generation can be found in Subsection 6.2.5 and on ECCS sump strainer design in Subsection 6.2.2.

REFERENCES: SECTION 6.1

1) FP&L letter L-80-246 from Dr. R E Uhrig (FP&L) to Mr. R L Baer (USNRC), dated August 1, 1980

2) USNRC Letter Docket No. 50-389 from Mr B J Youngblood (USNRC) to Dr R E Uhrig (FP&L) dated November 7, 1980.

TABLE 6.1-1

PRINCIPAL ESF PRESSURE RETAINING
MATERIALS AND ESF STRUCTURAL MATERIALS THAT
COULD BE EXPOSED TO CORE COOLING OR CONTAINMENT SPRAY SOLUTIONS

1. CONTAINMENT SPRAY		
a) Pumps		ASTM-A-296 (CA6NM) -A-276 TP 410T ASME-SA-182 (F 304) -SA-194 GR. 7 -SA-193 (GR. B-7)
b) Spray Nozzles		ASME-SA-351 GR CF8
c) Piping Flanges/Fittings		ASME-SA-312 GR TP 304, 316/316L -SA-358 CL 1 Type 304 -SA-182 GR F304 -SA-351 GR CF8 -SA-403 GR WP 304W, 316/316LW
d) Valves		ASME-SA-351 GR CF8M -SA-182 Type 316 -SA-564 Type 630
e) Refueling Water Tank		ASME-SA-240 Type 304 -SA-182 Type 304 -SA-479 Type 304 -SA-403 Type 304W
f) Sump Strainer		Stainless steel TP 304 Carbon steel
2. SAFETY INJECTION		
a) Piping Flange/Fittings		ASME-SA-312 GR TP 304 -SA-182 GR F 304 -SA-351 GR CF8 -SA-403 GR WP 304W
b) Pumps		ASME SA-182 F304 ASTM A-296 CA6NM
c) SDC Flow Control Valves		ASME SA-182 F316 ASME SA-351 CF8M
d) Safety Injection Tanks	Heads:	ASME SA-516 GR-70 (W Aust. Stainless steel clad)
	Shell:	ASME SA-516 GR-70 (W SA 240 type 304L RB clad)
e) Shutdown HX		ASME SA-249-TP-304

TABLE 6.1-2

PROTECTIVE COATINGS IN THE PRIMARY CONTAINMENT VESSEL[1]

SURFACE TO BE COATED	PRIMER OR SURFACER	DFT (mils)	FINISH COAT	DFT (mils)	APPROX. AREA (Sq. ft.)
A) Concrete walls, ceilings & equipment pedestals	Nu-Klad 1-14 Amercoat 90	Trace-7 2-4	Amercoat 90 Amercoat 90	4-8 3-5	66,927
B) Concrete floors	Nu-Klad 110AA	65-125	Amercoat 90	4-8	25,350
C) Concrete underwater sumps	Nu-Klad 110AA	65-125 (not over 250)	N/A	N/A	907
D) Elec. tunnel floors	Nu-Klad 110AA	As above	Amercoat 90	4-8	5,400
E) Elec. tunnel walls & ceiling	Nu-Klad 114	Trace-7	Amercoat 90 Amercoat 66	4-8 4-8	
F) Carbon steel					
1) Containment Vessel (interior)	Carbo Zinc 11	2-7	Phenoline 305	4-8	86,700
2) Structural Steel (Equipment Supports)	Carbo Zinc 11 Mobilzinc 7 (13-F-12)	3.5 avg. 3.5 avg.	Amercoat 90 Amercoat 90	4-8 4-8	17,320
3) Embedded Plates	Carbo weld 11 Dimetcote 6	1-1.5 2-4	Amercoat 90 Amercoat 90	4-8 4-8	6,212
4) Main NSSS equipment valves, supports refueling machine, fuel trans. syst, RC motor stand	Carbo zinc 11	2-7	N/A	N/A	3,600
Safety inj. tanks,	Dimetcote 6	2-4	Amercoat 90	4-8	5,800
Head lift rig	Dimetcote E-Z	2-4	Amercoat 90	4-8	
motor op. valves, support & yoke	Carbo zinc 11	2-7	Phenoline 305	4-10	350
Stm. Gen. Snubbers (4)	Carbo Zinc 11	2-7	Carboline 3912	4	80
Stm. Gen. Snubbers (12), RC Pump motor	Plasite 7155NP	3	Plasite 7155	4	2,040

TABLE 6.1-2 (Cont'd)

PROTECTIVE COATINGS IN THE PRIMARY CONTAINMENT VESSEL[1]

SURFACE TO BE COATED	PRIMER OR SURFACER	DFT (mils)	FINISH COAT	DFT mils)	APPROX. AREA (sq. ft.)
5) Miscellaneous Equipment including polar crane, maint. hatch shield door, bulkhead doors, pipe whip restraints, conduit supports, reactor coolant pump supports, elect. tray supports, restraints, PC pipe sleeves, equipment restraints.	Carbo zinc 11	2-7	Mobil 89 Series	4-8	127,381
	Carbo zinc 11	2-5	Amercoat 90	4-8	
	Mobilzinc 7 (13-F-12)	2.5-5	Amercoat 90	4-8	
	Dimetcote 6	2.5-5	Amercoat 90	4-8	
	Carbo zinc 11	3.5 avg	Amercoat 66	4-8	
	Carbo weld 11	1-2	Amercoat 66	4-8	
	Dimetcote E-7	2-4	Amercoat 66	4-8	
	Carbo weld 11	1-2	Amercoat 90	4-8	
	Dimetcote 6	2-4	Amercoat 66	4-8	
Exhaust & supply fans, cont. fan coolers, reactor cavity pressure dampers, cont.duct pressure relief dampers, CS valves, check valves,strainers, hangers & restraints,sump pump, lighting and misc. transformers.	Mobilzinc 7 (13-F-12)	3-5	Mobil 89 Series	4-8	
	Carbo zinc 11	2-7	Phenoline 305	4-10	
	Phenoline 305	2-5	Phenoline 305	2-5	
	Amercoat 90	4-8	Amercoat 90	4-8	
6) Galvanized steel, including electrical trays, exposed conduit electrical boxes, duct dampers and louvers, duct, cable trays.	Zinc	0.8-3.5	N/A	N/A	123,944

Notes:

(1) This table identifies the original coatings used within containment and is maintained here for informational purposes. Replacement coatings are required to be approved in accordance with the latest approved Engineering Specification for Service Level I coatings for application within the containment building.

6.2 CONTAINMENT SYSTEMS

The original analysis supporting the design of the St. Lucie Unit 2 containment was conducted by the Nuclear Steam Supply System (NSSS) vendor, Combustion Engineering, and the Architect Engineer (AE), EBASCO. Similar to the original design of most nuclear plant containments, the NSSS vendor supplied the mass and energy release data for a spectrum of Loss of Coolant Accidents (LOCAs) and Main Steam Line Breaks (MSLBs). This information was generated using NRC approved blowdown codes, now referenced in the Standard Review Plans (SRPs). This blowdown or mass and energy release information was used by the AE to generate the corresponding transient containment pressure and temperature response. For the St. Lucie Unit 2 dual containment (i.e. steel shell with annulus), the peak P/T values were considered with structural limitations to set the peak design pressure and temperature of the steel shell. The annulus design analyses, described later, were based on different assumptions assuming maximum heat transfer through the steel shell.

The peak containment design pressure is the pressure which the vapor space or containment atmosphere should not exceed. The peak containment design temperature, however, is the temperature which the steel shell should not exceed. While the forcing function for this temperature is the vapor space response, the equipment surface temperature is lagged and will generally not experience the elevated temperatures resulting from pure steam blowdown (i.e., MSLB) responses. The transient containment vapor space response is used as input to equipment qualification (EQ) analyses, which specify the environment within which Class 1E safety related equipment, must function. While the EQ analysis is described in other plant documentation, the temperature response produced during the design analysis, and presented in the UFSAR, is the prime input.

Due to the two-steam generator large inventory design of Combustion Engineering designed plants, the MSLB event typically produces the peak containment pressure and temperature. Depending on the analysis assumptions regarding the amount of moisture carryover in the steam release, the MSLB can also produce containment temperatures well in excess of the LOCA. These elevated temperatures, however, are typically of short duration and in most cases will not impact the operation of in-containment equipment or challenge the design structural temperature limit. The duration of the MSLB is also short, relative to the LOCA, and does not have the potential for a significant radiological release. As a result, the requirements for operation of post-accident safety related equipment are limited when compared to the long-term cooldown associated with the LOCA. For these reasons, the post-accident requirements for a LOCA were typically used as the basis for engineered safety feature equipment design, integrated leakage rate testing (ILRT) limits, and long term EQ requirements.

As can be inferred by the above discussion, the analysis of the LOCA requires a greater level of detail than the MSLB. While the MSLB transient largely consists of the blowdown phase, the LOCA is represented by four distinct phases. As described in greater detail in subsequent sections, these are characterized by the blowdown, reflood, post-reflood, and long term phases. As a result, the LOCA analysis requires the use of three different computer codes versus one for the MSLB.

The original analysis of the LOCA for St. Lucie Unit 2 plant was split by Combustion Engineering and EBASCO. A parametric study of break sizes was analyzed by Combustion Engineering using an early version of the NRC approved CEFLASH and FLOOD-MOD2 computer codes. These codes provided data from the initiation of the event through the end of blowdown for the hot leg breaks and end of post-reflood for the cold leg breaks. The mass and energy release data associated with a sample of breaks was provided to EBASCO for the containment pressure and temperature response calculation. The long term phase was also conducted by EBASCO utilizing the primary and secondary conditions present at the end of these intervals. The assumptions associated with these analyses were based on plant conditions and Regulatory requirements present at that time. The most significant of these was the initial power level of 2560 MWt.

Following the initial FSAR writeup, a number of plant changes, Regulatory requirements, and methodology and modeling enhancements occurred. While these changes were addressed, they had been performed on a piecemeal basis. The current analysis (Ref. 27) discussed below was performed to consolidate prior updates and changes. This included a recalculation of all mass and energy release data and both the short and long-term containment pressure and temperature response. While many enhancements in methodology, computer codes, and inputs were incorporated, the bounds of the analysis were also extended. This extension was the modeling of the Component Cooling Water (CCW) and Intake Cooling Water (ICW) systems.

The CCW/ICW modeling was developed to address limiting CCW/ICW parameters for the St. Lucie Unit 2 plant. As described in subsequent sections, this modeling was an extension of ABB C-E's NRC approved CONTRANS containment code. While past containment analyses have assumed a fixed CCW temperature, this model simulated the transfer of containment heat through the CCW and ICW loops back to the ultimate heat sink. This model was developed and tested using FPL supplied data for St. Lucie Units 1 & 2. In this manner, fan cooler heat and shutdown cooling heat exchanger (SDCHX) capacity was a time varying function based on the transient calculation of Component Cooling Water temperature. For St. Lucie Unit 2, the additional heat loads of High Pressure Safety Injection (HPSI) pump and control room A/C were also added to the CCW loop. The prior considerations of peak accident heat load required for the conservative hand calculations of the past were no longer necessary or considered to be a critical parameter.

With this approach, the analysis took on a more realistic quality which coupled the old analysis bounds with those plant components and systems commonly assumed in the past as fixed inputs. Important items such as the ICW heat exchanger overall heat transfer coefficients, which addressed fouling and tube pluggage, along with more accurate ICW flow rates and ultimate heat sink temperature all served as design inputs to the analysis. Therefore, along with the typical plant analysis criteria of peak containment pressure and EQ temperature, the peak CCW temperature was also added.

The consolidated analysis was done in general accordance with the SRPs, with enhanced methods and inputs, resulting in the results showing a less severe response than those originally predicted. The coupled modeling of mass and energy release to containment and CCW/ICW response has provided a unique analysis of the overall plant performance during a design basis LOCA. The detailed analysis including summary of inputs, assumptions, and their bases is discussed in Reference (27).

The St. Lucie Unit 2 SGs have been replaced. The LOCA containment peak pressure and temperature analysis and MSLB containment peak pressure analysis demonstrated that the analyses of record remained bounding (see Reference 39).

GOTHIC (Generation of Thermal-Hydraulic Information for Containments) is an integrated, general purpose thermal-hydraulics software package for design, licensing, safety and operating analysis of nuclear power plant containments and other confinement buildings. GOTHIC has been reviewed and approved by the NRC for use in containment analysis for analysis of LOCA and MSLB in large dry containments and was used for various analyses to support the replacement steam generators.

GOTHIC is further discussed in Reference 39.

St. Lucie Unit 2 has been evaluated for an EPU. The LOCA containment peak pressure and temperature analysis and MSLB containment peak pressure analysis have generated new bounding values as discussed below.

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

6.2.1.1 Containment Structure

6.2.1.1.1 Design Bases

The containment is designed to provide protection to the public from the consequences of a loss of coolant accident (LOCA) up to and including a double-ended rupture of the largest reactor coolant pipe assuming unobstructed discharge from both ends coincident with the loss of offsite power and any single active component failure. The containment structure and the Engineered Safety Features ensure that potential radiological doses to the public resulting from such an occurrence are below the guidelines established for design basis accidents.

The spectrum of postulated accidents considered in determining the design containment peak pressure and temperature, the subcompartment peak pressure and the containment differential pressure used in the original analysis are summarized in Table 6.2-1. The spectrum of break sizes was chosen to establish the upper bounds of containment pressure and temperature following a DBA. For the Stretch Power Analysis only the worst main steam line break (MSLB) case was analyzed for Cycle 2, as presented in Table 6.2-63A. The updated limiting LOCA case is also presented for completeness. The containment peak pressure and temperature have been reviewed at the stretch power rating and are still within technical specification limits.

The EPU evaluation of peak containment pressure is a complete reanalysis. The three single failures, loss of one containment cooling train, failure of a MFIV to close and the failure of a MSIV to close were evaluated over a spectrum of power levels so the maximum containment pressure was identified. The MSLB EQ temperature response was based on the peak pressure case with the maximum temperature. The MSLB cases analyzed for EPU are provided in Table 6.2-63B.

The containment systems protect the public from the consequences of a postulated break in the Reactor Coolant System. The containment systems consists of the steel containment vessel surrounded by the Shield Building, and the engineered safety feature systems which include the Safety Injection System (Emergency Core Cooling System), the Containment Heat Removal System (Containment Spray System and Containment Cooling System), the Shield Building Ventilation System, the Containment Isolation System, and the Combustible Gas Control System.

A description of the containment and the design criteria relating to construction techniques, static loads, and seismic loads coincident with loss of coolant accident induced loads are provided in Section 3.8. This section pertains to those aspects of containment design, testing, and evaluation that relate to the accident mitigation function.

The containment vessel is designed to withstand the pressure and temperature transients calculated to exist after a design basis accident (DBA). Post accident conditions are determined by evaluating the combined influence of the energy sources, heat sinks and Engineering Safety Features (ESF) operation.

The spectrum of postulated accidents considered in determining the design containment peak pressure and temperature, the subcompartment peak pressure, and the containment differential pressure are summarized in Table 6.2-1. The spectrum of break sizes was chosen to establish the upper bounds of containment pressure and temperature following a DBA. For postulated subcompartment pipe break accidents, a discussion of the criteria for selecting break locations is given in Subsection 6.2.1.2.

The accident controlling design for each of the categories of containment peak pressure, containment peak temperature, subcompartment peak pressure, containment differential pressure, and containment minimum pressure is that case which produces the most severe loadings for the spectrum of accidents postulated. The margin between values calculated for a DBA and design values is established by comparing Tables 6.2-2 and 6.2-3. Table 6.2-2 defines the DBA for each design category and Table 6.2-3 gives the margin and the formula for calculating margin.

For the containment peak pressure analysis, subcompartment peak pressure analyses, the containment peak temperature analysis, and the ECCS minimum containment pressure analysis, it is assumed that each accident is concurrent with the most limiting single failure. No two accidents are assumed to occur simultaneously or consecutively.

The time dependent mass and energy release for the postulated accidents under the categories of containment peak pressure and temperature analyses are referenced in Table 6.2-1. The computer codes and assumptions used in deriving each of the mass and energy release tables are discussed in Subsections 6.2.1.3 and 6.2.1.4.

Energy released to the containment atmosphere as a result of the postulated accidents is removed by the Containment Heat Removal System discussed in Subsection 6.2-2. Each train of the Containment Heat Removal System (consisting of one spray pump, one set of spray headers, one shutdown heat exchanger and two fan coolers) is designed for 100 percent of the required heat removal rate.

For the original containment analysis, the most limiting case with respect to containment peak pressure was the maximum safety injection case. The most restrictive single active failure was a loss of one spray train of the Containment Heat Removal System resulting in minimum heat removal capability. Only one containment spray train and four containment fan coolers of the Containment Heat Removal System were assumed to operate during the LOCA while the maximum safety injection conservatively maximized the LOCA energy release rate. The single active failure of one onsite diesel generator is not the most restrictive failure since safety injection would not be maximized. For the EPU containment analysis, the most limiting case with respect to containment peak pressure is the Double Ended Hot Leg Slot (DEHLS) break with the minimum safety injection case.

The current main steam line break (MSLB) DBA initiates from 0% power. The most limiting single active failure is determined to be the failure of one train of the Containment Heat Removal System with the availability of offsite power.

The containment peak temperature results from a DBA 100.3 percent power main steam line break. The most limiting single active failure is determined to be the failure of one main steam isolation valve (MSIV) with the availability of offsite power.

The analysis of containment minimum pressure is based on confirming the ECCS core reflood capability under the conservative set of assumptions that maximize the heat removal effectiveness of ESF systems, structural heat sinks, and other potential heat removal processes. These assumptions are discussed in Subsection 6.2.1.5.

6.2.1.1.2 Design Features

The design bases and design measures taken to assure that the containment structure is adequately protected against the dynamic effects of postulated missiles and pipe ruptures are discussed in Sections 3.5 and 3.6 respectively. The codes, standards, and guides applied in the design of the containment structure and internal structures are described in Section 3.8. Equipment environmental qualification is discussed in Section 3.11.

The accident analysis is based on Regulatory models such as those prescribed in the NRC's Standard Review Plan. Assumptions regarding safety injection and containment active heat removal devices have been incorporated into the analysis, within the standard methods for single failure assumptions. The containment is equipped with the following ESF equipment to mitigate the consequences of an inside containment break.

- Safety Injection
  - 4 safety injection tanks
  - 2 high pressure safety injection pumps
  - 2 low pressure safety injection pumps
- Primary Containment Cooling
  (Operational during Injection and Recirculation modes)
  - 4 containment fan coolers
  - 2 containment spray pumps
  - 2 shutdown cooling heat exchangers
- Secondary Containment Cooling
  - 2 Component Cooling Water loops
  - 2 CCW/ICW heat exchangers
  - 2 Intake Cooling Water loops

The accident analyses assume the most limiting combination of the above equipment both for offsite power available operation and for loss of offsite power. Both component and train failures are considered. For the updated LOCA analysis, the assumptions related to the CCW and ICW systems are selected to produce limiting results. For the LOCA event, both maximum and minimum safety injection are assumed with offsite power available and unavailable, respectively, for each break location. Maximum SI refers to the operation of two HPSI and two LPSI pumps with offsite power available. Minimum SI refers to the operation of one HPSI and one LPSI without offsite power. For each scenario, the loss of the limiting containment active cooling device (component or train) was assumed.

Following a LOCA, heat in the containment is rejected to the CCW by the air/water heat transfer in the containment fan coolers. The containment sprays take suction from the refueling water tank (RWT) and remove energy from the vapor space as the atomized spray water is heated. During the injection mode, the spray pump mini-recirculation line back to the RWT is open. Following the generation of the recirculation actuation signal (RAS), this line is isolated as the containment spray pumps take suction from the containment sump. While the generation of the RAS is based on a RWT low level trip, the LOCA analysis conservatively minimizes the amount of available RWT inventory. Following RAS, containment heat is rejected to the CCW via the water/water heat transfer in the shutdown cooling heat exchangers.

Other essential loads such as the HPSI pumps, containment spray pumps and control room A/C also transfer heat to the CCW. This heat is then transferred to the ultimate heat sink via the ICW heat exchanger. While not modeled in the containment P/T analysis, each of the two shield building ventilation system fan sub-systems is capable of removing the post-LOCA energy increase in the shield building annulus.

Two redundant containment vacuum breakers are provided for protection against loss of containment integrity under external loading conditions. Calculations of containment pressure following an inadvertent operation of the Containment Spray System results in pressures within the containment design differential allowable pressure. Details of this evaluation are provided in Subsection 6.2.1.1.3. The margin, between calculated and design pressure differentials is shown in Table 6.2-3.

The systems used for normal containment ventilation include the continuous Containment/Hydrogen Purge System and the Containment Cooling System. These systems are fully discussed in Section 9.4 and Subsection 6.2.2, respectively. No dependence is placed on the containment purge maintaining containment temperature and humidity. The limiting containment conditions for normal plant operation are contained in the Technical Specifications.

A containment sump is provided to collect water for the recirculation mode (refer to Subsection 6.2.2.2.3). All but approximately 2,100 gallons of the water drains to the containment sump following a LOCA or MSLB. This amount of water becomes trapped in the refueling canal since it lies below the level of the canal drains. The bottom of the containment sump is at elevation 7.58 feet MLW, which is below the lowest floor elevation inside containment except for the reactor vessel cavity and the reactor cavity sump.

6.2.1.1.3 Design Evaluation - Containment Pressure - Temperature Analysis

In the event of a postulated LOCA or MSLB inside containment, the release of coolant from the rupture area causes the high temperature, high pressure fluid to flash to steam. This release of mass and energy raises the temperature and pressure of the containment atmosphere. The severity of the resulting temperature and pressure peaks developed depends upon the nature, locations, and size of the postulated rupture.

In order to establish the controlling rupture for containment design, a spectrum of primary and secondary breaks were analyzed as part of the design basis calculations to determine their significance in selecting the containment design basis accidents. Table 6.2-4 presents a summary of results (the calculated pressure, temperature, time peak pressure, time to end of blowdown, and total peak containment energy release) of these analyses and the containment design basis accidents are noted on Table 6.2-2. Note that Table 6.2-2 also provides peak CCW temperature.

The calculated transients following a postulated accident are a direct consequence of the energy balance within the containment. Of particular importance are the initial conditions postulated at the start of the accident, the ability of the heat sinks within the containment to absorb energy during the accident, and the capability of the Containment Heat Removal System to reduce the total energy within the containment thus bringing the containment heat sinks, sump water, and atmosphere into thermal equilibrium. The listing of initial conditions is found on Table 6.2-5.

The containment pressure analysis input data are based upon plant design features. A conservative prediction of consequences is assured by determining upper and lower bounding values of containment initial conditions, geometric parameters, and thermodynamic properties, and by applying these values in the manner producing maximum pressure and temperature results.

The containment accident analyses for MSLB are performed using an the Westinghouse SGNIII computer code. The SGNIII code is NRC approved to generate containment MSLB mass and energy release data. The SGNIII code is run concurrently with the CONTRANS containment code to produce the containment response to the MSLB. The CONTRANS code is described in the following section. In order to establish the controlling conditions for containment design, the accidents in Table 6.2-63B were analyzed for EPU.

Containment Pressure/Temperature Calculations

The containment pressure/temperature analyses are performed using the CONTRANS computer program (Reference 31). The CONTRANS model predicts the pressure and temperature within the containment regions and the temperatures in the containment structures. The code takes input from separate blowdown and core thermal behavior studies which have determined mass and energy input rates such as that from a Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB).

The CONTRANS model analyzes both active and passive heat transfer within the containment following a LOCA or MSLB. The code can also be used for best estimate studies and has compared favorably with other industry codes such as CONTEMPT. The containment spray and fan cooler models as well as the passive heat sink numerical calculations enable an accurate calculation of their effect in reducing containment pressure and temperature.

CONTRANS calculates a pressure-time transient with stepwise integrations between the thermodynamic state points. The integrations are based on the laws of conservation of mass and energy together with their thermodynamic relationships. Superposition of heat input functions is assumed so that any combination of coolant release, metal-water reaction, decay heat generation, and sensible heat release can be used with appropriate ESF features to determine the containment pressure time history associated with a LOCA.

The program uses a two region containment model consisting of the containment atmosphere (vapor region), the sump (liquid region), and a primary system model (reactor vessel) which is used to calculate mass and energy release data after the cold leg LOCA post-reflood period or the hot leg LOCA blowdown period. Mass and energy is transferred between the liquid and vapor regions by boiling, condensation, and evaporation. The heat and mass transfer coefficients between these two regions are calculated in CONTRANS. Each region is assumed to be homogeneous, but a temperature difference can exist between regions. Any moisture condensed in the vapor region during a time increment is assumed to fall immediately into the liquid region. Noncondensible gases are included in the vapor region.

The thermodynamic assumptions used in CONTRANS and the modeling of the atmosphere and sump regions, reactor vessel region (long term release), safety injection system, containment spray system and shutdown cooling heat exchanger are discussed in Reference (31). This also includes the modeling of the heat transfer surfaces and coefficients. The cooling medium, typically the component cooling water (CCW), utilized for the original shutdown cooling heat exchanger model is assumed to be at a fixed inlet temperature.

An upgrade effort of the CONTRANS code was undertaken to support the St. Lucie Units 1 & 2 LOCA containment analyses. The intent was to calculate a time dependent CCW temperature. The updated coding incorporated a CCW and Intake Cooling Water (ICW) or salt water loop with common CCW heat exchanger. The new model allows for the time dependent addition of extra heat loads and associated flows such as that of the safety injection or spray pumps or control room A/C. The model is conservative since loop transient time delays or water volumes are not credited and as such is simplistic in nature. The model has been directly attached to the existing CONTRANS code and allows for inputs such as ocean temperature and CCW heat exchanger performance parameters. The enhanced model also updates containment fan cooler capacity as a function of changing CCW temperature.

LOCA Analysis

A sensitivity study[1] was performed as part of the original design basis analyses varying free volume and the size of the heat sinks within the containment. The results show that the containment free volume is the principal factor responsible for large changes in the peak containment pressure and temperature responses. In the sensitivity study, the surface area of the heat sinks in the containment was varied over a range of ±20 percent. Two sets of analyses were done, one in which only the surface area of the internal heat sinks was varied and the other in which all heat sinks, including the size of the containment, were varied with a proportional change in the free volume. For the change of ±20 percent in the surface area of the internal heat sinks the peak pressure was found to vary by ±5 percent. For the change of all the heat sinks and a proportional variation of the free volume of about 25 percent, the peak pressure varied by about ±25 percent.

The sizing of the Containment Heat Removal System is based on the heat removal rate necessary to keep the peak pressure reached during a LOCA less than the design pressure of the containment. The peak pressure occurs during the blowdown phase. This peak is larger than the rise in containment pressure that occurs during the recirculation period when the containment energy balance is coming into equilibrium.

The CONTRANS model and formulations have been shown to be applicable and conservative for the design of the containment vessel by simulated design basis accident tests such as the CVTR (Ref. 32) blowdown experiment. As shown in the topical report of Reference (31), test cases are presented which outline the CONTRANS methodology. As shown, the proper representation of heat sinks is important, therefore care must be exercised in choosing the correct nodal spacing.

The containment heat sink and heat source data used in accident analyses are described in Tables 6.2-7 and 6.2-8. Table 6.2-7 is a detailed list of the geometry of each heat sink and Table 6.2-8 describes the resulting simplified heat sink models used for computer input. Node spacing used for concrete, steel, and steel-lined concrete heat sinks is fine enough to ensure an accurate representation of the thermal gradient in each slab.

The given values for surface area and thickness reflect the total areas and surface-area weighted thicknesses for all steel exposed to the containment atmosphere from all sources. These sources include structural steel, polar crane and removable platform structures, instrumentation and control equipment (cabinet, tubes, etc), hydrogen recombiners, HVAC equipment (ducts, fan coolers, valves), refueling machine, miscellaneous piping, and containment penetration nozzles, conduits and cable trays.

Steel is coated with a layer of primer and finish coat. All steel and concrete is assumed to be insulated on one side and in contact with the containment atmosphere (by a condensing heat transfer correlation) or sump on the other. The initial temperature is 120 F.

The given values of the concrete for surface area and thickness reflect total areas and surface-area weighted thicknesses of all concrete within the containment. This concrete includes the primary and secondary shield walls, pressurizer cubicle, regenerative heat exchanger cubicle, pipe tunnel, containment sump pump wall, and the steam generator foundation.

A complete list of the heat sink thermophysical properties used in the analysis is given on Table 6.2-8 and Section 6.1.2. The node spacing within a heat sink is dependent upon the gradient of temperature within the sink. To ensure an accurate representation of the temperature gradient, a maximum number of node points are placed where the temperature gradient is at a maximum. Too few node points simulate a more gradual slope implying excessive energy stored within the heat sink.

The high thermal conductivity and relative thinness of the steel heat sinks results in a rapidly uniform temperature distribution throughout the sink. It is, therefore, only necessary to provide for sufficient nodes to adequately define its relative thickness (in relation to other heat sinks).

Containment Peak Pressure/Temperature Analysis

Loss of Coolant Accident

Input Assumptions

A updated peak and long term containment pressure and temperature Loss of Coolant Accident (LOCA) analysis was conducted for the EPU. This analysis was a blend of past limiting assumptions coupled with more recent Standard Review Plan Regulatory requirements. This approach maintained conformance with the Licensing requirements of Unit 2. The following cases were analyzed:

- 9.82 ft^2 Double-Ended Suction Leg Slot Break
- 9.82 ft^2 Double-Ended Discharge Leg Slot Break
- 19.24 ft^2 Double-Ended Hot Leg Slot Break

For consistency with the design basis calculations, all cases were analyzed assuming minimum and maximum safety injection flows. The design basis break size sensitivity analyses were utilized in the selection of limiting break size for each break location.

The pressure and temperature transient calculations are based on the following potential sources of mass and energy release into the containment:

1) stored heat from the reactor core and internal structures
2) fission coastdown and decay heat from the reactor core
3) stored heat in the reactor coolant pressure boundary and the steam generators.
4) reactor coolant inventory with contained fission products
5) fission products from the fuel elements in the core
6) metal-water reaction
7) safety injection water
8) containment spray water

The above sources were addressed in the CEFLASH-4A, FLOOD3, and CONTRANS long term release models described earlier. Since reflood is not part of the hot leg break methodology the FLOOD3 code was not used for the hot leg breaks. The mass and energy release calculation was therefore limited to the CEFLASH-4A and CONTRANS long term release models. For the cold leg breaks, the FLOOD3 mass and energy release was a continuation of the CEFLASH-4A blowdown. All mass and energy was input to the CONTRANS code for the transient calculation of containment pressure and temperature.

The containment related assumptions are as follows:

1) The containment free volume is 2,499,735 ft^3.

2) The initial relative humidity within the containment is 45 percent.

3) The initial containment pressure is 15.41 psia.

4) The initial containment, sump, and heat sink temperature is 120 °F, the maximum Tech. Spec. operating temperature of the containment.

5) No leakage into or out of the containment occurs.

6) A heat transfer coefficient as a function of ΔT is assumed for the transfer of heat to the annulus. The expression as presented in Reference (31) is:

$$h = 0.2\ \Delta T^{1/3} \qquad (\text{Btu/hr-ft}^2\text{-°F}).$$

7) No heat transfer to the sump, shield wall, or atmosphere is assumed.

8) The decay heat after reactor shutdown is calculated on the basis of infinite reactor operation at 3030 MWt (3020 MWt with 0.3% power measurement uncertainty).

9) For cold leg breaks, all energy is assumed to go out the break. For hot leg breaks all core and reactor vessel energy is assumed to go out the break. RCS and SG wall sensible heat loss is assumed to be insignificant relative to the long term release energy addition.

10) One containment spray pump operates and sprays a base flowrate of 2700 gpm of 104 °F (max. Tech. Spec. Temperature plus instrument inaccuracy) water into the containment until the recirculation actuation signal is generated. Following the closure of the mini-recirculation valves to the RWST after RAS, an additional 150 gpm is assumed.

11) Although the Tech. Spec. minimum Refueling Water Storage Tank (RWST) volume is 477,360 gallons, the volume remaining in the tank following RAS, the time associated with switching to the sump, and NPSH requirements for the safety injection and containment spray pumps must be considered. With these considerations, the design basis switch-over volume of 290,300 gallons was assumed for safety analysis to calculate time of RAS.

12) One shutdown cooling heat exchanger (SDCHX) operates during the recirculation mode to cool the containment spray water when one train of containment spray is running and two SDCHXs operate when both spray trains are running. A "service" U value of 274 Btu/hr-ft²-°F is assumed. The heat removal capacity is updated as a function of time by a changing CCW temperature.

13) The base fan cooler performance is consistent with the original design basis capacity as shown in Table 6.2-6.

14) The analyses are conducted assuming loss of offsite power. For the minimum SI cases, the loss of diesel generator results in the loss of one train of containment cooling which disables one containment spray and one train (i.e. two units) of fan coolers. The maximum SI cases require flow from both trains of safety injection. The maximum SI cases must assume both emergency diesel generators start. The failures for these cases are assumed to affect one spray train or one fan cooler train. The maximum SI cases are run twice for each break location. One case assumes the failure of one spray train while the other case assumes a failure of one fan cooler train. The minimum SI cases only require flow from one safety injection train. The assumed failure is the loss of a diesel generator and only one train of containment spray and one fan cooler train are assumed for the minimum SI cases.

15) A fan cooler containment high pressure actuation setpoint of 5.15 psig and a spray containment high high actuation setpoint of 7.05 psig were assumed. These analytical setpoints included allowance for instrument uncertainty and delay times.

The LOCA accident analyses are based upon the following additional overall assumptions:

1) Reactor thermal power is 3030 MWt (normal operation at 3020 MWt plus 0.3% power measurement uncertainty).

2) An initial T_{cold} of 551°F (plus 3°F for instrument uncertainty).

3) An initial RCS flowrate of 375,000 gpm.

4) For the discharge leg break, the contents of three safety injection tanks (SITs) discharge into the reactor vessel when reactor coolant system pressure drops below tank pressure. This assumes the entire contents of the safety injection tank in the ruptured leg does not reach the core. For the hot and suction leg cases the contents of four SITs is considered.

5) An average of the min/max Technical Specification SIT water volume per tank of 1488 ft^3 was assumed with a Technical Specification maximum initial cover gas pressure of 650 psig.

6) One high and low pressure safety injection pump are assumed during the injection mode for the minimum SI cases, while 2 HPSI and 2 LPSI pumps are assumed for the maximum SI cases. Following RAS, all LPSI pumps are shut off.

7) A CCW flowrate of 3719 gpm is assumed for one Shutdown Cooling Heat Exchanger.

8) A degraded ICW flowrate of 13,857 gpm is assumed with a maximum ocean temperature of 95°F for the LOCA containment pressure and temperature analysis. For the CCW temperature analysis, see Section 9.2.1.2.

9) The CCW heat exchanger performance for the peak pressure and temperature analyses is done with a U factor of 274 Btu/hr-ft^2-°F. The CCW analysis uses a U factor of 300 Mtu/hr-ft^2-°F.

10) The addition of HPSI pump, containment spray pump and control room A/C heat load was included in the CCW calculations.

11) Volumetric expansion due to primary and secondary conditions at hot full power operating conditions vs. "cold drawing" values was considered per SRP guidelines. This includes the components of pressure, temperature, and manufacturing tolerance uncertainties.

12) Initial normal pressurizer and steam generator water levels were assumed.

13) Bounding fuels related inputs were assumed in the blowdown analysis.

14) The LOCA containment pressure and temperature, and CCW temperature analyses are performed with no steam generator tube plugging (SGTP). This maximizes heat transfer from the secondary to primary system.

LOCA Analysis

As described earlier, the LOCA containment analysis is a multi-phase effort requiring the use of three different codes. The event is characterized by distinct phases which are a function of break location. For all break locations, there is an initial blowdown phase which is typically a 10 to 25 second interval and was analyzed with the CEFLASH-4A code. The outputs produced by the CEFLASH-4A code were the mass and energy release data for the interval and the RCS and steam generator conditions at the end of blowdown. For the suction and discharge cold leg breaks, the next phase is the reflood phase followed by the post-reflood phase. This was performed by the FLOOD3 code which used the end of blowdown conditions as a starting point for continuing the event. The primary purpose of the reflood and post-reflood periods is to account for the steam generator inventory energy, as well as the RCS and steam generator metal energy, all of which contribute to the exiting steam flow.

The product of the FLOOD3 code is similar to that of CEFLASH-4A in that mass and energy release data is generated and system conditions at the end of post-reflood are used as a starting point to the long term energy release model of the CONTRANS code. The CONTRANS code is used to produce the containment pressure / temperature response. As such the mass and energy release data produced by CEFLASH-4A and FLOOD3 are input to CONTRANS. As mentioned in the code section, 50% of the mass and energy release is subtracted from the FLOOD3 output during the time that the annulus is full to the point that the safety injection tanks are empty. During the long term phase of the event, CONTRANS also produces the mass and energy release data due to decay heat, residual metal energy from the RCS loops, reactor vessel, and steam generators. The steam generator inventory also heats up the exiting steam flow to the secondary temperature prior to release to containment.

For the hot leg break, the greatest quantity of two-phase mass which leaves the top of the core bypasses the steam generators and discharges directly into the containment. Since there is no mechanism for adding significant quantities of the steam generator secondary side fluid energy or RCS metal other than that of the reactor vessel to the break flow stream, the reflood and post-reflood analysis is not conducted for hot leg breaks. Therefore, reactor vessel temperature and heat loss conditions at the end of blowdown are directly input to the CONTRANS long term boiloff model.

The CONTRANS run is the final analytical phase of the analysis. As an input to the CONTRANS run, the Recirculation Actuation Signal (RAS) is required. As described in the assumptions section, this is a function of the amount of available RWST inventory compared with the assumed safety injection and containment spray flow rates. It is conservative to use the minimum spray flow rate for peak pressure and temperature analyses.

The critical time dependent parameters of containment pressure and temperature and, most recently, CCW temperature were the measure used to conclude that the analysis is acceptable.

LOCA Results

A summary of the containment peak pressure and temperature and peak CCW temperature results are provided in Table 6.2-2. Note that this is an overall summary table which compares the original design basis calculations with the updated analysis. Also listed are the relevant pressure and temperature criteria which were followed in the analysis. As can be seen from the summary table, the peak pressure and temperature from the updated analysis to support EPU were from the Double-Ended Hot Leg Slot break minimum SI case. The peak CCW temperature was from the Double-Ended Suction Leg break minimum SI case. Plots of the limiting peak containment pressure and temperature are provided in

Figures 6.2-2a through 6.2-3a. As can be seen, the limiting containment pressure and temperature cases are characterized by the peak right after the blowdown phase of the LOCA event. The small bump further into the transient represents the switch-over to the recirculation mode when relatively hot containment spray water is introduced to the containment.

Each case was shown to meet the peak containment pressure and CCW temperature criteria. The peak containment pressure and temperature cases fell below the original design basis analysis results largely due to improved methods and enhanced computer codes. All the cases analyzed to support the EPU were also shown to meet the SRP Section 6.2.1.3 criteria to demonstrate containment pressure reduced to less than half the peak calculated value by 24 hours.

Main Steam Line Breaks

Analyses are performed to show that the containment design pressure is not exceeded even if the following single active failures are postulated: (1) loss of one containment heat removal train (i.e., two fan coolers and one spray pump); or (2) MSIV failure to close; or (3) main feedwater isolation valve (MFIV) failure to close. The assumptions for each case are discussed more fully in Subsection 6.2.1.4.2.

The peak containment pressure is calculated to occur following the DBA 0.0 percent power MSLB with the failure of one of the trains of the Containment Heat Removal Systems with offsite power available. The peak containment temperature is calculated following the DBA 100.3 percent power with the failure of one MSIV assuming the availability of offsite power. The containment pressure and temperature transients for the most severe MSLB pressure and temperature cases are shown on Figures 6.2-9 through 6.2-12. Figures 6.2-11a and 6.2-12a show the long term pressure and temperature history of the containment temperature DBA. Figures 6.2-12 also shows the calculated transient containment vessel surface temperature for the containment temperature DBA. Pipe break locations, break areas, peak pressures and temperatures, times of peak pressure, initial power level, single active failure assumed and total energy released to the containment are summarized in Table 6.2-4 for each MSLB analyzed. The DBAs are identified in Table 6.2-2.

The heat transfer coefficient used for LOCA analyses initiates with Tagami and exponentially decreases to Uchida. The Uchida heat transfer coefficient contained in the SGNIII version of CONTRANS computer code is used for the analysis of all secondary system breaks.

The containment analyses for the MSLBs are performed using all the containment initial conditions, heat sinks and methodology assumed for the LOCA analyses except for the following:

a) For the MSIV and MFIV failure cases, two containment spray pumps operate and spray 5,400 gpm of water at 104°F into the containment.

b) Since the most severe MSLB containment transients are developed assuming the availability of offsite power, the containment sprays are assumed to reach full containment spray flow at 59.65 seconds after the containment pressure is calculated to reach the containment spray actuation signal (CSAS) analysis setpoint.

c) Consistent with the MSLB single active failure assumptions, four containment fan coolers and two containment spray pumps operate for the MSIV and MFIV failure cases. For the Containment Heat Removal System train failure case, two containment fan coolers and one containment spray pump are assumed to operate.

d) Consistent with the assumption of availability of offsite power, the containment fan coolers are assumed to be in full operation 11.15 seconds after the containment is calculated to reach the safety injection actuation signal (SIAS) analysis setpoint.

e) The vapor-sump heat and mass transfers are conservatively omitted for all MSLB analyses.

For all MSLBs analyzed, following blowdown of the ruptured steam generator unit, the RCS decay heat is transferred to the intact unit which, in turn, vents to the atmosphere when its main steam safety relief valves open. Therefore, there is no physical mechanism for the release of significant amounts of mass or energy to the containment after the end of blowdown. Main feedwater line breaks (MFWLB) are not analyzed since such breaks result in a blowdown less limiting than the MSLB because the pipe break mass flow for the MFWLB is limited by the feedwater line size. Fluid enthalpy for the MFWLB is also less than the enthalpy of the fluid in the MSLB.

A discussion of the computer codes, and the assumptions, including all assumed single active failures, used in deriving the MSLB mass and energy releases are discussed in Subsection 6.2.1.4. Accident chronologies for the most severe secondary system breaks are provided in Table 6.2-9. It is assumed that time equals zero at the start of the accident. The mass and energy release data is provided in Tables 6.2-12F and 6.2-12G.

The instrumentation provided to monitor and record the post-accident containment pressure and temperature, and the containment sump temperature is discussed in Subsection 7.5.1.8. This instrumentation is designed and qualified for the safe shutdown earthquake and the environmental conditions discussed in Sections 3.10 and 3.11.

For Cycle 2, the main steam line break (MSLB) analyses were performed to assure the identification of the most severe cases for Containment Analysis. Since the steam generator feedwater ring is below the initial steam generator water level, feedwater break cases will result in liquid entrainment for all but the most trivial break sizes. For this reason, feedwater line break cases are always less severe than steam line breaks at the same power level and a spectrum of feedwater line break cases is not required.

For the EPU (MSLB), a complete peak pressure analysis was done. In order to confirm that the most severe case is identified, the three single failure events were analyzed at 0%, 25%, 50%, 75%, and 100.3% power levels. The peak pressure case that results in the maximum peak temperature was run as an EQ case to provide the maximum containment temperature. The EQ case was rerun to identify the containment liner temperature.

Following a postulated MSLB main steam isolation and main feedwater isolation are initiated simultaneously by a MSIS from the Engineered Safety Features Actuation System. Feedwater flow is automatically terminated by the main feedwater isolation valves (MFIVs) or MFIV backup valves after receipt of a MSIS. The main steam isolation valves also close upon a MSIS.Table 6.2-65 provides the valve closure times and the assumptions used for the main steam line break analysis.

The feedwater hammer analysis was reviewed for the stretch power operation to demonstrate that the original analysis performed for the 100 percent power operation is conservative for the stretch power level as well.

Analyses have been performed to show that the containment design pressure is not exceeded even if the following single active failures are postulated: (1) loss of one containment heat removal train (i.e., two fan coolers and one spray pump); (2) MSIV failure to close; (3) MFIV failure to close. The specific assumptions for each case are given below:

Case 1 - Loss of One Containment Heat Removal Train

Since postulation of a steam line break plus the loss of one containment heat removal train assumes a single failure (e.g., loss of one diesel generator set), the MSIVs and MFIVs are postulated to function for this case. The main steam inventory in the line between the faulted steam generator nozzle and the nearest MSIV expands into the containment. The feedwater inventory between the faulted steam generator and its MFIV is assumed to flow into the steam generator where it is released to the containment. One containment spray train and two containment fan coolers are assumed to operate.

Case 2 - MSIV Failure

A single failure of one MSIV during a postulated main steam line break accident does not cause uncontrolled blowdown of more than one steam generator. The MSIV failure is postulated to occur at the faulted steam generator. Closure of the operational MSIV provides isolation of the intact steam generator. The MFIVs and both containment heat removal trains (two containment sprays and four containment fan coolers) function for this case. The steam inventory from the faulted steam generator up to the closed MSIV on the intact steam generator expands into the containment. The feedwater inventory between the faulted steam generator and its MFIV is assumed to flow into the steam generator and is released to the containment.

Case 3 - MFIV Failure

A single failure of one MFIV during a postulated main steam line break accident is accommodated by closure of the backup feedwater isolation valves upon receipt of a MSIS. The MSIVs and both containment heat removal trains (two containment sprays and four containment fan coolers) are postulated to operate. The MFIV nearest to the faulted steam generator is postulated to fail. Steam in the steam line between the break and the nearest MSIV expands to the containment. The feedwater inventory between the faulted steam generator and the feedwater backup valve flashes into the affected steam generator.

The feedwater flow rate to the faulted steam generator is determined using the calculated faulted and intact steam generator pressure responses, the feedwater pump characteristics and the feedwater isolation or regulating valve flow coefficients.

Containment External Pressure Analysis

Protection of the containment vessel against excessive external pressure is provided by two independent vacuum relief lines each sized to prevent the differential pressure between the containment and the Shield Building atmosphere from exceeding the design value of 0.70 psi. The vacuum system conforms to the requirements of para. NE-7116 of ASME Section III.

Each vacuum relief assembly consists of a check valve inside and an automatic air-operator butterfly valve outside the containment vessel. Actuation of the butterfly valve is controlled by differential pressure between the Shield Building annulus and the containment atmosphere. Redundant transmitters sense the differential pressure and provide a signal to the pilot solenoid on the air-operated butterfly valve to open. The butterfly valves which perform both a containment isolation function (closed) and vacuum relief function (open), will fail closed on loss of air. They are provided with accumulators which provide a minimum of three operations of the valve for vacuum relief in the event of loss of instrument air. The accumulator and piping for operation of the valves are designed as seismic Category I. Each accumulator is provided with test connections which allow testing of the accumulator and its associated check valves. Containment vacuum relief is also discussed in Subsection 3.8.2.3. The control diagram for the containment vacuum relief valves is shown on Figure 9.4-9. The P & I diagram for the instrument air connections to the vacuum relief valves is shown on Figure 9.3-2.

An analysis is made of the design basis accident for a containment differential pressure which is an actuation of the Containment Heat Removal System during normal plant operation. A sensitivity analysis was performed to compare the results of an accident assuming worst case winter conditions as opposed to one with worst case summer conditions. The highest pressure differential occurred under the winter conditions and is presented herein. The containment external pressure analysis was performed using the Ebasco modified version of the WATEMPT computer code (refer to Appendix 6.2B).

Assumptions used in the analysis of an inadvertent Containment Heat Removal System actuation are listed in Table 6.2-11. The calculated external pressure transient is shown as a function of time in Figures 6.2-16 and 17. The containment and Shield Building external pressure and design values and margin are given in Table 6.2-3.

There is no single failure spurious signal which can initiate both containment spray trains. Initiation of both containment spray subsystems is by coincidence of safety injection actuation signal and high-high containment pressure signal which indicates that a major energy source has been introduced into containment and that humidity is rapidly approaching 100 percent.

Main Steam Line Breaks Results (Cycle 2)

The peak containment pressure for stretch power was calculated to occur following a MSLB at 102 percent power with a failure of one MSIV assuming increased core inlet temperature and offsite power available. The peak containment temperature occurred following a MSLB at 102 percent power with the failure of either one train of the Containment Heat Removal System or one MSIV assuming increased core inlet temperature, decreased RCS flow and offsite power available. The mass and energy released to the containment are summarized in Tables 6.2-67A through F.

For all MSLBs analyzed, following blowdown of the ruptured steam generator unit, the RCS decay heat is transferred to the intact steam generator which, in turn, vents to the atmosphere when its main steam safety relief valves open. Therefore, there is no physical mechanism for the release of significant amounts of mass or energy to the containment after the end of blowdown.

The peak containment pressure/temperature was calculated to occur following a MSLB at 102 percent power and proven to be below the design conditions.

MSLB - Evaluation for Increased Steam Generator Tube Plugging and Reduced RCS Flow Rate

The most limiting mass and energy releases for an inside containment MSLB are generated assuming there are no steam generator tubes plugged and using the maximum Reactor Coolant System (RCS) flow rate. Since mass and energy release data generated for EPU are based on more conservative inputs (maximum RCS flow, no SGTP), the EPU mass and energy release data would bound those with the increased SGTP and reduced RCS flow rate.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Bases

Subcompartment pressurization due to RCS hot and cold leg pipe breaks was considered in the original containment structure design. Since then, however, the NRC revised General Design Criterion (GDC) 4 to eliminate the consideration of dynamic effects of a loss of coolant accident from the plant design bases. The dynamic effects of a LOCA include the effects of missiles, pipe whipping, discharging fluid (i.e., jet impingement), decompression waves within the ruptured pipe and dynamic or nonstatic pressurization in cavities, compartments, and subcompartments. The pipe rupture criteria used in the original (and latter) design are described below. The mechanical/structural loading effects or compartment pressurization from a circumferential (guillotine) or longitudinal (slot) break in the RCS hot or cold legs are no longer considered a design basis (References 28 and 29). The environmental qualification design basis for safety related equipment inside containment remains unchanged (Reference 27).

The existing analysis postulates high energy pipe ruptures in the annular area between the reactor vessel and the primary shield wall (called the reactor cavity), and the enclosed volume below the operating floor bounded by the primary and secondary shield walls (called the secondary shield wall area). This latter region contains the reactor coolant pumps and steam generators. The pressurizer area is analyzed separately for the pressure transient and uplift force.

Analyses are performed to determine the maximum pressure which could be produced by a broken high energy line discharging mass and energy into the subcompartments. Vent area is provided within each subcompartment to mitigate the effects of high energy line breaks.

Calculated DBA differential pressures are compared to the design differential pressure values used in the structural design of subcompartment walls and supports to insure that calculated values are less than design values.

Breaks are chosen for the analysis based upon the maximum break size that could be postulated given the structural stiffness characteristics of the Reactor Coolant System and its supports and pipe restraints. Restraints in the primary shield wall pipe penetrations help limit the size of breaks in the reactor cavity.

Table 6.2-13 is a summary of postulated pipe ruptures for the containment subcompartment analysis. Contained in this table is the pipe break location, description of the break, break area and release rate data and table numbers. The last column in this table, "Release Rate Data Table Numbers," refers to, for each compartment, a table of blowdown mass flow rate and energy release rates as a function of time for the break which was used for the component support evaluation.

Pipe break locations were chosen based upon the high stress point criteria in accordance with Regulatory Guide 1.46 (R0) and SRP 3.6.2, Refer to Subsection 3.6.2).

The peak loads tabulated from each of the pipe breaks listed on Table 6.2-13 was applied to the corresponding structure. In all cases it has been found that the structural design was adequate to withstand the differential forces resulting from the break. Table 6.2-3 provides a comparison between the peak calculated force and the structural design.

The major components of the RCS are designed to withstand the forces associated with the design basis pipe breaks. These pipe thrust forces at the break location, resultant subcompartment differential pressurization-forces and internal asymmetric hydraulic forces acting in the reactor internals. A complete description of the evaluation of the plant faulted condition for these components is provided in Subsection 3.6.2.

6.2.1.2.2 Design Features

The first two paragraphs of this section describe the vent paths that were modeled in the reactor cavity pressurization analysis that was performed for the plant operating License. The description of these vent paths is retained in the UFSAR for historical purposes only. Reactor cavity pressurization is no longer a design basis for the plant.

The reactor cavity is the annular air space between the primary shield wall, a heavily reinforced concrete structure, and the reactor vessel. This region contains the reactor vessel horizontal and vertical supports, located beneath the Reactor Coolant System piping. The major pressure relief paths for the cavity above the piping are the annular space between the reactor vessel flange and the primary shield wall at elevation 36 feet, and the annular spaces between the six primary shield wall pipe penetrations and the hot and discharge legs. In the lower cavity pressure relief is provided by the sump tunnel and the two tunnels leading to the electrical tunnel. These two tunnels are provided with pressure relief dampers as described in Subsection 6.2.1.2.3. All tunnels vent to the secondary shield wall area. Figures 6.2-18 to 21 show the details of the reactor cavity area.

Above elevation 36 feet a neutron shield wall obstructs flow into the refueling pool. About 180 ft^2 of vent area is provided from this enclosed region to the upper containment at the missile shield. See Figure 6.2-19 for details of this area.

The steam generator and pressurizer are enclosed by heavily reinforced concrete shield walls in a single large compartment below elevation 62 feet. See Figures 6.2-18 to 21 for details of the arrangement of subcompartments in this region. Pressure relief is provided by openings in the secondary shield wall to the annular areas between the secondary shield wall and the containment, and by flow through the operating floor where the steam generators pass through the floor to the upper containment.

The lower pressurizer and surge line piping are open to this region and venting is to an upper enclosed volume above elevation 62 feet. Pressure relief in this upper volume is provided by a doorway at the floor level.

Subcompartment free volumes and vent area information is discussed in Subsection 6.2.1.2.3.

6.2.1.2.3 Design Evaluation

The following description documents the reactor cavity pressurization analysis that was performed for the plant Operating License. It is retained in the UFSAR for historical purposes only (for example Tables 6.2-26 and 27). Reactor cavity pressurization is no longer a design basis for the plant.

The modified CEFLASH4 computer code was used to calculate the mass and energy release rates from postulated Reactor Coolant System pipe ruptures. This code is the same as the one containing the combination Henry-Fauske/Moody critical flow subroutine, described in Subsection 6.2.1.1.5 of the Arkansas Nuclear One Unit Two FSAR (Docket No. 50-368).

The "modified CEFLASH-4" code is, different from the CEFLASH-4A code used in the approved CESSAR PSAR. However, the "modified CEFLASH-4" code produces the same mass and energy releases as the CEFLASH-4A code.

The difference between CEFLASH-4A and CEFLASH-4 (not modified) is discussed in CESSAR-PSAR. In the "modified CEFLASH-4" there is the same method for calculation of critical mass flux as there is in CEFLASH-4A. For this reason, CEFLASH-4A and "modified CEFLASH-4" produce the same blowdown release rates.

The "modified CEFLASH-4" code was used because input data were available which satisfied input requirements of that version of the code. CEFLASH-4A input has a different format and would have required re-working the input data which would have been time consuming, while the blowdown release rates would not be changed.

The "modified CEFLASH-4" code was used for the Arkansas Power and Light Unit One-2 FSAR, which has been reviewed and approved by the NRC.

The subcompartment pressurization effects are dependent on the blowdown energy release rates. In order to provide adequate conservatism for design evaluation, the following methodology is used to generate short duration mass and energy release rates.

a) Subcooled and low quality break flow are computed using the Henry - Fauske correlation with a discharge coefficient of 1.0.

b) Break flow with quality above approximately 6.0 percent is computed using the Moody critical flow correlation with a discharge coefficient of 1.0.

c) The momentum flux terms are omitted when solving the conservation of momentum equation for flows within the Reactor Coolant System.

d) Initial Reactor Coolant System conditions are at 102 percent of nominal full power conditions.

e) Reactor Coolant System volumes are increased over nominal values.

f) The pressurizer water level is assumed to be above the normal full power operating water level.

The method used to calculate the mass and energy releases is similar to the one described in Subsection 6.2.1.1.4 of CESSAR (approved December, 1975).

The mass and energy release data that was utilized for the structural design is identical to that used for component support design verification. Therefore, the peak and transient forces provided in Figures 6.2-23 thru 6.2-30 were utilized for both the structural and component design, where applicable. The analysis of the RCS components (i.e., Reactor Vessel, Steam Generator, RC Pumps, Pressurizer and RC Piping) due to the asymmetric pressure loadings is provided in revised Subsection 3.9.1.4.1. A tabulation of results and comparison with the appropriate allowables is also provided. (See Subsection 3.9.1.4.1)

The component and support loads for the Steam Generator, Reactor Coolant Pump, and Pressurizer were determined by equivalent static analyses. A load factor of two on the calculated thrust, jet impingement, and subcompartment pressure loads is employed to account for the dynamic response of the structure. The model employed for static analysis is shown on Figure 3.9-18.

Table 6.2-13 provides a summary of the Reactor Coolant System postulated pipe ruptures. The mass/energy release rates of all ruptures are given in Tables 6.2-14A to 23B.

Since a portion of the main steam lines runs inside the secondary shield wall, the effects of a double ended guillotine rupture of the steam line are analyzed. The break is taken at node 16 of the line, as discussed in Appendix 3.6E and is shown on Figure 3.6E-1. The blowdown data used in the subcompartment analysis is derived from the RELAP 3[3] code used to generate the hydrodynamic forces. The assumptions and options used are the same as those used for the RELAP-4 computer code discussed below. Table 6.2-24 shows the mass/energy release rates as a function of time.

Analyses of the pressure transients in all the subcompartments are performed using the RELAP-4 Mod 5[4] computer code. The program options used in these studies are:

a) The RELAP-4 - CONTAINMENT option;

b) The thermal homogeneous equilibrium model (HEM) for determining the critical flow for air-steam-water mixtures;

c) The incompressible single stream form of the momentum equation, as these break cases produce relatively low subcompartment pressures, even within the break node.

The junction effective inertia (1/A) is calculated in a manner consistent with the methods described in RELAP-4. For a pair of volume v_i and v_k, with cross-sectional areas, A_i and A_k, and lengths in the direction of flow, l_i and l_k and a junction with area A_j and length l_j where $A_j \neq A_i$, A_k and $l_j << l_i$, l_k, the inertia coefficient 1/A is computed as:

$$\frac{l}{A} = \frac{l_i}{2A_i} + \frac{l_j}{A_j} + \frac{l_k}{2A_k} \qquad (1)$$

Flow coefficients for the subcompartment analysis are computed in a manner also consistent with the calculations done by the RELAP-4 code. The junction friction coefficient utilized in the analyses is a combination of the wall friction losses (K_f), and any irreversible friction losses (K_T) such as area changes, turns and gratings. The wall friction loss is computed as:

$$K_f = K_{f_i} + K_{f_j} + K_{f_k} \quad , \qquad (2)$$

$$K_{f_i} = \frac{f l_i}{2D_{H_i}} \left(\frac{A_j}{A_i}\right)^2 \qquad (3)$$

$$K_{fj} = \frac{fl_j}{D_{HJ}} \quad (4)$$

$$K_{fk} = \frac{fl_k}{2\,D_{Hk}} \left(\frac{A_j}{A_k}\right)^2 \quad (5)$$

Where D_{Hn} are the hydraulic diameters of the system of volumes i and k and junction j and f is conservatively assumed to be proportional to D_H - 1/4 (see References 5 and 6). This gives values of order 0.01. The coefficient for the total friction loss at a junction, K_{RELAP}, is then given by

$$K_{RELAP} = K_F + K_T \left(\frac{A_j}{A_T}\right)^2$$

where A_T represents the reference area to which K_T applies.

All subcompartments are assumed to be initially entirely filled with air, with no liquid present and with temperature, pressure and humidity conditions consistent with those used in the containment analysis (Table 6.2-5).

The conservatism of these conditions in maximizing the pressure is discussed in connection with the containment analysis.

For the Stretch Power Analysis, the mass and energy release data have been analyzed on Tables 6.2-66A through F. There is only a minor increase as compared to the original data provided for the same breaks at 2560 MWt.

For a similar mass and energy release increase, the containment peak pressure calculated for the Stretch Power resulted only in a minor rise (≤ .5 percent).

Taking into account the identical methodology used for calculating the containment peak pressure and the subcompartment loading, and also considering the existing design margins ≥ 100 percent) of the containment subcompartments (Table 6.2-2), it is concluded that for the Stretch Power, the subcompartment loads are below the design values.

Subcompartment Modeling

Divisions between subcompartments are determined by the physical flow restrictions within each compartment. A flow restriction is defined by the presence of an object in the flow path that changes the flow area in that direction, with the subdivision defined at the point of minimum flow area. This minimum flow area becomes the junction flow area used in the RELAP 4 analysis. For the models constructed for the reactor cavity and secondary shield wall area flow restrictions included the presence of steel and concrete supports, doorways, vent shafts and gratings, as well as large equipment such as the reactor vessel, primary piping, the steam generator, reactor coolant pumps and the pressurizer. By choosing node boundaries at the various physical flow restrictions, a method consistent with the lumped-parameter calculation model used by RELAP 4 and described above, calculated differential pressures and consequent support loads are realistically maximized. The nodalization sensitivity study performed for the Shearon Harris PSAR (Docket 50-400, 401, 402 and 403) shows that the peak calculated differential pressure is very sensitive to an increasing number of nodes until that number equals the number defined by physical flow restrictions. Increasing the subdivision of nodes beyond this point does not result in increased pressure differentials. The study concludes that further arbitrary subdivision of the compartment is unwarranted and can lead to unrealistic results if these "fictitious junctions" are modeled. The subcompartment models discussed below take account of all physical flow restrictions present in a manner identical to that shown to be optimum by the sensitivity study.

Table 6.2-25 list the results of the subcompartment analysis. In this table the peak node pressure, and peak differential pressure are listed. Along with these values a figure is referenced for both those values. Refer to Table 6.2-25 for a discussion on the peak and transient loading on the major components used to establish the adequacy of the support design.

The analysis performed for the major components support does not differ from the structural analysis model.

a) Reactor cavity

Due to the application of the leak-before-break criteria, the reactor cavity pressurization from the pipe breaks is not longer a design basis for the plant. The following discussion with associated tables and figures is retained only for historical purposes.

Pressure relief dampers were originally installed between the reactor cavity and the electrical tunnel to vent mass and energy released to the cavity area from RCS pipe breaks in the annular space between the reactor vessel and the primary shield wall. The safety related vent function, however, has since been eliminated from the damper design basis (via the 50.59 process) by taking credit for the leak-before-break qualities of the RCS main loop piping. The following paragraph documents performance characteristics of the dampers that were modeled in the reactor cavity pressurization analysis.

The reactor cavity model is shown on Figures 6.2-23 and 24. Volumes are defined by the reactor vessel flange (elevation 36 ft), the hot and discharge legs, the reactor vessel supports, the reactor vessel tangent line, and the pipe penetrations and tunnels. Tables 6.2-26 and 27 provides the various data values and flow coefficients used in the analysis. The relief dampers within the passageways leading to the electrical tunnel (junction 74 and 75) are assumed to open at a pressure differential of 1.1 psid with a 0.2 second delay, with a linear slope from closed to fully open by 0.0574 seconds after the delay. These performance data are the results of analyses performed by the damper manufacturer. Insulation on the reactor and piping is assumed to remain in place and is included in the calculation of occupied volume and obstructed flow area.

The breaks are assumed to occur in volumes above the reactor vessel horizontal supports (volume 2, discharge leg break and volume 4, hot leg break). The 200 square inch discharge leg guillotine break produces the highest pressures of the limited-area breaks. See Figures 6.2-25a to 25m, for the pressure response history for this break. Table 6.23 compares the pressures for the current set of breaks with the design pressures for the reactor cavity walls.

b) Secondary Shield Wall

Secondary shield wall area models are defined on Figures 6.2-26 and 27. Volumes are defined by the suction and discharge legs of the reactor coolant pump, the steam generator diameter, the shield walls and the operating floor. Tables 6.2-28 and 29 list the various data values and flow coefficients used in the analyses. Volume and flow areas are calculated conservatively from the "as built" design and take account of all major equipment and support structures. No flow is assumed to enter the reactor cavity to maximize the pressures.

The breaks analyzed are listed in Table 6.2-13. The 1400 square inch suction leg guillotine break at the steam generator nozzle (which has been eliminated by the application of the LBB criteria), represents the bounding mass and energy release into the compartment for all postulated RCS pipe breaks within the compartment. The mass and energy release at extended power uprate conditions resulting from all other smaller postulated RCS pipe breaks remains bounded by the mass and energy release utilized for the original 1400 square inch suction leg rupture. Thus the differential pressure across the secondary shield wall structure at extended power uprate conditions is bounded by original design basis. The following discussion with some of the associated tables and figures is retained only for historical purposes. For Reactor Coolant System breaks, flow is directed into the region between the steam generator and the primary shield wall (volume 1) away from the pressurizer volumes. This maximized the differential pressure across the Reactor Coolant System. Flow from suction and discharge leg breaks is also directed into the region between the steam generator and the secondary shield wall (volume 2) to find the maximum possible pressure differential across the secondary shield wall. The 1400 square inch suction leg guillotine break at the steam generator nozzle produces the highest differential pressures for the Reactor Coolant System breaks. See Figures 6.2-28a to 28y for the pressure response history for this break for both break flow in Volume 1 and Volume 2. Table 6.2-3 compares the pressures for this break with the design pressure for the compartment wall.

c) Pressurizer Area

The pressurizer area model is defined on Figure 6.2-29. For the pressurizer, a 161 in^2 surge line guillotine break is assumed to occur at elevation 40.67 ft at the pressurizer nozzle (volume 35). In addition, breaks in the spray and relief lines are assumed in the upper pressurizer volume (volume 34).

Upper Pressurizer Compartment

The pressure response in the upper pressurizer subcompartment above EL 62' was assessed for the extended power uprate (EPU) for the double-ended pressurizer relief line, pressurizer spray line and pressurizer surge line breaks, by comparing the estimated mass and energy (M&E) release rates at uprate conditions, with the corresponding values for original design.

It was determined that the M&E release rates for the pressurizer relief line break presented in Table 6.2-22 remain unchanged at EPU conditions. Consequently, the current peak wall differential pressure following the pressurizer relief line break remains unchanged at the EPU conditions.

Taking into consideration the lowest analyzed cold leg temperature of 543°F, the initial EPU M&E release rates following a pressurizer spray line guillotine break is higher than the M&E release rates associated with original design (Tables 6.2-23A and 6.2-23B) by 1.8% and 0.7% respectively. With respect to compartment pressurization, these minimal increases in the M&E release rates are more than compensated by the impact of the increased vent path of approximately 190 square feet due to removal of the missile shield roof of the pressurizer cavity which was included in the original design basis. Consequently, the current peak wall differential pressure following the pressurizer spray line break remains unchanged at the EPU conditions.

Taking into consideration the lowest analyzed cold leg temperature of 543°F, the initial EPU M&E release rates following a pressurizer surge line guillotine break is higher than the M&E release rates associated with original design (Tables 6.2-21A and 6.2-21B) by 0.8% and 0.4% respectively. With respect to compartment pressurization, these minimal increases in the M&E release rates are more than compensated by the impact of the increased vent path of approximately 190 square feet due to removal of the missile shield roof of the pressurizer cavity which was included in the original design basis. Consequently, the current peak wall differential pressure following the pressurizer surge line break remains unchanged at the EPU conditions.

Lower Pressure Compartment

The pressure response in the lower pressurizer subcompartment below EL 62' was assessed for the extended power uprate (EPU) for the double-ended pressurizer surge line break, by comparing the estimated mass and energy (M&E) release rates at uprate conditions, with the corresponding values for original design.

Taking into consideration the lowest analyzed cold leg temperature of 543°F. the initial EPU M&E release rates following a pressurizer surge line guillotine break is higher than the M&E release rates associated with original design (Tables 6.2-21A and 6.2-21B) by 0.8% and 0.4% respectively. Therefore, the maximum differential pressure is conservatively estimated to be approximately 22.7 psid (= 22.5 x 1.008) at the EPU conditions, which is below the design pressure of 24 psid. In conclusion, the design margin for the lower pressurizer compartment wall structure is not significantly impacted at EPU conditions and remains approximately 6% (= (24 - 22.7)/22.7 x 100) at the EPU conditions.

The 161 in^2 guillotine break in the pressurizer surge line at the pressurizer nozzle produces the highest differential pressures for the Reactor Coolant System breaks. Figures 6.2-30a to 30f show the results for the volumes of interest for this break. Table 6.2-3 compares the pressures for this break with the design pressure for the compartment wall.

6.2.1.3 Mass and Energy Release Calculations for Postulated Loss of Coolant Accident

6.2.1.3.1 Description of Blowdown Model

The CEFLASH computer code is used for the generation of LOCA blowdown mass and energy release data for St. Lucie Unit 2. The version used for the original design basis calculations was CEFLASH-4 which was derived from the FLASH-4 code developed by Bettis Atomic Power Laboratory. The EPU LOCA blowdown analysis was done with Version F4A.2.5 of the CEFLASH-4A/FII code. This code is the large break LOCA thermal-hydraulic systems code version and was developed in compliance with Appendix K for ECCS evaluation model analyses. The blowdown time interval when CEFLASH-4A is applied to containment analysis is typically 10 - 25 seconds when the RCS inventory is released to containment. The CEFLASH-4A code is used for both hot and cold leg breaks.

In accordance with the NRC's Standard Review Plan (Ref. 28), the CEFLASH-4A version has been tailored (through input) to maximize the core to coolant heat transfer. By this method, the core is allowed to stay in nucleate boiling except when in heat transfer to steam. In addition, it is allowed to return to nucleate boiling as conditions permit throughout the blowdown transient. Contrary to Appendix K requirements, maximum core to reactor coolant heat transfer is attained such that the energy release to the containment is also maximized.

The technical description of the CEFLASH-4A computer code model is provided in Reference (29). Provided below is a summary of the overall modelling assumptions and conservatisms.

1. The CEFLASH-4A model for the coolant to metal wall heat transfer in a node allows representation of only one wall per node. Accordingly, the thickness used for the overall heat transfer coefficient for each node wall is selected such that the energy released from the system is conservatively modeled.

2. The CEFLASH-4A lumped parameter wall representation uses the total heat capacitance of all the walls in the reactor coolant system that contact the coolant in a given node. This is conservative since, in reality, some of the walls will not participate as effectively as others in the heat transfer process for a large coolant node and heat loss to the ambient surroundings is neglected.

3. Although much of the steel facing the coolant in the reactor coolant system is stainless cladding ($K \approx 12$ Btu/hr-ft-°F), a conservative carbon steel conductivity ($K \approx 26$ Btu/hr-ft-°F) is used for the entire wall. This conservatively overpredicts the energy released from all such walls.

4. Wall to coolant heat transfer is assumed to be conduction limited for all walls; that is, wall surface convection heat transfer coefficients are assumed to be infinite which maximizes the heat transfer.

5. All water volumes are conservatively increased from their nominal design values in order to obtain an upper bound for the available mass and energy in the system prior to a LOCA. Pressure, temperature, and manufacturing tolerance related effects are included in the calculation of an overall volumetric expansion factor for both the RCS and steam generators. This is consistent with the SRPs which require blowdown calculations to be based on hot normal operating conditions.

6. An accepted Jens Lottes two phase heat transfer correlation is used for the core to coolant heat transfer whenever the flow through the core is not pure steam.

7. A maximum core inlet temperature (including uncertainty) and Technical Specification minimum core flowrate are assumed to maximize the available stored energy in the RCS.

8. The core is modelled as 1 radial zone and 5 axial zones. This allows sufficient nodal detail for containment related blowdown calculations.

9. Heat transfer across the steam generator tubes during the transient is modelled with the initial steady state full power overall heat transfer coefficient in both the forward and reverse directions. This is conservative since it maintains a forced convection heat transfer coefficient on the primary side and a nucleate boiling heat transfer coefficient on the secondary side during the LOCA blowdown. Although the RCPs are tripped at the initiation of the transient, this heat transfer coefficient is maintained throughout the transient as a conservative prediction of primary side heat transfer. On the secondary side, the reactor trip following the LOCA would in reality result in a turbine trip which would close the turbine stop valves and the secondary side heat transfer coefficient would decrease to a small natural convection coefficient. Using the initial steady state full power overall heat transfer coefficient conservatively maximizes the reverse heat transfer.

10. For this analysis, no tubes have been assumed to be plugged. Minimizing tube pluggage increases the secondary to primary heat transfer, increases the RCS volume, and minimizes the loop resistance. These effects contribute to a conservative mass and energy release.

11. The decay heat model used in the blowdown calculation is representative of the 1971 ANS standard.

12. The treatment of steam and feedwater flow is handled via input. The CONTRANS containment code was utilized to predict the time to containment high pressure. Adding delay and valve stroke times to this value produced a time dependent table which was utilized in the CEFLASH-4A input. Thus, steam flow is terminated by the closure of the stop valves following reactor trip on high containment pressure plus appropriate signal delays. Feedwater flow is added as a function of isolation valve position over time following MSIS on containment high pressure plus signal delays.

6.2.1.3.2 Description of Core Reflood and Post-Reflood Model

Reflood and Post-Reflood mass and energy release rates of the LOCA are calculated using the FLOOD3 computer code (Ref. 30). Since the refill period (the interval when the reactor vessel refills to the bottom of the active core) is conservatively omitted for containment calculations for containment heat removal considerations, reflood is assumed to follow the initial blowdown phase. The reflood phase is defined as the interval when the reactor vessel level rises, due to safety injection, from the bottom of the active core to the point that the core is quenched. Per SRP Section 6.2.1.3, this is the point at which the level rises to two feet below the top of the active core. The post-reflood period is the interval following the end of reflood to the point that the RCS and steam generators are essentially in equilibrium.

The FLOOD3 code is an extension of the NRC approved FLOOD-MOD2 code (Ref. 9) which was used for the initial design basis analyses. The FLOOD-MOD2 code required separate heat transfer considerations in order to select fluid densities and did not allow time dependent densities. Consequently, it was difficult if not impossible to analyze all possible parametric variations either rigorously or consistently when using FLOOD-MOD2. FLOOD3 uses the FLOOD-MOD2 hydraulics, and reproduces the FLOOD-MOD2 hydraulic solution when consistent sets of hydraulic data are used. FLOOD3 differs from its predecessor in that:

- Intact and broken loops are individually treated,
- Specific volumes of the fluids in the primary loop can be varied with time,
- A uniform methodology for both the Reflood and Post-Reflood time frames is used,
- A more rigorous heat transfer scheme is used for treating the transfer of energy from the loop and steam generator walls to the fluid in the primary loop,
- Containment back pressure may vary with time,
- Safety injection flow rates may be interpolated from a curve or computed explicitly.
- The fluid in the reactor vessel is heated mechanistically, as opposed to assuming saturated liquid in the core.

The intent of the reflood and post-reflood analysis is to extract heat from the hot steam generators following the blowdown phase. Since there is no mechanistic way for exiting steam to pass through the steam generator U-tubes for a hot leg break, the reflood analysis is only conducted for the cold leg breaks. This analysis assumes breaks in the suction and discharge legs.

For the reflood and post-reflood analysis, heat transfer is conservatively modeled for core, vessel walls, vessel internals, loop metal, steam generator tubes, steam generator inventory, and steam generator secondary walls. The FLOOD3 code hydraulics calculates loop flow rates and pressure. The heat transfer process predicts fluid enthalpies and fluid densities are calculated as functions of pressure and enthalpy. The conservatisms in the model are as follows:

A. A one-dimensional heat transfer model is used for all wall heat transfer calculations. It has been demonstrated in past Westinghouse C-E analyses via comparisons of one-dimensional models and otherwise identical two-dimensional models that one-dimensional modeling is conservative.

B. A nucleate boiling heat transfer coefficient of 10,000 Btu/hr-ft^2-°F is used to model the heat transfer from the steam generator tubes to the fluid. This coefficient represents an upper limit, and is conservatively used at all times throughout the tubes.

C. A conservative heat transfer coefficient is used to calculate the secondary to primary heat transfer.

D. The carryover rate fraction (CRF) (the mass ratio of liquid exiting the core to the liquid entering the core) used during reflood is 0.05 up to the 18 inch core level, increases to 0.8 at the 24 inch core level, and is kept at 0.8 until the 9.39 foot level is reached. The level of 9.39 feet is 2 feet below the top of the active core. Other variables, such as core inlet temperature, pressure, flow rate, linear heat rate, or other experimental data are not used to determine the CRF. The assumed CRF is consistent with the prescribed fractions set in the SRPs

E. Reflood is assumed to terminate when the 9.39 foot quench level in the core is reached.

F. During reflood, credit is taken for the condensation of steam in the annulus by the cold safety injection water. As a conservatism, credit is not taken unless the reactor vessel annulus is full. The safety injection flow is injected directly into the annulus. Also, as an additional conservatism, credit is not taken when the safety injection rate is too low to thermodynamically condense all of the steam in the annulus. The percentage of the total steam flow condensed varies slightly with time for each case. For the suction and discharge cold leg break cases, credit is taken for the condensation of 50% of the total steam flow when the annulus is full and the thermodynamic criteria are simultaneously met. No credit for condensation is taken after the SITs empty.

The post-reflood model is identical to the reflood model except that, at the end of reflood, the CRF is changed from 0.8 to 1.0. This conservatively increases the system flow rates due to the increased CRF. The flow rates are further enhanced by the fact that the core liquid height is now constrained at the 9.39 foot level, which maximizes the available driving head between the annulus level and the core in the FLOOD3 flooding equation. All heat transfer coefficients are kept at the values used for the reflood analysis. Condensation is analyzed as previously described, however, there is not sufficient spillage to completely thermodynamically condense the steam so that credit for condensation has not been taken.

6.2.1.3.3 Description of Long-Term Cooling Model

The heat generation rate from shutdown fissions, heavy isotope decay, and fission product decay is considered during this phase of the LOCA event. As mandated in the SRPs, the fission product decay heat model must be equal to or more conservative than the ANS 5.1 1979 decay heat model. This analysis utilized the ANS 5.1 1979 decay heat model $=2\sigma$ with long term actinides.

Following the post reflood period outlined above, the mass/energy source terms for long-term containment analysis are computed concurrently with the containment back pressure in the CONTRANS containment code. The steam flows out the break will be a function of the depressurization of the containment, decay heat (plus margin) and primary metal-to-primary fluid heat transfer. The steam generator secondary fluid, tube, thick and thin metal stored energy are used to superheat the steam prior to discharge into the containment.

The long-term energy release data is based on the containment pressure and the following calculational method:

The reactor coolant system is assumed to be a vessel containing a constant mass of saturated water. The pressure in the vessel is assumed to be the containment pressure. SIS water is injected in the vessel. Steam is formed at a rate determined by decay heat, RCS metal-to-coolant heat transfer and the rate of containment depressurization. Since the water in the vessel is saturated, boiling will occur even without decay heat or metal heat transfer as the containment pressure decreases. The difference between the safety injection rate and the steaming rate is the spillage rate to the sump. It is conservatively assumed that all of the decay and RCS metal heat transfer goes into creating steam. The spillage flow is assumed to have the same enthalpy as the SIS injection enthalpy.

The rate of steam production is

$$m_{stm} = \frac{q_{RCS} - M \frac{du_f}{dt}}{h_g - h_{in}}$$

Where:

m_{stm} = steam flow, lbm/sec

q_{RCS} = decay and RCS metal heat rates, Btu/sec

$M\frac{du_f}{dt}$ = rate of change of internal energy in RCS as a result of depressurization, Btu/sec

h_g = saturated steam enthalpy at containment pressure, Btu/lbm

h_{in} = SIS injection enthalpy, Btu/lbm

The steam created from the decay and RCS metal heat is saturated. For cold leg breaks it is conservatively assumed that all the steam passes through the steam generators and leaves at the secondary side temperature. The long-term energy release to the containment is given by

$$(m\ h)_{Release} = m_{stm}\ h_g + q_{SG}$$

Where:

$(m\ h)_{Release}$ = rate of steam energy release to containment, Btu/sec

q_{SG} = rate of heat transfer from steam generators to RCS steam, Btu/sec

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment

Mass and energy release analyses for postulated secondary system breaks inside containment are performed for a spectrum of break sizes and power levels in order to assure the identification of the most severe cases for containment analysis. The spectrum of breaks was run using the SGNIII computer code at initial power levels of 100.3, 75, 50, 25 and 0 percent. For each power level, analyses are performed for a full double area break and at decreasing break sizes until that largest break size for which no liquid entrainment is calculated could be identified.

At each power level the break size thus identified will allow the greatest mass and energy release and will therefore determine the most severe containment temperature and pressure conditions for the power level under consideration.

Since the steam generator feedwater ring is below the initial steam generator water level, feedwater break cases will result in liquid entrainment for all but the most trivial break sizes. For this reason, feedwater line break cases are always less severe than steam line breaks at the same power level and a spectrum of feedwater line break cases is not required.

6.2.1.4.1 Mass and Energy Release Data

Mass/energy release data for the most severe secondary system pipe rupture are given in Tables 6.2-12F and 12G.

6.2.1.4.2 Single Failure Analysis

Main steam isolation and main feedwater isolation are initiated simultaneously by a MSIS from the Engineered Safety Features Actuation System. Feedwater flow is automatically terminated by the main feedwater isolation valves (MFIVs) or MFIV backup valves after receipt of a MSIS. The main steam isolation valves also close upon a MSIS. Table 6.2-65 provides the valve closure times and the assumptions used for the main steam line break analysis. Figure 6.2-33 shows the main steam and feedwater piping schematic with pipe volumes between specified nodes.

Analyses have been performed to show that the containment design pressure is not exceeded even if the following single active failures are postulated: (1) loss of one containment heat removal train (i.e., two fan coolers and one spray pump); (2) MSIV failure to close; (3) MFIV failure to close. The assumptions for each case are given below.

Case 1 - Loss of One Containment Heat Removal Train

Since postulation of a steam line break plus the loss of one containment heat removal train assumes a single failure (e.g., loss of one diesel generator set), the MSIVs and MFIVs are postulated to function for this case. The main steam inventory in the line between the faulted steam generator nozzle and the nearest MSIV expands into the containment. The feedwater inventory between the faulted steam generator and its MFIV is assumed to flow into the steam generator where it is released to the containment. One containment spray train, and two containment fan coolers are assumed to operate.

Case 2 - MSIV Failure

A single failure of one MSIV during a postulated main steam line break accident does not cause uncontrolled blowdown of more than one steam generator. The MSIV failure is postulated to occur at the faulted steam generator. Closure of the operational MSIV provides isolation of the intact steam generator. The MFIVs and both containment heat removal trains (two containment sprays and four containment fan coolers) function for this case. The steam inventory from the faulted steam generator up to the closed MSIV on the intact steam generator expands into the containment. The feedwater inventory between the faulted steam generator and its MSIV is assumed to flow into the steam generator and is released to the containment.

<u>Case 3 - MFIV Failure</u>

A single failure of one MFIV during a postulated main steam line break accident is accommodated by closure of the backup feedwater isolation valves upon receipt of a MSIS. The MSIVs and both containment beat removal trains (two containment sprays and four containment fan coolers) are postulated to operate. The MFIV nearest to the faulted steam generator is postulated to fail. Steam in the steam line between the break and the nearest MSIV expands into the containment. The feedwater inventory between the faulted steam generator and the feedwater backup valve flashes into the affected steam generator.

The feedwater flowrates to the faulted steam generator are calculated for various power level and single active failure cases using the calculated faulted and intact steam generator pressure responses, the feedwater pump characteristics and the feedwater isolation or regulating valve flow coefficients (see Figure 6.2-34).

The feedwater flow rates were reanalyzed during the Stretch Power Uprate and found to be bounded by the values analyzed for Cycle 1. This was due to a 5% margin that was added to the Cycle 1 inventories in preparation for this future upgrade. (Reference 37).

The feedwater flow rates used for Cycle 1 continue to bound the feedwater flow rates for EPU conditions. This is largely due to the effect of the steam generator steam nozzle restrictors. The restrictors tend to reduce the rate of steam generator pressure reduction. The higher steam generator pressure reduces the feedwater flow.

A failure scenario involving the Auxiliary Feedwater System is not explicitly analyzed for the following reasons:

1) The safety grade AFWS to be installed in St. Lucie 2 in response to post TMI requirements is designed such that no single failure will result in delivery of feedwater to a ruptured generator.

2) The revised main feedwater system provides two Safety grade MFIVs in series in each line, so that isolation of main feedwater without additional inventory considerations will occur even if one of the MFIVs fails.

3) The ESFAS logic and the AFWS will feed auxiliary feedwater (as required) to the intact steam generator during a MSLB. The addition of this water to the intact unit was omitted from all MSLB mass/ energy release analyses since the addition of cold water to the intact unit, if considered, would cool the NSSS and hence would provide slower steaming rates from the ruptured unit to the containment.

Offsite power is assumed to be available for the analysis. Availability of offsite power allows the continuation of reactor coolant pump and feedwater pump flow. Maintaining reactor coolant and feedwater pump flow maximizes the rate of primary to secondary heat transfer which maximizes the rate of mass/energy release.

6.2.1.4.3 Initial Conditions

Reactor Coolant System parameters at 100.3 percent of full power are given in Table 6.2-5. The steam generator's pressure program is given in Section 10.1. Feedwater temperature as a function of power level is given on Figures 10.2-1 and 2. The initial steam generator inventory is calculated assuming manufacturing tolerances which maximize the initial inventory. The increase in the initial inventory resulting from thermal expansion of the steam generator is included.

6.2.1.4.4 Description of Blowdown Model

The SGNIII computer code, described in Reference 2, is used for the secondary system pipe break analysis. All significant equations including those for the calculation of primary to secondary, core to coolant and metal to coolant heat transfer and for the calculation of steam separation and moisture carryover are discussed in Reference 7. Experimental justification for all heat transfer coefficients, steam separation velocities, and two-phase flow correlations are provided in Reference 2.

Following closure of the MFIVs, there is an inventory of feedwater between the MFIV and the faulted steam generator. As the faulted steam generator depressurizes, this inventory starts to boil. As steam in the line expands, this feedwater inventory is pushed into the steam generator and is boiled off by primary to secondary heat transfer. The expansion of the feedwater inventory into the faulted steam generator is considered in the blowdown model. The expansion is assumed to be isentropic.

6.2.1.4.5 Deleted

6.2.1.4.6 Additional Information Required for Confirmatory Analyses

The flow area of the main steam lines are as indicated in Table 6.2-1. For the MSLB analysis, the postulated rupture is assumed to occur at the nozzle of one of the steam generators. Therefore, the fL/D from the affected unit to the break is conservatively assumed to be zero. In the MSLB analysis, the fL/D from the intact steam generator to the break is assumed to be approximately 7, which is related to the flow condition of the 34 inch. ID pipe at the steam generator nozzle. Main steam and main feedwater line volumes are given on Figure 6.2-33.

Feedwater flow to the affected steam generator for the most severe MSLB cases listed in Subsection 6.2.1.1 are shown on Figure 6.2-34.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling Systems

6.2.1.5.1 Introduction and Summary

Appendix K to 10 CFR 50 provides the required and acceptable features of Emergency Core Cooling System (ECCS) evaluation models[12]. Included is the requirement that the containment pressure assumed in the evaluation of ECCS performance not exceed a pressure calculated conservatively for that purpose. The result presented herein is obtained using the method and models approved by the Nuclear Regulatory Commission. This result was used in the ECCS performance analysis for the St. Lucie Unit 2 Cycle 1, which is presented in Subsection 6.3.3.

6.2.1.5.2 Method of Calculation

The calculations reported in this section were performed using the methods described in Reference 13 and approved in Reference 14. In this method, the CEFLASH-4A[15] computer program is used to determine the mass and energy released to the containment during the blowdown phase of a postulated LOCA. The COMPERC-II[16] computer program is used to determine both the mass and energy released to the containment during the reflood phase and the minimum containment pressure response to be used in the evaluation of the effectiveness of the Emergency Core Cooling System.

6.2.1.5.3 Input Parameters

6.2.1.5.3.1 Mass and Energy Release Data

The mass and energy released to the containment for the most severe LOCA, full double ended guillotine on pump discharge, was calculated as a function of time. The quantity of safety injection fluid that is assumed to spill from the break is discussed in Subsection 6.2.1.5.3.5.

6.2.1.5.3.2 Initial Containment Internal Conditions

The initial containment conditions which are used for this analysis are:

Temperature	60°F (minimum)
Pressure	14.33 psia (minimum)
Relative Humidity	100 percent (maximum)

For each parameter, the conservative direction with respect to minimizing the containment pressure appears in parentheses.

6.2.1.5.3.3 Containment Volume

The net free containment volume assumed for this analysis is 2,500,000 ft^3.

6.2.1.5.3.4 Active Heat Sinks

In order to conservatively maximize the heat removal capacity of the containment active heat sinks, both trains of containment spray pumps (two) and fan coolers (four) are assumed to be actuated in the shortest possible time following the break, and to operate at their maximum capacity assuming the minimum temperatures for both the spray water and cooling water. To minimize actuation time for these systems, offsite power is assumed to be available. (It should be noted that offsite power is assumed not to be available for the Safety Injection System). The heat removal capacity of a containment fan cooler is shown as a function of containment temperature in Figure 6.2-35.

The assumed operating parameters for the containment sprays are as follows:

Flow rate	6900 gpm (total, all pumps)
Temperature	55 F

The containment sprays are conservatively assumed to be actuated within 24 seconds from the time of the postulated LOCA. Three of the four fan coolers are assumed to be operating at the time of the postulated LOCA. The fourth fan cooler is actuated in 10.6 sec.

6.2.1.5.3.5 Steam Water Mixing

The effect of condensing containment steam with spilled ECCS water upon the containment pressure is calculated in the manner described in Section III.D.2 of Reference 13.

6.2.1.5.3.6 Passive Heat Sinks

The surface areas and thicknesses of all exposed containment passive heat sinks are listed in Tables 6.2-7 and 8. To conservatively maximize the heat transfer to these passive sinks, the surface areas are assumed to be at the maximum of their uncertainty ranges, and their thermal properties (conductivity and heat capacity) are maximized. The thermal properties assumed for this analysis are:

Material	Thermal Conductivity Btu/ft-h-F	Volumetric Heat Capacity Btu/ft^3-F
Paint on steel	1.67	57.6
Paint on concrete	0.375	47.1
Steel	25.9	53.57
Galvanized steel	64.0	40.60
Stainless steel	9.8	54.0
Concrete	1.0	32.4

6.2.1.5.3.7 Heat Transfer to Passive Heat Sinks

The condensing heat transfer coefficients between the containment atmosphere and the passive heat sinks have been calculated in the manner described in Section III.D.2 and Figure III.D.2-2 of Reference 13. The variation of the condensing heat transfer coefficients as a function of time is shown qualitatively on Figure 6.2-37.

6.2.1.5.3.8 Containment Purge System

The analysis presented in this subsection includes the effects of the eight inch continuous containment purge system. The continuous containment purge system is assumed to be operating at the time of the postulated LOCA. The purge system isolation valves are assumed to be fully closed 5.0 seconds after a containment isolation actuation signal was generated by high containment pressure (6.0 psig). It is conservatively assumed that only dry air is removed from the containment atmosphere.

6.2.1.5.4 Results

For the most severe LOCA, a full double ended guillotine on pump discharge, the minimum containment pressure response to be used in analyzing the effectiveness of the ECCS is shown on Figure 6.2-38. The responses of the containment atmosphere and containment sump temperatures are shown on Figures 6.2-39 and 40, respectively.

6.2.1.6 Testing and Inspection

6.2.1.6.1 Preoperational Testing and Inspection

Preoperational testing of the containment structure and associated systems and components is described in Section 14.0. A description of the pre-operational containment leak test program is given in Subsection 6.2.6.

6.2.1.6.2 Inservice Testing and Inspection

a) Containment Vessel

Periodic "Type A, B and C" leakage rate tests will be conducted in accordance with Appendix J, "Reactor Containment Leakage Testing for Water Cooled Power Reactors," 10 CFR 50, to verify the continued leak-tight integrity of the containment. Technical Specification Amendment 88, dated February 10, 1997, was approved which allows Appendix J, Option B testing. This testing allows the licensee to establish intervals for containment leak-testing based on the performance history and risk significance of the components (containment, penetrations, and valves). All Type A, B, and C testing requirements will be performed at intervals determined by the containment leakage rate testing program (See Technical Specification 6.8.4). The criteria for the new testing intervals is evaluated based on individual performance history and the permitted intervals specified in NEI 94-01.

After the initial preoperational leakage rate tests, a set of three "Type A" tests will be performed at approximately equal intervals during each 10 year service period, with the third test coinciding with the end of the service period (Note that these service intervals have changed based on Appendix J, Option B testing).

"Type B" tests will be performed during each major refueling but in no case at intervals greater than two years. Air locks and door seals will be tested in accordance with Technical Specifications. "Type C" tests will be performed during each reactor shutdown for major refueling but in no case at intervals greater than two years. The applicability of these tests to the containment isolation valves is discussed in Subsection 6.2.4 and Technical Specifications (Note that these service intervals have changed based on Appendix J, Option B testing. See also Technical Specification 6.8.4).

The results of "Type A, B and C" leakage rate tests will be reported in a summary Integrated Leak Rate Test report available for inspection at plant site. Leakage tests that fail to meet the acceptance criteria will be reported in a separate summary that includes an analysis and interpretation of the test data. Results and analyses of supplemental verification tests employed to demonstrate the validity of the leakage rate test measurements will also be included (Note: Reporting requirements are based on the containment leakage rate testing program).

A steel containment vessel, designed, fabricated, inspected and pressure tested in accordance with the ASME Section III and protected by the concrete Shield Building will offer continued structural integrity over the life of the unit. The vessel receives a code stamp from an authoritative body which represents the most recent developments in the techniques of pressure vessel design and fabrication that are backed up by years of research, testing and successful inservice experience. Therefore, it is contemplated that there will be no need for any special in service surveillance program other than visual inspection of the exposed interior and exterior surfaces of the containment vessel as described in the Technical Specifications and the containment leakage rate testing program.

b) Shield Building

Periodic tests of the Shield Building and its ventilation/filtration system will be conducted to verify both system capacity and performance and building leaktight integrity. Periodic testing is discussed in the plant Technical Specifications and will be performed per the frequency and acceptance criteria of the plant Technical Specifications and the Containment Leakage Rate Testing Program.

6.2.1.7 Instrumentation Application

Instrumentation required to monitor containment conditions and actuating those systems and components having safety functions are discussed in Sections 7.2 and 7.3.

6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

The Containment Heat Removal System which consists of the Containment Cooling System (CCS) and Containment Spray System (CSS), is provided to remove heat from the containment, thereby reducing the containment pressure and temperature, and maintaining them at acceptably low levels following an accident. The Containment Heat Removal System is arranged into two independent 100 percent capacity subsystems, each of which is comprised of one containment spray train and two fan coolers. Heat removal is achieved by the simultaneous operation of the two subsystems.

The CSS is designed to remove both heat and fission products from a postaccident containment atmosphere by spraying water from ring headers located high in the containment. The fission product removal capability of this system is discussed in Subsection 6.5.2.

The CCS fan coolers are designed to operate during normal plant operation and following a loss of coolant accident (LOCA) or a main steam line break (MSLB) inside containment. The outlets of the fan coolers are connected to a ductwork system which transports cooled air to various regions of the containment atmosphere.

6.2.2.1 Design Bases

The design bases for the Containment Heat Removal Systems include the following:

a) The Containment Heat Removal System is designed to ensure its capability to accommodate the source of energy, the energy release rate and the integrated energy release due to the postulated accidents outlined in Subsections 6.2.1.3 and 6.2.1.4.

b) The Containment Heat Removal System is part of the Engineered Safety Features (ESF) which will function immediately following an accident. Only one of the two independent subsystems (one CS pump and two fan coolers) is required to prevent the pressure and temperature in containment from reaching unacceptable levels following an accident.

c) Each Containment Heat Removal subsystem was procured to reduce post-accident containment pressure to less than one half its peak value within 24 hours.

d) The components of the Containment Heat Removal System located inside containment are designed to operate under accident conditions for extended periods. The post-accident environment and the environmental qualification of equipment is described in Section 3.11.

e) The Containment Heat Removal System is designed such that no single failure can prevent it from performing its design function. Two of four fan coolers and one CSS loop are powered from an independent safety bus and the other subsystem is powered from the other independent safety bus. Loss of one bus does not affect the ability of the Containment Heat Removal System to maintain containment temperature and pressure below the design values.

f) Components of the Containment Heat Removal System which are necessary to support the system safety functions and are located inside containment are designed to Quality Group B, seismic Category I requirements.

g) The Containment Heat Removal System is designed to withstand the effects associated with piping failures (see Section 3.6 and associated appendices).

h) The Containment Heat Removal System incorporates provisions for periodic inspection and testing as described in Subsection 6.2.2.4.

6.2.2.2 System Design

Containment heat removal is accomplished by the combined effect of the Containment Spray System and the Containment Cooling System. These systems are described separately below.

6.2.2.2.1 Containment Spray System

The Containment Spray System (CSS) consists of two independent and redundant trains each consisting of a spray pump, shutdown heat exchanger, piping, valves, and spray header as shown on Figure 6.2-41. The system has two modes of operation:

a) The initial injection mode, during which the system draws water from the refueling water tank (RWT).

b) The recirculation mode during which the CSS takes suction from the containment sump for long term cooling.

Containment spray is automatically initiated following receipt of the containment spray actuation signal (CSAS), which is a coincidence of the safety injection actuation signal (SIAS) and the high-high containment pressure signal of 5.4 psig (refer to Section 7.3). Upon receipt of the CSAS, the containment spray pumps are started and isolation valves FCV-07-1A and FCV-07-1B are opened. The system can be manually controlled both locally and from the control room. During this mode, the containment spray pumps take suction from the RWT and spray borated water directly into the containment atmosphere. Although the spray water first passes through the shutdown heat exchanger, no significant heat load is placed on the exchanger during this mode of operation. This period of injection lasts for a minimum of 20 minutes assuming system runout flow for two high pressure safety injection (HPSI) pumps, two low pressure safety injection (LPSI) pumps, and two containment spray (CS) pumps, with the RWT at the Technical Specification level (see Section 6.3 and Subsection 6.5.2). System delay times are given in Figures 6.2-42 and 43.

When a low level in the RWT is reached (see Section 7.3), the recirculation actuation signal (RAS) is initiated. This closes RWT isolation valves MV-07-1A and MV-07-1B in 90 seconds and opens containment sump isolation valves MV-07-2A and MV-07-2B in 30 seconds. These valves can be manually operated locally or from the control room. MV-07-1A, MV-07-1B, MV-07-2A and MV-07-2B are subject to the requirements of NRC Generic Letter 89-10. The motor operators on these valves are designed such that the sump isolation valves open faster than the RWT valves close, thus insuring continuous suction for the spray pumps. Further, enough water is maintained below the low level point in the RWT to maintain supply throughout the closure of the RWT isolation valves. Should, however, one of the RWT valves fail to close, the water seal created by the difference in elevation between the containment sump water level and RWT level would prevent air from being drawn into the system.

The setpoints for the actuation signals are described in Section 7.3 and the rationale for establishing the setpoints are based on Subsections 6.2.1.3 and 6.2.1.4 analyses.

As discussed in NRC Generic Letter 2008-01 and INPO SER 2-05, the presence of unanticipated gas voids within fluid systems can challenge the ability of systems to perform their design functions due to issues such as gas binding, water hammer, injection delay times, etc. Requirements for maintaining Containment Spray system operability with respect to gas intrusion are contained within Technical Specifications and Gas Accumulation Management Program procedures.

The containment sump is a large reservoir, designed in accordance with the Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems", June, 1974 (RO) provides for long-term recirculation.

Ongoing research and new guidance from the NRC have resulted in the need for upgrades to the containment sump system at St. Lucie Unit 2 (PSL-2). Operating experience identified new contributors to debris and possible blockage of PWR sumps, such as degraded or failed containment paint (coatings). Thus, NRC Generic Safety Issue 191 (GSI-191), "Assessment of Debris Accumulation on PWR Sumps Performance," was created in 1998.

In issuing report NUREG/CR-6762, NRC research determined that post-accident sump blockage is a generic concern for PWRs. On September 13, 2004, the NRC issued Generic Letter 2004-02 (GL 2004-02), which required all licensees to address planned actions to mitigate the potential impact of debris blockage on Emergency Recirculation during design basis accidents at Pressurized Water Reactors.

In response to GL 2004-02, FPL committed to modifications to the PSL-2 existing sump screen system to meet the applicable regulatory requirements discussed in the generic letter. According to the response, "Corrective actions will be implemented to ensure that the [ECCS] and [CSS] recirculation functions under debris loading conditions will be in compliance with the Applicable Regulatory Requirements section of [GL 2004-02] when all modifications are completed. The licensing basis will be updated to reflect the results of the analysis and modifications performed to demonstrate compliance with the regulatory requirements. This update will be performed in accordance with the requirements of 10 CFR 50.59."

A design modification implements "correction actions" (namely, the installation of new sump strainers) to support the ECCS and CSS recirculation functions under debris loading conditions. The installed sump strainer configuration meets the functional and design basis requirements of the existing sump and screen. New downstream effects concerns identified by GL 2004-02 will continue to be addressed through industry initiatives for resolution and closure of GL 2004-02.

Sump strainer configuration can be seen on Figure 6.2-44 and is discussed in Subsection 6.2.2.2.3.

During recirculation, spray flow drawn from the sump is first cooled by the shutdown heat exchanger before returning to the containment through the spray nozzles.

Each train of containment spray contains four spray ring headers located high in the containment. Each header contains numerous spray nozzles designed to produce widely dispersed sprays with fine droplets for enhanced heat removal capabilities. A complete discussion of the ring headers and spray nozzles is found in Subsection 6.2.2.3.

All portions of the CSS located inside containment are designed to operate in environmental conditions associated with a post-accident containment atmosphere. Environmental qualification of components is described in Section 3.11. All system components are fabricated of corrosion resistant materials as indicated in Table 6.2-38. Design codes and standards are also identified in Table 6.2-38.

6.2.2.2.2 Containment Cooling System

The Containment Cooling System (CCS), shown on Figures 9.4-1 and 9.4-9 consists of four coolers, a ducted air distribution system (Figures 6.2-46 and 6.2-47) and the associated instrumentation and controls. Each fan cooler consists of an axial flow fan, two speed motor, casing, two banks of cooling coils, and back draft damper. The design and performance data are shown in Table 6.2-38. The design codes and standards are also specified therein.

The system has two modes of operation which are:

a) Normal Operation - during which three fan coolers are operating at a rated flow of 60,000 CFM each, with high speed motor operation.

b) Accident Operation - during which at least two fan coolers are operating at a reduced flow of 39,600 CFM each, with low speed motor operation.

During normal operation three of the four fan coolers are operating at the higher of the two speeds. As discussed in Subsection 9.4.8.4.1, the Containment Fan Cooling system is required to maintain reactor supports within design limitations.

Upon receipt of SIAS, the four fans are automatically placed into low speed operation (39,600 CFM) within 10 seconds if offsite power is available and 23 seconds if unavailable. For system delay times, see Figures 6.2-42 and 43. For a discussion of the rationale behind selection of setpoints for CCS actuation, see Section 7.3. No operator intervention is required to initiate fan cooler emergency operation.

Each fan cooler unit is sized to remove one-third of the total normal containment heat load. Containment ambient temperature is limited to an average of 120 F when the units are supplied with 100 F component cooling water.

Each cooling coil bank is made up of coil sections connected to supply and return manifolds of the Component Cooling Water System (CCWS). Coils are horizontal tube, vertical plate-fin type, and are mounted on a structural frame. Cooling coil frame, cooling coils, coil header, stub and flange materials are given in Table 6.2-38. The flow to the coils from the CCWS is monitored by a low flow alarm.

Cooling coils are provided with individual stainless steel drain pans and drain piping to prevent flooding of lower coils by condensate cascading from upper coils. Condensation is drained to the reactor cavity sump. The cooling coils are designed to ASME Code, Section III, Code Class 2 requirements. Hydrostatic testing to 300 psig and bubble testing to 200 psig are performed.

The fans are axial type, direct motor driven, with air foil adjustable blades. The motors are totally enclosed air-over type, therefore the fan bearings do not contact the air stream. Bearings have a minimum B-10 life as defined by Anti-Friction Bearing Manufacturer's Association (AFBMA) of 100,000 hours under normal operating conditions. Fan housing is of welded steel construction and reinforced to withstand the transient pressure rise following an accident.

The casing of the containment fan coolers, ring header, and duct riser from the ring header to inside the secondary shield wall are required to remain intact following an accident. The casing of each fan cooler is reinforced to withstand a pressure differential of two psi in any direction. It is constructed of 3/8 inch carbon steel plate in the bottom and 3/16 inch plate for vertical walls.

Adequate quantity of relief dampers have been provided in the ring header and duct risers to limit the differential pressure to 20 in. wg or less. Relief dampers are designed in accordance with American Air Filter Topical Report AAF-TR-7101, "Design and Testing of Fan Coolers - Filter Systems for Nuclear Applications". This report was accepted by the NRC per letter dated November 10, 1972. Relief dampers are designed to withstand accident induced pressure transients as indicated in Subsection 6.2.1. To withstand a differential pressure of 20 in. wg, the ring header and duct risers are constructed with 14 gage galvanized steel with reinforcing angles. Blowout panels are provided on the duct risers between the fan coolers and ring header. These attenuate high pressure transmission from inside the secondary shield wall area through the duct. Blowout panels are engineered to open at a differential pressure of 1.25 psi. In addition, to prevent any damage to the fan coolers, gravity dampers are also provided at the discharge side of the fan. Gravity dampers are designed to withstand a differential back pressure of 1.5 psi.

The fan coolers are located outside the secondary shield wall in four different quadrants of the containment: three on elevation 45 ft. and one on elevation 62 ft. This arrangement provides separation and minimizes recirculation between units. The discharge side of the fan coolers are connected to the ring header manifold to provide mixing of air at elevation 105 feet. Adequate quantity of supply registers or air outlets are provided around the periphery of the ring header blowing air toward the center of containment. To enhance mixing of air above the ring header, some of these air outlets are directed upward to the top of the dome. The rest of the air outlets are located at various parts and inside of the secondary shield wall from elevation 45 ft. down to elevation 26 ft. For a given air flow, each register is sized to give a horizontal throw of approximately 40 ft. (average) that covers a wide area and gives an outlet velocity that induces mixing of containment air. Air mixing is also accomplished by convection. Cold air from the ring header will settle downward displacing hotter air. The colder air is heated causing it to rise toward the suction of the fan coolers. By verifying the air flow of each register and the cooling performance of the cooling coils during pre-operational testing, the adequacy of mixing the containment air by fan coolers is assured. See Figures 6.2-46 and 6.2-47 for plan and elevation views showing the ductwork and the location of outlet registers.

Provisions made to facilitate periodic testing of the Containment Cooling System are discussed in Subsection 6.2.2.4.

Environmental qualification of system components is addressed in Section 3.11.

The analysis of the stretch power impact on the Containment Cooling System has proved that the system is capable of operating without any modification at the increased power level (2710 MWt).

6.2.2.2.3 Containment Sump Design

The containment sump is a large collecting reservoir provided to supply water to the Containment Spray and Safety Injection Systems for long-term recirculation. Design of the containment sump, which follows the recommendations outlined in Regulatory Guide 1.82 (R0), is described in the following paragraphs:

a) A large capacity sump, enclosing the redundant suctions of the ECCS and the CSS which are separated by approximately 15 ft. is provided for continuous recirculation.

b) The sump is separated from the Reactor Coolant, Main Steam, and Feedwater Systems by the secondary shield wall. Where necessary, such as in the case of safety injection tank piping and charging lines, design provisions are made to protect the sump from effects of piping failure. For a more complete discussion, see Section 3.6.

The Reactor Drain Tank, also located in the sump, is designed to remain in place following an accident. The effect of uplift loads resulting from the submergence of an empty tank have been analyzed and found to be well within the capabilities of the hold down bolts. A Containment Isolation Actuation Signal (CIAS) isolates the tank and stops and drain pumps.

c) The sump is located at the lowest floor elevation in the containment (exclusive of the reactor cavity sump) and is shielded by two filtering devices; an outer trash rack, and a strainer system. The strainer system is composed of the following two primary assemblies:

1) Eight vertical strainer stacks composed of individual disks formed by perforated sheet, bolted together in vertical stacks with intermediate stiffener support plates, and a core tube for flow of water down to the plenum. Each stack is laterally supported via carbon steel brackets fastened to the south sump wall, which has steel embedment plates. The strainer stacks and plenum are fastened to the base structure to provide adequate gravity and seismic support without inducing significant thermal loading on the strainer assemblies.

2) A stainless steel framed substructure, which acts as a flow plenum that is sealed for the design debris particle size and supports the loads from the strainer stacks that rest on top of the plenum. The plenum also extends down to the sump floor at elevation 7'-7" at the ECCS suction pipe inlets. The plenum is mounted to the Elevation 7'-7" floor and the walls at Elevation 11'-0" concrete ledge.

With this strainer system design, as shown in Figure 6.2-44, fluid is filtered by the strainers prior to reaching the safeguards systems suction lines.

d) Drains from the various regions of the containment are directed to the sump via vent openings in the secondary shield wall. These vent openings have coarse grating acting as trash racks that prevent debris laden water from impinging on the screens that enclose the sump.

e) Large debris is prevented from reaching the sump by the trash racks placed at the secondary shield wall vent openings. These racks are designed to withstand the effects of the SSE and remain functional to the above design requirements. Small debris including insulation on piping and equipment is considered to be the primary source of post-accident debris inside containment, which could potentially clog the sump strainers.

There are two types of thermal insulation used inside the RCB. They are:

1) Metal Reflective

2) Metal Encapsulated

The metal reflective insulation is built of stainless steel panels. The panels consist of interior and exterior sheets. It is estimated that the total area of metal reflective insulation is 6182 square feet. This insulation is used on the Reactor Vessel and Reactor Coolant Pumps.

The second type of insulation (Type A), metal encapsulated, occurs in two forms. The predominant form consists of a fiberglass insulating wool blanket encapsulated in fiberglass cloth contained within a metal jacket. Fiberglass insulation encapsulated with metal is also used for anti-sweat insulation.

The estimated quantity of this type insulation used inside containment appears in Table 6.2-40.

Another form of metal encapsulated insulation (Type B) occurs on small bore lines and forms a part of some support/restraint structures. These pieces, or pipe shields, are half cylinders of calcium silicate and marinite held together around the pipe by its interlocking stainless steel jacket. Two stainless steel clamps are provided around each unit.
A small quantity of calcium silicate insulation is also used on piping located outside of the biological shield wall. This insulation is held together by stainless steel jackets and is designed to remain in place during and after a seismic event. The calcium silicate insulation used outside the biological shield wall is located above the maximum flood level and is not subject to transport to the sump strainers. Due to the small size of these units and their wide dispersion in the RCB these units present no hazard to plugging of the sump strainers.

Both types of insulation are designed to remain in place during and after a seismic event. All insulation is designed and constructed to not be adversely affected by postulated accident environmental conditions. Refer to Table 6.2-39. Following an accident the only insulation that will be dislodged is that in the vicinity of a high energy line break or in the direct path of the resultant jet. All other insulation will remain in place. Any dislodged metallic insulation will settle out of water flow in the bottom of containment. Approach flow velocities are expected to be too low to transport heavy metallic assemblies. In any case the trash racks at the secondary shield wall vent openings will prevent these assemblies from reaching the sump.

Fiberglass insulation assemblies have undergone substantial testing by Owens/Corning (Topical Report OCF-1). These tests were designed to determine the behavior of the insulation under accident conditions and to ensure that the containment sump operation is not impaired. The results of these tests demonstrate that this insulation is suitable for use inside the containment and does not obstruct the flow through containment sump strainers. This report was reviewed and endorsed by the Nuclear Regulatory Commission.

f) The containment sump strainer system encloses the sump suction lines. The overall gross surface area is approximately 5600 ft^2. The surface area is sufficient to provide adequate NPSH to the ECCS and CS pumps considering the postulated debris loading, including a limiting fiber bed condition. In addition, a seismically supported vertical sump divider plate is provided in the strainer plenum to separate the redundant suction lines. The perforated plate hole size for the plenum divider plate is sized to a nominal diameter of 0.0625 inches to avoid entrapment of particles greater than a size of 0.135 in the HPSI discharge throttle valves and allows flow between the two sides of the strainer plenum.

g) The strainer system is designed to withstand the effects of the SSE. Limiting the perforated plate hole size to a nominal diameter of 0.0625 inches controls the maximum allowable debris size that can pass through the strainers. Particle retention is 100% of particles larger than 0.06875 (+0). The perforated plate hole size was selected to avoid entrapment of particles greater than 0.090" in diameter (nominal value Ref. 36) in the fuel assembly spacer grids. The strainer system is made of Type 304 stainless steel (except support attachments) to resist corrosion and structural degradation.

h) Two carbon steel guard pipes extend from the containment sump inside containment to outside the Shield Building wall. Each pipe is directly welded to a steel containment vessel nozzle and is an extension of the nozzle in both directions. Passing through each carbon steel pipe is a stainless steel sump suction line which connects the containment sump to the suction of the engineered safety feature pumps. The stainless steel pipes are welded to the carbon steel pipes at the containment sump so that water cannot enter the annulus formed by the concentric pipes. Outside the containment the stainless steel pipes are sealed to the guard pipes by means of a stainless steel bellows. The bellows seal allows for anticipated differential movement due to thermal or seismic forces.

i) The containment sump strainers are designed to limit air ingestion to less than 2%. The containment sump strainers were evaluated taking into consideration NRC guidance regarding air ingestion and vortexing. Reg. Guide 1.82 states that vortexing can lead to air ingestion. Thus the evaluation considers any vortexing to be unacceptable. Reg Guide 1.82 recommends limiting air ingestion to less than 2% by volume.

The evaluation of the containment sump strainers for vortexting is based on PCI prototype testing and application of general vortexing limiting features as described in Reg. Guide 1.82. The test results showed no vortexing and use of the general vortexing limiting features in the construction of the PCI strainers preclude the formation of vorticles. Thus there are no vortexing concerns.

The containment sump evaluation and Table A-1 of Reg Guide 1.82 indicate that sump performance specifically related to air ingestion is a strong function of the Froude Number, Fr. By limiting the Froude Number to a maximum of 0.25, air ingestion can be maintained to <2%. The containment sump evaluation determined that the Froude Number for the PCI Sure Flow strainers is less than 0.25. Therefore due to the combination of a low Froude Number and lack of an air entrainment mechanism (i.e., vortext formation) in conjunction with the complete submergence of the strainer, air ingestion is not expected to occur.

The elevation in relation to structures can be seen on Figures 1.2-10 and 1.2-11. In compliance with NUREG-0737, the containment is provided with redundant means of post-accident water level indication consisting of one narrow range and two wide range level monitors. The wide range monitors are capable of measuring water level from the lowest point in the containment up to the maximum postulated flood level. These indicators are fully qualified for use inside containment, safety-related (Category 2 as defined in Reg. Guide 1.97) and seismic Category I. Further description can be found in Subsection 7.5.3.2.

j) Removable platforms are provided to permit visual inspection of the sump.

6.2.2.3 Design Evaluation

The structures and components inside containment absorb energy released after an accident, acting as heat sinks. However, this does not prevent containment pressures and temperatures from reaching unacceptable levels and therefore, some active means of heat removal must be provided. The Containment Spray System and the Containment Cooling System are provided as means for containment heat removal. Diverse and separate actuation signals for each system with fully redundant channels within each system have been provided for greater reliability. For a discussion of safety grade signals, see Section 7.3.

Accident analyses given in Subsection 6.2.1 demonstrate that any one of the two independent subsystems, consisting of one spray pump and two fan coolers, can maintain post-accident containment pressures and temperatures within acceptable levels. Also Subsection 6.2.1 gives the integrated energy content of the containment atmosphere, sump fluid, and heat sinks following an accident.

A single failure analysis is made on active components of the Containment Heat Removal System to show that minimum required heat removal capability remains after any single failure. Table 6.2-41 demonstrates that no single failure, up to and including that of an emergency diesel generator, can prevent the functioning of at least one containment pump and two fan coolers.

System design is guided by General Design Criteria 38, 39 and 40; and Regulatory Guides 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," November, 1970 (R0); 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants," February, 1976 (R3); 1.29, "Seismic Design Classification," February, 1976 (R2), and 1.82 (R0). Specific codes governing component design can be found in Table 6.2-38.

6.2.2.3.1 Containment Spray System Evaluation

Following receipt of CSAS, which occurs when containment pressure reaches 11 psig, both spray pumps automatically start and both containment spray isolation valves open. Since the RWT supply line isolation valves are in the normally open position, no active failure can prevent positive suction to the spray pumps. When operating in the injection mode, a net positive suction head (NPSH) of 48.5 feet is available at the pump. This assumes a maximum fluid temperature of 100°F and takes no credit for the height of fluid in the RWT. The NPSH required is 31.8 feet; for a tabulation of parameters used in the calculation of the available NPSH, see Table 6.2-42. Upon receipt of the CSAS, the spray pumps begin delivering flow through the spray nozzles in 23.5 seconds with offsite power available and in 45.5 seconds without offsite power. Full flow is achieved in 36.5 or 58.5 seconds, with and without offsite power respectively.

A CSS analysis is presented on Figure 6.2-48. During injection, the system delivers between a minimum of 2800 gpm and a maximum of 3020 gpm against a containment pressure of 44 psig, depending on RWT water level. When atmospheric pressure exists in containment, the system delivers between 3400 and 3600 gpm. Of the above flows, 150 gpm is used for continuous pump recirculation (injection phase only).

Following receipt of recirculation actuation signal pump suction is automatically switched to the containment sump. Following an accident, the pressure inside containment exceeds the saturation pressure corresponding to sump water temperature. No credit, however, is taken for this favorable situation. As required by Regulatory Guide 1.1 (R0), NPSH calculations are performed for the entire range of sump temperatures, 80F to 240F, and the minimum containment pressure physically possible, i.e., sump water saturation pressure. Further, it is assumed that the sump is filled to elevation 23.38 feet following the design basis LOCA. The minimum NPSH available at EPU conditions for the CS pump during recirculation, at the limiting plant specific containment sump water temperature of 192°F, for the design basis case is approximately 32.5 feet. Manufacturer's test results indicate that no more than 23.9 feet is required over the entire pump range demonstrating that adequate NPSH is available at all times (see Table 6.2-42).

While in the recirculation mode, containment spray essentially acts as a closed system delivering a fixed flow independent of containment pressure. As indicated on Figure 6.2-48, a fixed flow of 3560 gpm is maintained during recirculation.

cooling the spray flow from 240 to 172°F, assuming a component cooling water supply temperature of 120°F.

Each train of the CSS has a supply header and four spray nozzle rings located high in the containment. Each train has a minimum of 178 spray nozzles. The following CSS design provisions are made to maximize the heat removal capability of the system:

a) The nozzle rings are located as high as practicable inside containment (see Figures 6.5-4 and 6.5-5). Spray droplet fall heights range from 107 to 154 feet down to the operating floor at elevation 62 feet.

b) The nozzles produce widely dispersed 60 degree cone spray patterns. The typical drop size distribution and pattern distribution of a nozzle spraying vertically down are provided on Figures 6.5-6 and 7. Substantial spray pattern overlap insures adequate coverage of the containment cross section (see Subsection 6.5.2).

c) Nozzles are designed to produce small droplet sizes to enhance heat transfer characteristics. The selection of droplet size is also based on maximum fission product removal and is discussed in Subsection 6.5.2. Method employed to determine the mean spray drop size is described in Subsection 6.5.2.4.

d) The nozzles are passive devices which produce constant spray patterns without moving parts.

e) The nozzle orifice size is larger than the maximum particle size postulated in the flow path and thus are not subject to clogging.

The nozzles used are manufactured expressly for use in nuclear power plants. Nozzle design data appears in Table 6.2-43.

6.2.2.3.2 Containment Cooling System Evaluation

The heat removal rates of the fan coolers as a function of containment atmosphere temperature, for normal and accident conditions, are shown on Figures 6.2-49 and 6.2-50. Table 6.2-44 shows the heat and condensate removal rate for specific values of containment air/steam temperature and component cooling water temperature.

A fan cooler heat transfer test program was conducted by American Air Filter. The temperature, pressure and humidity were varied for each test by controlling the steam-to-air ratio and velocity of saturated air entering the coil. Heat removal capacity at the coil was measured for specified cooling water flowrates.

Since the fan coolers are serviced by the Component Cooling Water System, which is a closed system, no significant fouling effects are expected.

Cooling coils in the containment fan coolers have been designed with an anticipated water fouling factor of 0.0005. Fouling factor is primarily influenced by water chemistry, velocity and temperature. During accident conditioning only the water temperature will change, from approximately 102°F inlet to 206°F outlet. According to TEMA standards, for clean, treated water, fouling factor will not change over this temperature range. Therefore, the heat removal capability of 61.6 x 100 BTU/Hr, (original procurement value), based on component cooling water entering temperature of 100°F, of each fan cooler will remain unchanged during accident conditions. Note that this bounding capability reflects performance of a clean CCW Heat Exchanger as procured to the original specifications. The actual heat load from the analysis at any given time will be different.

6.2.2.4 Tests and Inspection

6.2.2.4.1 Containment Spray System

The containment spray pumps are qualified for operability by being subjected to tests by the manufacturer. Among the tests performed are:

a) Hydrostatic test of the pump casing at 150 percent of maximum operation pressure.

b) Suction pressure suppression test to verify NPSH requirements over entire pump capacity range.

c) Transient test of 40°F to 240°F in 10 seconds to conservatively simulate switch-over from RWT to sump suction.

d) Measurements to verify minimum pump casing wall thickness.

e) Seal leakage tests.

To verify that the system responds and performs its function as required, system tests are performed at intervals during reactor shutdowns for refueling or scheduled maintenance. The tests are performed with the isolation valves in the spray lines blocked and utilizing the miniflow test line connecting the pump discharge to the refueling water tank. Pressure and flow indicators are provided to verify measured pump performance characteristics. Operation of the system is initiated by tripping the actuation instrumentation. The test is considered satisfactory if visual observations indicate that all components operate as required. Surveillance requirements are a part of the Technical Specifications. As discussed in NRC Generic Letter 2008-01 and INPO SER 2-05, the presence of unanticipated gas voids within fluid systems can challenge the ability of systems to perform their design functions due to issues such as gas binding, water hammer, injection delay times, etc. Requirements for maintaining Containment Spray system operability with respect to gas intrusion are contained within Technical Specifications and Gas Accumulation Management Program procedures.

The spray nozzles are subjected to testing by the manufacturer. Tests to verify pressure drop, spray pattern, and droplet size are performed in the shop (see Subsection 6.5.2). In situ surveillance testing is performed in accordance with the Technical Specifications.

Active components of the CSS are required to remain functional during and subsequent to an accident. The equipment manufacturer is provided with the component design parameters, system transients, and seismic perturbations in order to qualify his equipment. The results of the operability tests are provided in Appendix 3.9A.

Removable inspection plates are located on both sides of the strainer plenum. The plates allow access to the inside of the strainer plenum to perform visual inspections of the strainer and ECCS suction lines.

6.2.2.4.2 Containment Cooling System

Components of the containment fan coolers are performance tested in the manufacturer's shop and then tested and balanced after installation in the plant to demonstrate the proper functioning of the system. Prototype performance tests for motors are conducted under simulated accident conditions in accordance with IEEE 334-1974, "IEEE Standard for Type Tests of Continuous Duty Class IE 1E Motors for Nuclear Power Generating Stations." Testing of the fan coolers is performed in accordance with American Air Filter Topical Report AAF-TR-701, "Design and Testing of Fan Cooler-Filter Systems for Nuclear Applications", ," February 1972.

For maintenance, servicing and visual inspection of containment fan coolers, access doors are provided in the housing. Fans, dampers and cooling coils are installed in such a way that these components can be removed for servicing. Flow ports and thermocouples are provided for testing the fan performance and heat removal capability of the cooling coils respectively.

The heat removal capability of the cooling coils is verified by noting the temperature readings in the control room.

An initial fan performance test is conducted.

6.2.2.4.3 Containment Sump Strainer

Performance testing was performed on a PCI Sure Flow prototype strainer module to determine head loss of the strainer that bound all plant design conditions. The performance testing of the PCI strainer incorporated the following:

a) The prototype strainer module represents one module of the 8 stack vertical strainers installed. The scaling of strainer module flow and debris amount was determined by calculation.

b) The debris input is representative of the various types of debris, amounts generated and transported based on the debris generation and transport calculations.

c) The following tests were performed:

    1. Clean Strainer Head Loss - determined the head loss of the clean strainer.

    2. Fibrous Debris Only - This test included only fibrous debris to establish transport characteristics of the fibers and evaluated how the fibrous debris bed is formed.

    3. Design Basis Debris Loaded Strainer Head Loss - determined the debris bed head loss for the design basis accident.

    4. Particulate Debris Only Head Loss - determined the debris bed head loss for only particulate debris.

An inspection program has been developed to verify that the containment is free of debris that may lead to blockage of or damage to the ECCS sump. These inspections assure that the containment and sump are in the "as licensed" state of cleanliness prior to each reactor startup. Debris resulting from maintenance performed during normal operation will be carefully collected and removed from containment as part of standard operating procedures.

The sump inspection program includes an examination of sump structures, such as intakes and strainers, as outlined in Regulatory Guide 1.82 (R0).

6.2.2.5 Instrument Requirements

The instrumentation and controls associated with automatic initiation of the CSS and CCS during an accident are designed in accordance with IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations." Instrumentation and controls provide the operator with information for monitoring all modes of operation. "RUN" and "STOP" indication is provided for each spray pump motors, including "CLOSE" and "OPEN" position indication for each isolation valve. "RUN FAST", "RUN SLOW" and "STOP" indication is provided for each cooling fan.

Containment spray is automatically initiated by the CSAS. Each one of the redundant containment spray trains is started by the redundant CSAS generated by a coincidence of the SIAS and the high-high containment pressure signal, as described in Section 7.3. The system is initiated by starting the pumps and by opening the isolation valves FCV-07-1A and FCV-07-1B at the same time. Manual control of the system can be done from the control room.

The recirculation mode of operation is automatically initiated by a recirculation actuation signal as described in Section 7.3.

The containment fan coolers are normally started manually from the control room where the operator selects three out of four units to run at the higher of the two fan speeds during normal operation. As discussed in Subsection 9.4.8.4.1, three out of four Containment Fan Coolers are required to maintain reactor supports within design limitation. Four-position "STOP-AUTO-START-TEST" control switches are provided for each fan at the HVAC control board, in the control room. "TEST" positions of the control switch permits testing or running the fans at the low speed.

Component cooling water flows through all four cooling coils. Table 6.2-45 provides a list of Containment Heat Removal System instrumentation application. Table 6.2-46 shows the alarms which are not associated with any indicator or recorder but provides information to the operator, e.g., motor overload, valve fails to close or open etc.

6.2.3 SECONDARY CONTAINMENT FUNCTIONAL DESIGN

The secondary containment systems consist of (a) a secondary concrete enclosure (i.e., Shield Building) which surrounds the primary containment vessel and (b) a Shield Building Ventilation System (SBVS).

6.2.3.1 Design Bases

The design bases of secondary containment systems are to:

a) limit the pressure rise in the Shield Building annulus following a LOCA so as not to exceed the Shield Building internal design pressure, assuming a single active failure, and controlling and collecting leakage from the containment penetrations;

b) establish and maintain a subatmosphereic pressure in the Shield Building annulus following LOCA, assuming a single active failure, and to resist the maximum potential for exfiltration under all wind loading conditions characteristic of the site;

c) provide means of reducing offsite doses resulting from post accident leakage by evacuating air from the annulus and routing through the SBVS filtration units, assuming a single active failure;

d) provide fission product removal capability based on guidelines established for design basis accidents, a containment vessel design leak rate of 0.5 volume percent per day for the first 24 hours and 0.25 volume percent per day thereafter;

e) withstand safe shutdown earthquake loads and external/internal pressure differential of three psi without loss of function;

f) withstand post-accident environmental conditions within the Shield Building without loss of function;

g) provide means of reducing offsite doses by providing fission product removal capability while evacuating air from the Fuel Handling Building as described in Subsection 9.4.2, assuming a single active failure; and

h) permit appropriate periodic inspection and periodic pressure and functional testing to assure system integrity and functional capability.

Additional design bases for the Shield Building are given in Subsection 3.8.4.1.1.

6.2.3.2 System Design

6.2.3.2.1 Shield Building

The Shield Building is designed as a seismic Category I structure and its design features are further described in Subsection 3.84. Plan and elevation drawings are provided in Section 1.2.

The design and performance data for the Shield Building is provided in Table 6.2-47. Industry codes and standards and Regulatory Guides applied in the design of the Shield Building are listed in Subsection 3.8.4.2.

The Containment Isolation System is discussed in Subsection 6.2.4 and isolation valve design features are described in Subsection 6.2.4.3. The penetration types are defined, and their design features are discussed in Subsection 3.8.2.

The containment isolation actuation signal which activates the isolation valves are listed in Table 6.2-53, where containment penetrations and their isolation valves are listed according to penetration types and isolation class.

The Shield Building design objectives include the establishment of an essentially leaktight barrier against uncontrolled release of radioactivity. The Shield Building provides the annular space to consolidate the primary containment outleakage, and the SBVS maintains a negative pressure inside the annulus and filters for removal of fission products, and thus prevents the leakage from flowing directly from the annular space through the Shield Building structure to the atmosphere.

A fraction of the primary containment outleakage occurs through penetrations which includes isolation valves, seals, gaskets and welds as possible sources for potential bypass leakage paths. These penetrations pass through the primary containment steel vessel as well as the Shield Building and are identified as potential bypass leakage paths in Subsection 6.2.4.4.

An evaluation of potential bypass leakage paths considering realistic equipment design limitations and test sensitivities is provided in Subsection 6.2.4.3.

Isolation valves which are subject to potential leakage are tested according to "Type C" tests, and seals, gaskets and welds are tested according to "Type B" tests defined in Appendix J to 10 CFR 50. Tests of bypass leakage paths and for the determination of the bypass leakage fraction are provided in the Technical Specifications.

6.2.3.2.2 Shield Building Ventilation System

The Shield Building Ventilation System is shown on Figures 9.4-1 and 9.4-11 and consists of two full capacity redundant fan and filter subsystems which share a common Shield Building duct intake. Each filter subsystem consists of an electric heater, demisters, electric heater, HEPA prefilter, charcoal adsorber and HEPA afterfilters enclosed in a common casing. Each of these components are discussed in Subsection 6.5.1.

Table 6.2-48 contains the design data and materials of construction for Shield Building Ventilation System components, including charcoal filters. Subsection 6.5.1 contains a comparison of the Shield Building Ventilation System design to Regulatory Guide 1.52 requirements. The SBVS fan performance curve is shown on Figure 6.2-52.

The Shield Building Ventilation System annulus air intake consists of a ring duct with inlets at approximate Elevation 62 ft. located in each quadrant and at the top of the Shield Building. Two separate 30 inch diameter lines from the ring duct penetrate the Shield Building walls to connect to their corresponding filter subsystems. The fan and filter subsystems are located in the Reactor Auxiliary Building (Elevation 43 ft). Outside air lines, 26 x 26 inches and each isolated by a check valve and a motor operated butterfly valve in series, are connected to the intake of the filter subsystems to provide cooling air to the filters when required. A 12 inch line with an isolating butterfly valve cross connects the filter subsystems downstream of the filter hanks and upstream of the fans to maintain flow through the inoperative filters in the event of failure of a fan. A gravity damper is located at the discharge of each fan to prevent loss of capacity of an operating fan due to recirculation through an inactive subsystem.

Following a LOCA, the resultant pressure and temperature expansion of the containment vessel and the heat transfer through the vessel walls cause a decrease in the Shield Building volume and an increase in pressure. The annulus pressure is rapidly drawn down by operation of the Shield Building Ventilation System. The motorized dampers downstream of the fans are normally open. Upon receipt of a CIAS, the fans are in full operation in 10 seconds assuming offsite power is available. In the case of a loss of offsite power, the fans receive a CIAS to start but are not actuated until they are loaded on to the diesel generators. Therefore, assuming a coincident loss of offsite power, the fans are in full operation in approximately 26 seconds after start of the diesel generators. The Shield Building annulus pressure transient analysis, provided in Subsection 6.2.3.3.1, assumes that following a postulated LOCA, there is a free venting period followed by fan start-up. Once in full operation, the fan initial exhaust rate is approximately 7500 cfm. Within 310 sec. after a LOCA, the Shield Building annulus pressure is below atmospheric. The fan continues exhausting at a decreasing rate until the pressure in the Shield Building is two in. wg negative with respect to atmospheric as sensed by a pressure differential transmitter. At this point, the motorized damper at the discharge of the fan closes to a preset position to throttle air flow to the continuous rated system flow of 6000 cfm. As the Shield Building annulus becomes evacuated and the heat transfer rate from the containment stabilizes, the amount of outflow from the annulus is essentially balanced by the Shield Building inleakage. At a differential pressure of 1.75 in. wg negative, the amount of inleakage is less than 100 cfm. In order to provide adequate cooling air flow through the filters, additional air is taken from the outside atmosphere through a makeup cooling air line located outside the annulus upstream of the filter train. The outside air makeup line motor operated butterfly valve is opened automatically when the annulus differential pressure reaches one inch wg negative. The check valve in the cooling line is designed to have a pressure drop of not more than 3.0 inch wg and to open at 1.0 inch wg negative to provide vacuum control in the system and to allow outside air to cool the filters.

The SBVS is also interconnected to the spent fuel pool area exhaust duct. Upon receipt of a high radiation signal in the fuel pool area, the exhaust air is directed to the SBVS filtration units. The motor operated butterfly valves FCV-25-30 and 31 open and the exhaust fans start automatically. Motor operated valves FCV-25-32 and 33 close to isolate the annulus. Although a fuel handling accident inside the FHB concurrent with a LOCA is not considered a design basis event, a CIAS overrides the Fuel Handling Building high radiation signal and initiates the depressurization of the Shield Building annulus. The Fuel Handling Building Ventilation System is further discussed in Subsection 9.4.2.

Each of the SBVS intake trains is also connected to the Continuous Containment/Hydrogen Purge System. This connection, manually initiated from the control room, provides hydrogen purge capability while minimizing offsite radiological consequences. The Continuous Containment/Hydrogen Purge System description is provided in Subsection 9.4.8.8.

Both SBVS subsystems are automatically started by a CIAS or high radiation signal from the Fuel Handling Building. One can be manually shut down and placed in the standby mode. The standby subsystem automatically restarts if the operating subsystem should fall. The cross connection valve is opened from the control room to assure air flow through the failed system. Detectors in the charcoal beds annunciate temperatures exceeding 200°F.

The design basis review to determine the maximum expected differential pressure has been completed for motor-operated butterfly valves FCV-25-32 and FCV-25-33 in response to NRC Generic Letter 89-10, Safety-Related Motor-Operated Valve Testing and Surveillance. FCV-25-29 and FCV-25-34 were removed from the GL 89-10 program.

6.2.3.3 Design Evaluation

6.2.3.3.1 Performance Requirements and Capabilities

Each of the two full capacity fan-filter trains of the Shield Building Ventilation System, along with the Shield Building, are designed to fulfill the performance requirements stated in the design bases in Subsection 6.2.3.1.

As a part of the original plant design bases, the analysis of the functional capability of the SBVS to depressurize and maintain a uniform negative pressure within the Shield Building annulus was performed for the 9.82 square foot double ended suction leg slot break LOCA using the WATEMPT computer code described in Appendix 6.2B (Historical). For the EPU condition, computer code GOTHIC is used to determine whether the SBVS meets the design criteria for the limiting LOCA case (DEHLS break, min SI, 1 CS, 2 CFCs). The description of the development of the pipe break mass and energy release rate and the containment initial conditions are contained in Subsection 6.2.1. Any additional initial conditions or changes from those listed in Subsection 6.2.1 are contained in Table 6.2-49.

Assumptions used in the analysis include the following:

a) Credit is not taken for heat transfer from the secondary containment structure to the environs.

b) Secondary containment leakage is considered into and out of the Shield Building.

As shown on Figure 3.8-9, containment penetrations for the Containment Purge System (P-10 and P-11) each contain a 3/4 inch bleed-off into the Shield Building annulus. The purpose of these bleed-offs is that after an accident inside containment that results in pressurization and subsequent containment isolation, any valve seat leakage through the first two isolation valves (see Figure 9.4-9) bleed-off to the annulus volume and protect the penetration assembly through the concrete Shield Building from exceeding its design pressure. Due to the limited amount of expected valve seat leakage (284 cc/hr at 45 psig for both valves) and the small size of these bleed-off openings, this leakage contribution is considered insignificant and therefore not factored into the annulus pressure transient analysis. Even considering a single active failure of the outboard isolation valve to close, the leakage from the outside environs through one 3/4 inch bleed-off as a result of a differential pressure (atmosphere and negative pressure in the annulus) is insignificant.

Following the postulated LOCA, the annulus is quickly pressurized due to the volume reduction resulting from the sudden expansion of the containment vessel caused by the pressure and temperature transient resulting from the blowdown. The annulus is further pressurized as the gases within the annulus are gradually heated up due to convection and radiation heat transfer through the steel wall of the containment vessel.

The containment pressurization, heating, and subsequent volume expansion was not implicitly modeled for the EPU assessment. Instead, the limiting LOCA EPU containment pressure and temperature profiles were treated as boundary conditions and the annulus volume reduction effect due to containment volume expansion was explicitly modeled based on the peak pre-EPU containment pressure for the DESLS break shown in Figure 6.2-53 (historical data). The annulus volume expansion effect is also explicitly modeled, based on the EPU containment pressure and temperature for the limiting LOCA case (DEHLS break, min SI, 1 CS, 2 CFCs).

The 4 times Tagami heat transfer coefficient from the containment vapor space to the vessel surfaces is applied. This approach is conservative since the containment pressure and temperature profiles calculated for the containment analysis are applied as boundary condition for the annulus pressurization. In reality, the containment pressure and temperature will be significantly decreased if the maximum Tagami condensation heat transfer coefficient is increased by 4 times. The containment vessel (steel) is heated up mainly due to the condensing heat transfer from the containment vapor space to the vessel surfaces.

Radiation heat transfer via the steel containment vessel to the annulus gas space and to the shield building concrete wall and ceiling were considered in the analysis. Natural convective heat transfer was also applied between the containment vessel outer surface and the annulus gas space, and between the annulus gas space and the shield building concrete wall and the inner surfaces of the ceiling. Heat transfer due to conduction within the containment vessel and the shield building concrete wall and ceiling was also considered.

The shield building ventilation system (SBVS) fan flow rate transient used in this analysis was conservatively based on fan performance assuming operation of the emergency diesel generator at 1% under frequency conditions.

The analysis results are as follows:

a) Pre-EPU containment pressure and temperature as function of time (Figures 6.2-53 and 56) (Historical)

b) Annulus pressure and temperature as function of time (Figures 6.2-54 and 56A)

c) Condensing Heat Transfer Coefficient as a function of time (Figure 6.2-57)

d) SBVS purge flow rate as a function of fan static pressure (Figure 6.2-52)

From the results above, it is demonstrated that each of the Shield Building Ventilation System centrifugal fans is adequately designed to independently decrease annulus pressure to slightly subatmospheric after a LOCA. Due to the subatmospheric annulus pressure the post-accident activity release from the containment vessel is routed through the filter train provided and not through the Shield Building as leakage.

Air flow due to wind around and over a building creates regions in which the static pressure is either above or below the static pressure in the undisturbed air stream. In general, pressures are positive on the windward side and negative on the leeward side. Pressures on the remaining sides may be negative or positive depending on the angle of the wind. The velocity head equivalent to a given wind speed can be expressed as: (17)

$$P_2 = 0.000482\ V^2_w$$

where P_2 = Velocity Head, inch wg

V_w = Wind Velocity, mph

As indicated in Table 2.3-11, the maximum daily average windspeed at the St. Lucie Site is 18 mph (8.1 m/sec). The corresponding velocity head is calculated to be -0.156 inch wg. According to Reference 17, surfaces when curved may act as airfoils, resulting in negative pressures two or more times the velocity head of the free air stream. Since the Shield Building is a curved surface, the velocity head can be assumed to be -0.468 inch wg. Since the Shield Building annulus is maintained at -2.0 inch wg., no exfiltration is expected through the concrete wall.

The SBVS heaters have the capacity to reduce air from 100 percent to 70 percent humidity at a 6000 cfm flow rate. Each moisture separator demister has an efficiency of 99 percent when exposed to entrained water particles of one to five microns in size. The 12 HEPA filters per train have an efficiency of at least 99.97 percent when tested with 0.3 micron dioctylphthalate (DOP) smoke. HEPA filter assemblies are constructed of materials capable of withstanding a temperature of 200°F. The charcoal adsorbers are potassium iodine (KI) impregnated and have the radioiodine removal efficiency of 99 percent minimum in accordance with ANSI N509-1976. The relative humidity of air entering the charcoal adsorber is below 70 percent because each filter train is provided with a 30 kw electric heating coil to assure that the relative humidity remains below 70 percent. A second heater is provided, with power from an alternate source should power to a train be lost, to assure 70 percent humidity of the cooling air, for removal of decay heat.

The amount of decay heat (refer to Figure 6.2-60) is very small when compared to the mass and large surface area of the charcoal adsorber and filter housing. Convection and conduction to ambient atmosphere can easily remove this decay heat without any cooling air. To enhance the removal of decay heat, cooling air is allowed to pass through the inoperative filter train. Figure 6.2-61 shows the charcoal temperature as a function of cooling air flow.

An evaluation was performed to determine the impact of the Extended Power Uprate (EPU) with regard to decay heat generation and dissipation. The above analysis identifies that the amount of decay heat is small when compared to the mass and large surface area of the charcoal adsorber and filter housing. UFSAR Figure 6.2-61 shows the charcoal temperature as a function of cooling air flow (no heat transfer assumed through the housing) and it is shown that with a flow rate as low as 100 cfm the temperature rise of the air at the time of maximum decay heat following a DBA would be 25°F at the minimum design flow rate of 300 cfm the temperature rise would only be approximately 8°F.

The existing margin between the currently predicted heat rise at a minimum cooling flow of 300 cfm (i.e., -8°F) and the alarm setpoint of 200°F is more than sufficient to accommodate the extended power uprate and the change in source terms from TID-14844 to alternate source term (AST). The application of AST reduced iodine inventory and associated heat load in the charcoal filters to less than that predicted to have been accumulated in the TID-14844 design basis analysis. Although the extended power uprate will increase the source term, the resulting iodine inventory and associated heat load in the charcoal filters will still remain below the design basis TID-14844 values.

The heat contribution to the charcoal media due gamma heating resulting from the particulate inventory accumulated in the HEPA filters is not significant since the associated gamma heating rate in the charcoal is less than that generated by the iodine in the charcoal during the first 30 days following the DBA when the decay heat rate is the highest.

Each filter train is monitored for temperature by redundant sensors as indicated in Subsection 6.2.3.5. Since charcoal adsorber temperature is alarmed and recorded in the control room and, since capability exists for utilizing the redundant filter train, and since cooling air is available for the shutdown filter train, even when any single active failure is assumed, the charcoal filter train temperature does not approach the desorption or combustion temperature. Thus, the SBVS design provides satisfactory cooling of the adsorber beds. The filter bed loading for a fuel handling accident would be less severe than for the above analysis.

6.2.3.3.2 Single Failure Analysis

Two redundant subsystems are provided, either of which is capable of meeting the requirements of the system design bases. Each subsystem is actuated by a separate CIAS actuation channel. As explained in Section 8.3, all redundant active components are powered from separate emergency bases. Consequently, the Shield Building Ventilation System design function is not compromised by any single failure. A failure mode and effect analyses is shown in Table 6.2-50.

An analysis is conducted to determine if a single active failure could result in an air flow through a recently shutdown filter train of less than the flow required to maintain a 200°F adsorber temperature. For any single active failure, a system air flow balance indicates that in the worst case at least 300 cfm of cooling air is drawn through the he inoperative train by means of a cross-connection to the remaining fan/ filter train. Assuming failure of the operating train of the Shield Building Ventilation System, train A for example, approximately 5700 cfm of air would be drawn through the train B air intake when the train B fan is automatically actuated. Of the 6000 cfm, a system air balance analysis demonstrates that approximately 300 cfm is drawn through the inoperable train A filters. This is accomplished by the normally open (control room operated) cross-connect valve (FCV-25-13) and ductwork which is located downstream of the filters and upstream of the fans. The cooling air from train A then joins the train B air stream prior to discharge by fan HVE-6B.

Therefore, the analysis shows that adequate air flow is available for filter cooling even when the minimum cooing flow requirements are based on upper limit calculations. Also a small heater, provided in the stand-by train, is powered from an operating train power source to assure humidity control of the cooling air in the stand-by train.

The single failure criterion is not applied to HVAC ductwork serving redundant safety related equipment. No ductwork operates at pressures greater than 22 inch wg (about 3/4 psi) so that rupture of the ductwork could not occur due to the pressure of the contained gases.

Common ductwork serving safety related equipment occurs at the following,:

a) Duct which cross-connects the two SPVS filter trains,

b) Annulus ring duct,

c) Exhaust riser from the top of dome connecting the two intake ring headers, and

d) Ductwork that connects fans to plant vent

In each case the ducting is located in areas where there is no high pressure piping that could result in pipe whip or jet impingement. Similarly the ducting is not located in areas where internally generated missiles or failure of non-seismic structures or components could affect system operation. Leakage from cracks in the ductwork do not prevent the vent system from performing its safety function. There is no high energy pipes that run inside the Shield Building annulus. All high energy piping passing through both the primary containment vessel and the Shield Building wall is provided with guard pipes.

6.2.3.4 Tests and Inspections

An overall system performance test of the Shield Building and SBVS is described in Section 14.0.

Periodic integrated functional testing of the Shield Building/SBVS and FHB/SBVS are performed in accordance with the Technical Specifications. Periodic testing of HVAC isolation valve operability is also performed.

Testing of the SBVS fans and filters is discussed in Subsection 6.5.1.

6.2.3.5 Instrumentation Requirements

Each redundant SBVS is provided with instrument and controls designed for proper functioning of SBVS in order to meet the requirement stated in the design basis in Subsection 6.2.3.1. The application of instrumentation is shown in Table 6.2-51.

A CIAS or high radiation signal from the FHB starts both exhaust fans HVE-6A and 6B. (The SBVS response for high radiation in the FHB is further discussed in Subsection 9.4.2.) Upon receipt of one in. wg. in negative differential pressure signal from annulus (PDS-25-7A & 7B) the outside cooling air inlet valves (FCV-25-11 and 12) automatically open. Normally the exhaust fan dampers (D-23 and D-24) are open. Upon receipt of two in. wg. negative differential pressure signal from PDS-25-7A, 7B the dampers partially close to reduce the air flow. Each damper is also controlled by the exhaust air flow, through a flow transmitter and controller (FT-25-20 and FIC-25-20, respectively). Air flow below the normal set point increases the opening of the damper. Air flow above the set point reduces the opening of the damper. Hence the air flow is maintained at preset value.

Operator intervention is required to place one train on a standby mode when both trains are started automatically. The standby fan is restarted automatically by a low flow signal from the discharger line of the operating fan.

The 30 kW electric heater is interlocked with the fan of the same filter train. The 1.5 kW electric heater is interlocked with the other fan to limit relative humidity of cooling air to a maximum of 70 percent for decay heat removal).

6.2.4 CONTAINMENT ISOLATION SYSTEM

The containment isolation system provides the means of isolating fluid systems that pass through containment penetrations such that any radioactivity that may be released into the containment atmosphere following a postulated design basis accident (DBA) is confined. There is no one particular system for complete containment isolation, but isolation design is provided applying acceptance criteria common to penetrations in many different fluid systems.

6.2.4.1 Design Bases

The design bases governing the containment isolation system are discussed below.

The containment isolation valves are designated seismic Category I and designed to ASME Code, Section III and Quality Croup B requirements. Containment isolation valves are designed to ensure leak-tightness and reliability of operation. Containment isolation globe, check and gate valves meet the requirements of manufacturers standards MSS-SP-61, "Hydrostatic Testing of Steel Valves," and containment isolation butterfly valves meet the requirements of manufacturers standards MSS-SP-67, "Butterfly Valves."

6.2.4.1.1 Conditions Requiring Containment Isolation

a) Automatic initiation of a containment isolation actuation signal (CIAS) occurs when a high containment pressure setpoint or a high containment radiation level setpoint is detected, or when a Safety Injection Actuation Signal is initiated. This provides diversity of parameters sensed for the initiation of containment isolation.

b) The CIAS closes fluid line penetration isolation valves not required for operation of the Engineered Safety Features.

c) The containment isolation system is designed such that no single active failure (in conjunction with loss of offsite power) could result in offsite doses or doses to operators in the control room in excess of guidelines established for design basis accidents.

d) The main steam and feedwater valves close on MSIS and the valves for the component cooling water for the reactor coolant pump motors close on SIAS (see Section 7.3 and Subsection 6.2.4.3.2).

6.2.4.1.2 Criteria for Isolation of Fluid System Penetrating the Containment

a) The containment isolation provisions for the fluid system penetrations (excluding the ESF systems) are designed in accordance with General Design Criteria 54, 55, 56 and 57 (refer to Table 6.2-52). Exceptions to GDC provisions are discussed in Subsection 6.2.4.3.

b) Containment isolation valves on lines serving engineered safety feature systems are not closed automatically by the CIAS, may be closed by remote manual operation from the control room.

c) The piping penetration assemblies are designed to withstand at least a pressure and temperature equal to the containment vessel design internal pressure and temperature and to withstand the post accident transient environment. Two barriers are provided between the containment atmosphere and the outside atmosphere, so that failure of one valve to close does not prevent isolation. The design of containment penetrations accommodate thermal pressurization concerns due to environmental heating of trapped fluids (Reference NRC Generic Letter 96-06).

6.2.4.1.3 Criteria for Isolation of Fluid Instrument Lines

The containment isolation provision for the fluid system instrument line penetrating the containment conform to the Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment," March 1971 (R0). The instrument tubing and containment penetrations 45 and 53 are designed to be ASME Class 2; the shield building penetrations 45B and 53B are designed to be ASME Class 3.

6.2.4.1.4 Design Requirement for Containment Isolation Barrier

a) Penetration assemblies and containment isolation valves are designed as seismic Category I in accordance with Regulatory Guide 1.29, "Seismic Design Classification," February, 1976 (R2) and are designed to Quality Group B in accordance with Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," February, 1976 (R3) as delineated in Section 3.2.

b) Provisions for Type B and C leak testing of the penetrations are provided utilizing the guidance of 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," and/or Appendix J, Option B performance based requirements.

c) Containment isolation valves are designed to ensure leak tightness in accordance with manufacturer standards (see Subsection 6.2.4.3).

d) The containment isolation system is able to perform its design function following any natural phenomenon.

e) The containment isolation system is designed to withstand pipe rupture effects.

6.2.4.2 <u>System Design</u>

Table 6.2-52 provides the design information regarding the containment isolation provisions for the fluid system lines and fluid instrument lines penetrating the containment.

Fluid lines penetrating the containment are classified according to their function during normal and post-accident operation. Various combinations of valving and pressure boundary barriers are used to establish isolation of each penetration class. The penetration classes correspond to GDC 55, 56 and 57 and Regulatory Guide 1.11 (R0). In each of the penetration classes at least two barriers are provided between the containment atmosphere and the outside atmosphere so that no single failure can prevent containment isolation. The explicit mode of compliance with these requirements is illustrated below:

a) GDC 56 "Primary Containment Isolation"

GDC 56 delineates containment isolation provisions for lines that connect directly to the containment atmosphere and penetrate containment vessel. The St. Lucie Unit 2 design complies with GDC 56 as follows:

Penetrations in Class A1 are for lines through which fluid normally flows or may flow during power operation, which are not connected to the reactor coolant pressure boundary (RCPB), and are either:

1) open directly to the containment atmosphere and connected to non-seismic piping or duct work outside the containment; or

2) connected to non-seismic piping on both sides of the containment.

These lines have two isolation valves in series, one located inside and one located outside containment, which close automatically when containment isolation is required. Check valves are considered automatic only when located inside containment. Class A1 valves except check valves can be closed from the control room.

Class A2

Penetrations in Class A2 are for lines which are normally closed and not opened during power operation, which are not connected to the reactor coolant pressure boundary, and are either:

1) open directly to the containment atmosphere and connected to non-seismic piping outside the containment; or

2) connected to non-seismic piping on both sides of the containment

Class A2 Lines are provided with either:

1) two manual isolation valves (normally closed) in series; or

2) one manual isolation valve (normally closed) and one blind flange in series; or

3) one manual isolation valve (normally closed) and check valve in series

4) two automatic isolation valves in series.

In each of the above cases, one isolation valve is located inside containment and one isolation valve is located outside containment. A check valve is considered as an automatic isolation valve only when located inside containment.

A lock is provided on each normally closed valve and administrative procedures ensure that the valve is not left open after maintenance or inadvertently opened during power operation.

b) GDC 55 "Reactor Coolant Pressure Boundary Penetrating Containment"

GDC 55 provides containment isolation provisions for lines that are part of the RCPB and that penetrates primary reactor containment. The design complies with GDC 55 as follows:

Class B1

Penetrations in this class are for lines which are connected to the Reactor Coolant System and through which fluid normally flows or may flow during power operation. Two valves in series, one located inside and one located outside containment, are provided which close automatically when containment isolation is required. Class B1 valves, except check valves also can be closed from the control room. The charging line (penetration 27) is discussed in Subsection 6.2.4.3.

Class B2

Penetrations in this class are for lines which are connected to the Reactor Coolant System and are normally closed during power operation. These lines have two manual valves or two valves capable of remote manual operation in series, one located inside and one located outside containment, which are provided with locks to ensure that a valve is not left open after maintenance or inadvertently opened during power operation.

c) GDC 57 "Closed System Isolation Valves"

GDC 57 provides containment isolation provisions for lines that penetrate the containment vessel and are neither part of the RCPB nor connected directly to the containment atmosphere. The design complies with the requirements of GDC 57 as follows:

Class C

Penetrations in this class are for lines which are neither connected to the Reactor Coolant System nor open to the containment atmosphere but connected to a closed seismic Category I system inside containment. These lines are provided with at least one containment isolation valve outside the containment which is either manual locked closed, automatic, or capable of remote manual operation.

d) Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment (R0)"

Regulatory Guide 1.11 (R0) specifies containment isolation provisions for fluid instrument lines penetrating primary containment. The design conforms to requirements of Regulatory Guide 1.11 (R0) as follows:

Class D

Penetrations in this class are for fluid instrument sensing lines that are connected to the Reactor Coolant System or connected directly to the containment atmosphere. These lines are provided with one isolation valve capable of automatic or remote manual operation located outside the containment and as close to the containment as practical. A self actuated excess flow check valve is considered an automatically actuated valve.

Fluid instrument lines penetrating the containment are in complete conformance with Regulatory Guide 1.11 (R0) and are isolated by means of a self actuated excess flow check valve or one remote manual valve located outside containment.

e) Systems Performing an ESF Function

Class E

Penetrations in this class are for lines that are designed to be open following a DBA to mitigate the effect of the accident. These lines are provided with either (1) automatic valve located inside containment in series with one remote manually actuated valve or (2) one remote manually actuated valve or one check valve located inside containment and a remote manual valve located outside containment.

The only line that utilizes only one isolation valve outside containment and no isolation valve inside is the containment sump suction (Penetrations 32 and 33). It should be noted that the reliability of this system is increased with this arrangement. Further the system is closed outside containment and any single active failure can be accommodated. The closed system outside containment is seismic Category I, Quality Group B and has a design temperature and pressure rating greater than that of the containment. The outside line is always maintained full of water and hence no leak testing is required.

The emergency sump suction penetrations process lines are enclosed in a leak-tight housing, (i.e., carbon steel guard pipes) which extend from the sump inside containment to the Containment Isolation Valve located outside containment. Each guard pipe is directly welded to a steel containment vessel nozzle and acts as an extension of the containment in both directions. Passing through each guard pipe is the stainless steel sump suction line. These lines are welded to the guard pipe in the sump so that water cannot enter the

the annulus formed by the concentric pipes. Outside containment the suction lines are sealed to the guard pipes by means of a stainless steel bellows to allow for thermal movement. Figure 3.8-6 provides a detailed description of this type IV penetrations. The containment isolation valves are located in the Reactor Auxiliary Building pipe tunnel which is a controlled leakage area. Leakages from these systems are directed to the ECCS room sump which is provided with safety grade, seismic Category I level indications. A backup seismic Category I level indicator is also provided in each ECCS room sump to alert the operator of any abnormal condition. The ECCS area is also provided with two safety-related radiation monitors to measure the airborne effluent. A description of these monitors is provided in Subsection 11.5.2.2.10.

6.2.4.2.1 System Design Requirements

Design requirements for the containment isolation barriers are imposed as follows:

a) Containment isolation valve closure times are selected to assure rapid closure of the containment following postulated accidents.

Valve closure times for all containment isolation valves are provided in Table 6.2-53.

b) The containment isolation valves are designed in compliance with Regulatory Guides 1.26 (R3) and 1.29 (R2).

c) Containment isolation valves, actuators and controls are located in areas which are enclosed. These penetration area enclosures are also designed to withstand the missiles postulated in Section 3.5. The piping in the penetration area is designed such that the failure of any pipe will not cause pipe whip damage to other piping nearby. This design concept assures that the failure of a nonseismic pipe will not result in loss of containment isolation. Under accident conditions the penetration area is continuously vented by the Emergency Core Cooling Area Ventilation System (Subsection 9.4.3) such that the service operating environment will not be detrimental to the equipment. Isolation valves are located above the maximum water level in the containment thereby preventing flooding and any debris from inhibiting the isolation function of the valve. The only exception is valve V6341 Reactor Drain Tank (RDT) Isolation Valve. This valve is located in the RDT pit and will be submerged during a LOCA. The operability of this valve has been analyzed and it has been determined that its containment isolation capabilities will not be compromised by being submerged.

The fuel transfer tube has a double gasketed blind flange on the containment side permitting pressure leak testing between the gaskets, and a gate valve at the end inside the Fuel Handling Building. The transfer tube and attachments are designed to withstand the forces resulting from the safe shutdown earthquake. The double gasketed blind flange serves as the primary containment seal.

d) In general, the design utilizes air-operated "fail-safe" containment isolation valves. There are also active automatically actuated isolation valves located inside the containment which are required to perform an isolation function in the event of a DBA. A list of active valves is provided in Subsection 3.9.3. Assurance of the operability of the valve is discussed in Subsection 3.9.3.

e) The main function of containment isolation is to minimize the radiological offsite doses. Therefore, the main qualification of a valve as an isolation barrier is its leak prevention capabilities. Applicable containment isolation valves are designed and factory tested so that the maximum allowable leakage does not exceed 10 cc of water per hour per inch of nominal valve diameter when subjected to containment maximum internal pressure (see Subsection 6.2.4.3).

Leakage testing of isolation valves is described in Subsection 6.2.6. Design provisions are provided to perform the required 10 CFR 50 Appendix J and/or Appendix J, Option B, Type C, containment isolation valve testing. The system schematic drawings depicting the various design provisions for leak testing of the containment isolation valves are provided in Subsection 6.2.6.

Specific information on isolation valve testing, bypass leakage, penetration testing and acceptance criteria is presented in Subsection 6.2.6.

f) Wherever two automatic isolation valves are in series, each valve operator is actuated from a separate and redundant CIAS channel, A or B, with power supplied to each valve from a separate emergency bus, A or B, as explained in Section 8.3. Table 6.2-53 lists the CIAS actuation channel for redundant isolation valves.

Valves isolating penetrations serving engineered safety features systems will not automatically close with the CIAS, but can be closed by remote manual operation from the control room to isolate any engineered safety features system that may malfunction.

g) Secondary mode of operation of isolation valves required to change position upon initiation of CIAS are listed in Table 6.2-53.

h) Fluid systems that have a post-accident safety function are provided with remote-manual switches in the control room. Flow measuring devices and other instrumentation are provided for the operator to know when to isolate the fluid system. The ESF instrumentation and display are discussed in Sections 7.3 and 7.5 respectively.

The Engineered Safety Features' areas are provided with sumps and corresponding pumps which are automatically initiated on high water levels with annunciations in the control room. This design feature provides the operator with the information required to isolate an ESF train should leakage occur.

i) The containment isolation penetrations that utilize water filled seal systems are: containment sump suction (Penetrations 32 and 33) containment spray (Penetrations 34 and 35), safety injection (Penetrations 36 to 39), shutdown cooling (Penetrations 40 and 64) and hot leg injection (Penetrations 69 and 70). These systems form closed seismic Category I systems outside containment and perform ESF functions post LOCA. They are designed to Quality Group B requirements and are protected from the dynamic effects of pipe rupture to ensure a fluid seal under all conditions. The closed systems inside the containment are designed as seismic Category I and Quality Group B and are considered as an extension of the containment. These penetrations are not subjected to type C or bypass leakage testing.

j) The environmental qualification tests for mechanical and electrical components exposed to accident environment inside the containment are described in Section 3.11.

6.2.4.3 Design Evaluation

As indicated in Table 6.2-52 the systems conforming to GDCs 55 and 56 are provided with two containment isolation valves, one inside and the other outside the containment. The containment penetrations which are required to close after an accident are provided with automatic isolation valves which close on CIAS. Whenever two automatic isolation valves are in series, as shown in Table 6.2-53, each valve operator is actuated from a separate and redundant CIAS channel, with power supplied to each valve from a separate safety-related bus, as explained in Section 8.3. This arrangement ensures containment isolation in the event of a single electrical failure.

The GDC 57 penetrations as depicted in Tables 6.2-52 and 53 are provided with at least one isolation valve outside the containment. A single outside valve does not jeopardize the containment isolation because the system inside the containment is closed, seismic Category I, Quality Group B and is considered as an extension of the containment.

Various systems perform post-accident functions and thus the valving arrangement must permit flow to the required areas (e.g., reactor, containment; etc). These penetration provisions are consistent with NRC guidelines in that they provide for the safe shutdown of the plant and aid to mitigate the consequences of an accident. The containment isolation provisions for lines required for post-accident or ESF related systems are as follows:

1) One automatic valve inside containment and one remote-manual valve outside containment. The systems that utilize this concept are:

   (a) Containment Spray System - Penetrations 34 and 35
   (b) Safety Injection System -Penetrations 36 to 39

   To comply with a regulatory staff request, the following isolation valve changes are made:

   Penetration 27 (chemical and volume control) - a check valve is added inside containment in line 2-CH-331 (charging line) as close to the containment as practicable.

   Penetration 36 to 39 (safety injection) - Check valves are added in these lines inside the containment as close as practicable to the containment boundary.

2) One remote-manual valve inside the containment and one remote-manual valve outside containment.

   The only ESF system that utilizes this concept is the H_2 sample system (Penetrations 48 and 51).

The isolation valves which are required to close after an accident are designed with a rapid closure time as indicated in Table 6.2-53. The Continuous Containment/Hydrogen Purge System is designed with a rapid (5 seconds) isolation valve closure time.

Table 6.2-53 also indicates the primary and secondary modes of valve closure, shutdown, post accident valve position and the valve position and the valve failure position. A case by case evaluation of this table confirms that if the primary mode of closure fails the valve could still be closed, if required, with an acceptable time frame by the secondary mode of closure.

The component cooling water isolation valves (Penetrations 23 and 24) do not close on a CIAS but do not jeopardize containment isolation. The component cooling water isolation valves are closed by SIAS. Component cooling water to the reactor coolant pumps and motors and CEDM coolers is interrupted by closure of any one of these isolation valves and inadvertent loss of the cooling water would require plant shutdown. The use of the SIAS to close the isolation valves does not compromise containment isolation since the same containment pressure setpoint (3.5 psig) is used for generating the CIAS and SIAS and affords additional protection for the reactor coolant pumps since the high containment pressure also trips the reactor. Furthermore, at NRC request, a delayed reactor trip is installed which trips the reactor upon loss of CCW flow to the reactor coolant pumps as discussed in Subsection 9.2.2.

There are no lines carrying radioactive water outside the containment (except the ESF systems required post-LOCA) which could be open before an accident and which do not close on a CIAS (e.g., reactor coolant drain tank pump suction line closes on a CIAS).

The motor operated valves fail in the "as is" position. These valves assume the required status following a DBA; thus containment isolation is not affected if ac power is lost temporarily.

As a minimum, the containment isolation valves are designed and factory tested so that the maximum allowable leakage does not exceed 1/10 of a standard cubic feet of air per hour per inch of nominal valve diameter when subjected to containment maximum internal pressure. The offsite doses are discussed in Chapter 15. The surveillance requirements for containment isolation valves are a part of the Technical Specifications.

6.2.4.4 Tests and Inspections

The program for the initial and subsequent functional testing for both the 10 CFR 50, Appendix J, Type C valves and valves that qualify as a source of bypass leakage is provided in Subsection 6.2.6 and Chapter 14 (Note that Technical Specification, Amendment 88, allows Appendix J, Option B testing. This testing allows the licensee to establish intervals for leak-testing based on the performance history and risk significance of the components (containment, penetrations, and valves).

The Containment Leakage Rate Testing Program includes those penetration isolation valves considered as possible sources of bypass leakage and Type C testing. Isolation valves not considered as a possible source are listed below.

a) Penetrations that form a closed system inside containment in accordance with GDC 57 are not subject to bypass leakage or Type C testing. These systems are designed seismic Category I, Quality Group B systems inside containment and designed for a higher pressure than the containment design pressure. These systems are also protected from the dynamic effects of pipe rupture as outlined in Section 3.6. The penetrations that utilize this concept are as follows:

Penetrations	1,2	- Main Steam
	3,4	- Feedwater
	5,6	- Steam Generator Blowdown
	30,49	- Steam Generator Blowdown Sampling
	15-22	- CC Water to Containment Fan Coolers

During accident conditions, the Main Steam and Feedwater Isolation valves will close upon receipt of a MSIS (high containment pressure or low steam generator pressure) while the Steam Generator Blowdown and Blowdown Sampling Isolation valves will close upon receipt of a CIAS (high containment pressure or high containment radiation or SIAS). Figure 6.2-2a provides the containment pressure response for the worst break scenario and illustrates that the containment atmosphere rises to a peak of 43.43 psig and reduces to less than half the peak pressure within the first day post-accident. Therefore, the steam generator inventory that existed prior to the accident will be available post-accident and will act as a steam seal at the onset of the accident. Subsequent to the decay of the steam generator pressure and level, the Auxiliary Feedwater System will automatically maintain the steam generator level to guarantee the pressure of a water seal and thereby preclude bypass leakage.

b) Penetrations that are used post-accident or for ESF - related function are not subject to bypass leakage or type testing. These lines form a closed seismic Category I system outside containment. They are designed as seismic Category I, Quality Group B systems and are protected from the dynamic effects of the pipe rupture as outlined in Section 3.6.

These include:

Penetrations	32,33-	Containment Sump Suction
	34,35-	Containment Spray
	36-39-	Safety Injection
	40,64-	Shutdown Cooling
	69,70-	Hot leg injection

c) Various penetrations that are filtered by the Shield Building Ventilation System and thus are not subject to bypass leakage. The following valves follow this criteria:

Penetrations 10,11 - Containment Purge/Exhaust
67,68 - Containment Vacuum Relief

d) The penetrations for the instrument sensing lines which are isolated in accordance with Regulatory Guide 1.11 (R0) are connected to a closed seismic Category I piping outside containment and, therefore, not subject to bypass leakage or Type C testing.

e) The charging line (Penetration 27) is not considered a credible source of bypass leakage following a LOCA. Charging Pumps 2A and 2B are automatically started following receipt of a Safety Injection Actuation Signal (SIAS) and are powered by the emergency diesel generators. Thus, after an accident flow is directed into containment through this penetration precluding bypass leakage by establishing a water seal. If the pumps were not operating radioactive contaminants are prevented from reaching the environment by a minimum of three seismically qualified, Safety Class 2 check valves in series. These design features virtually eliminate any possibility of bypass leakage.

The criteria determining for which penetration isolation valves bypass leakage could occur are:

a) a pressure or water seal could not be guaranteed.

b) the bypass leakage could not be guaranteed to exit in a leakage control area (i.e., Shield Building annulus, ECCS pump area or piping penetration tunnel).

c) the lines are connected to neither a closed seismic Category I system inside nor outside of containment.

The criteria for determining which containment isolation valves require Type C testing is as follows:

a) Lines that provide a direct connection between the inside and outside atmospheres of the primary reactor containment during normal operation.

b) Lines that are required to close automatically upon receipt of a CIAS.

c) Lines that are required to operate intermittently under post-accident operations.

Specific information on isolation valve testing, bypass leakage, penetration testing and acceptance criteria is presented in Subsection 6.2.6.

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

Systems are provided in accordance with General Design Criterion 41 to control the concentration of hydrogen that may be released into containment following a LOCA. These systems are the containment hydrogen analyzers, the containment hydrogen recombiners and containment hydrogen purge.

The hydrogen build-up inside containment is controlled by the Containment Combustible Gas Control System. This system provides the capability to monitor and maintain hydrogen concentrations within safe limits after a LOCA and is made up of the following subsystems: containment hydrogen analyzers, containment hydrogen recombiners and containment hydrogen purge.

1

The stretch power impact on the post LOCA hydrogen build-up inside containment has been analyzed.

A number of changes have been incorporated into the current model, including an updated long-term temperature transient, increased radiolytic hydrogen due to the increased reactor power level, additional zinc-based paint, additional galvanized steel and additional aluminum ladders.

The analysis proved that the Containment Combustible Gas Control System is capable, without any modification, of preventing the hydrogen gas concentration inside containment from exceeding the lower flammability limit of four volume percent, which complies with the existing system design criteria.

6.2.5.1 Design Bases

The capability to uniformly mix the containment atmosphere and to prevent high concentration of combustible gases from forming locally is provided by:

a) forced circulation by the containment fan coolers and their associated ductwork

b) mixing of containment air by the Containment Spray System

c) the process of natural diffusion of combustible gas within the containment air as discussed in Subsection 6.2.5.3

The H_2 analyzers and H_2 recombiners comprise the redundant 100 percent capacity Combustible Gas Control System (CGCS). The design bases for these systems, which are Engineered Safety Features, are:

a) To permit the sampling and measurement of hydrogen concentration at several points within the containment, and annunciate in the control room if the hydrogen concentration is above three volume percent.

b) To prevent hydrogen gas concentration inside containment from exceeding the lower flammability limit of four volume percent.

c) To perform the above functions assuming a single failure.

d) To be designed to seismic Category I requirements and applicable Quality Group B criteria.

e) To withstand the dynamic affects associated with pipe rupture as discussed in Section 3.6.

f) To be operable when subjected to the post-accident environment as discussed in Section 3.11.

g) To provide for testing and inspection during normal plant operation.

The hydrogen recombiners are fixed systems inside the containment and there is no sharing of the Combustible Gas Control System between St. Lucie Units 1 and 2.

In addition to the redundant CGCS, the Continuous Containment Purge/hydrogen Purge System is available for fission product removal and hydrogen purge following a LOCA.

6.2.5.2 System Design

6.2.5.2.1 Containment Hydrogen Analyzer Subsystem

The Containment Hydrogen Analyzer System consists of two redundant subsystems as shown on Figure 12.3-14, consisting of the sample and return piping, associated valves, hydrogen analyzer, grab sample cylinder, sample pump, moisture separator, cooler, instruments, calibration gas line and reagent gas line.

Each of the redundant subsystems is physically separate and operates independently of the other, and is powered from an independent onsite power source. No single failure can result in a total loss of hydrogen concentration measurement capability. Failure of one train is annunciated in the control room.

Components of the system are accessible for periodic inspection and maintenance. The system is designed to permit remote calibration at periodic intervals with a reference hydrogen gas standard (span gas) and oxygen. The system is independent of any system used during normal plant operation so that plant operation does not impose restrictions on such testing.

The Hydrogen Analyzer System piping, from the sample points within the Containment and piping returning the sample to the Containment, up to and including all containment isolation valves, are designed and fabricated in accordance with ASME Section III Class 2 and N-stamped. The hydrogen analyzer package contains instrumentation elements which are inherently non-ASME, code items (e.g., flowmeters, pressure gages, and the analyzer element). Therefore, the hydrogen analyzer package is classified as a Class 1E instrument. Instrumentation, controls and electric equipment associated with the system will be Class 1E. Conformance to applicable IEEE Standards is discussed in Chapter 7.

The system is initiated by manual operator action from the control room. No action outside the control room is necessary for system operation.

Once initiated, the system draws a continuous air sample from one of the sample points inside containment. Sampling valves can be manually controlled to analyze any sample point. The air is passed through the detector, analyzed, and pumped back into containment. Analyzer readings are recorded in the control room, and an alarm is actuated if concentration is above three percent. Alarm is also provided for low flow and low temperature in the analyzer hot box. Design and performance date for the analyzer is listed in Table 6.2-54.

The system is designed and the components are qualified to operate under the applicable environmental conditions as described in Section 3.11.

The operating principle of the hydrogen analyzer is thermal conductivity of the sample. Air samples are drawn from any of the following sample points

a) Containment dome

b) Upper Containment

c) Pressurizer enclosure

d) Vicinity of reactor coolant pump (RCP) 2A1

e) Vicinity of reactor coolant pump 2A2

f) Vicinity of reactor coolant pump 2B1

g) Vicinity of reactor coolant pump 2B2

These points provide broad coverage of the containment for hydrogen monitoring and constitute a redundant independent H_2 Sampling System. Sampling lines originating from the containment dome, pressurizer, RCP 2A1 and RCP 2A2 areas constitute one independent train of the H_2 Sampling System. The other train consists of sampling lines originating from the upper containment, RCP 2B1 and RCP 2B2 areas. Each train of the sampling lines has a common header inside the containment and penetrates the containment in a separate penetration assembly.

As discussed in Subsection 6.2.2.2, there is adequate mixing of containment atmosphere so that local stratification or pocketing of hydrogen does not occur. The analyzer cubicles are located at elevation 43.0 ft of the Reactor Auxiliary Building (RAB). The analyzer system control panel is located in the control room.

A grab sample cylinder located at elevation 43.0 ft of the RAB is provided to permit hydrogen concentration measurement independent of the containment hydrogen analyzer detector.

6.2.5.2.2 Containment Hydrogen Recombiner Subsystem

The containment hydrogen recombiners control hydrogen in containment by using heat to cause recombination of liberated hydrogen with free oxygen in the air to form water.

The hydrogen recombiner system is described in Westinghouse Topical Report WCAP 7709-L[18] and shown on Figure 6.2-63. Supplement 1 through 4 of WCAP 7709-L were accepted by NRC on May 1, 1976. It is designed seismic Category I and Quality Group B requirements.

Each recombiner consists of a thermally insulated vertical metal duct with electric resistance metal sheathed heaters provided to heat a continuous flow of containment air to a temperature which is sufficient to cause a reaction between the hydrogen and the oxygen in the air. The recombiner is provided with an outer enclosure to provide protection from water spray coming from the Containment Spray System. The recombiner consists of an inlet preheater section, a heater-recombination section, a mixing chamber, and a cooling/exhaust section. Mixing of containment air is by the containment fan coolers and their associated ductwork as discussed in Subsection 6.2.2.2.2 by the turbulence introduced by the containment sprays, and by the process of natural diffusion of combustible gas with the containment air. Air is drawn into the recombiner by natural convection and passes first through the preheater section. This section consists of a shroud placed around the central heater section to take advantage of heat conduction through the walls to preheat the incoming air. This accomplishes the dual functions of reducing heat losses from the recombiner and of preheating the air.

The warmed air passes through an orifice plate and then enters the electric heater section where it is heated to approximately 1150°F to 1400°F causing recombination to occur. The flow then enters the cooling/exhausting section where the stream is mixed and diluted with cooler containment air in order to discharge the stream back into the containment atmosphere at a lower temperature.

Each hydrogen recombiner system has a removal capacity which is sufficient to limit concentrations of gases within the containment to safe concentrations; i.e., concentrations below the flammability limits. After a three hour startup period, the recombiner efficiency is 99-100 percent and the effluent does not exceed 100°F above ambient.

The unit is manufactured primarily of corrosion-resistant, high-temperature material for major structural components, except for the base which is steel. The electric hydrogen recombiner used conventional type electric resistance heaters sheathed with Incoloy-800 which is an excellent corrosion resistant material for this service. These heaters are designed to operate with sheath temperatures equal to those used in certain commercial heaters; however, these recombiner heaters operate at significantly lower power densities than in commercial practice.

The recombiners are located on the elevation 62.0 feet of the containment. They are inaccessible following a LOCA, and as such there is no sharing of recombiners among St. Lucie Units 1 and 2 or with other facilities. The hydrogen recombiners are designed for 60 years normal and one year post LOCA conditions. Design and performance data for the recombiners are listed in Table 6.2-55.

Each of the two recombiners is 100 percent capacity, and is connected to a separate onsite power source so that no single failure results in a total loss of recombiner function.

The recombiner is started by the operator by manual action from the control room. The operator is alerted when the containment H_2 level reaches three volume percent as signaled by the redundant Class 1E alarms of the Containment Hydrogen Analyzer System. Plant procedures provide guidance to the operator on when to start the hydrogen recombiner following a LOCA.

6.2.5.2.3 Containment Hydrogen Purge System

The Continuous Containment Purge/Hydrogen Purge System is provided as a further possible means of controlling hydrogen inside the containment following a LOCA. This system is provided as required by the NRC, although no single failure following a LOCA would necessitate its use. Therefore, the system is non-safety-related except for the containment penetrations and isolation valves which are seismic Category I and Quality Group B. See Table 6.2-57 for a failure mode and effects analysis.

The only redundancy required in the system is to assure the containment isolation function, as discussed in Subsection 6.2.4. Functional and operational redundancy of the system is not provided, as the system serves only as a diverse means of backup to the already redundant containment hydrogen recombiners. However, the system is capable of controlling hydrogen inside containment following a LOCA independent of the operation of recombiners. Table 6.2-56 shows the design data and materials for hydrogen purge system.

The system consists of a purge makeup penetration line, an exhaust penetration line, two exhaust fans and interconnecting ductwork between the fan discharge and suction header of the Shield Building Ventilation System (see Figure 9.4-11).

The system is actuated by manual operator action from the control room by opening the isolation valves on the exhaust and makeup lines and starting the exhaust fan. The operator makes the decision to use the purge system based on readings from the containment hydrogen analyzers.

The system draws air from the dome and from elevation 51.0 ft. of the containment by the system exhaust fan through the continuous containment purge air cleaning unit where it is filtered prior to release to the environment. Air can also be drawn by the exhaust fan and discharged to the Shield Building Ventilation System where it is filtered prior to release to the environment as discussed in Subsection 6.2.3. A separate makeup air line to the containment is located in the Reactor Auxiliary Building.

The system is located in the Reactor Auxiliary Building at elevation 43.0 ft. with the exception of the piping and isolation valves which are inside the containment.

The portions of system piping penetrating containment and associated containment isolation valves are designed to the codes and standards specified in Subsection 6.2.4.3. The ductwork and filters of the Shield Building Ventilation System, which treat the purge exhaust, are designed to the codes and standards specified in Subsection 6.5.1. The continuous containment purge clean-up unit and the hydrogen purge exhaust ductwork connecting the SBVS is not designed as safety related because the hydrogen purge system is not the main system for combustible gas control. However, the air clean-up unit and ductwork is seismically supported. The only portion of the system which would be exposed to the post-LOCA environment in containment is the inboard containment isolation valves and associated piping. These valves are active containment isolation valves, and as such will be qualified as specified in Subsection 6.2.4.2.

6.2.5.3 Design Evaluation

6.2.5.3.1 Functional Evaluation

The Containment Combustible Gas Control System (CGCS) provides the capability to monitor and maintain hydrogen concentrations within safe limits following a LOCA. The CGCS design conforms to the Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," September 1976 (R1).

Instrumentation provided for the system and discussed in Subsection 6.2.5.5 provides ample monitoring and indicating capabilities for complete control of the hydrogen control and sampling operations. Complete separation of redundant containment isolation valve control switches and wiring is provided. Internal wiring associated with redundant systems are separated by the metal enclosures which are permanently marked with identifying letters. In no way is exposed wiring from one redundant system be in close proximity with the exposed wiring of the other redundant system.

Sufficient concrete shielding is provided for the hydrogen analyzers to allow access for inspection and repair. Table 12.3A-7 provides access requirements for the H_2 Analyzer during accident conditions.

The hydrogen recombiner and hydrogen analyzers are designed as seismic Category I and are located in the containment and Reactor Auxiliary Building, respectively. A failure mode and effects analysis for the combustible gas control system is shown in Table 6.2-57.

6.2.5.3.2 Analysis of Hydrogen Generation

Following a LOCA, hydrogen gas is generated inside the containment by the following sources:

- Release of hydrogen dissolved in the reactor coolant
- Metal-water reaction involving the zirconium fuel cladding and the reactor coolant.
- Radiolytic decomposition of water.
- Corrosion of metals and paints by solutions used for emergency core cooling and containment spray.
- Radiolytic decomposition of solid organic materials.

a) Dissolved Hydrogen

During normal plant operation, some hydrogen is dissolved in the Reactor Coolant System water. The concentration can range from 10 to 50 cc (STP)/kg (H_2O) (see Subsection 9.3.4). Assuming that the Reactor Coolant system contains as much as 500,000 lbm of water at the maximum hydrogen concentration, 423 scf of hydrogen are dissolved in the coolant.

$$500{,}000 \text{ lbm} \times 0.4536 \frac{\text{kg}}{\text{lbm}} \times 50 \frac{\text{cc}}{\text{kg}} (\text{STP}) \times 3.631 \times 10^{-5} \frac{\text{ft}^3}{\text{cc}} \times$$

$$1.057 \frac{\text{scf}}{\text{ft}^3 (\text{STP})} = 423 \text{ scf}$$

This is assumed to be released instantaneously to the containment, but is only a minor source of hydrogen. Similarly, the concentration of dissolved oxygen in the primary coolant is less than 0.1 ppm. This represents 0.04 lbm of dissolved oxygen, a very small amount.

b) Metal-Water Reaction

As a result of a LOCA, elevated fuel cladding temperatures cause the zirconium to react with steam according to the following reaction:

$$Zr + 2\,H_2O \rightarrow ZrO_2 + 2H_2$$

Regulatory Guide 1.7 (R1) stipulates the amount of hydrogen that should be assumed to result from this metal-water reaction, for determining the performance requirements for the hydrogen recombiners. That amount is five times the maximum amount calculated in accordance with 10 CFR 50.46, but no less than the amount that would result from reaction of all the metal in the outside surfaces of the cladding cylinders surrounding the fuel (excluding the cladding surrounding the plenum volume) to a depth of 0.00023 inches. The maximum amount of zirconium calculated to react during ECCS operation is one percent. With the required factor of five, the amount assumed to react for purposes of hydrogen generation analysis is five percent. This five percent reaction is the governing requirement, rather than the 0.00023 inches reaction depth. The St. Lucie Unit 2 active core contains approximately 48,000 lbm of zircaloy, and for the hydrogen generation analysis it is assumed that five percent of this reacts instantaneously.

c) Radiolytic Hydrogen Generation

Water is decomposed into free hydrogen and oxygen by the absorption of energy emitted by fission products contained in the fuel and fission products intimately mixed with the water. The process is described by the conservative value $G(H_2) = 0.5$ molecule per 100 ev as specified in Regulatory Guide 1.7 (R1). The assumed distribution of fission products is the distribution specified in Regulatory Guide 1.7 (R1).

According to this distribution, one percent of the solids and 50 percent of the halogens present in the core are intimately mixed with the reactor coolant water. The noble gases are assumed to be released to the atmosphere. For fission products intimately mixed with the water, the fraction of the radiation absorbed by the water is 100 percent for both betas and gammas. For fission products remaining in the core, the fraction absorbed is 10 percent for gammas and zero for betas. These assumptions are summarized in Table 6.2-58. The beta and gamma releases were combined according to this model in calculating the amount of hydrogen generated by radiolysis.

The normalized decay power curve is provided on Figures 6.2-31 and 6.2-32. The thermal power assumed for this analysis is 2700 Mwt.

d) Corrosion of Metals

The use of aluminum, zinc and zinc-base paint has been restricted inside the St. Lucie Unit 2 containment.

For calculations of hydrogen generated due to corrosion, the long term containment temperature history used is given on Figure 6.2-64. It is an extension of the DBA LOCA transient for the containment atmosphere shown on Figure 6.2-39. It is assumed that late in the accident the curve levels off and the temperature remains constant.

The sources of aluminum are listed in Table 6.2-59. The hydrogen generation due to corrosion of aluminum was determined from the containment temperature transient and the aluminum corrosion rate data. The aluminum corrosion rate was taken from Reference 19. It is for a spray solution which contains sodium hydroxide and hence is much more corrosive than the hydrazine spray solution used in St. Lucie Unit 2. Thus the corrosion rate used here is extremely conservative and complies with Regulatory Guide 1.7 (R1). The aluminum corrosion rate as a function of time is shown on Figure 6.2-65.

The exposed zinc is found in wire rope restraint socket speltering (for the reactor coolant pumps) and in galvanized steel. Galvanized steel is present mainly in ductwork, cable trays, conduits, tube supports/hangers, instrument racks and platforms. The sources of zinc are itemized in Table 6.2-59. The hydrogen generation due to corrosion of metallic zinc was obtained by superimposing the containment temperature transient for the design basis LOCA on the zinc corrosion rate data. The zinc corrosion rate was determined from the results of tests reported in Reference 20 for the concentration of boric acid which is used in the spray solution. The zinc corrosion rate as a function of time is shown on Figure 6.2-66.

Other surfaces, including platform framing, restraint structures, equipment supports, miscellaneous steel, and the polar crane, are coated rather than galvanized. A list of coated surfaces is given in Table 6.1-2. All uninsulated surfaces coated with a zinc-base primer are top coated with a non-zinc finish coat. The hydrogen generation rate due to zinc in the primer is negligible. Although the zinc-base primer is not directly exposed to the spray solution because it is covered with a non-zinc-base finish coat, an assumption made for the hydrogen generation analysis includes a hydrogen contribution as a result of the corrosion of all zinc-base paint primers exposed to a LOCA environment. The paint corrosion rate used is taken from Figure 9 of D. Van Rooyen's report [21] which is derived from the paper by Zittel.[22] This corrosion correlation is combined with the containment temperature transient to obtain the paint hydrogen generation rate (see Figure 6.2-67). It is assumed that zinc-based paint is a continuously present source for hydrogen in containment throughout the DBA (i.e., an indefinite thickness of paint was assumed).

e) Radiolysis of Solid Organic Materials

Hydrogen evolution due to radiolysis of solid organic materials is a negligible source inside the containment as compared to radiolytic hydrogen generation by water. Solid organic materials inside containment consist primarily of cable material and a small amount of unqualified organic materials described in Subsection 6.1.2.

A review has been conducted to determine the total quantity of cable material (jacket and insulation) inside the containment. The amount of cable material inside containment is 10,779 pounds of power cable; 15,138 pounds of control cable; and 27,945 pounds of low-level cable for a total of 53,862 pounds. Adding a 10 percent contingency factor the total quantity of cable material is 59,249 pounds. PC/M 05128 replaced existing ICI MI (Mineral Insulated) cable with organic cable increasing the weight of cable material for low-level cable from 27,945 pounds to 28,813 pounds. This in turn increases the total weight of cable material inside containment from 53,862 pounds to 54,731 pounds. As a result the 10 percent contingency factor is reduced to 8 percent.

To illustrate the order of magnitude of hydrogen evolution due to radiolysis of cable insulation, a conservative analysis was performed. There are approximately 59,000 pounds of cable insulation and jacket material in the containment. The major ingredient is a type of vulcanized chlorinated rubber compound. Gas yields from organic material reported in the literature are very conservative since tests are performed on thin sections in a vacuum; while diffusion through thick cable and jacket to the surface causes gas trapping and hydrogen recombination. Reference 25 indicates that elastomers and rubbers have G values ranging from .15 to .8 (G value = molecules of gas evolved / 100 ev). Using a conservative high G value of .8, and a total 30-day dose of 1.4×10^8 rads (Appendix D, NUREG-0588), results in a total hydrogen evolution of 2260 ft^3. This amount is approximately, 0.9 percent of the containment volume, and insignificant compared to metal-water reaction, water radiolysis and metal corrosion sources. The effect is further reduced when the time scale for gas evolution is considered, and the fact that more realistic estimates would probably reduce the volume almost an order of magnitude.

The total amount of protective coatings and organic materials used inside the containment that do not meet the requirements of ANSI N101.2 (1972) and Regulatory Guide 1.54 is approximately six cubic feet. The total quantity of combustible gases that can be formed from these unqualified organic materials under DBA conditions is 4.2 pounds maximum of 1625 cubic feet at STP conditions. These figures are based on the following:

Using the conservative assumption that all of the hydrogen in the unqualified materials will be released as hydrogen gas under DBA conditions and the conservative postulation of 1 percent hydrogen by weight in these materials for an average density of 70 pounds per cubic foot, the calculation indicated that a maximum weight of 4.2 lbs. of hydrogen would be evolved into containment. This translates to 1624.6 cubic feet at STP conditions. For the total containment volume of 2.5×10^6 cubic feet, the hydrogen percent by volume would be about 0.06 percent. This value is insignificant compared to the steady state maximum value of 3.2 percent by volume accumulated hydrogen during the operation of the hydrogen recombiner as shown on Figure 6.2-68.

Figure 6.2-68 shows the hydrogen production as a function of time for each individual source and for the total.

Regulatory Guide 1.7 (R1) sets a limit of four volume percent for the hydrogen concentration. Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident", (11/24/75) also recommends a margin, such as 0.5 volume percent, between the hydrogen concentration limit and hydrogen concentration at which the equipment would be activated. Based on these considerations, operation of a hydrogen recombiner is assumed to begin at a hydrogen concentration of 3.0 volume percent. Figure 6.2-68 shows that this occurs approximately 3.5 days after the accident.

Although two recombiners are provided inside the containment, one is sufficient to maintain the hydrogen concentration below the required level. The Continuous Containment Purge/Hydrogen Purge System is a backup system which will most likely never be operated. The flow rate for each recombiner is 100 scfm. The flow rate for the Continuous Containment Purge/Hydrogen Purge System can be throttled to 100 cfm. Figure 6.2-68 shows the hydrogen concentration as a function of time with the recombiner(s) or the purge system in operation. The concentration history is shown for operation of one recombiner or the Continuous Containment Purge/Hydrogen Purge System in any of its modes serves to keep the hydrogen concentration below the four volume percent limit of Regulatory Guide 1.7 (R1).

6.2.5.3.3 Hydrogen Mixing Inside Containment

The containment fan cooler system provided for cooling and adequate mixing of containment air is discussed in Subsection 6.2.2.2.2. Fan coolers are provided which adequately mix the containment air. Adequate ducting is provided to all the containment compartments and subcompartments which assures no accumulation of combustible gas in pockets. Mixing of containment air is also accomplished by the process of natural diffusion of combustible gas with the containment air.

The long term operating requirements for the Containment Cooling System is to provide for containment atmosphere blending to prevent stratification and possible pocketing of hydrogen gas and not so much the heat removal. Air is continuously introduced upwards into the dome from outlets on the ring header. Air and hydrogen are thus mixed and displaced to lower levels of the containment.

6.2.5.3.4 EPU Impact on Containment Combustible Gas Control

The EPU has no impact on the fundamental mixing mechanisms discussed in section 6.2.5.3.3 or on the capability of the hydrogen monitoring system to diagnose beyond design-basis hydrogen concentrations.

6.2.5.4 Tests and Inspections

Preoperational tests and manufacturer's parameters form the basis for establishing the Combustible Gas Control System (CGCS) reference characteristic. This information is used to determine the required frequency of testing as well as verifying test results.

Visual inspection and functional testing are performed at specified intervals as part of the Technical Specifications. Visual inspection is made of the containment Penetrations and all piping associated with the CGCS as required by the ASME Code, Section XI.

Seismic and environmental qualification of the H_2 recombiner is shown in Sections 3.10 and 3.11.

6.2.5.5 Instrumentation Requirements

System instrumentation and control, as indicated in Table 6.2-60, is provided in the control room for post-LOCA monitoring. Indicating lights show the operating status of the hydrogen analyzers and hydrogen recombiners in the control room. Also provided in the control room are position indication for the hydrogen analyzer sampling line isolation valves, hydrogen concentration recording device, qualified panel meter, indication of the temperature of the heating elements and the amount of power drawn by the hydrogen recombiner. The system is to be manually controlled after an accident based on information obtained from the hydrogen analyzer. The analyzer controls located in the control room enable the operator to open and close the sample valves within the containment in sequence and record the hydrogen concentration. Sample valve positions are indicated in the control room panel. The analyzers have a setpoint for an alarm, at three volume percent hydrogen.

The operation of the hydrogen recombiner is continuously monitored in the control room. Thermocouples are provided to continuously monitor the recombiner, and proper recombiner operation after an accident is assured by measuring the amount of electric power to the recombiner from the power supply which is located in the Reactor Auxiliary Building. The proper amount of air flow through the recombiner is fixed during initial tests and checkout by an orifice plate on the recombiner inlet.

The containment hydrogen purge has individual control switches for all fans and isolation valves in the control room. Valve position indications are also provided in the control room.

6.2.6 CONTAINMENT LEAKAGE TESTING

The containment including its associated penetrations, piping and isolation valves, has been designed to limit leakage to 0.5 percent of the weight of the contained air per day at the peak accident pressure during a DBA. Per Reference 26, the DBA peak accident pressure (Pa) for the containment test pressure is to be considered the analyzed LOCA containment peak pressure. As required by GDC 52, 53 and 54, preoperational and periodic leakage testing of the primary containment and penetrations will be performed. The requirements for leakage rate testing are given in Appendix J to 10 CFR 50. Note that Technical Specification Amendment 88, dated February 10, 1997, was approved which allows Appendix J, Option B testing. This testing allows the licensee to establish intervals for containment leak-testing based on the performance history and risk significance of the components (containment, penetrations, and valves). All Type A, B, and C testing requirements will be performed at intervals determined by the containment leakage rate testing program (See Technical Specification 6.8.4). The criteria for the new testing intervals is evaluated based on individual performance history and the permitted intervals specified in NRC letter dated August 27, 2015. See Reference 41 (page 6.2-81).

The program for leak rate testing is performed in two parts. The integrated leak rate test (ILRT) pressurizes the primary containment and tests the containment vessel, penetrations and isolation valves simultaneously; the local leakage rate test is performed on a more frequent basis and tests applicable penetrations and isolation valves on an individual basis. This program of leakage rate testing verifies the leaktight integrity of the containment vessel, penetrations and isolation valves during the life of the plant.

6.2.6.1 Containment Integrated Leak Rate Test

After installation of penetrations, appropriate piping and isolation valves, but prior to core loading, a preoperational containment integrated leak rate test (Type A test) is performed as well as an overload pressure test (see Subsection 3.8.2). Periodic Type A tests are performed during the life of the unit to verify the continued leaktight integrity of the containment systems.

Type A tests are conducted in accordance with the provisions of Appendix J, Option B to 10 CFR 50.

Prior to the preoperational or periodic Type A leak rate tests, a general inspection of the accessible interior and exterior surfaces of the containment vessel is performed to ensure that there is no evidence of structural deterioration which may affect either the containment structural integrity or leaktightness. If there is evidence of structural deterioration, the Type A leak rate test is not performed until corrective action is taken in accordance with repair procedures, nondestructive examinations and tests as specified in the applicable codes. Results of the inspections, any deterioration found and any corrective actions taken are documented and reported per the containment leakage rate testing program, and available for inspection at plant site.

During the period of the performance of the Type A test no repairs or adjustments are made so that the containment is tested in as close to the "as is" condition as practical. During the period between completion of one Type A test and initiation of the containment inspection for the subsequent Type A test, repairs or adjustments are made to components whose leakage exceeds that specified in the Technical Specifications, as soon as practical after identification. If during a Type A test potential excessive leakage paths are identified which will interfere with satisfactory completion of the test, or which result in the Type A test not meeting the acceptance criteria, the Type A test is terminated and the leakage through such paths measured using local leakage testing methods. Repairs and/or adjustments to equipment are made and the Type A test re-performed. The corrective actions taken and the change in leakage rate determined are included in the test report available for inspection at the plant site.

Containment systems are lined up to simulate post-accident conditions for the Type A test with exceptions as permitted under 10 CFR 50 Appendix J, Option B. The reactor and Reactor Coolant System are maintained in a cold shutdown condition. The closure of the containment isolation valves for the Type A test is accomplished by normal operation and without any preliminary exercising or adjustments (e.g., no tightening of valve after closure by valve motor). Repairs to malfunctioning or leaking valves are made as necessary. Information on valve closure malfunction or valve leakage that requires correction before the test is included in the test report available for inspection at the plant site.

Those portions of systems which potentially are open to the containment atmosphere following a LOCA (i.e., GDC 56 systems) and systems connected to the reactor coolant pressure boundary (GDC 55 systems) are opened or vented to the atmosphere for the Type A test. Vented systems are drained of water or other fluids to the extent necessary to assure exposure of the system containment isolation valves to containment air test pressure and to assure that they are subjected to the peak differential pressure or a penalty is added to the Type A test results based on LLRT results for the system. Systems required to maintain the unit in a cold shutdown condition are operable in their normal mode and are not vented or drained. However, any of these systems that are containment leakage paths and require Type C or bypass local leakage tests as defined in Subsection 6.2.4 have the results of the local leakage tests added to the result of the Type A tests. Systems used during the Type A test for pressurizing the containment or sensing the leakage are not lined up in the post-accident positions but any leakage from the isolation valves in these systems is determined by local methods and the results are added to the Type A test. Systems that operate in post-accident conditions and filled with water are not vented or drained for the Type A test. Systems which form closed seismic Category I systems inside containment (as defined by GDC 57) are not vented to the containment atmosphere.

The air used to pressurize the containment is conditioned for temperature and humidity to the extent necessary to prevent moisture condensation in the containment at the test pressure. The air used to pressurize the containment is essentially oil free to prevent coating of the containment walls with oil or interfering with the test instrumentation.

The method selected for the determination of leakage is consistent with the requirements of Appendix J, and/or Appendix J, Option B of 10 CFR 50 with the option of using NRC approved methods for performing the leak rate testing and calculating the results.

Instrumentation is provided to monitor the containment during leakage rate testing. This instrumentation is designed, calibrated and tested to provide high accuracy to ensure that the containment atmosphere parameters are precisely measured.

The test instrumentation is specified in the test procedure. This includes instrumentation to measure containment pressure, temperature and humidity.

An error analysis of these instruments is made to establish the sensitivity of the instrumentation and to verify that the instrumentation system selected is acceptable for the measurement of leakage rates in the range desired.

Sensing devices are located at different locations in the containment to measure average temperature and humidity. Location of the temperature and humidity sensors is made with consideration of the temperature and humidity patterns in the containment. These patterns are employed in determination of the mean representative temperature and humidity for the absolute method of leakage rate testing. The use of fans to minimize temperature variations in regions of the containment is allowed but the fans are of limited horsepower to minimize potential heat input.

The precision pressure measuring device is located outside of the containment and is connected to the containment via a test penetration.

The preoperational Type A test is conducted at the peak accident pressure (P_a) and a reduced pressure selected for periodic testing (P_t). P_t is not less than one-half of P_a. Periodic type tests are conducted at P_a or P_t.

The containment environmental conditions are stabilized for a period of about four hours prior to the start of a leakage rate test. Leakage rate tests are not started until temperature equilibrium is attained.

The leakage rate test period extends to 24 hours of sustained internal pressure. If it is demonstrated to the satisfaction of the NRC that the leakage rate is accurately determined during a shorter test period, the agreed upon shorter period is used. 10 CFR 50 Appendix J, Option B approved by T.S. Amendment #88 allows use of methodology for a shorter duration test.

Pressure, temperature and humidity observations are made within the containment and recorded during the course of the leakage rate test at hourly intervals as a minimum. Pressure and temperature measurements of the outside atmosphere are made at the same time interval. The times of the observations are noted in hours and minutes. A dated log of events and pertinent observations is maintained during the leakage rate test. The correctness of data is attested to by those responsible for the test and, where specified by the test procedure, by a qualified witness.

At the conclusion of the leakage rate test, the accuracy of the Type A test is verified by a supplemental test method. The supplemental test injects into or bleeds from the containment an accurately measured amount of air. The supplemental test method selected is conducted for a sufficient duration to establish accurately the change in leakage rate between the Type A test and the supplemental test. The quantity of air injected into or bled from the containment during the supplemental test is between 0.75 and 1.25 of the allowable leakage rate (La). Leakage rate acceptance criteria are as follows:

The peak pressure (P_a) measured leakage rate (L_{am}) is less than 0.75 of the allowable leakage rate (L_a). The containment vessel and its penetrations are designed to limit leakage (L_a) to no more than 0.5 weight percent of the internal free volume of the containment in 24 hours at the peak accident pressure (P_a).

The reduced pressure (P_t) measured leakage rate (L_{tm}) is less than 0.75 of the allowable reduced pressure leakage rate (L_t). L_t is established during the preoperational Type A test and is determined by:

- if; $L_{tm}/L_{am} \leq 0.7$

$$L_t = L_a (L_{tm}/L_{am})$$

- if; $L_{tm}/L_{am} > 0.7$

$$L_t = L_a (P_t/P_a)^{1/2}$$

The results of the supplement verification test are acceptable provided the difference between the leakage determined during the Type A test and the supplemental test is within 25 percent of the allowable leakage rate (L_a or L_t).

If a Type A test fails the acceptance criteria or the results of the supplemental test do not agree within 25 percent of the allowable leakage rate, the reason is determined, corrective action is taken and a successful Type A or supplemental test, as applicable, is performed.

6.2.6.2 Containment Penetration Leakage Rate Test

Penetration local leakage rate tests, referred to as Type B tests, are performed on the containment penetrations, closure hatches, and airlocks that incorporate flexible bellows, resilient seats, flexible metal seats or gaskets as a part of the primary containment boundary. All closure seats of the type mentioned are provided with test taps for local leakage rate testing without pressurization of the entire containment volume. All bellows are individually testable.

Subsection 3.8.2.1.1 describes the mechanical penetrations and electrical penetrations of the containment.

Of the piping penetrations, only the main steam and feedwater penetrations have flexible expansion bellows. All other process piping penetrations have the process pipe welded directly to the steel containment vessel and are leak tested during the Type A test. The fuel transfer tube has a durable gasketed flanged closure to permit Type B leakage rate testing. In addition, flexible expansion bellows connect the fuel transfer tube to the containment vessel walls and require Type B testing.

The electrical penetrations are seated cannisters which require Type B leakage testing.

The construction hatch is a seal welded assembly which is leak tested during the Type A test. No Type B testing is required. The maintenance hatch closure is double gasketed to permit Type B leak testing.

Two types of leak testing are performed on the personnel airlock and the emergency escape airlock. The airlock doors are provided with double gasketed seals to permit leak testing. In addition, the entire airlock is pressurized and leak tested on a periodic basis to leak test the seals and closures other than the doors. Strongbacks are placed on the inner door during this test to prevent unseating.

All Type B local leakage rate testing is performed at the peak accident pressure P_a.

Local leakage rate testing is performed by pressurizing the test space to the required pressure with air and either:

a) measuring the makeup flow required to maintain a constant pressure or,

b) measuring the rate of pressure drop with the pressure source isolated.

The method used depends upon the magnitude of the leakage found.

The acceptance criteria for this leakage rate testing is combined with the isolation valve leakage rate testing (Type C leak testing) discussed in Subsection 6.2.6.3. The acceptance criteria is that the combined leakage rate of penetrations and isolation valves subject to Type B and Type C testing is less than 0.60 L_a.

Sources of leakage in excess of the acceptance criteria are repaired and the corrective actions taken are reported in the test report available for inspection at the plant site.

6.2.6.3 Containment Isolation Valve Leakage Rate Test

Containment isolation valve leakage rate tests, referred to as Type C tests, are performed on the containment isolation valves that are required to close and limit containment leakage during LOCA. Table 6.2-62 identifies the containment isolation valves that require Type C leak testing. Subsection 6.2.4.4 defines the bypass leakage paths and identifies the containment isolation valves subject to bypass leakage testing, and also provides justification for not doing so.

All Type C and bypass leak rate tests are performed at the peak accident pressure, P_a . The test method is identical to that described in Subsection 6.2.6.2 for Type B leak testing. The provisions for leak testing are shown in Table 6.2-62. The Type C tests and bypass leakage tests are similar with the exception of the acceptance criteria.

The combined leakage rate of penetrations and isolation valves subject to Type B and Type C testing is less than 0.60 L_a.

The combined leakage rate of all isolation valves subject to bypass leakage testing is less than .12L_a.

Corrective actions taken are reported to the NRC as a part of the test report available for inspection at the plant site.

6.2.6.4 Scheduling and Reporting of Periodic Tests

After the preoperational leakage rate tests, a set of three Type A tests is performed at approximately equal intervals during each 10 year service period. The final test of each set is conducted when the plant is shutdown for the 10 year plant in-service inspections (Note that these service intervals have changed based on Appendix J, Option B testing).

Type B tests, except tests for airlocks, are performed during each reactor shutdown for refueling or other convenient intervals, but in no case at intervals greater than two years. Airlocks are tested at six month intervals. Airlock door gasket leak rate tests are performed after each opening of the door except when the airlock is being used for multiple entries when such a test is performed once per three days (Note that these service intervals have changed based on Appendix J, Option B testing).

Type C tests are performed during each reactor shutdown for refueling but in no case at intervals greater than two years (Note that these service intervals have changed based on Appendix J, Option B testing).

The preoperational and subsequent periodic tests are the subject of a summary technical report available for inspection at the plant site.

These reports are prepared in accordance with the requirements of 10 CFR 50, Appendix J and/or Appendix J, Option B. The report shall contain an analysis and interpretation of the leakage rate test data for the Type A test results to the extent necessary to demonstrate the acceptability of the containment leakage rate in meeting the acceptance criteria.

Leakage test results from Type A, B, and C tests that failed to meet the acceptance criteria are reported in a separate accompanying summary report that includes an analysis and interpretation of the test data. Results and analysis of the supplemental verification test employed to demonstrate the validity of the leakage rate measurement are also included.

6.2.6.5 Special Test Requirements

Any major modification or replacement of a component which is part of the primary reactor containment boundary, or resealing of a seal welded door, is performed after the preoperational leakage rate test is followed by either a Type A, Type B, Type C or bypass leakage test (as applicable) for the area affected by the modification. The measured leakage from this test is included in a test report submitted to the NRC.

However modifications, replacements or resealing of seal welded doors, performed directly prior to the conduct of scheduled leakage tests do not require a separate test.

Bypass leakage testing and the secondary containment functional design are given in Subsections 6.2.4 and 6.2.3, respectively.

SECTION 6.2: REFERENCES

1. Waterford Steam Electric Station Unit 3 PSAR, Docket 50-382, Louisiana Power & Light Company letter LPL 2656 dated December 14, 1973 from E Rodrique to K Kniel.

2. Cessar, Combustion Engineering Standard Safety Analysis Report, Combustion Engineering, Inc. NRC Docket No. STN-50-470.

3. Rettig, W. H., et al, RELAP-3 - A Computer Program for Reactor Blowdown Analysis, Idaho Nuclear Corporation, IN-1321, June 1970.

4. RELAP-4/Mod 5, A Computer Program for Transient Thermal-Hydraulic Analysis of Nuclear Reactor and Related Systems, Volumes I, II, III, Aerojet Nuclear Company, ANCR-NUREG-1335, September 1976.

5. Idelchik, J. E., Handbook of Hydraulic Resistance and of Friction, translation, Israel Program for Scientific Translation, Jerusalem, 1966, AEC-TR-6630.

6. Flow of Fluid Through Valves, Fittings and Pipe, Crane Company, Chicago, 1957, Technical Paper No. 410.

7. CENPD-26, Description of Loss of Coolant Calculational Procedures.

8. W. H. McAdams, Heat Transmission, McGraw-Hill, 1954.

9. FLOOD-MOD2. A-Code to Determine the Core Reflood Rate for a PWR Plant With Two Core Vessel Outlet Legs and Four Core Vessel Inlet Legs, Interim Report, Aerojet Nuclear Company, November 2, 1972.

10. Jacob, Max, Heat Transfer, Vol. 1, John Wiley & Sons, New York, 1949, pp 251-291.

11. Carryout Rate Correlation, G. F. Malon, ANS Transactions, V 15, No. 2, 1972, Winter Meeting.

12. Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Cooled Nuclear Power Reactors, Federal Register, Vol. 39, No. 3 - Friday, January 4, 1974.

13. "Calculative Methods for the C-E Large Break LOCA Evaluation Model," CENPD-132, August, 1974 (Proprietary).

    "Calculational Methods for the C-E Large Break LOCA Evaluation Model," CENPD-132, Supplement 1, February, 1975 (Proprietary).

    "Calculational Methods for the C-E Large Break LOCA Evaluation Model," CENPD-132, Supplement 2, July, 1975 (Proprietary).

    "Calculational Methods for the C-E Nuclear Power Large Break LOCA Evaluation Model," CENPD-132, Supplement 4, March, 2001 (Proprietary).

14. Letter, 0. D. Parr (NRC) to F. M. Stern (C-E), June 11, 1975.
Letter, 0. D. Parr (NRC) to A. E. Scherer (C-E), December 9, 1975.
Stuart A. Richards (NRC) to P.W. Richardson (WEC), "Safety Evaluation of Topical Report CENPD-132, Supplement 4, Revision 1, 'Calculative Methods for the CE Engineering Technology Large Break LOCA Evaluation Model,' (TAC No. MA5660)," December 15, 2000.

15. "CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis," CENPD-133, August, 1974 (Proprietary).

"CEFLASH-4A, A FORTRAN-IV Digital Computer Program for Reactor Blowdown Analysis (Modifications)," CENPD-133, Supplement 2, February, 1975 (Proprietary).

16. "COMPERC-11, A Program for Emergency Refill-Reflood of the Core," CENPD-134, August, 1974 (Proprietary).

"COMPERC-II, A Program for Emergency Refill-Reflood of the Core (Modifications)," CENPD-134, Supplement 1, February, 1975 (Proprietary).

17. Ashrae, Handbook of Fundamentals, 1972, pp. 333.

18. Westinghouse Topical. Report WCAP 7709-L dated July, 1971.

Supplement 1 - Electric hydrogen recombiner for PWR containments.

Supplement 2 - Electric hydrogen recombiner for PWR containments equipment qualification report.

Supplement 3 - Electric hydrogen recombiner for PWR containments long term tests.

Supplement 4 - Electric hydrogen recombiner for PWR containments.

Supplement 5 - Electric hydrogen recombiner special tests.

Supplement 6 - Electric hydrogen recombiner IEEE 323-1974 qualification.

Supplement 7 - Electric hydrogen recombiner LWR containments supplemental test No. 2.

19. Greiss, J. C. and Bacarella, A., "Design Considerations of Reactor Containment Spray Systems," Part III "The Corrosion of Materials by Spray Solutions," ORNL-TM-2412, December, 1969.

20. Hutchinson Island Plant, Unit No. 1, PSAR Docket No. 50-335, October, 1969 (Amendment 4).

21. Van Rooyen, D., "Hydrogen Release Rates from Corrosion of Zinc and Aluminum," Brookhaven National Laboratory, BNL-NUREG-24532, May, 1978.

22. Zittel, H. E., "Post-Accident Hydrogen Generation from Protective Coatings in Power Reactors," Nuclear Technology, Volume 17, February, 1973.

23. USNRC letter dated December 8, 1978 from Mr R. L. Baer (NRC) to Mr G. Dinsky (Owens Corning).

24. FP&L Letter L-82-431 dated October 6, 1982 from Dr R. E. Uhrig (FP&L) to Mr D. Eisenhut (NRC).

25. Bolt., R and Carrol J, "Radiation Effects on Organic Materials" Table 6.3, Academic Press, New York, 1963.

26. Letter from E. G. Tourigny (NRC) to W. F. Conway (FPL), "Issuance of Amendment Re: Containment Leakage Rate Test Pressure Change," dated December 16, 1988.

27. ABB C-E calculation 016-AS93-C-006 Rev. 0, dated March 17, 1993, St. Lucie 2 LOCA Containment Peak Pressure/Temperature (P/T) Analysis at 102% Power (2754 Mwt).

28. NRC Standard Review Plan, NUREG-0800, Section 6.2.1, July, 1981.

29. CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis, CENPD-133, Supplement 5-P, June 1985 and previous supplements (Proprietary).

30. Computer Code Description and Verification Report for FLOOD3 Mod 2, Engineering Inc., dated July 7, 1999.

31. Combustion Engineering Topical Report CENPD-140-A, dated June 1976, Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis.

    Software Change Specification and Validation for CONTRANS2 Revision ctn2m1.0702 dated August 30, 2002.

32. J. A. Norberg, et al, "Simulated Design Basis Accident Tests of the Carolina's Virginia Tube Reactor Containment -- Preliminary Results," IN-1325.

33. Federal Register, Volume 57, dated October 27, 1987.

34. USNRC letter dated March 5, 1993 from J. A. Norris (NRC) to G. H. Goldberg (FP&L).

35. FP&L letter L-92-231 dated August 26, 1992 from D. A. Sager (FP&L) to USNRC.

36. St. Lucie Unit 2 Safety Evaluation, PSL-ENG-SENS-97-044, Containment Sump Screens, Revision 1.

37. EBASCO Calculation NSSS-041, Rev. 1, dated 11-07-83, Feedwater Spiking

38. FPL Engineering Evaluation PSL-ENG-SEMS-99-021, Rev. 0 (Tracking No. 99067), "Inputs for Updated CCW/ICW Peak Temperature Analysis - St. Lucie Nuclear Plant Unit 2 (Enclosure to FPL Letter ENG-PSL-99-0271 dated June 28, 1999).

39. AREVA NP Document 77-5069878-001, "Replacement Steam Generator Report for Florida Power and Light St. Lucie Unit 2."

40. GL-2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents of Pressurized Water Reactors."

41. NRC letter, F.E. Saba to M. Nazar, dated August 27, 2015, "St. Lucie Plant, Unit Nos. 1 and 2 – Issuance of Amendments Regarding Change to Technical Specification 6.8.4.h, Containment Leakage Rate Testing Program."

TABLE 6.2-1

POSTULATED ACCIDENTS FOR CONTAINMENT DESIGN

Containment Design Parameter	Postulated Accidents	Mass and Energy Release Reference
A. Containment Peak Pressure Temp-erature	<u>Limiting Updated Loss-of-coolant accident (LOCA)</u>	
	Double-ended hot leg slot (DEHLS), 19.24 ft^2 area (minimum safety injection)	
	<u>Extended Power Uprate Main steam line breaks (MSLB)</u>	
	5.41 ft^2 MSLB, 100.3% power Containment Cooling Train Failure	
	5.41 ft^2 MSLB, 75% power, Containment Cooling Train Failure	-
	5.41 ft^2 MSLB, 50% power, Containment Cooling Train Failure	-

TABLE 6.2-1 (Con't)

	Containment Design Parameter	Postulated Accidents	Mass and Energy Release Reference
A.	Containment Peak Pressure Temperature (Cont'd)	5.41 ft^2 MSLB, 25% power, Containment Cooling Train Failure	-
		5.41 ft^2 MSLB, 0% power, Containment Cooling Train Failure	-
		5.41 ft^2 MSLB, 100.3% power, MFIV Failure	-
		5.41 ft^2 MSLB, 75% power, MFIV Failure	-
		5.41 ft^2 MSLB, 50% power, MFIV Failure	-
		5.41 ft^2 MSLB, 25% power MFIV Failure	-
		5.41 ft^2 MSLB, 0% power, MFIV Failure	-
		5.41 ft^2 MSLB, 100.3% power, MSIV Failure	
		5.41 ft^2 MSLB, 75% power, MSIV Failure	-
		5.41 ft^2 MSLB, 50% power, MSIV Failure	-
		5.41 ft^2 MSLB, 25% power, MSIV Failure	-
		5.41 ft^2 MSLB, 0% power, MSIV Failure	-
		5.41 ft^2 MSLB, 100.3% power, EQ (peak temperature) MSIV Failure	Table 6.2-12G
B.	Subcompartment Peak Pressure	Reactor Cavity	
		100 in^2 hot leg guillotine (HLG)	Table 6.2-14A and B

TABLE 6.2-1 (cont'd)

Containment Design Parameter	Postulated Accidents	Mass and Energy Release Reference
B. Subcompartment Peak Pressure (1) (Cont'd)	200 in^2 discharge leg guillotine (DLG)	Table 6.2-15A and B
	Steam generator compartment	
	1000 in^2 hot leg guillotine (HLG) (2)	Table 6.2-16A and B
	1400 in^2 suction leg guillotine (SLG) (SG Nozzle) (2)	Table 6.2-17A and B
	1400 in^2 suction leg guillotine (SLG) (RCP Nozzle) (2)	Table 6.2-18A and B
	532 in^2 suction leg slot (SLS) (RCP elbow) (2)	Table 6.2-19
	532 in^2 suction leg slot (SLS) (S.G. elbow) (2)	Table 6.2-19
	1400 in^2 discharge leg guillotine (DLG) (RCP Nozzle) (2)	Table 6.2-20A and B
	1559 in^2 main steam line guillotine (5)	Table 6.2-24
	Pressurizer Subcompartment	
	161 in^2 double ended pressurizer surge line guillotine break (Pressurizer nozzle) (4)	Table 6.2-21A and B
	9.28 in^2 double ended pressurizer-relief line guillotine break (Pressurizer nozzle) (3)	Table 6.2-22
	11.87 in^2 double ended spray line guillotine-break (Pressurizer nozzle) (4)	Table 6.2-23A and B
C. Containment differential pressure	Inadvertent operation of the Containment Heat Removal System (Both trains)	N/A
D. Containment ECCS Minimum pressure-	Refer to Subsection 6.2.1.5	N/A

(1) Analysis is no longer required due to application of the LBB criteria (refer to UFSAR 6.2.1.2.3)

(2) This break is eliminated due to application of the LBB criteria (refer to UFSAR 6.2.1.2.3)

(3) The pre-EPU M&E release data remains unchanged at EPU conditions (refer to UFSAR 6.2.1.2.3)

(4) The pre-EPU M&E release data slightly increase at EPU conditions (refer to UFSAR 6.2.1.2.3)

(5) The pre-EPU M&E release data for this break is not the limiting case

TABLE 6.2-2

CALCULATED VALUES FOR CONTAINMENT PARAMETERS

Parameter		Design Basis Accident	Calculated Value
Peak Containment Atmosphere Pressure			
a.	LOCA (original)	9.82 ft^2 DESLS (max. SI)	41.8 psig
a1.	LOCA (EPU)	19.24 ft^2 DEHLS (min. SI)	43.43 psig
b.	MSLB	5.41 ft^2 MSLB, 0% Failure of one Containment Cooling Train	37.49 psig
Peak Containment Atmosphere Temperature			
a.	LOCA (original)	9.82 ft^2 DESLS (max. SI)	271.4°F
a1.	LOCA (EPU)	19.24 ft^2 DEHLS (min. SI)	266.66°F
b.	MSLB	5.41 ft^2 EQ-MSLB, 100.3% Failure of a MSIV to close	377.33°F
Peak CCW Temperature			
a.	LOCA (EPU)	9.82 ft^2 DESLS (min. SI)	119.93°F
Peak Subcompartment Pressure			
a.	Reactor Cavity (1)	200 in^2 discharge leg guillotine	86.0 psid
b.	Steam Generator	1400 in^2 suction leg guillotine at nozzle (2)	5.9 psid
c.	Pressurizer	161 in^2 surge line guillotine at nozzle (3)	1.3 psid (El. 62') 22.7 psid (El. 29.5')
Differential Pressure			
a.	Containment	Inadvertent operation of the Containment Heat Removal System	0.6 psid
b.	Shield Building	Inadvertent operation of the Containment Heat Removal System	0.615 psid
Minimum Pressure		DEDLG	See Subsection 6.2.1.5

(1) Analysis is no longer required due to application of the LBB criteria (refer to UFSAR 6.2.1.2.3)

(2) This break is eliminated due to application of the LBB criteria (refer to UFSAR 6.2.1.2.3)

(3) The pre-EPU M&E release data slightly increase at EPU conditions (refer to UFSAR 6.2.1.2.3)

TABLE 6.2-3

PRINCIPAL CONTAINMENT DESIGN PARAMETERS

Parameter	Design	Margin[1]
Containment		
Internal design pressure, psig		
(LOCA original)	44.0	5.26%
(LOCA analysis to support EPU	44.0	1.29%
(MSLB)	44.0	14.80%
Differential design pressure, psid	0.70	16.6%
ILRT	43.48	
Net free volume, 10^6 ft^3	2.50	Not applicable
Design leak rate, percent free volume per day at 44.0 psig	0.5	Not applicable
Shield Building		
External design pressure, psig	3.0	387%
Subcompartments		
Reactor cavity design wall loading Spacially varied peak valve, psid	172	100%
Steam generator compartment design wall loading, psid	24.0	307%
Pressurizer compartment design upper wall loading, psid	14.0	977%
Pressurizer compartment design lower wall loading, psid	24.0	6%

NOTES:

(1) Margin (%) $\frac{\text{design value - peak calculated value}}{\text{peak calculated value}} \times 100$

Actual margin, i.e., the margin between design values and peak calculated values when using realistic or median parameter values would be much larger.

TABLE 6.2-4

SUMMARY OF CALCULATED CONTAINMENT PRESSURE AND TEMPERATURE

A. LOCA

	DESLS (max. SI, original)	DEHLS (min. SI, analysis to support EPU)
Break Size, ft^2	9.82	19.24
Peak Pressure, psig	41.8	43.43
Peak Temperature, F	271.4	266.66
Time of Peak Pressure, sec	210	19.8
Time to End of Blowdown, sec	20.4	11.70
Energy Release to Containment at End of Blowdown BTU x 10^{6*}	287.13	316.99

* Includes energy addition due to metal-water reaction.

TABLE 6.2-4 (Cont'd)

C. MSLB-COOLING TRAIN FAILURE

% Power	100.3	75	50	25	0
MSLB Break Size, ft^2	5.41	5.41	5.41	5.41	5.41
Peak Pressure, psig	35.06	35.59	33.51	34.11	37.49
Peak Temperature, F	380.97	377.28	374.32	371.47	367.56
Time of Peak Pressure, sec	63.74	63.60	123.91	126.21	204.50
Time of End of Blowdown, sec	104.0	119.0	139.0	179.0	229.0
Energy release to Containment at End of Blow-down BTU x 10^6	253.6	259.3	269.9	289.6	335.9

D. MSLB-MAIN FEEDWATER ISOLATION VALVE FAILURE

% Power	100.3	75	50	25	0
MSLB Break Size, ft^2	5.41	5.41	5.41	5.41	5.41
Peak Pressure, psig	33.89	32.60	32.23	32.75	35.05
Peak Temperature, F	379.78	376.05	373.08	370.21	366.32
Time of Peak Pressure, sec	57.74	103.00	119.74	123.03	196.16
Time of End of Blowdown, sec	104.0	119.0	144.0	184.0	229.0
Energy release to Containment at-End of Blowdown BTU x 10^6	254.9	260.6	271.4	291.4	337.2

TABLE 6.2-4 (Cont'd)

E. MSLB-MAIN STEAM ISOLATION VALVE FAILURE

% Power	100.3	75	50	25	0
MSLB Break Size, ft^2	5.41	5.41	5.41	5.41	5.41
Peak Pressure, psig	35.05	33.53	32.98	33.50	35.63
Peak Temperature, F-	382.99	379.25	376.25	373.37	339.31
Time of Peak Pressure, sec	57.74	57.60	117.28	122.71	193.79
Time to End of Blowdown, sec	104.4	119.0	139.0	179.0	229.0
Energy release to Containment of-End of Blowdown BTU x 10^6	261.3	266.6	276.9	296.5	342.2

TABLE 6.2-5

INITIAL CONDITIONS FOR LOCA CONTAINMENT PEAK PRESSURE-TEMPERATURE ANALYSIS

Parameter	Value
Reactor Coolant System	
Reactor power level, MWt[1]	3030
Vessel inlet cold leg temperature, F	554
Pressurizer pressure, psia	2,250
Containment	
Pressure, psia	15.41
Temperature, F	120
Relative humidity, %	45
Component cooling water base temperature, F	105
Refueling water storage pool temperature, F	104
Net free volume, x 10^6 ft^3	2.499735
Stored Water	
Usable refueling water storage pool (minimum), gal	290,300 credited for safety analysis; 386,735 for ECCS/CS strainer submergence
Safety injection tanks, ft^3	5,952

NOTES:

(1) At design overpower to 100.3 percent and normal liquid levels.

TABLE 6.2-6

ENGINEERED SAFETY FEATURE SYSTEMS OPERATING ASSUMPTIONS FOR LOCA CONTAINMENT PEAK PRESSURE ANALYSIS

System/Item*	Design Capacity	Value Used for LOCA Peak Pressure
Passive Safety Injection System		
Number of accumulators (safety injection tanks)	4	4
Pressure, psig	500	650
Volume, ft^3/tank (total available)	1850	1488
Active Safety Injection Systems		
High-pressure safety injection		
Number of headers/pump	2	2
No. of lines/header	4	4
Number of pumps	2	2
Low-pressure safety injection		
No. of headers/pump	1	1
Number of lines/header	2	2
Number of pumps	2	2
Containment Spray System		
Number of lines	2	1
Number of pumps	2	1
Number of headers	2	1
Flow rate, gpm/pump	3450 (runout)	2700
Containment Fan Coolers		
Number of Units	4	2

* See analysis (Ref. 41) for additional inputs

TABLE 6.2-6 (Cont'd)

System/Item	Design Capacity	Value Used for LOCA Peak Pressure Analyses
Heat Removal Rate, 10^6 BTU/hr	277.2 @ 280 F 128.16 @ 200 F 15.66 @ 120 F	267.2* @ 280 F 120.8* @ 200 F 11.76* @ 120 F *Accounts for 1% voltage under frequency
Cooling Water flowrates, gpm/unit	1200	1518
Source of Cooling Water	Component Cooling Water	Component Cooling Water
Base Cooling Water Temperature, F	100	105
Heat Exchangers		
Shutdown heat exchangers (shell and U-Tube)		
Number	2	1
Heat transfer area, ft^2	5790/Hx	5790 / Hx
Overall heat transfer coefficient, Btu/hr-ft^2-F	274.0	274.0
Flow rates:		
Tube side, gpm	3450	2850
Shell side, gpm	4820/Hx	3719/Hx
Temperature:		
Time Dependent	---	---
Source of cooling water	Component cooling water	Component cooling water

TABLE 6.2-7

CONTAINMENT PASSIVE HEAT SINKS

Description	Material	Thickness	Surface Area (Ft^2)	Exposure Side 1	Exposure Side 2	Thermal Conductivity (BTU/hr-ft-F)	Volumetric Heat Capacity (BTU/ft^3-F)
1. Containment Concrete Structures (walls and floor slabs (4)	Paint	4-7 mils	All concrete surfaces are painted			.25-.375	47.1
	Surfaces	4-7 mils				.16	28.0
	Concrete Surfaces					1.0	31.9
	Secondary Shield	4'	1451	Cont liquid	*		
	Wall	4'	20438	Cont vapor	*		
	Primary Shield	7'3"	814	Cont liquid	*		
	Wall	7'3"	1943	Cont vapor	*		
		5'11"	76	Cont liquid	*		
		5'11"	9894	Cont vapor	*		
		3'	21	Cont liquid	*		
		3'	1195	Cont vapor	*		
	Massive Wall (interior surface only)	20'	725	Cont liquid	Ground		
	Shield Wall	2'	540	Cont liquid	*		
		2'	3180	Cont vapor	*		
	Ground Floor at El 23.0' and El 18.0' Heavy Foundation (interior surface only)	20'	5485	Cont liquid	Ground		
			6035	Cont vapor			
	Operating Floor at El 62.0	4'	13450	Cont vapor	*		
	Steam Generator Shield Wall	4'	6390	Cont vapor	*		
	Pressurizer Shield Wall	2'	3050	Cont vapor	*		
	Missile Shield Cover	3'	851	Cont vapor	*		
	Hatch Cover	4'	670	Cont vapor	*		
	Ring Wall	2'	2760	Cont vapor	*		
	Reactor Cavity Floor	22'	380	Cont liquid	Ground		
	Sump Pit	3'	907	Cont liquid	(6)		
	Misc Walls	2'	1250	Cont vapor	*		

TABLE 6.2-7(Cont'd)

Description	Material	Thickness	Surface Area(Ft^2)	Exposure Side 1	Exposure Side 2	Thermal Conductivity (BTU/hr-ft-°F)	Volumetric Heat Capacity (BTU/ft^3-°F)
1. (Cont'd)	Concrete Steam Generator Base Vertical Wall	1.81 1.81	370 520	Cont vapor Cont liquid	(7) (7)	1.6	31.9
	Connecting Web	2'	1632	Cont vapor	*	1.0	31.9
	Electrical Tunnel	3'	5400	Cont liquid	(6)		
	Air Tunnel	3'	710	Cont liquid	(6)		
	Access Way	3 '	1717	Cont liquid	(6)		
	Vertical Surface Area for trenches Outside Secondary Shield Wall	3'	6410	Cont liquid	(6)	1.0	31.9
2. Uninsulated Pipe Stainless Steel (SS) Carbon Steel (CS)	SS 12" sch 40	0.406 in	519	Cont vapor		9.8	54.0
	SS 12" sch 40	0.406 in	183	Cont liquid		9.8	54.0
	SS 10" sch 40S	0.365 in	521	Cont vapor		9.8	54.0
	SS 8" sch 40S	0.322 in	3903	Cont vapor		9.8	54.0
	SS 6" sch 40S	0.280 in	1545	Cont vapor		9.8	54.0
	SS 1" sch 160S	0.250 in	5	Cont vapor		9.8	54.0
	SS 4" sch 40S	0.237 in	1245	Cont vapor		9.8	54.0
	SS 4" sch 40S	0.237 in	445(1)	Cont vapor		9.8	54.0
	SS 3/4" sch 160S	0.219 in	13	Cont vapor		9.8	54.0
	SS 2" sch 80S	0.218 in	374	Cont vapor		9.8	54.0
	SS 2" sch 80S	0.218 in	221	Cont liquid		9.8	54.0
	SS 3" sch 40S	0.216 in	512	Cont vapor		9.8	54.0
	SS 3" sch 40S	0.216 in	262	Cont liquid		9.8	54.0
	SS 1" sch 80S	0.179 in	643	Cont liquid		9.8	54
	SS 1" sch 80S	0.179 in	43	Cont vapor		9.8	54.0
	CS 2" sch 40	0.154 in	635	Cont liquid		25.9	53.57
	CS 2" sch 40	0.154 in	749(1)	Cont vapor		25.9	53.57
	SS 2" sch 40S	0.154 in	2	Cont vapor		9.8	54.0
	SS 1/2" sch 80S	0.147 in	3	Cont vapor		9.8	54.0
	CS 1" sch 40	0.133 in	163	Cont vapor		25.9	53.57
	SS 2-1/2", 3", 3-1/2", 4" sch 10S	0.120 in	763	Cont vapor		9.8	54.0
	SS 3/4" sch 40S	0.113 in	8	Cont vapor		9.8	54.0
	CS 1/2" sch 40	0.109 in	136	Cont vapor		25.9	53.57
3. Safety Injection Tanks(4)	Stainless Steel	1.785 in	4905	Cont vapor	(7)	9.8	54
4. Reactor Dr. Tk.	SS	5/16 in	230	Cont liquid	(7)	9.8	54

TABLE 6.2-7 (Cont'd)

	Description	Material	Thickness(in)	Surface Area (Ft^2)	Exposure Side 1	Exposure Side 2	Thermal Conductivity (BTU/hr-ft-F)	Volumetric Heat Capacity (BTU/ft^3-F)
5.	Quench Tank (painted)	SS	5/16	300	Cont vapor	(7)	9.8	54
6.	H_2 Recombiner	SS	2-1/2	200	Cont vapor	(7)		
7.	Refueling Machine	Steel	2-1/2	945	Cont vapor	Cont vapor	25.9	53.57
8.	Crane Steel	Steel	3/16	500	Cont vapor	Cont vapor	25.9	53.57
			7/32	5000	Cont vapor	Cont vapor	25.9	53.57
			1/4	120	Cont vapor	Cont vapor	25.9	53.57
			3/8	4800	Cont vapor	Cont vapor	25.9	53.57
			1/2	1100	Cont vapor	Cont vapor	25.9	53.57
			1-3/8	1900	Cont vapor	Cont vapor	25.9	53.57
9.	Instr. Housing (Painted)	Cast Aluminum	0.25	9	Cont vapor	(7)	120	34.3
10.	Instr. Racks	Galvanized	3.5 mils	310	Cont vapor	(7)	611.0	40.6
		Steel	.154				25.9	53.57
		Galvanized	3.5 mils	550	Cont vapor	Cont vapor	611.0	40.6
		Steel	.39				25.9	53.57
		Galvanized	3.5 mils				64.0	40.6
		Galvanized	3.5 mils	175	Cont vapor	Cont vapor	611.0	40.6
		Steel	.25				25.9	53.57
		Galvanized	3.5 mils				611.0	40.6
11.	Tubing	316 SS	0.065	3000	Cont vapor	Process fluid	9.8	54.0
12.	Fittings	316 SS	0.065	300	Cont vapor	Process fluid	9.8	54.0
13.	Instr Manifolds	316 SS	1.25	25	Cont vapor	Process fluid	9.8	54.0
14.	Piping seismic restraints and hangers	CS	.75	20000	Cont vapor	Cont vapor	25.9	53.57
		CS	.75	5000	Cont liquid	Cont liquid	25.9	53.57
15.	Electrical Conduit	Galvanized	.8 mil	31089	Cont vapor	(7)	64.0	40.6
		Steel	.154			(7)	25.9	53.57
		Galvanized	.8 mil	10328	Cont vapor	(7)	64.0	40.6
		Steel	.133			(7)	25.9	53.57
16.	Electrical boxes	Galvanized	2 mil	2185	Cont vapor	(7)	64.0	40.6
		Steel	.083			(7)	25.9	53.57

TABLE 6.2-7 (Cont'd)

Description	Material	Thickness(in)		Surface Area (Ft^2)	Exposure Side 1	Exposure Side 2	Thermal Conductivity (BTU/hr-ft-F)	Volumetric Heat Capacity (BTU/ft^3-F)
17. Cable Trays (without covers)	Galvanized	2.5 mil		13000	Cont vapor	Cont vapor	64.0	40.6
	Steel	.065					25.9	53.57
	Galvanized	2.5 mil					64.0	40.6
Cable Trays (with covers)	Galvanized	2.5 mil		14666	Cont vapor	(7)	64.0	40.
	Steel	.065		8666			25.9	53.57
	Steel	.049		6000			25.9	53.57
18. Tube Supports	Galvanized	3.5 mil		1875	Cont vapor	Cont vapor	64.0	40.6
	Steel	.109					25.9	53.57
	Galvanized	3.5 mil					64.0	40.6
19. Cable Tray Restraints	Steel	.25		7836	Cont vapor	Cont vapor	25.9	53.57
	Steel	.3125		718	Cont vapor	Cont vapor	25.9	53.57
	Steel	.375		315	Cont vapor	Cont vapor	25.9	53.57
	Steel	.1875		1479	Cont vapor	Cont vapor	25.9	53.57
	Steel	.235		314	Cont vapor	Cont vapor	25.9	53.57
	Steel	.269		628	Cont vapor	Cont vapor	25.9	53.57
	Steel	.5		141	Cont vapor	Cont vapor	25.9	53.57
	Steel	1.0		316	Cont vapor	Cont vapor	25.9	53.57
20. Cont Vessel Cylinder (3)	Steel	1.92		55900	Cont vapor	Annulus	25.9	53.57
21. Cont Vessel Dome (3)	Steel	0.96		30800	Cont vapor	Annulus	25.9	53.57
22. Refueling Pool Exposed Steel	Stainless Steel	0.1875	3' Conc Wall	1000	Cont vapor	Concrete	25.9	53.57
			6' Conc Wall	3300				
			7' Conc Wall	1800				
			20' Conc Wall	2200				
23. Gratings, Ladders, Misc.	Galvanized	3.5 mils		18383	Cont vapor	Cont vapor	64.0	40.6
	Steel	.1875		13674			25.9	53.57
	Steel	.3125		3610			25.9	53.57
	Steel	.375		681			25.9	53.57
	Steel	.5		418			25.9	53.57
	Galvanized	3.5 mils					64.0	40.6
	Galvanized	3.5 mils		660	Cont vapor	(7)	64.0	40.6
	Steel	.1875		207			25.9	53.57
	Steel	2.0		453			25.9	53.57
24. Handrails and Treads	Galvanized	3.5 mils		5260	Cont vapor	(7)	64.0	40.6
	Steel	.14					25.9	53.57
	Aluminum	.4375		674	Cont vapor	Cont vapor	120.	34.3

TABLE 6.2-7 (Cont'd)

Description	Material	Thickness(in)	Surface Area (Ft^2)	Exposure Side 1	Exposure Side 2	Thermal Conductivity (BTU/hr-ft-F)	Volumetric heat Capacity (BTU/ft^3-F)
25. Trench Covers	Steel	0.25	465	Cont vapor	Cont vapor	25.9	53.57
26. Elevator	Steel	0.25	2675	Cont vapor	Cont vapor	25.9	53.57
27. HVAC Duct	Galvanized	1.53 mil	23418	Cont vapor	Cont vapor	64.0	40.6
	Steel	.1046				25.9	53.57
	Galvanized	1.53 mil				64.0	40.6
28. HVAC Casings	Galvanized	1.53 mil	2045	Cont vapor	Cont vapor	64.0	40.6
	Steel	.1345				25.9	53.57
	Galvanized	1.53 mil				64.0	40.6
29. HVAC Fans	Steel	.375	5629	Cont vapor	Cont vapor	25.9	53.57
30. Embedded Steel(8)	Steel	.75	724	Cont vapor	Concrete	25.9	53.57
	Steel	1.0	2211	Cont vapor	Concrete	25.9	53.57
	Steel	1.5	15	Cont vapor	Concrete	25.9	53.57
	Steel	1.375	1583	Cont vapor	Concrete	25.9	53.57
	Steel	.75	180	Cont liquid	Concrete	25.9	53.57
	Steel	1.0	1458	Cont liquid	Concrete	25.9	53.57
		1.5	41	Cont liquid	Concrete	25.9	53.57
31. Pipe Restraints	Steel	.25	490	Cont vapor	Cont vapor	25.9	53.57
	Steel	.375	1248	Cont vapor	Cont vapor	25.9	53.57
	Steel	.42	1305	Cont vapor	Cont vapor	25.9	53.57
	Steel	.5	898	Cont vapor	Cont vapor	25.9	53.57
	Steel	.5625	208	Cont vapor	Cont vapor	25.9	53.57
	Steel	.625	1759	Cont vapor	Cont vapor	25.9	53.57
	Steel	.688	967	Cont vapor	Cont vapor	25.9	53.57
	Steel	.72	533	Cont vapor	Cont vapor	25.9	53.57
	Steel	.75	2161	Cont vapor	Cont vapor	25.9	53.57
	Steel	.8125	179	Cont vapor	Cont vapor	25.9	53.57
	Steel	1.0	3613	Cont vapor	Cont vapor	25.9	53.57
	Steel	1.25	1352	Cont vapor	Cont vapor	25.9	53.57
	Steel	1.375	1200	Cont vapor	Cont vapor	25.9	53.57
	Steel	1.5	4074	Cont vapor	Cont vapor	25.9	53.57
	Steel	1.75	96	Cont vapor	Cont vapor	25.9	53.57
	Steel	2.0	788	Cont vapor	Cont vapor	25.9	53.57
	Steel	2.25	530	Cont vapor	Cont vapor	25.9	53.57
	Steel	2.5	274	Cont vapor	Cont vapor	25.9	53.57
	Steel	2.75	280	Cont vapor	Cont vapor	25.9	53.57
	Steel	3.25	150	Cont vapor	Cont vapor	25.9	53.57
		3.5	60	Cont vapor	Cont vapor		

TABLE 6.2-7 (Cont'd)

Description	Material	Thickness(in)	Surface Area (Ft^2)	Exposure Side 1	Exposure Side 2	Thermal Conductivity (BTU/hr-ft-F)	Volumetric Heat Capacity (BTU/ft^3-F)
31. (Cont'd)	Steel(8)	4.0	56	Cont vapor	Concrete	25.9	53.57
	Steel(8)	2.25	18	Cont vapor	Concrete	25.9	53.57
	Steel(8)	1.5	2117	Cont vapor	Concrete	25.9	53.57
	Steel(8)	2.5	130	Cont vapor	Concrete	25.9	53.57
	Steel(8)	2.0	420	Cont vapor	Concrete	25.9	53.57
	Steel(8)	1.0	268	Cont vapor	Concrete	25.9	53.57
	Steel(8)	1.25	20	Cont vapor	Concrete	25.9	53.57
	Steel(8)	1.75	51	Cont vapor	Concrete	25.9	53.57
	Steel(8)	3.0	15	Cont vapor	Concrete	25.9	53.57
	Steel(8)	1.625	17	Cont vapor	Concrete	25.9	53.57
	Steel(8)	2.25	20	Cont liquid	Concrete	25.9	53.57
	Steel(8)	1.0	155	Cont liquid	Concrete	25.9	53.57
	Steel(8)	2.25	31	Cont liquid	Cont liquid	25.9	53.57
	Steel(8)	2.0	278	Cont liquid	Cont liquid	25.9	53.57
	Steel(8)	1.0	50	Cont liquid	Cont liquid	25.9	53.57
	Steel(8)	1.25	20	Cont liquid	Cont liquid	25.9	53.57
	Steel(8)	.683	30	Cont liquid	Cont liquid	25.9	53.57
	Steel(8)	.5	32	Cont liquid	Cont liquid	25.9	53.57
	Steel(8)	.375	40	Cont liquid	Cont liquid	25.9	53.57
32. Equipment Supports	Steel	.38	212	Cont vapor	Cont vapor	25.9	53.57
	Steel	.5	222	Cont vapor	Cont vapor	25.9	53.57
	Steel	.69	72	Cont vapor	Cont vapor	25.9	53.57
	Steel	.75	445.8	Cont vapor	Cont vapor	25.9	53.57
	Steel	1.0	341	Cont vapor	Cont vapor	25.9	53.57
	Steel	1.12	35	Cont vapor	Cont vapor	25.9	53.57
	Steel	1.5	763.5	Cont vapor	Cont vapor	25.9	53.57
	Steel	1.75	39	Cont vapor	Cont vapor	25.9	53.57
	Steel	2.0	2202.6	Cont vapor	Cont vapor	25.9	53.57
	Steel	2.5	289	Cont vapor	Cont vapor	25.9	53.57
	Steel	4.0	504.8	Cont vapor	Cont vapor	25.9	53.57
	Steel	5.0	78	Cont vapor	Cont vapor	25.9	53.57
	Steel	3.0	853.4	Cont vapor	Cont vapor	25.9	53.57
	Steel	.25	21	Cont vapor	Cont vapor	25.9	53.57
	Steel	1.25	570.6	Cont vapor	Cont vapor	25.9	53.57
	Steel	2.0	482.6	Cont vapor	Insulated	25.9	53.57
	Steel	4.0	99.2	Cont vapor	Insulated	25.9	53.57
	Steel	1.0	500	Cont liquid	Cont liquid	25.9	53.57
	Steel	1.5	397.6	Cont liquid	Cont liquid	25.9	53.57
	Steel	2.5	112	Cont liquid	Cont liquid	25.9	53.57
	Steel	3.0	681	Cont liquid	Cont liquid	25.9	53.57
	Steel	4.0	1039	Cont liquid	Cont liquid	25.9	53.57
	Steel	5.0	357	Cont liquid	Cont liquid	25.9	53.57
	Steel	6.0	288	Cont liquid	Cont liquid	25.9	53.57
	Steel	3.5	185	Cont liquid	Cont liquid	25.9	53.57
	Steel	2.0	206.7	Cont liquid	Cont liquid	25.9	53.57

TABLE 6.2-7(Cont'd)

Description	Material	Thickness (in)	Surface Area(Ft^2)	Exposure Side 1	Exposure Side 2	Thermal Conductivity (BTU/hr-ft-F)	Volumetric Heat Capacity (BTU/ft^3-F)
32. (Cont'd)	Steel	.25	10	Cont liquid	Cont liquid	25.9	53.57
	Steel	1.25	31	Cont liquid	Cont liquid	25.9	53.57
	Steel	.5	377	Cont liquid	Cont liquid	25.9	53.57
	Steel	1.75	107	Cont liquid	Cont liquid	25.9	53.57
	Steel	2.75	336	Cont liquid	Cont liquid	25.9	53.57
	Steel	2.625	76	Cont liquid	Cont liquid	25.9	53.57
	Steel	4.25	382	Cont liquid	Cont liquid	25.9	53.57
	Steel	.75	5.3	Cont liquid	Cont liquid	25.9	53.57
	Steel[8]	1.25	159.5	Cont vapor	Concrete	25.9	53.57
	Steel[8]	1.0	266.8	Cont vapor	Concrete	25.9	53.57
	Steel[8]	1.5	90.2	Cont vapor	Concrete	25.9	53.57
	Steel[8]	3.0	46	Cont vapor	Concrete	25.9	53.57
	Steel[8]	.5	25.5	Cont vapor	Concrete	25.9	53.57
	Steel[8]	2.0	16.4	Cont vapor	Concrete	25.9	53-57
	Steel[8]	.75	2.7	Cont vapor	Concrete	25.9	53.57
	Steel[8]	2.5	21.7	Cont vapor	Concrete	25.9	53.57
	Steel[8]	.5	3868.0	Cont liquid	Concrete	25.9	53.57
	Steel[8]	1.0	2.3	Cont liquid	Concrete	25.9	53.57
	Steel[8]	2.0	94.6	Cont liquid	Concrete	25.9	53.57
	Steel[8]	3.0	116.1	Cont liquid	Concrete	25.9	53.57
			142.2	Cont liquid	Concrete	25.9	53.57
	Steel	4.0					
	Steel[8]	.75	1.4	Cont liquid	Concrete	25.9	53.57
	Steel[8]	12.0	144.0	Cont liquid	Concrete	25.9	53.57
33. HVAC Duct Restraints	Steel	.1875	81	Cont vapor	Cont vapor	25.9	53.57
		.25	6721	Cont vapor	Cont vapor	25.9	53.57
		.288	206	Cont vapor	Cont vapor	25.9	53.57
		.3125	536	Cont vapor	Cont vapor	25.9	53.57
		.345	293	Cont vapor	Cont vapor	25.9	53.57
		.5	414	Cont vapor	Cont vapor	25.9	53.57
		.433	264	Cont vapor	Cont vapor	25.9	53.57
		.375	335	Cont vapor	Cont vapor	25.9	53.57
37. Containment Sump Strainer	SS	.048	5607	Cont liquid	Cont liquid	9.8	54
	SS	.375	25	Cont liquid	Cont liquid	9.8	54
	SS	.5	214	Cont liquid	Cont liquid	9.8	54
	SS	.5	27	Cont liquid	Concrete	9.8	54

TABLE 6.2-7 (Cont'd)

Notes:

(1) Piping is painted

(2) For all carbon steel surfaces see Section 6.1.2 for paint system specifications.

(3) For the steel containment vessel see Section 6.1.2 for paint system specifications.

(4) The given surface area for a concrete wall has been reduced to represent both sides of the wall since the faces often have different areas. A "*" in the exposure side 2 column identifies where this has been done and should be taken to mean both sides are exposed to the environment of exposure side one. The given thickness is that of the full wall.

(5) When both sides are exposed to containment environment - the area is of one side only.

(6) Exposure on side 2 is not applicable since it ranges from surfaces previously given to ground.

(7) The interior surface is not readily subject to the exposure of side 1 since that environment has no free path to the inner surface. The surface area represents the outer surface only and the heat sink is considered to be the full thickness.

(8) Where the steel is buried in concrete or backed by concrete this is indicated in the exposure side 2 column. In these cases the surface area represents that area exposed to the environment of the exposure side 1 column.

TABLE 6.2-8

SUMMARY OF PASSIVE HEAT SINKS
USED FOR THE CONTAINMENT ANALYSIS

Description	Thickness	Surface Area(Ft^2)	Exposure Side 1	Exposure Side 2	Thermal Conductivity (BTU/hr-ft-F)	Volumetric Heat Capacity (BTU/ft^3-F)
1. Containment Primary Cylinder	Top Coat - *	55,900	Cont vapor	Annulus	*	*
	Bottom coat - *				*	*
	Steel shell - 1.92				25.9	53.57
	Paint film - *				*	*
2. Containment Primary Dome	Top coat - *	30,800	Cont vapor	Annulus	*	*
	Bottom coat - *				*	*
	Steel dome - .96				25.9	53.57
	Paint film - *				*	*
3. Ground Floor - Concrete	Paint film - *	5,733.3	Cont vapor	Ground	*	*
	Surfacer - *				*	*
	Concrete - 2. ft				1.0	31.9
4. Concrete (shield walls concrete pads)	Paint film - *	64,023.4	Cont vapor	Insulated	*	*
	Surfacer - *				*	*
	Concrete - 2. ft				1.0	31.9
5. Concrete exposed to Containment Sump Water	Paint film - *	23,898.2	Cont liquid	Insulated	*	*
	Surfacer - *				*	*
	Concrete - 3. ft				1.0	31.9
6. Stainless Steel -	Stainless Steel -	8,300	Cont vapor	Insulated	9.8	54.0
	Concrete - 4. ft				1.0	31.9
7. Stainless Steel (piping other components)	Stainless Steel -	15,618	Cont vapor	Insulated	9.8	54.0
8. Galvanized steel and cable trays)	a) Zinc - .8 mil	41,417	Cont vapor	Insulated	64.0	40.6
	Steel - .1488 in				25.9	53.57
	b) Zinc - 1.53 mil	25,463	Cont vapor	Cont vapor	64.0	40.6
	Steel - .107 in				25.9	53.57
	Zinc - 1.53 mil				64.0	40.6
	c) Zinc - 3 mil	91,047	Cont vapor	Insulated	64.0	40.6
	Steel - .09041 in				25.9	53.57

* Coating specifications are described in Section 6.1.2.

TABLE 6.2-8 (cont'd)
SUMMARY OF PASSIVE HEAT SINKS USED FOR THE CONTAINMENT ANALYSIS

Description	Thickness		Surface Area(Ft^2)	Exposure Side 1	Exposure Side 2	Thermal Conductivity (BTU/hr-ft-F)	Volumetric Heat Capacity (BTU/ft^3-F)
9. Structural and Miscellaneous Exposed Steel	a)	Top coat - * Bottom coat - * Steel - .25 in Bottom coat - * Top coat - *	22,769	Cont vapor	Cont vapor	* * 25.9 * *	* * 53.57 * *
	b)	Top coat - * Bottom coat - * Steel - .375 in Bottom coat - * Top coat - *	12,539	Cont vapor	Cont vapor	* * 25.9 * *	* * 53.57 * *
	c)	Top coat - * Bottom coat - * Steel - .5 in Bottom coat - * Top coat - *	6,311	Cont vapor	Cont vapor	* * 25.9 * *	* * 53.57 * *
	d)	Top coat - * Bottom coat - * Steel - .75 in Bottom coat - * Top coat - *	24,357.8	Cont vapor	Cont vapor	* * 25.9 * *	* * 53.57 * *
	e)	Top coat - * Bottom coat - * Steel - 1.52141 in Bottom coat - * Top coat - *	19,328.7	Cont vapor	Cont vapor	* * 25.9 * *	* * 53.57 * *
	f)	Top coat - * Bottom coat - * Steel - 3.34176 in Bottom coat - * Top coat - *	1,926.2	Cont vapor	Cont vapor	* * 25.9 * *	* * 53.57 * *
	g)	Top coat - * Bottom coat - * Steel - .11206 in	12,276	Cont vapor	Insulated	* * 25.9	* * 53.57
	h)	Top coat - * Bottom coat - * Steel - 1.39856 in	6,970.6	Cont vapor	Insulated	* * 25.9	* * 53.57

See Analysis Reference (27) for assumed coating value. Coating specifications are described in Section 6.1.2.

TABLE 6.2-9

ACCIDENT CHRONOLOGIES

Limiting Double Ended Hot Leg Slot Break Sequence of Events
(Minimum SI Flow, 1 Containment Spray, 2 Containment Fan Coolers

TIME (SECONDS)	EVENT	SETPOINT/VALUE
0.0	Break Occurs, Reactor Trip and Loss of Offsite Power are Assumed	19.24 ft² break
0.59	Containment High-High Pressure Setpoint Reached	7.05 psig
11.702	End of Blowdown Phase	
19.80	Containment Peak Pressure and Temperature Occurs	43.43 psig, 266.66°F
24.15	Containment Fan Coolers Begin Removing Heat from Containment	
29.45	Containment Spray Flow Begins	
75.45	Full Containment Spray Flow	
2412.03	Switchover to sump recirculation as RWT reaches low level. Containment spray water cooled by Shutdown Cooling heat exchanger.	@ RWT injected volume of 290,300 gallons credited for safety analysis
1.0e+06	Transient Modeling Terminated	

TABLE 6.2-9 (cont'd)

ACCIDENT CHRONOLOGIES

Page 2 of this table has been deleted.

TABLE 6.2-9 (cont'd)

ACCIDENT CHRONOLOGIES

EPU

D. Worst Pressure Case MSLB

Time (seconds)	EVENT
0.0	Break occurs
3.14	Main Steam Isolation Setpoint
3.14	Reactor Trip Setpoint
3.14	Main Feedwater Isolation Setpoint
3.55	Containment Spray Setpoint
9.44	Main Feedwater Isolation Valves Closed
11.04	Main Steam Isolation Valves Closed
13.55	Full Fan Cooler Operation
63.2	Containment Spray on Full
204.50	Containment Peak Pressure
229.0	End of Blowdown

E. Worst Temperature Case MSLB

Time (seconds)	EVENT
0.0	Break Occurs
4.7	Main Steam Isolation Setpoint
4.7	Reactor Trip Setpoint
4.7	Main Feedwater Isolation Setpoint
5.22	Containment Spray Setpoint
11.0	Main Feedwater Isolation Valves Closed
12.6	Main Steam Isolation Valves Closed
14.87	Full Fan Cooler Operation
51.87	Containment Peak Temperature
64.87	Containment Spray on Full
110.0	End of Blowdown

PAGE INTENTIONALLY LEFT BLANK

Table 6.2-10
This table has been deleted.

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

TABLE 6.2-11

ASSUMPTIONS USED IN ANALYSIS OF INADVERTENT CONTAINMENTS

SPRAY SYSTEM ACTUATION

Cycle 1 | 1

Item	Assumed Value
Containment	
Initial temperature, F	90
Initial pressure, psia	14.4292
Relative humidity, %	20
Net free volume (minimum), $ft.^3$	2.50×10^6
Passive heat sinks	ignored for conservatism
Containment Spray System	
Number of trains in operation	2
Flowrate per train, gpm (runout flow)	3450
Refueling water temperature, F	55
Spray efficiency, %	100
Containment Fan Coolers	
Number of units in operation	4
Component Cooling Water temperature, F	60
Fan Cooler Heat Removal/unit (10^6Btu)	11.40 @ 120 F air 6.053 @ 100 F air 1.190 @ 70 F air
Containment Vacuum Breakers	
No. of Vacuum Breaker	2
No. assumed failed	1
Setpoint differential pressure to open Vacuum Breakers, psid	0.3683
Delay time to start opening Vacuum Breakers, sec	1.15
Vacuum Break System flow area, $ft.^2$	1.77
Loss coefficient	1.64

TABLE 6.2-11 (Cont'd)

Item	Assumed Value
Shield Building	
Initial temperature, F	56
Initial pressure, psia	14.7
Relative humidity, %	16
Net free volume (minimum), ft^3	547,000
Passive heat sinks	ignored for conservatism

Table 6.2-12A
This table has been deleted.

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

Table 6.2-12B
This table has been deleted.

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

Table 6.2-12D
This table has been deleted.

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

Table 6.2-12E
This table has been deleted

TABLE 6.2-12F

0% POWER MSLB, WITH ONE CONTAINMENT COOLING TRAIN (EPU)

TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC		TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC
0.00	8029.28	9.60E+06		7.999	5121.61	6.17E+06
0.199	7772.40	9.30E+06		8.199	5090.55	6.13E+06
0.399	7599.16	9.10E+06		8.399	5060.12	6.09E+06
0.599	7459.57	8.93E+06		8.599	5030.31	6.06E+06
0.799	7338.00	8.79E+06		8.799	5001.09	6.02E+06
0.999	7227.34	8.66E+06		8.999	4972.45	5.99E+06
1.199	7123.94	8.54E+06		9.199	4944.36	5.95E+06
1.399	7025.80	8.42E+06		9.399	4916.79	5.92E+06
1.599	6931.82	8.31E+06		9.599	4895.66	5.89E+06
1.799	6841.32	8.21E+06		9.799	4876.68	5.87E+06
1.999	6753.85	8.10E+06		9.999	4858.35	5.85E+06
2.199	6669.14	8.00E+06		10.199	4840.63	5.83E+06
2.399	6586.99	7.91E+06		10.399	4823.03	5.81E+06
2.599	6507.25	7.81E+06		10.599	4805.81	5.79E+06
2.799	6430.98	7.72E+06		10.799	4789.09	5.77E+06
2.999	6357.53	7.64E+06		10.999	4772.81	5.75E+06
3.199	6286.07	7.55E+06		11.199	2315.28	2.79E+06
3.399	6216.60	7.47E+06		11.399	2310.09	2.78E+06
3.599	6149.08	7.39E+06		11.599	2304.80	2.77E+06
3.799	6083.48	7.31E+06		11.799	2299.40	2.77E+06
3.999	6020.23	7.24E+06		11.999	2293.89	2.76E+06
4.199	5959.79	7.16E+06		12.199	2288.27	2.75E+06
4.399	5901.15	7.09E+06		12.399	2282.55	2.75E+06
4.599	5844.29	7.03E+06		12.599	2276.72	7.24E+06
4.799	5789.18	6.96E+06		12.799	2270.80	2.73E+06
4.999	5735.80	6.90E+06		12.999	2264.77	2.73E+06
5.199	5684.12	6.84E+06		13.199	2258.66	2.72E+06
5.399	5634.72	6.78E+06		13.399	2252.45	2.71E+06
5.599	5586.90	6.72E+06		13.599	2246.14	2.70E+06
5.799	5540.51	6.67E+06		13.799	2239.75	2.70E+06
5.999	5495.59	6.61E+06		13.999	2233.27	2.69E+06
6.199	5452.15	6.56E+06		14.199	2226.70	2.68E+06
6.399	5410.16	6.51E+06		14.399	2220.05	2.67E+06
6.599	5369.59	6.46E+06		14.599	2213.31	2.67E+06
6.799	5330.42	6.41E+06		14.799	2206.48	2.66E+06
6.999	5292.58	6.37E+06		14.999	2199.57	2.65E+06
7.199	5256.03	6.33E+06		15.199	2192.58	2.64E+06
7.399	5220.72	6.28E+06		15.399	2185.51	2.63E+06
7.599	5186.59	6.24E+06		15.599	2178.36	2.62E+06
7.799	5153.57	6.20E+06		15.799	2171.13	2.61E+06
15.999	2163.84	2.61E+06		59.995	1350.80	1.63E+06
16.199	2156.47	2.60E+06		61.995	1333.29	1.60E+06
16.399	2149.04	2.59E+06		63.995	1316.20	1.58E+06
16.599	2141.56	2.58E+06		65.995	1299.49	1.56E+06

TABLE 6.2-12F (continued)

0% POWER MSLB, WITH ONE CONTAINMENT COOLING TRAIN (EPU)

TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC		TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC
16.799	2134.02	2.57E+06		67.995	1283.15	1.54E+06
16.999	2126.43	2.56E+06		69.995	1267.16	1.52E+06
17.199	2118.80	2.55E+06		71.995	1251.50	1.50E+06
17.399	2111.14	2.54E+06		73.995	1236.16	1.49E+06
17.599	2103.46	2.53E+06		75.995	1221.14	1.47E+06
17.799	2095.49	2.52E+06		77.995	1206.41	1.45E+06
17.999	2087.39	2.51E+06		79.995	1191.97	1.43E+06
18.199	2079.29	2.50E+06		81.995	1177.80	1.42E+06
18.399	2071.20	2.49E+06		83.995	1163.89	1.40E+06
18.599	2063.13	2.48E+06		85.995	1150.10	1.38E+06
18.799	2055.08	2.48E+06		87.995	1135.94	1.36E+06
18.999	2047.07	2.47E+06		89.995	1121.71	1.35E+06
19.199	2039.10	2.46E+06		91.995	1107.47	1.33E+06
19.399	2031.18	2.45E+06		93.995	1093.28	1.31E+06
19.599	2023.31	2.44E+06		95.995	1079.28	1.30E+06
19.799	2015.51	2.43E+06		97.995	1065.55	1.28E+06
19.999	2007.77	2.42E+06		99.995	1052.10	1.26E+06
21.995	1935.40	2.33E+06		104.995	1019.59	1.22E+06
23.995	1872.89	2.26E+06		109.995	988.681	1.19E+06
25.995	1819.09	2.19E+06		114.995	959.245	1.15E+06
27.995	1771.23	2.13E+06		119.995	931.102	1.12E+06
29.995	1727.58	2.08E+06		124.995	904.243	1.08E+06
31.995	1687.73	2.03E+06		129.995	877.773	1.05E+06
33.995	1651.73	1.99E+06		134.995	852.363	1.02E+06
35.995	1619.30	1.95E+06		139.995	828.001	991238
37.995	1589.79	1.91E+06		144.995	804.709	962997
39.995	1562.45	1.88E+06		149.995	782.502	936074
41.995	1536.74	1.85E+06		154.995	760.458	909352
43.995	1512.39	1.82E+06		159.995	733.124	876223
45.995	1489.27	1.79E+06		164.995	703.800	840688
47.995	1467.26	1.77E+06		169.995	674.115	804725
49.995	1446.20	1.74E+06		174.995	642.957	766990
51.995	1425.92	1.72E+06		179.995	610.019	727112
53.995	1406.29	1.69E+06		184.995	574.609	684261
55.995	1387.26	1.67E+06		189.995	538.914	641088
57.995	1368.77	1.65E+06		194.995	502.725	597346
199.995	459.135	544661		399.995	0.803132	999.887
204.995	407.664	482454		404.995	0.786561	979.344
209.995	347.018	409264		409.995	0.770528	959.464
214.995	196.073	230433		414.995	0.755003	940.211
219.995	128.902	151448		419.995	0.739968	921.562
224.995	94.3073	110783		424.995	0.725401	903.49
229.995	2.92246	3601.28		429.995	0.711282	885.972
234.995	2.28154	2813.74		434.995	0.697593	868.985

TABLE 6.2-12F (continued)

0% POWER MSLB, WITH ONE CONTAINMENT COOLING TRAIN (EPU)

TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC		TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC
239.995	2.18982	2703.4		444.995	0.671436	836.519
244.995	1.99445	2463.65		449.995	0.658934	820.998
249.995	1.97833	2445.48		454.995	0.646796	805.927
254.995	1.81987	2250.79		459.99	0.574523	715.922
259.995	1.80126	2228.99		464.99	0.565348	704.532
264.995	1.68609	2087.44		469.99	0.557058	694.243
269.995	1.65446	2049.19		474.99	0.548922	684.143
274.995	1.57072	1946.26		479.99	0.541051	674.372
279.995	1.53023	1896.8		484.99	0.533355	664.817
284.995	1.46718	1819.28		489.99	0.525874	655.528
289.995	1.42317	1765.28		494.99	0.518577	646.468
294.995	1.37243	1702.86		499.99	0.511466	637.638
299.995	1.32971	1650.31		504.99	0.504975	629.578
304.995	1.28703	1597.76		509.99	0.499434	622.703
309.995	1.24751	1549.07		514.99	0.493985	615.941
314.995	1.21251	1505.95		519.99	0.488614	609.275
319.995	1.17860	1464.15		524.99	0.483330	602.716
324.995	1.14690	1425.05		529.99	0.478132	596.264
329.995	1.11605	1386.98		534.99	0.473022	589.92
334.995	1.08702	1351.14		539.99	0.467999	583.684
339.995	1.05888	1316.38		544.99	0.463061	577.553
344.995	1.03218	1283.4		549.99	0.458207	571.526
349.995	1.00643	1251.56		554.99	0.453435	565.601
354.995	0.981836	1221.15		559.99	0.448744	559.776
359.995	0.958811	1192.67		564.99	0.444133	554.05
364.995	0.936754	1165.38		569.99	0.439599	548.419
369.995	0.915514	1139.09		574.99	0.435140	542.882
374.995	0.895063	1113.78		579.99	0.430756	537.437
379.995	0.875370	1089.39		584.99	0.426445	532.082
384.995	0.856362	1065.85		589.99	0.422205	526.815
389.995	0.838007	1043.11		594.99	0.418034	521.634
394.995	0.820270	1021.13		599.99	0.413932	516.538
439.995	0.684317	852.507		600	0.413924	516.528

TABLE 6.2-12G

EQ 100.3% POWER MSLB, WITH ONE FAILED MSIV (EPU)

TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC		TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC
0.00	7869.13	9.41E+06		8.799	5562.57	6.69E+06
0.199	7484.04	8.96E+06		8.999	5600.89	6.73E+06
0.399	7179.86	8.60E+06		9.199	5635.77	6.77E+06
0.599	6915.73	8.23E+06		9.399	5667.55	6.81E+06
0.799	6690.32	8.02E+06		9.599	5696.54	6.85E+06
0.999	6491.27	7.79E+06		9.799	5723.02	6.88E+06
1.199	6315.41	7.58E+06		9.999	5747.24	6.91E+06
1.399	6159.88	7.39E+06		10.199	5769.39	6.93E+06
1.599	6022.14	7.23E+06		10.399	5789.65	6.96E+06
1.799	5897.86	7.08E+06		10.599	5808.16	7.00E+06
2.199	5685.94	6.83E+06		10.999	5840.36	7.02E+06
2.399	5597.66	6.72E+06		11.199	5857.85	7.04E+06
2.599	5519.10	6.63E+06		11.399	5873.88	7.06E+06
2.799	5448.98	6.54E+06		11.599	5888.49	7.08E+06
2.999	5386.20	6.47E+06		11.799	5901.68	7.09E+06
3.199	5329.82	6.40E+06		11.999	5913.46	7.11E+06
3.399	5279.02	6.34E+06		12.199	5923.81	7.12E+06
3.599	5233.11	6.28E+06		12.399	5932.71	7.13E+06
3.799	5191.47	6.23E+06		12.599	5937.69	7.14E+06
3.999	5153.59	6.19E+06		12.799	5709.43	6.86E+06
4.199	5119.01	6.15E+06		12.999	5496.58	6.61E+06
4.399	5087.33	6.11E+06		13.199	5290.39	6.36E+06
4.599	5058.21	6.07E+06		13.399	5099.78	6.13E+06
4.799	5031.34	6.04E+06		13.599	4923.58	5.92E+06
4.999	5006.45	6.01E+06		13.799	4760.71	5.72E+06
5.199	4983.31	5.98E+06		13.999	4610.13	5.54E+06
5.399	4961.71	5.96E+06		14.199	4470.87	5.37E+06
5.599	4941.47	5.93E+06		14.399	4341.99	5.21E+06
5.799	4922.42	5.91E+06		14.599	4222.66	5.07E+06
5.999	4904.43	5.89E+06		14.799	4112.08	4.94E+06
6.199	4887.37	5.87E+06		14.999	4009.30	4.83E+06
6.399	4871.14	5.85E+06		15.199	3910.22	4.71E+06
6.599	4855.66	5.83E+06		15.399	3818.50	4.59E+06
6.799	4840.84	5.81E+06		15.599	3733.47	4.49E+06
6.999	4945.00	5.94E+06		15.799	3654.51	4.40E+06
7.199	5043.63	6.06E+06		15.999	3581.09	4.31E+06
7.399	5133.10	6.17E+06		16.199	3512.71	4.22E+06
7.599	5214.20	6.27E+06		16.399	3448.97	4.15E+06
7.799	5287.67	6.36E+06		16.599	3392.58	4.08E+06
7.999	5354.20	6.44E+06		16.799	3340.97	4.02E+06
8.199	5414.42	6.51E+06		16.999	3292.28	3.96E+06
8.399	5468.91	6.57E+06		17.199	3246.32	3.90E+06
8.599	5518.22	6.63E+06		17.399	3202.86	3.85E+06

TABLE 6.2-12G (continued)

EQ 100.3% POWER MSLB, WITH ONE FAILED MSIV (EPU)

TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC		TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC
17.599	3161.74	3.80E+06		83.995	954.202	1.19E+06
17.799	3122.80	3.75E+06		85.995	869.034	1.09E+06
17.999	3042.82	3.66E+06		87.995	779.495	981485
18.199	2968.85	3.57E+06		89.995	690.829	872158
18.399	2900.91	3.49E+06		91.995	605.228	766052
18.599	2837.40	3.41E+06		93.995	524.635	665623
18.799	2776.39	3.34E+06		95.995	452.847	575762
18.999	2714.08	3.26E+06		97.995	390.897	497934
19.199	2678.24	3.22E+06		99.995	335.540	428131
19.399	2665.36	3.20E+06		109.985	3.57043	4573.04
19.599	2652.66	3.19E+06		119.985	3.09985	3997.02
19.799	2640.16	3.17E+06		129.985	2.80184	3599.05
19.999	2627.87	3.16E+06		139.975	1.96532	2527.21
21.995	2518.30	3.03E+06		149.975	1.83404	2360.85
23.995	2431.30	2.93E+06		159.975	1.71201	2204.93
25.995	2359.30	3.84E+06		169.975	1.59999	2062.05
27.995	2293.28	2.80E+06		179.995	1.49787	1931.66
29.995	2230.09	2.72E+06		189.975	1.40686	1815.37
31.995	2172.25	2.66E+06		199.975	1.32225	1707.25
33.995	2120.12	2.59E+06		249.855	0.987842	1277.33
35.995	2071.64	2.53E+06		299.855	0.859872	1113.17
37.995	2026.47	2.48E+06		349.855	0.717463	930.736
39.995	1984.24	2.43E+06		399.855	0.614194	798.365
41.995	1944.14	2.38E+06		449.855	0.538752	701.65
43.995	1905.76	2.33E+06		499.855	0.478275	624.051
45.995	1868.92	2.29E+06		549.855	0.427640	558.999
47.995	1833.53	2.25E+06		599.855	0.384955	504.096
49.995	1799.37	2.20E+06		649.855	0.326078	427.53
51.995	1766.13	2.16E+06		699.855	0.290165	380.585
53.995	1733.87	2.13E+06		749.855	0.270007	354.193
55.995	1702.63	2.09E+06		799.855	0.252464	331.206
57.995	1672.48	2.05E+06		849.855	0.236177	309.858
59.995	1630.83	2.00E+06		899.855	0.219200	287.585
61.995	1574.52	1.94E+06		949.855	0.204117	267.793
63.995	1514.79	1.87E+06		999.855	0.189794	248.997
65.995	1477.23	1.82E+06		1049.85	0.177185	232.452
67.995	1438.26	1.77E+06		1099.85	0.164943	216.387
69.995	1397.24	1.72E+06		1149.85	0.153169	200.939
71.995	1353.21	1.67E+06		1199.85	0.143205	187.869
73.995	1302.67	1.61E+06		1249.85	0.131584	172.632
75.995	1244.21	1.54E+06		1299.85	0.120867	158.576
77.995	1179.03	1.47E+06		1349.85	0.110670	145.203

TABLE 6.2-12G (continued)

EQ 100.3% POWER MSLB, WITH ONE FAILED MSIV (EPU)

TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC		TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC
79.995	1108.81	1.38E+06		1399.85	0.100967	132.476
81.995	1033.93	1.29E+06		1449.85	9.08E-02	119.085
1499.85	8.22E-02	107.819		4399.71	1.15E-03	1.51044
1549.85	7.41E-02	97.1706		4499.71	6.77E-04	0.888537
1599.85	6.65E-02	87.2442		4599.71	1.2E-03	1.63853
1649.85	5.95E-02	78.0414		4699.71	1.06E-03	1.39328
1699.85	5.32E-02	69.769		4799.71	9.62E-04	1.26145
1749.85	4.78E-02	62.6604		4899.71	3.21E-04	0.421335
1799.85	4.23E-02	55.465		4999.71	-1.19E-01	1311.73
1849.85	3.74E-02	49.0424		5099.71	1.64E-03	2.15187
1899.85	3.30E-02	43.2941		5199.71	6.82E-04	0.894091
1949.85	2.72E-02	35.6669		5299.71	5.81E-04	0.762093
1999.85	2.48E-02	32.584		5399.71	-6.80E-01	1311.71
2049.85	2.18E-02	28.5982		5499.71	-9.18E-01	1311.71
2099.85	1.91E-02	25.0817		5599.71	1.32E-03	1.72716
2149.85	1.68E-02	21.989		5699.71	4.00E-04	0.524308
2199.85	1.58E-02	20.694		5799.71	2.98E-04	0.391065
2249.85	1.36E-02	17.8233		5899.71	-1.45E-03	-1.90563
2299.85	1.27E-02	16.6073		5999.71	-1.20E-03	-1.56807
2349.85	1.18e-02	15.5378		6099.71	1.68E-03	2.19994
2399.85	1.11E-02	14.5599		6199.71	1.93E-04	0.252607
2449.86	9.15E-03	12.0008		6299.71	9.22E-05	0.120938
2499.86	9.03E-03	11.8475		6399.71	-1.79E-03	-2.35158
2549.86	8.50E-03	11.1555		6499.71	-1.59E-03	-2.08458
2599.86	7.99E-03	10.4776		6599.71	1.79E-03	2.34213
2649.86	7.50E-03	9.84253		6699.71	6.71E-05	8.80E-02
2699.86	7.65E-03	10.0349		6799.71	-6.60E-02	1311.59
2749.86	7.12E-03	9.34323		6899.71	-1.72E-03	-2.2559
2799.86	6.68E-03	8.75927		6999.71	-1.92E-03	-2.52068
2849.86	6.30E-03	8.25934		7099.71	2.19E-03	2.86837
2899.86	5.95E-03	7.80592		7199.71	-2.15E-02	1311.37
2949.86	4.75E-03	6.23707		7299.71	-2.20E-01	1311.37
2999.86	4.65E-03	6.09755		7399.71	-1.80E-03	-2.36743
3099.71	4.20E-03	5.514		7499.71	-2.24E-03	-2.93238
3199.71	4.33E-03	5.67999		7599.71	2.35E-03	3.0761
3299.71	3.82E-03	5.01187		7699.71	-5.76E-02	1311.26
3399.71	3.47E-03	4.55734		7799.71	-3.42E-01	1311.26
3499.71	2.54E-03	3.32712		7899.71	-3.32E-03	-4.34957
3599.71	2.35E-03	3.07693		7999.71	-2.52E-03	-3.30405
3699.71	2.62E-03	3.44224		8099.71	5.69E-03	7.46329
3799.71	2.35E-03	3.09034		8199.71	9.01E-06	1.18E-02
3899.71	7.62E-04	1.00022		8299.71	-4.39E-01	1311.09

TABLE 6.2-12G (continued)

EQ 100.3% POWER MSLB, WITH ONE FAILED MSIV (EPU)

TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC		TIME (SEC)	BREAK MASS RATE LBM/SEC	BREAK ENERGY RATE BTU/SEC
3999.71	1.41E-03	1.85447		8399.71	-4.66E-01	1311.09
4099.71	4.58E-03	6.00605		8499.71	-2.78E-03	-3.6472
4199.71	1.66E-03	2.17636		8599.71	-1.98E-01	1311.09
4299.71	1.51E-03	1.97773		8699.71	2.14E-04	0.280302
8799.71	-5.09E-01	1311.45		8999.71	-3.03E-03	-3.9747
8899.71	-5.56E-01	1311.45		9000.01	-3.03E-03	-3.97141

Integral 222533.0 Lbm 270.716E+06 BTU

TABLE 6.2-13

SUMMARY OF POSTULATED PIPE RUPTURES FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS

Pipe		ID Inches	Location	Description	Break Area* (Sq. Inches)	Original Design Basis Release Rate Data Table Numbers**
	Reactor Cavity (1)					
Hot Leg		42.0	Reactor Vessel Nozzle	Guillotine Break	100.	6.2-14A and 14B
Discharge Leg		30.0	Reactor Vessel Nozzle	Guillotine Break	200.	6.2-15A and 15B
	Secondary Shield Wall Area					
Hot Leg (2)		42.0	Steam Generator Nozzle	Guillotine Break	1000.	6.2-16A and 16B
Suction Leg (2)		30.0	Steam Generator Nozzle	Guillotine Break	1400.	6.2-17A and 17B
Suction Leg (2)		30.0	Pump Nozzle	Circumferential Break	1400.	6.2-18A and 18B
Suction Leg (2)		30.0	Pump Elbow	Slot Break within ±90° from Elbow Crotch	532.	6.2-19
Suction Leg (2)		30.0	Steam Generator Elbow	Slot Break within ±90° from Elbow Crotch	532.	6.2-19
Discharge Leg (2)		30.0	Pump Nozzle	Guillotine Break	1400.	6.2-20A and 20B
Main Steam (5)		31.5	Point 16 (Figure 36E.1)	Full Double Ended Guillotine	1559	6.2-24
	Pressurizer Area					
Pressurizer Surge Line (4)		10.1	Pressurizer Nozzle	Guillotine Break	161	6.2-21A and 21B
Pressurizer Relief Line (3)		3.44	Pressurizer Nozzle	Guillotine Break	9.28	6.2-22
Pressurizer Spray Line (4)		3.44	Pressurizer Nozzle	Guillotine Break	11.87	6.2-23A and 23B

* The Break Area given is total area available for flow from both sides of a given break location.

** Note that where two tables are given for circumferential break release rates, the total blowdown at a given time is equal to the sum of the rates given in the two tables

(1) Analysis is no longer required due to application of the LBB criteria (refer to UFSAR 6.2.1.2.3)
(2) This break is eliminated due to the application of LBB criteria (refer to UFSAR 6.2.1.2.3)
(3) The pre-EPU M&E release data remains unchanged at EPU conditions (refer to UFSAR 6.2.1.2.3)
(4) The pre-EPU M&E release data slightly increase at EPU conditions (refer to UFSAR 6.2.1.2.3)
(5) The pre-EPU M&E release data for this break is not the limiting case

TABLE 6.2-14A

(Historical)

MASS/ENERGY RELEASE RATES FOR 100 SQ IN
HOT LEG GUILLOTINE BREAK AT RV NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM RV SIDE)

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	609.10	0.
.00100	330.5	609.10	204962.
.00200	668.6	609.00	407177.
.00300	998.6	609.00	608147.
.00400	1332.0	609.00	811188.
.00500	1669.0	609.00	1016421.
.00600	2001.0	609.00	1218609.
.00700	2323.0	608.90	1414475
.00800	2630.0	608.90	1605060.
.00900	2952.0	608.80	1797178.
.01000	3281.0	608.80	1997473.
.01200	3951.0	608.90	2405764.
.01400	4571.0	608.80	2782825.
.01600	5216.0	608.80	3175501.
.01800	5889.0	608.80	3582179.
.02000	6488.0	608.70	3949246.
.02200	6518.0	608.80	3968158.
.02400	6528.0	608.80	3974246.
.02600	6466.0	608.70	3935854.
.02800	6487.0	608.70	3948637.
.03000	6497.0	608.70	3954724.
.03200	6456.0	608.70	3929767.
.03400	6455.0	608.70	3929159.
.03600	6454.0	608.70	3928550.
.03800	6419.0	608.60	3906603.
.04000	6408.0	608.60	3698692.
.04200	6405.0	608.60	898083..
.04400	6385.0	608.60	3885911.
.04600	6375.0	608.60	3879825.
.04800	6374.0	608.60	3879216.
.05000	6361.0	608.50	3870669.
.05500	6335.0	608.50	3854848.
.06000	6290.0	608.40	3826836.
.06500	6237.0	608.40	3794591.
.07000	6184.0	608.30	3761727.
.07500	6118.0	608.20	3720968.
.08000	6041.0	608.10	3673532.
.08500	5988.0	608.00	3640704.
.09000	5966.0	608.00	3626720.
.09500	5939.0	607.90	3610318.
.10000	5902.0	607.90	3587826.
.11000	5819.0	607.80	3536788.
.12000	5714.0	607.60	3471826
.13000	5615.0	607.50	3411113.
.14000	5527.0	607.40	3357100.

(CONTINUED)

TABLE 6.2-14A (CÓNT.)
(Historical)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	5468.0	607.30	3320716.
.16000	5418.0	607.30	3290351.
.17000	5360.0	607.20	3254592.
.18000	5258.0	607.10	3192132.
.19000	5168.0	607.00	3136976.
.20000	5098.0	608.90	3093976.
.22000	5017.0	506.80	3044316.
.24000	4953.0	606.70	3004985.
.26000	4860.0	606.50	2947590.
.28000	4769.0	606.40	2888890.
.30000	4726.0	606.40	2865846.
.32000	4678.0	606.30	2836271.
.34000	4636.0	606.30	2810807.
.36000	4613.0	606.30	2796862.
.38000	4560.0	606.20	2764272.
.40000	4548.0	606.20	2756998.
.42000	4568.0	606.20	2767909.
.44000	4547.0	606.20	2756391.
.46000	4525.0	606.20	2743055.
.48000	4524.0	606.20	2742449.
.50000	4509.0	606.20	2731537.
.55000	4486.0	606.20	2719413.
.60000	4518.0	606.30	2739263.
.65000	4567.0	606.40	2769429.
.70000	4612.0	606.50	2797178.
.75000	4609.0	606.50	2795359.
.80000	4624.0	606.60	2804918.
.85000	4648.0	606.70	2819942.
.90000	4656.0	606.80	2825261.
.95000	4650.0	606.80	2821620.
1.00000	4651.0	606.90	2822692.
1.10000	4644.0	607.00	2818908.
1.20000	4825.0	607.10	2806623.
1.30000	4564.0	607.20	2789447.
1.40000	4590.0	507.30	2787507.
1.50000	4625.0	607.50	2808473.
1.60000	4601.0	607.60	2795564.
1.70000	4608.0	607.80	2800742.
1.80000	4597.0	607.90	2794516.
1.90000	4582.0	608.10	2786314.
2.00000	4567.0	608.20	2777649.
2.50000	4441.0	609.00	2704569.
3.00000	4271.0	609.90	2604883.
3.50000	4128.0	611.10	2522621.
4.00000	4013.0	612.50	2457963.

TABLE 6.2-14B
(Historical)
MASS/ENERGY RELEASE RATES FOR 100 SQ IN
HOT LEG GUILLOTINE BREAK AT RV NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM SG SIDE)

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	609.10	0.
.00100	337.1	609.10	205328.
.00200	672.1	609.00	409309.
.00300	1005.0	609.00	610827.
.00400	1328.0	608.90	808619.
.00500	1647.0	608.80	1002694.
.00600	1959.0	608.80	1192639.
.00700	2268.0	608.70	1380532.
.00800	2577.0	608.70	1568620.
.00900	2888.0	608.60	1757637.
.01000	3200.0	608.60	1947520.
.01200	3837.0	608.60	2335198.
.01400	4494.0	608.60	2735048.
.01600	5151.0	608.70	3135414.
.01800	5789.0	608.70	3523764.
.02000	6400.0	608.60	3895040.
.02200	6371.0	608.60	3877391.
.02400	6383.0	608.60	3884694.
.02600	6425.0	608.70	3910898.
.02800	6463.0	608.70	3934028.
.03000	6482.0	608.70	3945593.
.03200	6478.0	608.70	3943159.
.03400	6480.0	608.70	3926115.
.03600	6416.0	608.60	3904778.
.03800	6392.0	608.60	3890171.
.04000	6382.0	608.60	3884085.
.04200	6388.0	608.60	3887737.
.04400	6402.0	608.60	3896257.
.04600	6413.0	608.60	3902952.
.04800	6413.0	608.60	3902952.
.05000	6402.0	608.60	3896257.
.05500	6345.0	608.60	3861567.
.06000	6289.0	608.50	3826857.
.06500	6224.0	608.40	3786682.
.07000	6157.0	608.30	3745303.
.07500	6079.0	608.20	3697248.
.08000	5977.0	608.10	3634614.
.08500	5935.0	608.00	3608480.
.09000	5961.0	608.10	3624884.
.09500	5973.0	608.10	3632181.
.10000	5971.0	608.10	3630965.
.11000	5900.0	608.00	3587200.
.12000	5757.0	607.80	3499105.
.13000	5571.0	607.60	3384940.
.14000	5496.0	607.50	3338020.

TABLE 6.2-14B (CONT.)

(Historical)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	5421.0	607.40	3292715.
.16000	5362.0	607.30	3256343.
.17000	5347.0	607.30	3247233.
.18000	5226.0	607.20	3173227.
.19000	5174.0	607.10	3141135.
.20000	5121.0	607.00	3108447.
.22000	5040.0	606.90	3058776.
.24000	4936.0	606.80	2995165.
.26000	4844.0	606.60	2938370.
.28000	4728.0	606.50	2867532.
.30000	4736.0	606.50	2872384.
.32000	4695.0	606.40	2874048.
.34000	4615.0	606.30	2798075.
.36000	4621.0	606.30	2801712.
.38000	4545.0	606.20	2755179.
.40000	4522.0	606.20	2741236.
.42000	4571.0	606.20	2770940.
.44000	4556.0	606.20	2761847.
.46000	4528.0	606.20	2744874.
.48000	4540.0	606.20	2752148.
.50000	4479.0	606.10	2714722.
.55000	4497.0	606.20	2726081.
.60000	4503.0	606.20	2729719.
.65000	4577.0	606.30	2775035.
.70000	4611.0	606.40	2796110.
.75000	4611.0	606.40	2796110.
.80000	4620.0	606.40	2801568.
.85000	4652.0	606.50	2821438.
.90000	4659.0	606.60	2826149.
.95000	4652.0	606.60	2821903.
1.00000	4655.0	606.60	2823723.
1.10000	4648.0	606.70	2819942.
1.20000	4628.0	606.80	2808270.
1.30000	4600.0	606.90	2791740.
1.40000	4593.0	607.00	2787951.
1.50000	4630.0	607.20	2811336.
1.60000	4607.0	607.30	2797831.
1.70000	4616.0	607.40	2803758.
1.80000	4604.0	607.50	2796930.
1.90000	4589.0	607.70	2788735.
2.00000	4575.0	607.80	2780685.
2.50000	4451.0	608.50	2708434.
3.00000	4284.0	609.30	2610241.
3.50000	4145.0	610.30	2529694.
4.00000	4032.0	611.60	2465971.

TABLE 6.2-15A

(Historical)

MASS/ENERGY RELEASE RATES FOR 200 SQ IN
DISCHARGE LEG GUILL. BREAK AT RV NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM PUMP SIDE)
Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	544.70	0.
.00100	4692.0	544.50	2554794.
.00200	8930.0	544.00	4857920.
.00300	11870.0	543.10	6446597.
.00400	12100.0	542.00	6558200.
.00500	10110.0	541.00	5469510.
.00600	8483.0	540.20	4582517.
.00700	7359.0	539.80	3972388.
.00800	6841.0	539.60	3691404.
.00900	6855.0	539.60	3698958.
.01000	7270.0	539.80	3924346.
.01200	8782.0	540.40	4745793.
.01400	10420.0	541.20	5639304.
.01600	11870.0	541.90	6432353.
.01800	12970.0	542.60	7037522.
.02000	13940.0	543.10	7570814.
.02200	14720.0	543.50	8000320.
.02400	15060.0	543.70	8188122.
.02600	15240.0	543.80	8287512.
.02800	15350.0	543.90	8348865.
.03000	15440.0	543.90	8397816.
.03200	15540.0	543.00	8453760.
.03400	15660.0	544.10	8530606.
.03600	15790.0	544.20	8592918.
.03800	15910.0	544.30	8659813.
.04000	15950.0	544.30	8681585.
.04200	15890.0	544.30	8648927.
.04400	15720.0	544.20	8554824.
.04600	15480.0	544.00	8421120.
.04800	15080.0	543.70	8198996.
.05000	14630.0	543.50	7951405.
.05500	13780.0	543.10	7483918.
.06000	13800.0	543.10	7494780.
.06500	14390.0	543.40	7819526.
.07000	14690.0	543.50	7984015.
.07500	14290.0	543.30	7763757.
.08000	13350.0	542.90	7247715.
.08500	12540.0	542.30	6800442.
.09000	12130.0	542.10	6575673.
.09500	12290.0	542.20	6663638.
.10000	12720.0	542.50	6900600.
.11000	12900.0	542.60	6999540.
.12000	12380.0	542.30	6713674.
.13000	11870.0	542.00	6433540.
.14000	10610.0	541.40	5852534.

(CONTINUED)

TABLE 6.2-15A (CONT.)

(Historical)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	10310.0	541.20	5579772.
.16000	10820.0	541.40	5857948.
.17000	10880.0	541.50	5891520.
.18000	10770.0	541.40	5830878.
.19000	11310.0	541.70	6126627.
.20000	11470.0	541.80	6214446.
.22000	10690.0	541.40	5787566.
.24000	10820.0	541.50	5859030.
.26000	10860.0	541.50	5880690.
.28000	11000.0	541.60	5957600.
.30000	11080.0	541.60	6006344.
.32000	10830.0	541.50	5864445.
.34000	10960.0	541.60	5935936.
.36000	11030.0	541.60	5973848.
.38000	11000.0	541.60	5957600.
.40000	10910.0	541.60	5908856.
.42000	10900.0	541.60	5903440.
.44000	11020.0	541.70	5969534.
.46000	10970.0	541.70	5942449.
.48000	10930.0	541.70	5920781.
.50000	11010.0	541.70	5964117.
.55000	10980.0	541.80	5948964.
.60000	11000.0	541.90	5960900.
.65000	11100.0	542.00	6016200.
.70000	10990.0	542.00	5956580.
.75000	10970.0	542.10	5946837.
.80000	11020.0	542.30	5976146.
.85000	10980.0	542.40	5955552.
.90000	11000.0	542.50	5967500.
.95000	11000.0	542.60	5968600.
1.00000	10990.0	542.80	5965372.
1.10000	10990.0	543.10	5968669.
1.20000	10990.0	543.40	5971966.
1.30000	10970.0	543.70	5964369.
1.40000	10960.0	544.10	5963336.
1.50000	10960.0	544.50	5967720.
1.60000	10960.0	544.90	5972104.
1.70000	10950.0	545.30	5971035.
1.80000	10930.0	545.70	5964561.
1.90000	10920.0	546.10	5963412.
2.00000	10890.0	546.50	5951385.
2.50000	10740.0	548.60	5891964.
3.00000	10530.0	550.50	5796765.
3.50000	10310.0	552.40	5695244.
4.00000	10030.0	554.30	5559629.

TABLE 6.2-15B

(Historical)
MASS/ENERGY RELEASE RATES FOR 200 SQ IN
DISCHARGE LEG GUILL. BREAK AT RV NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM RV SIDE)
Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	544.70	0.
.00100	4292.0	543.60	2333131.
.00200	6455.0	541.60	3496028.
.00300	8017.0	540.70	4334792.
.00400	9982.0	541.00	5400262.
.00500	12100.0	541.00	6558200.
.00600	13460.0	542.90	7307434.
.00700	13940.0	543.10	7570814.
.00800	13560.0	542.90	7361724.
.00900	13010.0	542.50	7057925.
.01000	12550.0	542.20	6804610.
.01200	12390.0	542.10	6716619.
.01400	12860.0	542.50	6976550.
.01600	13330.0	542.80	7235524.
.01800	13720.0	543.00	7449960.
.02000	14140.0	543.20	7680848.
.02200	14570.0	543.40	7917338.
.02400	14890.0	543.60	8094204.
.02600	15090.0	543.70	8204433.
.02800	15190.0	543.80	8260322.
.03000	15280.0	543.80	8309264.
.03200	15360.0	543.90	8354304.
.03400	15430.0	543.90	8392377.
.03600	15500.0	544.00	8432000.
.03800	15540.0	544.00	8453760.
.04000	15540.0	544.00	8453760.
.04200	15510.0	544.00	8437440.
.04400	15420.0	543.90	8386938.
.04600	15280.0	543.80	8309264.
.04800	15090.0	543.70	8204433.
.05000	14880.0	543.60	8088768.
.05500	14480.0	543.40	7868432.
.06000	14340.0	543.30	7790922.
.06500	14360.0	543.30	7801788.
.07000	14260.0	543.30	7747458.
.07500	13820.0	543.10	7505642.
.08000	13210.0	542.70	7169067.
.08500	12670.0	542.40	6872208.
.09000	12310.0	542.10	6673251.
.09500	12300.0	542.10	6667830.
.10000	12520.0	542.30	6789596.
.11000	12700.0	542.40	6888480.
.12000	12220.0	542.10	6624462.
.13000	11780.0	541.80	6382404.
.14000	11100.0	541.50	6010650.

(CONTINUED)

TABLE 6.2-15B (CONT.)

(Historical)

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	10560.0	541.20	5715072.
.16000	10730.0	541.30	5808149.
.17000	10960.0	541.40	5933744.
.18000	11010.0	541.50	5961915.
.19000	11160.0	541.60	6044256.
.20000	11230.0	541.60	6082168.
.22000	10860.0	541.40	5879604.
.24000	10850.0	541.40	5874190.
.26000	10830.0	541.40	5863362.
.28000	11080.0	541.50	5999820.
.30000	11010.0	541.50	5961915.
.32000	10910.0	541.40	5906674.
.34000	11030.0	541.50	5972745.
.36000	10960.0	541.50	5934840.
.38000	11040.0	541.50	5978160.
.40000	10950.0	541.50	5929425
.42000	10920.0	541.50	5913180.
.44000	11000.0	541.50	5956500.
.46000	10980.0	541.50	5945670.
.48000	10970.0	541.50	5940255.
.50000	11010.0	541.50	5961915.
.55000	10990.0	541.50	5951085.
.60000	11000.0	541.50	5956500.
.65000	11110.0	541.60	6017176.
.70000	11020.0	541.60	5968432.
.75000	11020.0	541.60	5968432.
.80000	11050.0	541.60	5984680.
.85000	11020.0	541.60	5968432.
.90000	11040.0	541.60	5979264.
.95000	11040.0	541.70	5980368.
1.00000	11040.0	541.70	5980368.
1.10000	11050.0	541.80	5986890.
1.20000	11050.0	541.80	5986890.
1.30000	11050.0	541.90	5987995.
1.40000	11060.0	542.00	5994520.
1.50000	11070.0	542.20	6002154.
1.60000	11080.0	542.30	6008684.
1.70000	11080.0	542.40	6009792.
1.80000	11080.0	542.60	6012005.
1.90000	11070.0	542.80	6008796.
2.00000	11060.0	543.00	6005580.
2.50000	10950.0	544.00	5956800.
3.00000	10770.0	545.30	5872881.
3.50000	10560.0	546.90	5775264.
4.00000	10280.0	548.70	5640636.

TABLE 6.2-16A
(Historical)
MASS/ENERGY RELEASE RATES FOR 1000 SQ IN
HOT LEG GUILL. BREAK AT SG NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM RV SIDE)

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	609.10	0.
.00100	5325.0	609.00	3242925.
.00200	10140.0	608.50	6170190.
.00300	13980.0	607.90	8498442.
.00400	16680.0	607.20	10128096.
.00500	18940.0	606.60	11489004.
.00600	20400.0	606.00	12362400.
.00700	22190.0	605.70	13440463.
.00800	25350.0	605.70	15354495.
.00900	28510.0	605.70	17268507.
.01000	31660.0	605.70	19176462.
.01200	37920.0	605.70	22968144.
.01400	39420.0	605.70	23876694.
.01600	39360.0	605.80	23844288.
.01800	39310.0	605.80	23813998.
.02000	39280.0	605.80	23795824.
.02200	39270.0	605.90	23793693.
.02400	39260.0	605.90	23787634.
.02600	39270.0	605.90	23793693.
.02800	39290.0	605.90	23805811.
.03000	39310.0	606.00	23821860.
.03200	39350.0	606.00	23846100.
.03400	39400.0	606.00	23876400.
.03600	39460.0	606.10	23916706.
.03800	39520.0	606.10	23953072.
.04000	39600.0	606.10	24001560.
.04200	39660.0	606.10	24037926.
.04400	45510.0	606.80	27615468.
.04600	47180.0	607.00	28638260
.04800	45680.0	606.80	27718624.
.05000	43830.0	606.60	26587278.
.05500	40110.0	606.20	24314682.
.06000	40050.0	606.20	24278310.
.06500	39660.0	606.10	24037926.
.07000	39640.0	606.00	24021840.
.07500	39600.0	606.00	23997600.
.00800	39510.0	606.00	23943060.
.08500	39380.0	606.00	23864280.
.09000	39220.0	605.90	23763398.
.09500	39060.0	605.90	23666454.
.10000	38880.0	606.80	23553504.
.11000	38500.0	605.80	23323300.
.12000	38100.0	605.70	23077170.
.13000	3770.0	605.70	22834890.
.14000	37360.0	605.60	22625216.

(CONTINUED)

TABLE 6.2-16A (CONT)

(Historical)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	37080.0	605.60	22455648
.16000	36870.0	605.50	22324785.
.17000	36720.0	605.50	22233960.
.18000	36600.0	605.50	22161300.
.19000	36460.0	605.50	22076530.
.20000	36320.0	605.50	21991760.
.22000	36020.0	605.50	21810110.
.24000	35620.0	605.50	21567910.
.26000	35030.0	605.50	21210665.
.28000	34290.0	605.50	20762595.
.30000	33580.0	605.50	20332690.
.32000	33100.0	605.50	20042050.
.34000	32840.0	605.60	19887904.
.36000	32650.0	605.60	19772840.
.38000	32390.0	605.70	19618623.
.40000	32010.0	605.80	19391658.
.42000	31590.0	605.80	19137222.
.44000	31240.0	605.90	18928316.
.46000	30990.0	606.00	18779940.
.48000	30810.0	606.10	18673941.
.50000	30130.0	606.10	18564843.
.55000	29870.0	606.30	18110181.
.60000	29100.0	606.50	17649150.
.65000	28570.0	606.80	17336276.
.70000	28400.0	607.30	17247320.
.75000	28230.0	607.80	17158194
.80000	28040.0	608.30	17056732.
.85000	27840.0	608.70	16946208.
.90000	27630.0	609.10	16829433.
.95000	27400.0	609.30	16694820.
1.00000	27160.0	609.60	16556736.
1.10000	26790.0	610.50	16355295.
1.20000	26340.0	611.60	16109544.
1.30000	25830.0	612.30	15815709.
1.40000	25390.0	612.70	15556453.
1.50000	24990.0	613.00	15318870.
1.60000	24600.0	613.50	15092100.
1.70000	24290.0	614.10	14916489.
1.80000	23980.0	614.50	14735710.
1.90000	23630.0	614.60	14522998.
2.00000	23320.0	614.80	14337136.
2.50000	22120.0	616.40	13634768.
3.00000	21360.0	617.20	13183392.
3.50000	20970.0	614.50	12886065.
4.00000	20870.0	611.00	12629370.

TABLE 6.2-16B
(Historical)
MASS/ENERGY RELEASE RATES FOR 1000 SQ IN
HOT LEG GUILL. BREAK AT SG NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM SG SIDE)

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	609.10	0.
.00100	5044.0	608.50	3069274.
.00200	9611.0	608.10	5844449.
.00300	14580.0	608.20	8867556.
.00400	18050.0	607.70	10968985.
.00500	20410.0	607.00	12388870.
.00600	23080.0	606.60	14000328
.00700	25020.0	606.20	15167124.
.00800	26520.0	605.80	16065816.
.00900	28530.0	605.60	17277768.
.01000	31670.0	605.60	19179352.
.01200	37700.0	605.60	22831120.
.01400	38810.0	605.60	23503336.
.01600	38410.0	605.60	23261096
.01800	38090.0	605.50	23063495.
.02000	37870.0	605.50	22930285.
.02200	37770.0	605.50	22869735.
.02400	37770.0	605.50	22869735.
.02600	37860.0	605.60	22928016.
.02800	38030.0	605.60	23030968.
.03000	38250.0	605.60	23164200.
.03200	38490.0	605.60	23309544.
.03400	38730.0	605.70	23458761.
.03600	38950.0	605.70	23592015.
.03800	39140.0	605.70	23707098.
.04000	39280.0	605.70	23791896.
.04200	39360.0	605.70	23840352.
.04400	39420.0	605.80	23880636.
.04600	39620.0	605.80	24001796.
.04800	43160.0	606.20	26163592.
.05000	40520.0	606.00	24555120.
.05500	39610.0	605.90	23999699.
.06000	39170.0	605.90	23733103.
.06500	38460.0	605.90	23302914.
.07000	37730.0	605.90	22860607.
.07500	37230.0	605.90	22557657.
.08000	37070.0	605.90	22460713.
.08500	37170.0	605.90	22521303.
.09000	37340.0	605.90	22624306.
.09500	37400.0	605.90	22660660.
.10000	37320.0	605.90	22612188.
.11000	36990.0	605.80	22408542.
.12000	36810.0	605.80	22299498.
.13000	36800.0	605.80	22293440.
.14000	36790.0	605.70	22283703.

(CONTINUED)

TABLE 6.2-16B (CONT.)

(Historical)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	36670.0	605.70	22211019.
.16000	36430.0	605.60	22862008.
.17000	36660.0	605.60	21837935.
.18000	35640.0	605.50	21580020.
.19000	35420.0	605.50	21446810.
.20000	35370.0	605.50	21416535.
.22000	34420.0	605.50	21144060.
.24000	34120.0	605.40	20656248.
.26000	33340.0	605.40	20186036.
.28000	32760.0	605.40	19645012.
.30000	32530.0	605.40	19683662.
.32000	32190.0	605.50	19491045.
.34000	31760.0	605.50	19230660.
.36000	31230.0	605.50	18909765.
.38000	30720.0	605.60	18604032.
.40000	30320.0	605.60	18361702.
.42000	30005.0	605.70	18261265.
.44000	29620.0	605.80	18060955.
.46000	29590.0	605.90	17928581.
.48000	29320.0	605.90	17764968.
.50000	28890.0	606.00	17567940.
.55000	28520.0	606.20	1728824.
.60000	28380.0	606.40	17209632.
.65000	28830.0	606.70	17127141.
.70000	28650.0	607.10	17035226.
.75000	27870.0	607.50	16931025.
.80000	27870.0	608.00	16823366.
.85000	27460.0	608.50	16709410.
.90000	27230.0	608.90	16580347.
.95000	26990.0	609.20	16442308.
1.00000	26700.0	609.50	16273650.
1.10000	26630.0	610.30	16068169.
1.20000	25760.0	611.40	15749664.
1.30000	25250.0	612.20	15458050.
1.40000	24830.0	612.80	15210858.
1.50000	24440.0	612.90	14979276.
1.60000	24050.0	613.40	14752270.
1.70000	23750.0	614.60	14588641.
1.80000	23460.0	614.40	14426112.
1.90000	23150.0	614.60	14227990.
2.00000	22850.0	615.70	14045895.
2.50000	21741.0	615.90	13389666.
3.00000	21000.0	616.30	12942300.
3.50000	20840.0	616.90	12731156.
4.00000	20590.0	600.40	12485776.

TABLE 6.2-17A
(Historical)
MASS/ENERGY RELEASE RATES FOR 1400 SQ IN
SUCTION LEG GUILL. BREAK AT SG NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM SG SIDE)

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	544.70	0.
.00100	11140.0	544.50	6065730.
.00200	21050.0	543.90	11449095.
.00300	27890.0	543.10	15147059.
.00400	32390.0	542.00	17555380.
.00500	33170.0	541.00	17944970.
.00600	31760.0	540.20	17156752.
.00700	33290.0	539.90	17973271.
.00800	38040.0	539.90	20537796.
.00900	42780.0	539.90	23096922.
.01000	47530.0	540.00	25666200.
.01200	47550.0	540.00	25677000.
.01400	47580.0	540.00	25693200.
.01600	50440.0	540.20	27247688.
.01800	53010.0	540.40	28646604.
.02000	53030.0	540.40	28657412.
.02200	52770.0	540.50	28522185.
.02400	52820.0	540.50	28549210.
.02600	52930.0	540.50	28608685.
.02800	52710.0	540.50	28489755.
.03000	52050.0	540.50	28133025.
.03200	51370.0	540.50	27765485.
.03400	51280.0	540.60	27721968.
.03600	51900.0	540.60	28057140.
.03800	52740.0	540.70	28516518.
.04000	53150.0	540.80	28743520.
.04200	53050.0	540.80	28689440.
.04400	52980.0	540.80	28651584.
.04600	53680.0	540.90	29035512.
.04800	55620.0	541.10	30095982.
.05000	58290.0	541.30	31552377.
.05500	63190.0	541.70	34230023.
.06000	66000.0	542.00	35772000.
.06500	67960.0	542.20	36847912.
.07000	68190.0	542.30	36979437.
.07500	64920.0	542.20	35199624.
.08000	61730.0	542.10	33463833.
.08500	62740.0	542.30	34023902.
.09000	61400.0	542.30	33297220.
.09500	60190.0	542.30	32641037.
.10000	59800.0	542.40	32435520.
.11000	62550.0	542.80	33952140.
.12000	63330.0	543.00	34388190.
.13000	62440.0	543.20	33917408.
.14000	61730.0	543.40	33544082.

(CONTINUED)

TABLE 6.2-17A (CONT.)

(Historical)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	62200.0	543.70	33818140.
.16000	62400.0	543.90	33939360.
.17000	62600.0	544.10	34060660.
.18000	61720.0	544.30	33594196.
.19000	62670.0	544.60	34130082.
.20000	60800.0	544.70	33008820.
.22000	59340.0	545.10	32346234.
.24000	61890.0	545.80	33779562.
.26000	59980.0	546.10	32755078.
.28000	50400.0	546.60	33014640.
.30000	59240.0	547.10	32410204.
.32000	60040.0	547.60	32877904.
.34000	58520.0	548.50	32068960.
.36000	59230.0	548.90	32487655.
.38000	57820.0	548.90	31737398.
.40000	59090.0	549.50	32469955.
.42000	57270.0	549.90	31492773.
.44000	57840.0	550.40	31835136.
.46000	56480.0	550.80	31109184.
.48000	56930.0	551.30	31385509.
.50000	56620.0	551.70	31237254.
.55000	55740.0	552.80	30813072.
.60000	54680.0	553.80	30281784.
.65000	54220.0	554.80	30081256.
.70000	53500.0	555.60	29724600.
.75000	52750.0	556.40	29350100.
.80000	52310.0	557.20	29147132.
.85000	51900.0	557.80	28949820.
.90000	51330.0	558.40	28662872.
.95000	50770.0	559.00	28380430.
1.00000	50220.0	559.50	28098090.
1.10000	50000.0	560.40	28020000.
1.20000	48920.0	561.00	27444120.
1.30000	47600.0	561.60	26732160.
1.40000	46290.0	562.20	26024238.
1.50000	45100.0	562.80	25382280.
1.60000	44020.0	563.40	24800868.
1.70000	42960.0	564.10	24233736.
1.80000	41870.0	564.80	23648176.
1.90000	40790.0	565.50	23066745.
2.00000	39770.0	566.20	22517774.
2.50000	36280.0	570.50	20697740.
3.00000	34890.0	577.30	20141997.
3.50000	32940.0	589.80	19428012.
4.00000	30890.0	602.30	18484587.

TABLE 6.2-17B

(Historical)
MASS/ENERGY RELEASE RATES FOR 1400 SQ IN
SUCTION LEG GUILL. BREAK AT SG NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM PUMP SIDE)

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	544.70	0.
.00100	11010.0	544.40	5993844.
.00200	19630.0	543.50	10668905.
.00300	25280.0	542.20	13706816.
.00400	26470.0	541.00	14320270.
.00500	24900.0	540.00	13446000.
.00600	28510.0	539.80	15389698.
.00700	33240.0	539.80	17942952.
.00800	37950.0	539.80	20485410.
.00900	42640.0	539.80	23017072.
.01000	47300.0	539.80	25532540.
.01200	46940.0	539.80	25338212.
.01400	46590.0	539.80	25149282.
.01600	46240.0	539.80	24960352.
.01800	45890.0	539.80	24771422.
.02000	45540.0	539.80	24582492.
.02200	45190.0	539.70	24389043.
.02400	44830.0	539.70	24194751.
.02600	44480.0	539.70	24005856.
.02800	44130.0	539.70	23816961.
.03000	43780.0	539.70	23628066.
.03200	43440.0	539.70	23444588.
.03400	43100.0	539.70	23261070.
.03600	42760.0	539.70	23077572.
.03800	42440.0	539.70	22904868.
.04000	42120.0	539.70	22732164.
.04200	41800.0	539.70	22559460.
.04400	41490.0	539.70	22392153.
.04600	41190.0	539.70	22230243.
.04800	40890.0	539.70	22068333.
.05000	40590.0	539.60	21902364.
.05500	39860.0	539.60	21508456.
.06000	39140.0	539.60	21119944.
.06500	38430.0	539.60	20736828.
.07000	37720.0	539.60	20353712.
.07500	37030.0	539.60	19981388.
.08000	36340.0	539.60	19609064.
.08500	36190.0	539.60	19528124.
.09000	36030.0	539.60	19441788.
.09500	35860.0	539.60	19350056.
.10000	35670.0	539.50	19243965.
.11000	35220.0	539.50	19001190.
.12000	34670.0	539.50	18704465.
.13000	34030.0	539.50	18359185.
.14000	33250.0	539.50	17938375.

(CONTINUED)

TABLE 6.2-17B (CONT.)

(Historical)

TIME (SECONDS)	FLOWRATE (LB/SEC)	ENTHLAY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	32620.0	539.50	17598490.
.16000	32310.0	539.60	17434476.
.17000	32040.0	539.60	17288784.
.18000	31750.0	539.60	17132300.
.19000	31410.0	539.60	16948836.
.20000	31040.0	539.60	16749184.
.22000	30390.0	539.70	16401483.
.24000	29840.0	539.80	16107632.
.26000	29270.0	539.80	15799946.
.28000	28700.0	539.90	15495130.
.30000	28130.0	539.90	15187387.
.32000	27580.0	540.00	14893200.
.34000	27080.0	540.10	14625908.
.36000	26560.0	540.10	14345056.
.38000	26050.0	540.20	14072210.
.40000	25590.0	540.30	13826277.
.42000	25180.0	540.30	13804754.
.44000	24800.0	540.40	13401920.
.46000	24450.0	540.50	13215225.
.48000	24110.0	540.50	13031455.
.50000	23830.0	540.60	12882498.
.55000	22970.0	540.70	12419879.
.60000	22290.0	540.80	12054432.
.65000	21650.0	540.90	11710485.
.70000	21030.0	541.00	11377230.
.75000	20450.0	541.10	11065495.
.80000	19900.0	541.10	10767890.
.85000	19400.0	541.20	10499280.
.90000	18950.0	541.30	10257635.
.95000	18550.0	541.30	10041115.
1.00000	18190.0	541.40	9848066.
1.10000	17650.0	541.50	9557475.
1.20000	17300.0	541.60	9369680.
1.30000	17050.0	541.70	9235985.
1.40000	16800.0	541.80	9102240.
1.50000	16560.0	541.80	8972208.
1.60000	16360.0	541.90	8865484.
1.70000	16200.0	542.00	8780400.
1.80000	16040.0	542.00	8683680.
1.90000	15880.0	542.10	8608548.
2.00000	15720.0	542.30	8524956.
2.50000	15020.0	543.00	8155860.
3.00000	14570.0	544.20	7928994.
3.50000	14010.0	545.80	7646658.
4.00000	12980.0	547.90	7111742.

TABLE 6.2-18A
(Historical)
MASS/ENERGY RELEASE RATES FOR 1400 SQ IN
SUCTION LEG GUILL. BREAK AT PUMP NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM SG SIDE)

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	544.70	0.
.00100	10870.0	544.20	5915454.
.00200	17990.0	542.80	9764972.
.00300	18930.0	540.80	10237344.
.00400	19000.0	539.80	10256200.
.00500	23720.0	539.80	12804056.
.00600	28410.0	539.80	15335718.
.00700	33060.0	539.90	17849094.
.00800	37640.0	539.90	20321836.
.00900	42170.0	539.90	22767583.
.01000	46060.0	539.90	25191734.
.01200	46360.0	539.90	25029764.
.01400	46090.0	539.90	24883991.
.01600	45840.0	539.90	24749016.
.01800	45600.0	540.00	24624000.
.02000	45380.0	540.00	24505200.
.02200	45180.0	540.00	24397200.
.02400	44990.0	540.10	24299099.
.02600	44820.0	540.10	24207282.
.02800	44670.0	540.10	24126267.
.03000	44530.0	540.10	24050653.
.03200	44400.0	540.20	23984880.
.03400	44280.0	540.20	23920056.
.03600	44180.0	540.20	23866036.
.03800	44090.0	540.20	23817418.
.04000	44000.0	540.30	23773200.
.04200	43930.0	540.30	23735379.
.04400	43870.0	540.30	23702961.
.04600	43820.0	540.30	23675946.
.04800	43770.0	540.30	23648931.
.05000	43730.0	540.40	23631692.
.05500	43670.0	540.40	23599268.
.06000	43650.0	540.40	23588460.
.06500	43660.0	540.50	23598230.
.07000	43690.0	540.50	23614445.
.07500	43740.0	540.50	23641470.
.08000	43810.0	540.60	23683686.
.08500	43980.0	540.60	23732340.
.09000	44020.0	540.70	23801614.
.09500	44160.0	540.70	23973664.
.10000	44330.0	540.80	23973664
.11000	44750.0	541.00	24209750.
.12000	45260.0	541.10	24490186.
.13000	45830.0	541.30	24807779.
.14000	46430.0	541.40	25137202.

(CONTINUED)

TABLE 6.2-18A (CONT)

(Historical)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	47000.0	541.50	25472160.
.16000	47630.0	541.70	25801171.
.17000	59510.0	542.50	32284175.
.18000	59730.0	542.70	32415471.
.19000	58200.0	542.80	32676560.
.20000	60020.0	542.90	32584858.
.22000	59270.0	543.10	32189537.
.24000	59400.0	543.40	32277960.
.26000	59990.0	543.90	32628561.
.28000	59250.0	544.10	32237925.
.30000	58840.0	544.50	32038380.
.32000	58430.0	544.90	31838507.
.34000	58760.0	545.30	32041828.
.36000	57980.0	545.70	31639686.
.38000	57910.0	546.10	31624651.
.40000	57680.0	546.50	31511190.
.42000	57450.0	546.90	31419405.
.44000	56980.0	547.40	31190852.
.46000	56590.0	547.80	31000002.
.48000	56350.0	548.20	30891070.
.50000	56280.0	548.60	30875208.
.55000	55450.0	579.70	30480865.
.60000	54710.0	550.70	30128797.
.65000	54030.0	551.70	29808351.
.70000	53520.0	552.70	29580504.
.75000	53040.0	553.70	29368248.
.80000	52410.0	554.50	29061345.
.85000	51840.0	555.40	28791936.
.90000	51430.0	556.20	28605366.
.95000	50980.0	556.90	28390762.
1.00000	50460.0	557.60	28136496
1.10000	49830.0	558.80	27845004.
1.20000	49200.0	559.70	27537240.
1.30000	48170.0	560.50	26999285.
1.40000	46870.0	561.10	26298757.
1.50000	45480.0	561.70	25546116.
1.60000	44210.0	562.30	24859283.
1.70000	43130.0	562.90	24277877.
1.80000	42170.0	563.60	23767012.
1.90000	41220.0	564.20	23256324.
2.00000	40240.0	564.90	22731576.
2.50000	36270.0	568.80	20630376.
3.00000	34900.0	574.70	20057030
3.50000	33230.0	585.30	19449519.
4.00000	30090.0	601.80	18108162.

TABLE 6.2-18B
(Historical)
MASS/ENERGY RELEASE RATES FOR 1400 SQ IN
SUCTION LEG GUILL BREAK AT PUMP NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM PUMP SIDE)

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	544.70	0.
.00100	10220.0	543.70	5556614.
.00200	15360.0	541.70	8320512.
.00300	15250.0	540.00	8235000.
.00400	19000.0	539.80	10256200.
.00500	23720.0	539.80	12804056.
.00600	28420.0	539.80	15341116.
.00700	33070.0	539.80	17851186.
.00800	37630.0	539.80	20312674.
.00900	42110.0	539.80	22730978.
.01000	46470.0	539.80	25084506.
.01200	45520.0	539.80	24571696.
.01400	44550.0	539.70	24043635.
.01600	43550.0	539.70	23503935.
.01800	42500.0	539.70	22937250.
.02000	41420.0	539.60	22350232.
.02200	40300.0	539.60	21745880.
.02400	39160.0	539.50	21126820.
.02600	38040.0	539.50	20522580.
.02800	36960.0	539.50	19939920.
.03000	36260.0	539.50	19562270.
.03200	36070.0	539.50	19459765.
.03400	35880.0	539.60	19360848.
.03600	35690.0	539.60	19258324.
.03800	35510.0	539.70	19164747.
.04000	35310.0	539.70	19056807.
.04200	35100.0	539.80	18946980.
.04400	34870.0	539.80	18822826.
.04600	34800.0	539.80	18677080.
.04800	34290.0	539.80	18509742.
.05000	33950.0	539.80	18326210.
.05500	32990.0	539.90	17811301.
.06000	32540.0	540.10	17574854.
.06500	32290.0	540.30	17446287.
.07000	32040.0	540.40	17314416.
.07500	31750.0	540.40	17157700.
.08000	31400.0	540.50	16971700.
.08500	30930.0	540.50	16717665.
.09000	30580.0	540.50	16528490.
.09500	30190.0	540.40	16314646.
.10000	29750.0	540.40	16076900.
.11000	28760.0	540.20	15536152.
.12000	27890.0	540.20	15066178.
.13000	27160.0	540.20	14671832.
.14000	26460.0	540.20	14293692.

(CONTINUED)

TABLE 6.2-18B (CONT)

(Historical)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	25820.0	540.30	13950546.
.16000	25450.0	540.50	13755725.
.17000	25370.0	540.80	13720096.
.18000	25160.0	540.90	13609044.
.19000	24830.0	540.90	13430547.
.20000	24590.0	540.90	13300731.
.22000	23970.0	540.80	12962976.
.24000	23950.0	541.10	12959345.
.26000	23960.0	541.20	12967152.
.28000	24060.0	541.40	13026084.
.30000	23710.0	541.20	12831852.
.32000	23600.0	541.30	12774680.
.34000	23620.0	541.40	12787868.
.36000	23500.0	541.30	12720550.
.38000	23290.0	541.30	12606877.
.40000	23060.0	541.30	12482378.
.42000	22960.0	541.40	12430544.
.44000	22740.0	541.40	12311436.
.46000	22430.0	541.30	12141359.
.48000	22180.0	541.40	12008252.
.50000	22080.0	541.40	11943264.
.55000	21480.0	541.40	11629272.
.60000	20990.0	541.50	11366085.
.65000	20450.0	541.50	11073675.
.70000	20070.0	541.60	10869912.
.75000	19710.0	541.70	10676907.
.80000	19340.0	541.70	10476478.
.85000	19020.0	541.70	10303134.
.90000	18730.0	541.80	10147914.
.95000	18450.0	541.80	9996210.
1.00000	18210.0	541.90	9867999.
1.10000	17850.0	542.00	9674700.
1.20000	17630.0	542.10	9557223.
1.30000	17430.0	542.20	9450546.
1.40000	17170.0	542.20	9309574.
1.50000	16920.0	542.30	9175716.
1.60000	16710.0	542.40	9063504.
1.70000	16520.0	542.50	8962100.
1.80000	16380.0	542.60	8887788.
1.90000	16240.0	542.70	8813448.
2.00000	16090.0	542.80	8733652.
2.50000	15420.0	543.60	8382312.
3.00000	14990.0	544.80	8166552.
3.50000	14390.0	546.60	7865574.
4.00000	13290.0	549.00	7296210.

TABLE 6.2-19
(Historical)
MASS/ENERGY RELEASE RATES FOR 532 SQ IN
SUCTION LEG SLOT BREAK
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	544.70	0.
.00100	16530.0	544.20	8995626.
.00200	27930.0	542.90	15163197.
.00300	33450.0	541.50	18113175.
.00400	33770.0	540.30	18245931.
.00500	36140.0	539.80	19508372.
.00600	36130.0	539.90	19506587.
.00700	36140.0	539.90	19511986.
.00800	39770.0	540.20	21483754.
.00900	43360.0	540.50	23436080.
.01000	45120.0	540.60	24391872.
.01200	45050.0	540.60	24354030.
.01400	44000.0	540.50	23782000.
.01600	44111.0	540.50	24245910.
.01800	44850.0	540.60	24391872.
.02000	45120.0	540.60	24391872.
.02200	44970.0	540.60	24310782.
.02400	45220.0	540.60	24445932.
.02600	45950.0	540.70	24845165.
.02800	46300.0	540.70	25034410.
.03000	45850.0	540.70	24791095.
.03200	45260.0	540.60	24467556.
.03400	45550.0	540.70	24628885.
.03600	46820.0	540.80	25320256.
.03800	48020.0	540.90	25974018.
.04000	48530.0	540.90	26249877.
.04200	48800.0	541.00	26400800.
.04400	49950.0	541.10	27027945.
.04600	52310.0	541.30	28315403.
.04800	54890.0	541.60	29728424.
.05000	56450.0	541.70	30578905.
.05500	58000.0	541.90	31430200.
.06000	60470.0	542.20	32786834.
.06500	60750.0	542.20	32938650.
.07000	59590.0	542.10	32303739.
.07500	56060.0	541.80	30373308.
.08000	57830.0	542.00	31343800.
.08500	53120.0	541.60	28769792.
.09000	49530.0	541.30	26810589.
.09500	49940.0	541.30	27032522.
.10000	54990.0	541.80	29793582.
.11000	58370.0	542.20	31648214.
.12000	56780.0	542.10	30780438.
.13000	49660.0	541.60	26895856.
.14000	50410.0	541.70	27307097.

(CONTINUED)

TABLE 6.2-19 (CONT)

(Historical)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	52850.0	542.00	28644700.
.16000	52700.0	542.10	28568670.
.17000	52350.0	542.10	28378935.
.18000	52800.0	542.30	28655132.
.19000	52840.0	542.30	28655132.
.20000	58840.0	542.30	27570532.
.22000	49270.0	542.30	26719121.
.24000	52860.0	542.80	28692408.
.26000	51550.0	542.90	27986495.
.28000	50500.0	543.00	27421500.
.30000	50330.0	543.20	27339256.
.32000	49770.0	543.30	27044041.
.34000	51040.0	543.70	27750448.
.36000	50410.0	543.80	27412958.
.38000	49270.0	543.90	26797953.
.40000	49410.0	544.10	26883981.
.42000	49650.0	544.40	27029460.
.44000	49440.0	544.50	26920080.
.46000	49280.0	544.70	26842816.
.48000	48710.0	544.90	26542079.
.50000	49110.0	545.10	26769861.
.55000	48350.0	545.50	26374925.
.60000	48340.0	546.00	26393640.
.65000	47800.0	546.40	26117920.
.70000	47790.0	546.90	26136351.
.75000	47560.0	547.30	26029588.
.80000	47300.0	547.70	25906210.
.85000	47180.0	548.00	25854640.
.90000	46920.0	548.40	25730928.
.95000	46740.0	548.80	25650912.
1.00000	46520.0	549.10	25544132.
1.10000	46170.0	549.80	25384266.
1.20000	45890.0	550.40	25257856.
1.30000	45590.0	550.90	25115531.
1.40000	45420.0	551.40	25044588.
1.50000	45260.0	551.80	24974468.
1.60000	45060.0	552.30	24886638.
1.70000	44880.0	552.60	24800688.
1.80000	44690.0	553.00	24713570.
1.90000	44430.0	553.30	24583119.
2.00000	44290.0	553.70	24523373.
2.50000	42150.0	555.30	23739075.
3.00000	41220.0	557.30	22971906.
3.50000	39850.0	560.30	22327955.
4.00000	38300.0	564.70	21628010.

TABLE 6.2-20A
(Historical)
MASS/ENERGY RELEASE RATES FOR 1400 SQ IN
DISCHARGE LEG GUILL BREAK AT PUMP NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM PUMP SIDE)

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	544.70	0.
.00100	10680.0	543.70	5806716.
.00200	15810.0	541.60	8562696.
.00300	15210.0	539.70	8208837.
.00400	18940.0	539.50	10218130.
.00500	23640.0	539.60	12756144.
.00600	28290.0	539.60	15265284.
.00700	32890.0	539.60	17736652.
.00800	37390.0	539.60	20175644.
.00900	41820.0	539.60	22566072.
.01000	46150.0	539.70	24907155.
.01200	45520.0	539.70	24567144.
.01400	44910.0	539.70	24237927.
.01600	44280.0	539.70	23897916.
.01800	43640.0	539.70	23552508.
.02000	43000.0	539.70	23207100.
.02200	42360.0	539.70	22861692.
.02400	41720.0	539.80	22520456.
.02600	41110.0	539.80	22191178.
.02800	40530.0	539.90	21882147.
.03000	39980.0	539.90	21585202.
.03200	39460.0	540.00	21308400.
.03400	38970.0	540.10	21047697.
.03600	38490.0	540.10	20788449.
.03800	38010.0	540.20	20533002.
.04000	37540.0	540.20	20279108.
.04200	37060.0	540.30	20023518.
.04400	36570.0	540.30	19758771.
.04600	36250.0	540.30	19585875.
.04800	36150.0	540.40	19535460.
.05000	36030.0	540.40	19470612.
.05500	35740.0	540.50	19317470.
.06000	35380.0	540.60	19126428.
.06500	34990.0	540.70	18919093.
.07000	34510.0	540.80	18663008.
.07500	33880.0	540.80	18322304.
.08000	33000.0	540.80	17846400.
.08500	32480.0	540.80	17565184.
.09000	32100.0	540.80	17359680.
.09500	31700.0	540.80	17143360.
.10000	31110.0	540.70	16821177.
.11000	30200.0	540.50	16323100.
.12000	29210.0	540.40	15785084.
.13000	28320.0	540.40	15304128.
.14000	27550.0	540.40	14888020.

(CONTINUED)

TABLE 6.2-20A (CONT)

(Historical)

TIME (SECONDS)	FLOW.RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	26910.0	540.50	14544855.
.16000	26480.0	540.70	14306922.
.17000	26100.0	540.80	14114880.
.18000	25690.0	540.90	13895721.
.19000	25340.0	541.00	13708940.
.20000	24990.0	541.00	13519590.
.22000	24440.0	541.20	13226928.
.24000	24140.0	541.40	13069396.
.26000	23930.0	541.50	12958095.
.28000	23600.0	541.60	12781760.
.30000	23260.0	541.70	12599942.
.32000	22930.0	541.70	12421181.
.34000	22520.0	541.90	12203588.
.36000	22560.0	542.00	12119120.
.38000	22010.0	542.00	11929420.
.40000	21720.0	542.10	11774412.
.42000	21440.0	542.20	11624768.
.44000	21160.0	542.30	11475068.
.46000	20760.0	542.30	11258148.
.48000	20410.0	542.40	11070384.
.50000	20160.0	542.60	10938816.
.55000	19640.0	542.90	10662556.
.60000	19210.0	543.20	10434872.
.65000	18890.0	543.60	10268604.
.70000	18650.0	543.90	10143735.
.75000	18380.0	544.30	10004234.
.80000	18110.0	544.60	9862706.
.85000	17850.0	544.90	9726465.
.90000	17610.0	545.30	9602733.
.95000	17350.0	545.60	9466160.
1.00000	17080.0	545.90	9323972.
1.10000	16580.0	546.80	9051696.
1.20000	16230.0	547.30	8882679.
1.30000	15780.0	547.90	8645862.
1.40000	15340.0	548.50	8413990.
1.50000	15000.0	549.20	8238000.
1.60000	14750.0	549.80	8109550.
1.70000	14430.0	550.30	7940829.
1.80000	14190.0	550.90	7817271.
1.90000	14010.0	551.40	7725114.
2.00000	13830.0	551.90	7632777.
2.50000	13250.0	554.00	7340500.
3.00000	14200.0	555.90	7893780
3.50000	14930.0	558.50	8338405.
4.00000	15580.0	562.80	8768424.

TABLE 6.2-20B
(Historical)
MASS/ENERGY RELEASE RATES FOR 1400 SQ IN
DISCHARGE LEG GUILL BREAK AT PUMP NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM RV SIDE)

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	544.70	0.
.00100	11080.0	544.10	6028628.
.00200	18070.0	542.50	9802975.
.00300	18280.0	540.30	9876684.
.00400	18930.0	539.50	10212735.
.00500	23030.0	539.50	12748385.
.00600	28260.0	539.50	15246270.
.00700	32850.0	539.50	17722575.
.00800	37380.0	539.50	20166510.
.00900	41860.0	539.50	22583470.
.01000	46270.0	539.50	24962665.
.01200	45710.0	539.40	24655974.
.01400	45180.0	539.40	24370092.
.01600	44660.0	539.40	24089604.
.01800	44170.0	539.40	23825298.
.02000	43690.0	539.40	23566386.
.02200	43230.0	539.40	23318262.
.02400	42790.0	539.50	23085205.
.02600	42380.0	539.50	22864010.
.02800	42000.0	539.50	22659000.
.03000	41640.0	539.50	22464780.
.03200	41320.0	539.60	22296272.
.03400	41020.0	539.60	22134392.
.03600	40750.0	539.60	21988700.
.03800	40520.0	539.70	21868644.
.04000	40320.0	539.70	21760704.
.04200	40140.0	539.80	21667572.
.04400	40000.0	539.80	21592000.
.04600	39880.0	539.90	21531212.
.04800	39800.0	539.90	21488020.
.05000	39740.0	540.00	21459600.
.05500	39700.0	540.10	21441970.
.06000	39810.0	540.30	21509343.
.06500	40040.0	540.40	21637616.
.07000	40350.0	540.50	21809175.
.07500	40730.0	540.60	22018638.
.08000	41140.0	540.60	22240284.
.08500	41550.0	540.60	22461930.
.09000	41960.0	540.60	22683576.
.09500	42370.0	540.60	22905222.
.10000	42780.0	540.70	23131146.
.11000	43640.0	540.70	23596148.
.12000	44550.0	540.80	24092640.
.13000	45470.0	540.90	24594723.
.14000	46390.0	540.90	25092351.

(CONTINUED)

TABLE 6.2-20B (Cont.)

(Historical)

TIME (SECONDS)	FLOW.RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	47330.0	541.10	25610263.
.16000	64470.0	542.10	34949187.
.17000	63760.0	542.00	34557920.
.18000	62720.0	541.80	33981696.
.19000	62140.0	541.70	33661238.
.20000	61360.0	541.50	33226440.
.22000	61040.0	541.40	33047056.
.24000	61310.0	541.40	33193234.
.26000	60880.0	541.40	32960432.
.28000	60520.0	541.30	32759476.
.30000	59810.0	541.20	32369172.
.32000	60140.0	541.30	32553782.
.34000	60030.0	541.30	32494239.
.36000	59330.0	541.20	32109396.
.38000	59220.0	541.20	32049864.
.40000	59150.0	541.30	32017895.
.42000	59330.0	541.30	32115329.
.44000	58770.0	541.30	31812201.
.46000	58380.0	541.30	31601094.
.48000	58380.0	541.30	31601094.
.50000	58250.0	541.40	31536550.
.55000	57660.0	541.40	31217124.
.60000	57090.0	541.50	30914235.
.65000	56900.0	541.70	30822730.
.70000	56200.0	541.80	30470832.
.75000	55760.0	541.90	30216344.
.80000	55350.0	542.10	30005235.
.85000	54820.0	542.30	29728886.
.90000	54390.0	542.50	29506575.
.95000	53970.0	542.70	29289519.
1.00000	53580.0	542.90	29088582.
1.10000	52830.0	543.40	28707822.
1.20000	51980.0	543.90	28271922.
1.30000	51050.0	544.40	27791620.
1.40000	50270.0	545.00	27397150.
1.50000	49900.0	545.60	27225440.
1.60000	49380.0	546.20	26971356.
1.70000	48780.0	546.80	26672904.
1.80000	48410.0	547.30	26494793.
1.90000	48120.0	547.90	26364948.
2.00000	47440.0	548.30	26011352.
2.50000	43830.0	550.10	24110883.
3.00000	40930.0	551.30	22564709.
3.50000	37330.0	553.00	20643490.
4.00000	35990.0	554.70	19963653.

TABLE 6.2-21A

MASS/ENERGY RELEASE RATES FOR 161 SQ IN
SURGE LINE GUIL BREAK AT PRESS. NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM HOT LEG SIDE) 1

Cycle 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	700.70	0.
.00100	7057.0	700.60	4944134.
.00200	6676.0	700.30	4675203.
.00300	6331.0	700.00	4431700.
.00400	6017.0	699.60	4209493.
.00500	5732.0	699.30	4008388.
.00600	5530.0	698.90	3864917.
.00700	5394.0	698.40	3767170.
.00800	5263.0	697.90	3673048.
.00900	5137.0	697.40	3582544.
.01000	5016.0	696.90	3495650.
.01200	4770.0	695.70	3318489.
.01400	4564.0	694.50	3169698.
.01600	4375.0	693.30	3033188.
.01800	4229.0	692.30	2927737.
.02000	4131.0	691.50	2856587.
.02200	4079.0	691.10	2818997.
.02400	4067.0	690.90	2809890.
.02600	4088.0	691.00	2824808.
.02800	4135.0	691.10	2857699.
.03000	4198.0	691.30	2902077.
.03200	4269.0	691.50	2952014.
.03400	4330.0	691.40	2993762.
.03600	4368.0	691.00	3018288.
.03800	4387.0	690.50	3029224.
.04000	4395.0	689.80	3031671.
.04200	4394.0	689.00	3027466.
.04400	4383.0	688.10	3015942.
.04600	4362.0	687.10	2997130.
.04800	4331.0	686.20	2971932.
.05000	4293.0	685.30	2941993.
.05500	4175.0	683.40	2853195.
.06000	4081.0	682.40	2784874.
.06500	4047.0	682.60	2762482.
.07000	4092.0	683.70	2797700.
.07500	4205.0	685.00	2880425.
.08000	4355.0	685.70	2986224.
.08500	4491.0	685.50	3078581.
.09000	4579.0	684.30	3133410.
.09500	4627.0	682.40	3157465.
.10000	4639.0	680.10	3154984.
.11000	4546.0	674.80	3067641.
.12000	4346.0	670.10	2912255.
.13000	4167.0	668.30	2784806.
.14000	4132.0	669.70	2767200.

(CONTINUED)

TABLE 6.2-21A(CONTD)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	4260.0	671.10	2858886.
.16000	4455.0	669.70	2983514.
.17000	4624.0	665.80	3078659.
.18000	4733.0	660.10	3124253.
.19000	4788.0	653.30	3128000.
.20000	4801.0	646.00	3101446.
.22000	4851.0	631.30	3062436.
.24000	4739.0	620.30	2939602.
.26000	4519.0	619.00	2797261.
.28000	4566.0	622.40	2841878.
.30000	4920.0	624.30	3071556.
.32000	5377.0	622.80	3348796.
.34000	5704.0	620.60	3539902.
.36000	5846.0	620.30	3626274.
.38000	5907.0	618.40	3652869.
.40000	5851.0	616.20	3605386.
.42000	5840.0	615.90	3596856.
.44000	5858.0	614.50	3599741.
.46000	5871.0	612.90	3598336.
.48000	5866.0	612.20	3591165.
.50000	5815.0	610.50	3550058.
.55000	5807.0	608.40	3532979.
.60000	5786.0	606.60	3509788.
.65000	5615.0	605.20	3398198.
.70000	5382.0	604.10	3251266.
.75000	5090.0	602.50	3066725.
.80000	4390.0	602.60	2645414.
.85000	4906.0	599.70	2942128.
.90000	4920.0	597.50	2939700.
.95000	3632.0	590.10	2143243.
1.00000	4519.0	595.60	2691516.
1.10000	4565.0	595.40	2718001.
1.20000	4679.0	595.80	2787748.
1.30000	3553.0	598.10	2125049.
1.40000	4715.0	595.60	2808254.
1.50000	4339.0	595.60	2584308.
1.60000	4321.0	595.90	2574884.
1.70000	3941.0	596.60	2351201.
1.80000	3956.0	595.20	2354611.
1.90000	3973.0	595.30	2365127.
2.00000	3233.0	612.70	1980859.
2.50000	2953.0	588.90	1739022.
3.00000	3397.0	618.20	2100025.
3.50000	3134.0	593.70	1860656.
4.00000	3460.0	605.00	2093300.

TABLE 6.2-21B

MASS/ENERGY RELEASE RATES FOR 161 SQ IN
SURGE LINE GUIL BREAK AT PRESS. NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM PRESSURIZER SIDE)
Cycle 1 | 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	700.70	0.
.00100	7250.0	700.70	5080075.
.00200	7249.0	700.70	5079374.
.00300	7249.0	700.70	5079374.
.00400	7248.0	700.70	5078674.
.00500	7247.0	700.70	5077973.
.00600	7247.0	700.70	5077973.
.00700	7246.0	700.70	5077272.
.00800	7246.0	700.70	5077272.
.00900	7245.0	700.70	5076572.
.01000	7244.0	700.70	5075871.
.01200	7243.0	700.70	5075170.
.01400	7242.0	700.70	5074469.
.01600	7240.0	700.60	5072344.
.01800	7239.0	700.60	5071643.
.02000	7238.0	700.60	5070943.
.02200	7236.0	700.60	5069542.
.02400	7235.0	700.60	5068841.
.02600	7234.0	700.60	5068140.
.02800	7233.0	700.60	5067440.
.03000	7231.0	700.60	5066039.
.03200	7230.0	700.60	5065338.
.03400	7229.0	700.60	5064637.
.03600	7227.0	700.60	5063236.
.03800	7226.0	700.60	5062536.
.04000	7225.0	700.60	5061835.
.04200	7224.0	700.60	5061134.
.04400	7222.0	700.60	5059733.
.04600	7221.0	700.50	5058311.
.04800	7220.0	700.50	5057610.
.05000	7218.0	700.50	5056209.
.05500	7215.0	700.50	5054108.
.06000	7212.0	700.50	5052006.
.06500	7209.0	700.50	5049905.
.07000	7205.0	700.50	5047103.
.07500	7202.0	700.50	5045001.
.08000	7198.0	700.40	5041479.
.08500	7193.0	700.40	5037977.
.09000	7189.0	700.40	5035176.
.09500	7184.0	700.40	5031674.
.10000	7179.0	700.40	5028172.
.11000	7169.0	700.30	5020451.
.12000	7160.0	700.30	5014148.
.13000	7150.0	700.30	5007145.
.14000	7141.0	700.20	5000128.

(CONTINUED)

TABLE 6.2-21B (CONT.)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	7131.0	700.20	4993126.
.16000	7122.0	700.20	4986824.
.17000	7112.0	700.10	4979111.
.18000	7103.0	700.10	4972810.
.19000	7093.0	700.10	4965809.
.20000	7084.0	700.00	4958800.
.22000	7064.0	700.00	4944800.
.24000	7045.0	699.90	4930796.
.26000	7026.0	699.80	4916795.
.28000	7007.0	699.80	4903499.
.30000	6989.0	699.70	4890203.
.32000	6970.0	699.60	4876212.
.34000	6950.0	699.50	4861525.
.36000	6929.0	699.50	4846836.
.38000	6909.0	699.40	4832165.
.40000	6889.0	699.30	4817478.
.42000	6870.0	699.30	4804191.
.44000	6850.0	699.20	4789520.
.46000	6831.0	699.10	4775552.
.48000	6811.0	699.00	4760889.
.50000	6792.0	699.00	4747608.
.55000	6740.0	698.80	4709912.
.60000	6693.0	698.60	4675730.
.65000	6645.0	698.40	4640868.
.70000	6599.0	698.20	4607422.
.75000	6552.0	698.10	4573951.
.80000	6500.0	697.90	4540537.
.85000	6461.0	697.70	4507840.
.90000	6416.0	697.50	4475160.
.95000	6371.0	697.30	4442498.
1.00000	6326.0	697.10	4409855.
1.10000	6238.0	696.70	4346015.
1.20000	6151.0	696.30	4282941.
1.30000	6065.0	695.80	4220027.
1.40000	5980.0	695.40	4158492.
1.50000	5896.0	695.00	4097720.
1.60000	5813.0	694.50	4037129.
1.70000	5724.0	694.10	3973028.
1.80000	5663.0	693.60	3927857.
1.90000	5609.0	693.10	3687598.
2.00000	5559.0	692.60	3850103.
2.50000	5338.0	689.90	3682686.
3.00000	5123.0	686.70	3517964.
3.50000	4903.0	682.90	3348259.
4.00000	4705.0	678.10	3190461.

TABLE 6.2-22

MASS/ENERGY RELEASE RATE FOR 9.284 SQ IN
PRESSURIZER RELIEF NOZZLE QUILL BREAK
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
(FLOW FROM PRESSURIZER SIDE)

Cycle 1 | 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	751.00	0.
.00100	675.3	751.00	507150.
.00200	675.3	751.00	507150.
.00300	675.3	751.00	507150.
.00400	675.3	751.00	507150.
.00500	675.3	751.00	507150.
.00600	675.3	751.00	507150.
.00700	675.2	751.00	507075.
.00800	675.2	751.00	507075.
.00900	675.2	751.00	507075.
.01000	675.2	751.00	507075.
.01200	675.2	751.00	507075.
.01400	675.2	751.00	507075.
.01600	675.2	751.00	507075.
.01800	675.1	751.00	507000.
.02000	675.1	751.00	507000.
.02200	675.1	751.00	507000.
.02400	675.1	751.00	507000.
.02600	675.1	751.00	507000.
.02800	675.0	751.00	506925.
.03000	675.0	751.00	506925.
.03200	675.0	751.00	506925.
.03400	675.0	751.00	506925.
.03600	675.0	751.00	506925.
.03800	674.9	751.00	506850.
.04000	674.9	751.00	506850.
.04200	674.9	751.00	506850.
.04400	674.9	751.00	506850.
.04600	674.9	751.00	506850.
.04800	674.9	751.00	506850.
.05000	674.8	751.00	506775.
.05500	674.8	751.00	506775.
.06000	674.7	751.00	506700.
.06500	674.7	751.00	506700.
.07000	674.6	751.00	506625.
.07500	674.6	751.00	506625.
.08000	674.5	751.00	506550.
.08500	674.5	751.00	506550.
.09000	674.4	751.00	506474.
.09500	674.4	751.00	506474.
.10000	674.4	751.00	506474.
.11000	674.3	751.00	506399.
.12000	674.2	751.00	506324.
.13000	674.1	751.00	506249.
.14000	674.0	751.00	506174.

(CONTINUED)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	673.9	751.00	506099.
.16000	673.8	751.00	506024.
.17000	673.7	751.00	505949.
.18000	673.6	751.00	505874.
.19000	673.5	751.00	505799.
.20000	673.4	750.90	505656.
.22000	673.2	750.90	505506.
.24000	673.0	750.90	505356.
.26000	672.8	750.90	505206.
.28000	672.7	750.90	505130.
.30000	672.5	750.90	504980.
.32000	672.3	750.90	504830.
.34000	672.1	750.90	504680.
.36000	671.9	750.90	504530.
.38000	671.7	750.90	504380.
.40000	671.5	750.90	504229.
.42000	671.4	750.90	504154.
.44000	671.2	750.90	504004.
.46000	671.0	750.90	503854.
.48000	670.8	750.90	503704.
.50000	670.6	750.90	503554.
.55000	670.1	750.90	503178.
.60000	669.7	750.80	502811.
.65000	669.2	750.80	502435.
.70000	668.8	750.80	502135.
.75000	668.3	750.80	501760.
.80000	667.9	750.80	501459.
.85000	667.4	750.70	501017.
.90000	667.0	750.70	500717.
.95000	666.5	750.70	500342.
1.00000	666.1	750.70	500041.
1.10000	665.2	750.60	499299.
1.20000	664.3	750.60	498624.
1.30000	663.4	750.60	497948.
1.40000	662.5	750.50	497206.
1.50000	661.6	750.50	496531.
1.60000	660.7	750.50	495855.
1.70000	659.8	750.40	495114.
1.80000	658.9	750.40	494439.
1.90000	658.1	750.40	493838.
2.00000	657.3	750.30	493172.
2.50000	653.8	749.80	490219.
3.00000	651.9	748.80	488143.
3.50000	651.1	747.60	486762.
4.00000	650.6	746.40	485608.

TABLE 6.2-23A

MASS/ENERGY RELEASE RATES FOR SPRAY LINE
GUILLOTINE BREAK AT PRESSURIZER NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
11.871 SQ IN BREAK AREA
(FLOW FROM PRESSURIZER SIDE)
Cycle 1 | 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	751.00	0.
.00100	188.2	751.00	141338.
.00200	188.2	751.00	141338.
.00300	188.2	751.00	141338.
.00400	188.2	751.00	141338.
.00500	188.2	751.00	141338.
.00600	188.2	751.00	141338.
.00700	188.2	751.00	141338.
.00800	188.2	751.00	141338.
.00900	188.2	751.00	141338.
.01000	188.2	751.00	141338.
.01200	188.2	751.00	141338.
.01400	188.2	751.00	141338.
.01600	188.2	751.00	141338.
.01800	188.2	751.00	141338.
.02000	188.2	751.00	141338.
.02200	188.2	751.00	141338.
.02400	188.2	751.00	141338.
.02600	188.2	751.00	141338.
.02800	188.2	751.00	141338.
.03000	188.2	751.00	141338.
.03200	188.2	751.00	141338.
.03400	188.2	751.00	141338.
.03600	188.2	751.00	141338.
.03800	188.2	751.00	141338.
.04000	188.2	751.00	141338.
.04200	188.2	751.00	141338.
.04400	188.2	751.00	141338.
.04600	188.2	751.00	141338.
.04800	188.2	751.00	141338.
.05000	188.2	751.00	141338.
.05500	188.2	751.00	141338.
.06000	188.2	751.00	141338.
.06500	188.2	751.00	141338.
.07000	188.2	751.00	141338.
.07500	188.2	751.00	141338.
.08000	188.2	751.00	141338.
.08500	188.2	751.00	141338.
.09000	188.2	751.00	141338.
.09500	188.2	751.00	141338.
.10000	188.2	751.00	141338.
.11000	188.1	751.00	141263.
.12000	188.1	751.00	141263.
.13000	188.1	751.00	141263.
.14000	188.1	751.00	141263.

(CONTINUED)

TABLE 6.2-23A (CONT.)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	188.1	751.00	141263.
.16000	188.1	751.00	141263.
.17000	188.1	751.00	141263.
.18000	188.1	751.00	141263.
.19000	188.1	751.00	141263.
.20000	188.0	751.00	141188.
.22000	188.0	751.00	141188.
.24000	188.0	751.00	141188.
.26000	188.0	751.00	141188.
.28000	187.9	751.00	141113.
.30000	187.9	751.00	141113.
.32000	187.8	751.00	141038.
.34000	187.8	751.00	141038.
.36000	187.7	751.00	140963.
.38000	187.7	751.00	140963.
.40000	187.6	750.90	140869.
.42000	187.6	750.90	140869.
.44000	187.5	750.90	140794.
.46000	187.4	750.90	140719.
.48000	187.4	750.90	140719.
.50000	187.3	750.90	140644.
.55000	187.0	750.90	140418.
.60000	186.8	750.90	140268.
.65000	186.5	750.90	140043.
.70000	186.2	750.90	139818.
.75000	185.9	750.80	139574.
.80000	185.5	750.80	139273.
.85000	185.2	750.80	139048.
.90000	184.8	750.80	138748.
.95000	184.4	750.70	138429.
1.00000	184.0	750.70	138129.
1.10000	183.2	750.60	137510.
1.20000	182.3	750.60	136834.
1.30000	181.5	750.50	136216.
1.40000	180.8	750.50	135690.
1.50000	180.2	750.40	135222.
1.60000	179.7	750.40	134847.
1.70000	179.3	750.40	134547.
1.80000	178.9	750.30	134229.
1.90000	178.7	750.30	134079.
2.00000	178.6	750.30	134004.
2.50000	178.2	750.30	133703.
3.00000	176.7	750.10	132543.
3.50000	174.2	750.00	130650.
4.00000	172.9	749.90	129658.

TABLE 6.2-23B

MASS/ENERGY RELEASE RATES FOR SPRAY LINE
GUILLOTINE BREAK AT PRESSURIZER NOZZLE
FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS
11.871 SQ IN BREAK AREA
(FLOW FROM COLD LEG SIDE)
Cycle 1 | 1

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
0.00000	0.0	544.70	0.
.00100	1546.0	544.60	841952.
.00200	1530.0	544.50	833085.
.00300	1516.0	544.50	825462.
.00400	1505.0	544.40	819322.
.00500	1495.0	544.30	813729.
.00600	1486.0	544.20	808681.
.00700	1480.0	544.20	805416.
.00800	1477.0	544.20	803783.
.00900	1476.0	544.20	803239.
.01000	1477.0	544.20	803783.
.01200	1486.0	544.30	808830.
.01400	1503.0	544.40	818233.
.01600	1522.0	544.50	828729.
.01800	1542.0	544.60	839773.
.02000	1561.0	544.70	850277.
.02200	1575.0	544.80	858060.
.02400	1583.0	544.90	862577.
.02600	1584.0	544.90	863122.
.02800	1580.0	544.80	860784.
.03000	1572.0	544.80	856426.
.03200	1563.0	544.70	851366.
.03400	1554.0	544.70	846464.
.03600	1548.0	544.70	843196.
.03800	1544.0	544.60	840862.
.04000	1542.0	544.60	839773.
.04200	1541.0	544.60	839229.
.04400	1540.0	544.60	838684.
.04600	1539.0	544.60	838139.
.04800	1536.0	544.60	836506.
.05000	1533.0	544.60	834872.
.05500	1523.0	544.50	829274.
.06000	1522.0	544.50	828729.
.06500	1538.0	544.60	837595.
.07000	1558.0	544.70	848643.
.07500	1561.0	544.70	850277.
.08000	1540.0	544.60	838684.
.08500	1514.0	544.40	824222.
.09000	1504.0	544.40	818778.
.09500	1517.0	544.50	826007.
.10000	1539.0	544.60	838139.
.11000	1545.0	544.60	841407.
.12000	1522.0	544.50	828729.
.13000	1518.0	544.50	826551.
.14000	1505.0	544.40	819322.

(CONTINUED)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)
.15000	1522.0	544.50	828729.
.16000	1539.0	544.60	838139.
.17000	1507.0	544.40	820411.
.18000	1499.0	544.30	815906.
.19000	1522.0	544.50	828729.
.20000	1523.0	544.50	829274.
.22000	1495.0	544.30	813729.
.24000	1524.0	544.50	829818.
.26000	1483.0	544.30	807197.
.28000	1514.0	544.50	824373.
.30000	1490.0	544.30	811007.
.32000	1508.0	544.40	820955.
.34000	1484.0	544.30	807741.
.36000	1506.0	544.40	819866.
.38000	1484.0	544.30	807741.
.40000	1501.0	544.40	817144.
.42000	1490.0	544.40	811156.
.44000	1496.0	544.40	814422.
.46000	1493.0	544.40	812789.
.48000	1497.0	544.40	814967.
.50000	1490.0	544.40	811156.
.55000	1494.0	544.50	813483.
.60000	1504.0	544.60	819078.
.65000	1510.0	544.60	822346.
.70000	1513.0	544.70	824131.
.75000	1520.0	544.80	826096.
.80000	1527.0	544.90	832062.
.85000	1535.0	545.00	836575.
.90000	1542.0	545.10	840544.
.95000	1548.0	545.20	843970.
1.00000	1554.0	545.30	847396.
1.10000	1563.0	545.50	852617.
1.20000	1566.0	545.70	854566.
1.30000	1562.0	545.80	852540.
1.40000	1554.0	546.00	848484.
1.50000	1542.0	546.10	842086.
1.60000	1528.0	546.20	834594.
1.70000	1513.0	546.20	826401.
1.80000	1500.0	546.30	819450.
1.90000	1489.0	546.50	813739.
2.00000	1481.0	546.60	809515.
2.50000	1493.0	547.60	817567.
3.00000	1508.0	548.70	827440.
3.50000	1470.0	549.30	807471.
4.00000	1455.0	550.20	800541.

TABLE 6.2-24

FULL DE GUILLOTINE MAIN STEAM LINE BREAK

(PT.16) FOR CONTAINMENT SUBCOMPARTMENT ANALYSIS

Cycle 1

1

FLOW FROM SG SIDE			FLOW FROM TURBINE SIDE		
Time (Sec)	Mass Flowrate (lbm/sec)	Enthalpy (Btu/Lbm)	Time (sec)	Mass Flowrate (lbm/sec)	Enthalpy (Btu/lbm)
0.0	0.0	1.19933×10^3	0.0	0.0	1.19939×10^3
.0011	8.05056×10^3	1.19111×10^3	.00110	7.07809×10^3	1.17993×10^3
.0025	6.04424×10^3	1.16928×10^3	.00270	3.62910×10^3	1.11925×10^3
.0038	5.6487×10^3	1.17067×10^3	.00470	1.45258×10^3	1.04429×10^3
.0060	5.74435×10^3	1.17226×10^3	.00600	0.818096×10^3	1.00633×10^3
.0080	5.46778×10^3	1.17143×10^3	.00740	0.64944×10^3	1.03556×10^3
.0100	5.54377×10^3	1.17953×10^3	.01000	1.08101×10^3	1.17254×10^3
.0122	5.82598×10^3	1.17752×10^3	.01900	3.73205×10^3	1.21186×10^3
.0140	5.55595×10^3	1.16532×10^3	.02700	5.34259×10^3	1.19658×10^3
.0170	4.69763×10^3	1.15526×10^3	.03300	5.98437×10^3	1.1868×10^3
.0181	4.60677×10^3	1.16087×10^3	.04300	6.30914×10^3	1.17335×10^3
.0200	4.78633×10^3	1.17681×10^3	.05300	6.13839×10^3	1.16209×10^3
.0240	5.48536×10^3	1.19944×10^3	.06000	5.88923×10^3	1.1558×10^3
.0260	5.82745×10^3	1.20299×10^3	.1000	4.46640×10^3	1.13503×10^3
.0280	6.08692×10^3	1.20101×10^3	.1250	3.93183×10^3	1.13242×10^3
.0301	6.17320×10^3	1.19509×10^3	.1540	3.6913×10^3	1.14996×10^3
.0430	5.69838×10^3	1.16705×10^3	.1800	3.86306×10^3	1.17082×10^3
.0500	5.21566×10^3	1.15847×10^3	.2150	4.27972×10^3	1.17981×10^3
.06200	4.90710×10^3	1.16638×10^3	.2500	4.43253×10^3	1.17696×10^3
.09150	5.42350×10^3	1.18471×10^3	.2750	4.39358×10^3	1.17435×10^3
.11950	5.04370×10^3	1.17606×10^3	.3860	4.15551×10^3	1.17436×10^3
.14000	5.20843×10^3	1.18766×10^3	.410	4.18010×10^3	1.17886×10^3
.1550	5.45279×10^3	1.20054×10^3	.490	4.46476×10^3	1.19291×10^3
.1800	6.10725×10^3	1.21321×10^3	.565	4.73559×10^3	1.19987×10^3
.20600	6.31569×10^3	1.21054×10^3	.625	4.87102×10^3	1.20151×10^3
.2400	6.21232×10^3	1.20318×10^3	.725	4.9739×10^3	1.2012×10^3
.2850	6.18929×10^3	1.20065×10^3	1.00	4.96876×10^3	1.19962×10^3
.340	6.13367×10^3	1.19839×10^3			
.400	6.11393×10^3	1.19835×10^3			
1.000	6.09481×10^3	1.19815×10^3			

6.2-25 TABLE

RESULTS OF SUBCOMPARTMENT ANALYSIS

Cycle 1

	Peak Node Pressure (psia)	Reference Figure*	Peak Wall Differential Pressure (psi)	Reference Figure*
REACTOR CAVITY (1)				
1 Hot leg guillotine Break at R.V. Nozzle	19.1	--	30.5	-
2 Discharge Leg Guillotine Break at R.V. Nozzle	100.1	6.2-25B	86.	6.2-251
STEAM GENERATOR				
3 Hot Leg Guillotine Freak at S.G. Nozzle (2)	20.2	-	5.6	-
4 Suction Leg Guillotine Break at S.G. Nozzle (2)	20.6	6.2-28A	5.9	6.2-28n
5 Suction Leg Guillotine Break at Pump Nozzle (2)	20.0	-	5.3	-
6 & 6a Suction Leg slot break at Pump Elbow & st S G Elbow (2)	18.0	-	3.2	-
7 Discharge Leg Guillotine Break at Pump Nozzle (2)	20.0	-	5.3	-
8 Main Steam Line Double Ended Guillotine Break at Point 16 (5)	17.3	-	2.4	-
PRESSURIZER AREA				
9 Pressurizer surge line guillotine Break at Pressurizer Nozzle (4)	37.4 (EL 29.5') 16.0 (EL 62')	 -	22.7 (6) 1.3	6.2-30e 6.2-30f
10 Pressurizer Safety Relief Line Guillotine Break at Pressurizer Nozzle (3)	14.8 (EL 62')	-	0.041	-
11 Pressurizer Spray line Guillotine Break at Pressurizer Nozzle (4)	14.9 (EL 62')	-	0.064	-

* Only worst cases presented

Note: Subcompartment walls are designed to withstand the maximum pressure differential resulting from the design basis subcompartment break.

(1) Analysis is no longer required due to application of the LBB criteria (refer to UFSAR 6.2.1.2.3)
(2) This break is eliminated due to the application of LBB criteria (refer to UFSAR 6.2.1.2.3)
(3) The pre-EPU M&E release data remains unchanged at EPU conditions (refer to UFSAR 6.2.1.2.3)
(4) The pre-EPU M&E release data slightly increases at EPU conditions (refer to UFSAR 6.2.1.2.3)
(5) The pre-EPU M&E release data for this break is not the limiting case
(6) 22.5 x 1.008 = 22.7 (based on Figure 6.2-30e)

TABLE 6.2-26
(Historical)
REACTOR CAVITY SUBCOMPARTMENT ANALYSIS
VENT PATH DESCRIPTION

Junction Number	From Vol. (defines Forward)	To Vol	Area (ft^2)	Elevation (ft)	Inertia Coefficient (L/A) (ft^{-1})	Total Head Forward	Loss Coefficient Reverse	Comments
1	1	7	34.36	25.5	0.1343	0.2229	0.2113	
2	2	8	16.726	"	0.4365	0.9132	1.0801	
3	3	9	16.726	"	0.3363	0.8955	1.0624	From Cavity to Pipe Penetrations
4	4	10	34.36	"	0.1343	0.2229	0.2113	
5	5	11	16.726	"	0.3114	0.8892	1.0560	
6	6	12	16.726	"	0.2362	0.8689	1.0360	
7	7	32	31.3	"	0.1243	1.0065	0.54	
8	8	32	16.726	"	0.4265	1.0756	0.60	
9	9	33	66.4	"	0.0916	0.9455	0.55	From Pipe Penetrations to Sec. Shld. Wall Area
10	10	33	31.3	"	0.1243	1.0065	0.54	
11	11	33	16.726	"	0.3014	1.074	0.60	
12	12	32	16.726	25.5	0.2262	1.0539	0.58	
13	1	41	24.94	36.0	0.2510	0.582	0.361	
14	2	41	24.94	"	"	"	"	
15	3	41	24.94	"	"	"	"	
16	4	41	24.94	"	"	"	"	Vent Path to Upper Cavity
17	5	41	24.94	"	"	"	"	
18	6	41	24.94	36.0	0.2510	0.582	0.361	
19	1	13	13.33	24.5	0.72220	0.308	0.300	
20	2	14	5.7	"	1.556	0.516	0.490	
21	2	15	5.7	"	1.556	0.516	0.490	
22	3	16	16.2	"	0.6460	0.213	0.248	
23	4	17	2.834	"	2.664	1.027	0.801	
24	4	18	2.834	"	2.664	1.027	0.801	
25	5	19	16.2	"	0.6460	1.213	0.248	
26	6	20	5.7	"	1.556	0.516	0.490	
27	6	21	5.7	24.5	1.556	0.516	0.490	
28	1	2	16.39	24.5	0.574	0.455	0.494	
29	2	3	14.01	"	0.543	0.371	0.439	
30	3	4	11.14	"	0.606	0.571	0.436	
31	5	4	11.14	"	0.606	0.571	0.436	
32	6	5	14.01	"	0.543	0.371	0.439	
33	1	6	16.39	24.5	0.574	0.455	0.494	
34	13	14	35.31	10.0	0.2176	0.035	0.035	
35	14	15	11.14	"	0.23	0.843	0.843	
36	15	16	35.31	"	0.2176	0.051	0.051	
37	16	17	35.31	"	0.2176	0.051	0.051	
38	18	17	11.14	"	0.23	0.843	0.843	
39	19	18	35.31	"	0.2176	0.051	0.051	
40	20	19	35.31	"	0.2176	0.051	0.051	
41	21	20	11.14	"	0.23	0.843	0.843	
42	13	21	35.31	"	0.2176	0.051	0.051	
43	13	22	24.94	"	0.4624	0.152	0.137	
44	14	23	11.417	"	1.01	0.223	0.207	
45	15	24	11.417	"	1.01	0.223	0.207	

TABLE 6.2-26 (Cont'd)
(Historical)

Junction Number	From Vol. To Vol. (defines Forward)		Area (ft^2)	Elevation (ft)	Inertia Coefficient (L/A) (ft^{-1})	Total Head Loss Coefficient Forward	Reverse	COMMENTS
46	16	25	24.94	10.0	0.4624	0.152	0.137	
47	17	26	11.417	"	1.01	0.223	0.207	
48	18	27	11.417	"	1.01	0.223	0.207	
49	19	28	24.94	"	0.4624	0.152	0.137	
50	20	29	11.417	"	1.01	0.223	0.207	
51	21	30	11.417	10.0	1.01	0.223	0.207	
52	22	23	36.6	1.4349	0.210	0.0841	0.0841	
53	23	24	22.32	"	0.229	0.373	0.373	
54	24	25	36.6	"	0.210	0.0841	0.0341	
55	25	26	36.6	"	0.210	0.0841	0.0841	
56	26	27	22.32	"	0.229	0.373	0.373	
57	28	27	36.6	"	0.210	0.0841	0.0841	
58	29	28	36.6	"	0.210	0.0841	0.0841	
59	30	29	22.32	"	0.229	0.373	0.373	
60	22	30	36.6	"	0.210	0.0841	0.0841	
61	22	31	63.36	"	0.1034	0.777	0.411	
62	23	31	30.7	"	0.20	0.8938	0.44	
63	24	31	30.7	"	0.20	0.8938	0.44	
64	25	31	63.36	"	0.1034	0.777	0.411	
65	26	31	30.7	"	0.20	0.8938	0.44	
66	27	31	30.7	"	0.20	0.8938	0.44	
67	28	31	63.36	"	0.1034	0.777	0.411	
68	29	31	30.7	"	0.20	0.8938	0.44	
69	30	31	30.7	1.4349	0.20	0.8938	0.44	
70	31	40	16.3216	-2.92	0.258	0.8357	1.006	
71	40	32	49.0	18.0	0.1935	1.013	0.4968	
72	22	34	43.125	1.4349	0.2255	0.335	0.307	
73	25	35	43.125	1.4349	0.2255	0.335	0.307	
74	34	36	20.21	0.0	0.2549	2.555	2.555	Relief
75	35	37	25.875	"	0.2549	2.554	2.554	Dampers
76	36	38	43.125	"	0.1343	0.271	0.271	
77	37	38	43.125	"	0.1343	0.271	0.271	
78	38	39	78.0	0.0	0.192	1.012	0.512	
79	39	43	78.0	7.5	0.1204	0.47	0.824	
80	41	42	144.0	62.0	0.0239	1.3812	1.1711	
81	33	42	218.0	62.0	0.0177	0.65	0.397	
82	32	42	218.0	62.0	0.0177	0.65	0.397	
83	43	32	78.0	18.0	0.1230	1.020	0.5313	

TABLE 6.2-27
(Historical)
REACTOR CAVITY SUBCOMPARTMENT ANALYSIS
NODE DESCRIPTION

Node Number	Volume (ft^3)	Height (ft)	Elevation (ft)
01	228.24	11.501	24.5
02	239.95	11.501	24.5
03	249.92	11.501	24.5
04	218.27	11.501	24.5
05	249.92	11.501	24.5
06	239.95	11.501	24.5
07	216.48	8.0	25.5
08	354.0	7.0	25.5
09	497.52	7.0	25.5
10	216.48	8.0	25.5
11	247.04	7.0	25.5
12	208.31	7.0	25.5
13	343.6	14.501	10.0
14	113.43	14.501	10.0
15	113.43	14.501	10.0
16	343.6	14.501	10.0
17	113.43	14.501	10.0
18	113.43	14.501	10.0
19	343.6	14.501	10.0
20	113.43	14.501	10.0
21	113.43	14.501	10.0
22	307.15	8.5652	1.4349
23	138.5	8.5652	1.4349
24	138.5	8.5652	1.4349
25	307.15	8.5652	1.4349
26	138.5	8.5652	1.4349
27	138.5	8.5652	1.4349
28	307.15	8.5652	1.4349
29	138.5	8.5652	1.4349
30	138.5	8.5652	1.4349
31	1483.65	4.3550	-2.92
32	73316.0	44.0	18.0
33	73316.0	44.0	18.0
34	582.0	7.50	0.0
35	453.0	7.50	0.0
36	336.0	7.50	0.0
37	455.0	7.50	0.0
38	396.0	7.50	0.0
39	5558.0	7.501	0.0
40	1556.1	25.001	-7.0
41	155102+5	32.5	36.0
42	1.485+6	157.0	62.0
43	1638.0	10.5	7.5

TABLE 6.2-28

SECONDARY SHIELD WALL SUBCOMPARTMENT ANALYSIS

VENT PATH DESCRIPTIONS

Cycle 1 | 1

JUN NUM	FROM VOL	TO VOL	JUNCTION FLOW AREA (FT**2)	JUNCTION ELEVATION (FT)	JUNCTION INERTIA (L/A) (FT-1)	TOTAL HEAD LOSS COEF. (FORWARD)	TOTAL HEAD LOSS COEF. (REVERSE)
1	1	2	.448140E+03	.180000E+02	.442530E-01	.627090E-01	.508850E-01
2	1	3	.280620E+03	.360000E+02	.750440E-01	.124250E+00	.104955E+00
3	2	4	.174686E+04	.360000E+02	.127500E-01	.765180E-01	.593040E-01
4	3	4	.309860E+03	.360000E+02	.502280E-01	.326620E+00	.317262E+00
5	3	5	.111130E+03	.580000E+02	.637770E-01	.439480E+00	.598859E+00
6	4	6	.174900E+03	.580000E+02	.284880E-01	.461340E+00	.804227E+00
7	5	6	.361600E+02	.580000E+02	.247617E+00	.256280E+00	.248869E+00
8	5	7	.143270E+03	.620000E+02	.687300E-01	.942980E-01	.857130E-01
9	6	8	.147410E+03	.620000E+02	.679680E-01	.114250E+00	.736980E-01
10	7	8	.400500E+02	.620000E+02	.140000E+00	.961640E+00	.947378E+00
11	7	19	.105190E+03	.760000E+02	.692870E-01	.117450E+01	.630984E+00
12	8	19	.113500E+03	.760000E+02	.678950E-01	.114800E+01	.596697E+00
13	1	9	.208250E+03	.180000E+02	.475000E-01	.107230E+01	.770253E+00
14	2	9	.148650E+03	.180000E+02	.572590E-01	.119450E+01	.947378E+00
15	2	10	.176180E+03	.180000E+02	.444800E-01	.113580E+01	.858540E+00
16	3	9	.189700E+03	.360000E+02	.493780E-01	.110990E+01	.820577E+00
17	4	9	.145720E+03	.360000E+02	.578000E-01	.120090E+01	.951836E+00
18	4	10	.193100E+03	.360000E+02	.434900E-01	.110460E+01	.811898E+00
19	4	5	.596200E+02	.580000E+02	.516450E-01	.889620E+00	.125658E+01
20	4	19	.989570E+03	.580000E+02	.176450E-01	.105880E+01	.685130E+00
21	6	8	.216400E+02	.620000E+02	.146150E+00	.120400E+01	.119557E+01
22	9	11	.221990E+03	.180000E+02	.438530E-01	.828510E+00	.104159E+01
23	9	12	.147410E+03	.180000E+02	.574850E-01	.102490E+01	.119320E+01
24	10	12	.260600E+02	.180000E+02	.109874E+00	.143040E+01	.145284E+01
25	9	13	.189700E+03	.360000E+02	.493780E-01	.823760E+00	.110674E+01
26	9	14	.144180E+03	.360000E+02	.580930E-01	.960900E+00	.119999E+01
27	10	14	.232170E+03	.360000E+02	.417420E-01	.714000E+00	.102173E+01
28	14	19	.994120E+03	.580000E+02	.176270E-01	.105680E+01	.682620E+00
29	20	26	.250000E+02	.180000E+02	.330000E+00	.338281E+00	.338281E+00
30	23	26	.250000E+02	.180000E+02	.330000E+00	.338281E+00	.338281E+00
31	11	12	.544790E+03	.180000E+02	.366110E-01	.585970E-01	.585970E-01
32	11	13	.278620E+03	.360000E+02	.727910E-01	.162900E+00	.136563E+00
33	12	14	.106390E+04	.360000E+02	.142200E-01	.436090E+00	.424450E+00
34	13	14	.151910E+03	.360000E+02	.703610E-01	.839200E+00	.824091E+00
35	14	15	.596200E+02	.580000E+02	.516450E-01	.889620E+00	.125658E+01
36	13	15	.111130E+03	.580000E+02	.637770E-01	.439480E+00	.598859E+00
37	15	16	.361600E+02	.580000E+02	.247617E+00	.256280E+00	.248869E+00
38	16	18	.147220E+03	.620000E+02	.679850E-01	.115000E+00	.745320E-01

TABLE 6.2-28 (Cont'd)

JUN NUM	FROM VOL	TO VOL	JUNCTION FLOW AREA (FT**2)	JUNCTION ELEVATION (FT)	JUNCTION DIAMETER (FT)	JUNCTION INERTIA (L/A) (FT-1)	TOTAL HEAD LOSS COEF. (FORWARD)	TOTAL HEAD LOSS COEF. (REVERSE)
39	15	17	.143270E+03	.620000E+02		.687300E-01	.942980E-01	.857130E-01
40	17	18	.400500E+02	.620000E+02		.140000E+00	.961640E+00	.947378E+00
41	18	19	.113500E+03	.760000E+02	.120213E+02	.678950E-01	.114800E+01	.596697E+00
42	17	19	.105190E+03	.760000E+02	.115729E+02	.692870E-01	.117450E+01	.630984E+00
43	9	10	.534470E+03	.480000E+02	.260866E+02	.342980E-01	.849210E+00	.846940E+00
44	2	20	.148000E+03	.180000E+02	.137273E+02	.700840E-01	.400100E+00	.400100E+00
45	12	23	.876000E+02	.180000E+02	.105611E+02	.859100E-01	.715530E+00	.715530E+00
46	20	21	.505080E+03	.230000E+02	.253592E+02	.199520E-01	.150450E+01	.150450E+01
47	21	22	.124300E+04	.450000E+02	.397824E+02	.106910E-01	.169150E+01	.168904E+01
48	23	24	.441050E+03	.230000E+02	.236973E+02	.219830E-01	.156200E+01	.156200E+01
49	24	25	.135130E+04	.450000E+02	.414793E+02	.103520E-01	.164460E+01	.164209E+01
50	10	26	.187400E+03	.180000E+02	.154468E+02	.646070E-01	.456530E+00	.456530E+00
51	26	27	.449360E+03	.230000E+02	.239195E+02	.231700E-01	.146200E+01	.146200E+01
52	27	28	.894700E+03	.450000E+02	.337516E+02	.137530E-01	.176290E+01	.175967E+01
53	28	19	.116930E+04	.620000E+02	.385850E+02	.162920E-01	.224030E+01	.183911E+01
54	25	19	.116730E+04	.620000E+02	.385519E+02	.151440E-01	.231970E+01	.196317E+01
55	22	19	.140630E+04	.620000E+02	.423150E+02	.146530E-01	.221560E+01	.185021E+01
56	4	22	.170000E+03	.542500E+02	.147123E+02	.369056E+00	.147940E+01	.155736E+01
57	10	28	.190000E+03	.542500E+02	.155536E+02	.358736E+00	.156910E+01	.190675E+01
58	14	25	.107500E+03	.542500E+02	.116993E+02	.427196E+00	.177640E+01	.185214E+01
59	21	27	.350400E+03	.230000E+02	.211221E+02	.401790E+00	.155380E+00	.751700E-01
60	27	24	.350400E+03	.230000E+02	.211221E+02	.401790E+00	.155380E+00	.751700E-01
61	22	28	.270400E+03	.450000E+02	.185549E+02	.519983E+00	.170410E+00	.808710E-01
62	28	25	.270400E+03	.450000E+02	.185549E+02	.519983E+00	.170410E+00	.808710E-01
63	4	6	.216400E+02	.580000E+02	.524909E+01	.387415E+00	.130840E+01	.142766E+01
64	14	16	.216400E+02	.580000E+02	.524909E+01	.387415E+00	.130840E+01	.142766E+01
65	16	18	.216400E+02	.620000E+02	.524909E+01	.239243E+00	.121590E+01	.119963E+01
66	14	16	.174900E+03	.580000E+02	.149228E+02	.291640E-01	.461340E+00	.804227E+00
67	29	30	.230400E+03	.295000E+02	.171276E+02	.122010E+00	.592357E+00	.432210E+00
68	30	12	.166720E+03	.295000E+02	.145696E+02	.644460E-01	.435452E+00	.332340E+00
69	30	31	.150000E+02	.399200E+02	.437019E+01	.359280E+00	.833508E+00	.817140E+00
70	31	32	.253000E+02	.580000E+02	.567565E+01	.259988E+00	.262170E+00	.261890E+00
71	32	33	.253000E+02	.620000E+02	.567565E+01	.114644E+00	.666609E+00	.414870E+00
72	33	34	.131200E+03	.715000E+02	.129247E+02	.712700E-01	.105755E+00	.163330E+00
73	33	19	.280000E+02	.620000E+02	.597082E+01	.563058E+00	.111976E+01	.568280E+00
74	31	14	.130000E+03	.489200E+02	.128655E+02	.105777E+00	.476882E+00	.487610E+00
75	35	29	.579600E+02	.399200E+02	.859052E+01	.201316E+00	.217310E-01	.600120E-01

TABLE 6.2-29

SECONDARY SHIELD WALL SUBCOMPARTMENT ANALYSIS
NODE DESCRIPTION

VOL NUM	VOLUME (FT**3)	HEIGHT (FT)	ELEVATION (FT)	
1	.365390E+04	.180010E+02	.180000E+02	
2	.297148E+05	.180010E+02	.180000E+02	
3	.427430E+04	.220010E+02	.360000E+02	
4	.344284E+05	.220010E+02	.360000E+02	
5	.615400E+03	.400100E+01	.580000E+02	NORTH STEAM GENERATOR AREA
6	.631900E+03	.400100E+01	.580000E+02	
7	.157420E+04	.140010E+02	.620000E+02	
8	.165110E+04	.140010E+02	.620000E+02	
9	.288513E+05	.400000E+02	.180000E+02	
10	.307145E+05	.400000E+02	.180000E+02	
11	.330950E+04	.180010E+02	.180000E+02	
12	.171973E+05	.180010E+02	.180000E+02	
13	.426190E+04	.220010E+02	.360000E+02	
14	.196305E+05	.220010E+02	.360000E+02	
15	.615400E+03	.400100E+01	.580000E+02	SOUTH STEAM GENERATOR AREA
16	.631900E+03	.400100E+01	.580000E+02	
17	.157420E+04	.140010E+02	.620000E+02	
18	.165110E+04	.140010E+02	.620000E+02	
19	.144000E+07	.157000E+03	.580000E+02	UPPER CONTAINMENT
20	.820360E+04	.110010E+02	.120000E+02	
21	.435412E+05	.220010E+02	.230000E+02	
22	.343250E+05	.170010E+02	.450000E+02	
23	.872530E+04	.110010E+02	.120000E+02	
24	.465777E+05	.220010E+02	.230000E+02	
25	.338135E+05	.170010E+02	.450000E+02	VOLUMES OUTSIDE SECONDARY SHIELD WALL
26	.905170E+04	.110010E+02	.120000E+02	
27	.364271E+05	.220010E+02	.230000E+02	
28	.276762E+05	.170010E+02	.450000E+02	
29	.635460E+03	.104210E+02	.295000E+02	
30	.635700E+03	.104210E+02	.295000E+02	
31	.899540E+03	.180810E+02	.399200E+02	PRESSURIZER AREA VOLUMES
32	.101960E+03	.400100E+01	.580000E+02	
33	.150750E+04	.950100E+01	.620000E+02	
34	.240690E+04	.135010E+02	.715000E+02	
35	.165489E+03	.500000E+01	.399200E+02	

TABLE 6.2-30

APPROXIMATE NSSS HEIGHTS AND FLOW AREAS

Location	Height (Ft)
Bottom of lower plenum	0.0
Bottom of annulus	3.139
Bottom of active core	9.663
Top of active core	21.055
Bottom of fuel alignment plate	22.271
Center of reactor vessel	27.040
Bottom of upper guide structure support plate	31.218
Top of Reactor Vessel upper head	40.005
Bottom of Suction Leg at steam generator	19.790
Bottom of steam generator plenum	27.398
Bottom of tube sheet	33.945
Top of tube sheet	35.822
Top of innermost tube bundle	58.081
Top of outermost tube bundle	64.733
Top of steam generator	86.354
Bottom of pressurizer	40.019
Top of pressurizer	69.878

AREAS

Location	Flow Area (Ft^2)
Pressurizer	51.05
Reactor vessel annulus	40.173
Active core	58.731
Hot Leg	9.62
SG active tubes	21.64
Cold leg	4.91

Table 6.2-31

This table has been deleted.

Table 6.2-32
This table has been deleted.

Table 6.2-33
This table has been deleted.

Table 6.2-34
This table has been deleted.

Table 6.2-35
This table has been deleted.

TABLE 6.2-36
VALVE CLOSURE TIMES AND ASSUMPTIONS USED IN MSLB ANALYSIS
EPU

Valve Closure Times (from receipt of signal)

1.	MSIV (I-HCV-08-1A and T-HCV-08-1B) closure time, sec	5.6
2.	MFIV (I-HCV-09-1A and I-HCV-09-2A) closure time, sec	4.0
3.	Backup feedwater isolation valve (I-HCV-09-1B,2B) closure time, sec	4.0
4.	Reactor trip on low steam generator pressure, psia	546
5.	Reactor trip on high containment pressure, psig	6.4
6.	Instrumentation delay time, sec	1.15

Assumptions

1. Credit is not taken for the coastdown of the main feedwater pumps or condensate pumps in the MSLB analysis. Therefore, no degradation of the feedwater flow occur until the closure of the main feedwater isolation or backup valves.

2. Offsite power is assumed to be available for the analysis. Availability of offsite power allows the continuation of reactor coolant pump and feedwater pump flows. Maintaining reactor coolant and feedwater pump flow maximizes the rate of primary to secondary heat transfer which maximizes the rate of mass/energy release.

Table 6.2-37
This table has been deleted.

PAGE LEFT INTENTIONALLY BLANK

TABLE 6.2-38

DESIGN DATA FOR CONTAINMENT SPRAY SYSTEM COMPONENTS

1. Containment Spray Pumps

Quantity	2
Type	Single stage vertical centrifugal
Number of Stages	1
Rotational speed, rpm	1780
Material	Casing - ASME SA182 F304 Impeller - ASTM A296 CA6NM Stainless steel for pressure parts in contact with borated water
Codes & Standards	Standards of the Hydraulic Institute, ASME Section III, 1974 Edition Winter Addenda Class 2
Liquid Pumped	Borated water (concentration of 1900 ppm boron buffered to a pH of 7.0 to 8.1)
Temperature Maximum, F Minimum, F	 300 40
Design Pressure, psig	300
Capacity each, gpm Total Dynamic Head, ft	2850* 470
Connections Discharge, in. Suction, in.	 8 14

2. Motors

Quantity	One per pump
Type	Constant speed, 500 hp, 4000 volt, 60 Hz, 3-phase induction, 1.15 service factor
Insulation	Class F
Enclosure	WP II

* Includes 150 gpm for minimum recirculation flow to refueling water tank.

TABLE 6.2-38 (Cont'd)

2. Motors(Cont'd)

Shaft	Solid vertical
Code	NEMA

3. Piping, Fittings, and Valves

a. Suction Side

Material, Piping	Type 304 or 316 stainless steel
Pressure, psig	60
Temperature, F	300
Pipe Sizes	
2 in. and smaller	Schedule 40 S or higher wall thickness
2-1/2 in. through 12.in	Schedule 10 S or higher wall thickness
14 in. through 24 in.	0.250 in. nominal wall thickness
Connections	
2 in. and smaller	Socket weld
2-1/2 in. and larger	Butt weld
Code, Piping	ANSI B31.1, ASME III, Class 2 1971 Edition, Summer 1973 Addenda
Valves	
2 in. and smaller	Stainless steel, 600 lb rating
2-1/2 in. and larger	Stainless steel, 150 and 300 lb rating
Code, Valves	ANSI B16.5, ASME III, Class 2, 1974 Edition 1974 Edition (Summer 1975 Addenda) 1971 Edition (Winter 1973 Addenda)

b. Discharge Side

Material, piping	Type 304 or 316 stainless steel
Pressure, psig	300
Temperature, F	300
Pipe Sizes	
6 in. and smaller	Schedule 40 S or higher wall thickness
8 in. through 12 in.	0.250 in. nominal wall thickness

TABLE 6.2-38(Cont'd)

b.	Discharge Side (Cont'd)	
	Connections	
	2 in. and smaller	Socket weld
	2-1/2 in. and larger	Butt weld
	Code, Piping	Same as Suction Piping
3.	Piping, Fittings, and Valves (Cont'd)	
	Valves	
	2 in. and smaller	Stainless steel 600 lb rating
	2- ½ in. and larger	Stainless steel 150 to 300 lb rating
	Code, valves	Same as Suction Piping
4.	Spray Nozzles	Also refer to Table 6.2-43 Spray Nozzle Design Data
	Quantity	178 per train (minimum)
	Type	Open Throat
	Material	Type 304 stainless steel
	Flow per nozzle, gpm	15.2
	Pressure drop, psi	40 when passing 15.2 gpm
	Spray droplet size, microns	700 (3.93×10^{-5} in. mean value)
	Orifice size, in.	0.375
5.	Shutdown Heat Exchanger	Refer to Table 5.4-4 Shutdown Cooling Heat Exchanger Data

TABLE 6.2-38 (Cont'd)

DESIGN DATA AND MATERIALS FOR CONTAINMENT COOLING SYSTEM COMPONENTS (HVS-1A,1B,1C, & 1D)

1. Fans

Quantity	4	
Type	Vaneaxial, airfoil blades (adjustable) direct motor driven	
	Normal Conditions	Post-LOCA Conditions
Air flow each, cfm	60,000	39,600
Air mixture density, lb/ft^{3}pressure	0.075	0.187
Fan external static inches wg	4.0	3.7
Fan speed, rpm	1770	1170
Outlet velocity, FPM	5454	3600
Material: Housing	ASTM A 283 carbon steel	
Material: Blades	1704 pH stainless steel	
Bearings	Anti friction ball type, 100,000 hour B-10 Rating Life	
Codes	ASME IX and DWS D1.1 for welding	

2. Fan Motors

Quantity	One per fan
Horsepower	125 HP/83 HP
Voltage and type	460 volt, 60 Hz, 3 phase induction type.
Insulation	Class H Type RN (tested to radiation exposure of 2 x 10^8 rad)
Enclosure and Ventilation	Totally enclosed air over motor
Speed RPM	1770/1170
Start up time for full speed	Max. 10 seconds

3. Cooling Coils

Number of coil banks	Two banks of 4 coils each
Type	Horizontal tube, vertical plate fin

3. Cooling Coils (cont'd)

Face area, ft^2	112
Tube material	ASME SB-75 Alloy 122
Tube O.D., in.	5/8
Minimum wall thickness in.	0.049
Fins per inch	6
Coil Frame Material	ASME SA-240 Type 304
Coil Header Material	ASME SB-42 Alloy 122
Stub and Flange Material	ASME SB-466 Alloy 706
Code	ASME III, Class 2

	Normal Conditions	Post-LOCA** Condition
Heat removal capacity, BTU/hr/unit	1×10^6	61.6×10^6
Air Temperature, inlet/outlet	120/103.2	264/257.3
Component water temperature, inlet/outlet /F	100/101.8	100/206.4
Component cooling water flow each, gpm	1200	1200
Fouling factor	.001	.0005

4. Other

Lubricants	Bearing lubricants have been tested to radiation exposure of 2×10^8 rad
Housing	Constructed of carbon steel with structural steel reinforcing

** Note that these values represent conditions specified for procurement purposes. The actual heat load will vary based on the accident Analyses (Ref. 27).

TABLE 6.2-39

MATERIAL OF CONSTRUCTION FOR INSULATION USED INSIDE CONTAINMENT

Component	Materials
Reflective Insulation (R)	
Outside Jacket	Stainless steel
Insulation	Stainless steel foil
Inside Jacket	Stainless steel
Straps	Stainless steel
Fasteners	Stainless steel
Buckles	Stainless steel
Buckle springs	Stainless steel
Support collars	Stainless steel
Support collar springs	Stainless steel
Reactor vessel insulation support structure	Carbon steel and stainless steel
Bolts, nuts, washers	Stainless steel
Encapsulated Insulation (E)	
Type A	
Jacket	Stainless steel
Insulation	Fiberglass
Fasteners	Stainless steel
Type B	
Jacket	Stainless Steel
Insulation	
Pipe Shield	Calcium Silicate/Marinite
Piping Outside Bioshield	Calcium Silicate
Clamps	Stainless Steel

TABLE 6.2-40

LOCATION AND QUANTITY OF MAJOR THERMAL INSULATION MATERIALS INSIDE CONTAINMENT

A. Primary System Equipment

Location	Insulation Type*	Total Estimated Quantity ft^2
Reactor vessel (excluding closure head)	R	1420
R/C pumps	R	4632
Steam Generators	R	6840
Pressurizer	E	1250
Regenerative heat exchanger	E	175
Reactor closure head (L-Panel)	E	292
Reactor closure head (dome)	R	208

B.

Piping Size	Insulation Type*	Total Estimated Quantity linear ft
42 in.	E	30
36 in.	E	22
34 in.	E	300
30 in.	E	150
20 in.	E	200
18 in.	E	32

General Note - All quantities are approximate.

*Note: R = Reflective insulation
E = Encapsulated insulation

TABLE 6.2-40 (Cont'd)

B. Piping Size (Cont'd)	Insulation Type*	Total Estimated Quantity linear ft
12 in.	E	465
10 in.	E	446
8 in.	E	440
6 in.	E	1300
4 in.	E	274
3 in.	E	270
2.5 in.	E	87
2 in.	E	1039
1.5 in.	E	56
1.0 in.	E	1500
3/4 in.	E	2420
1/2 in.	E	77
3/8 in.	E	450

THIS PAGE IS
INTENTIONALLY
BLANK

TABLE 6.2-41

FAILURE MODE AND EFFECTS ANALYSIS - CONTAINMENT HEAT REMOVAL SYSTEM

Component Identification And Quantity	Failure	Effect on System	Method of Detection	Monitor	Remarks
Containment spray pump (2)	Fails to start	Loss of 100% capacity spray substation	Pump start failure/ CSAS override and pump motor overload alarm/trip	CRI	One 100% capacity spray subsystem operable supplemented by the containment cooling system
Spray Line Flow Control Valve FCV-07-1A/B (pneumatic) (2)	Loss of air supply	No effect on system-fail opens valves	Full flow indication upstream of valve and valve position indicating llights.	CRI	
	Fails to open	Loss of one 100% capacity spray subsystem	No flow indication upstream of valve and valve position indicating lights. Low Header Pressure alarm	CRI	One 100% capacity spray subsystem remains operable supplemented by containment cooling system.
Sump recirculation suction line valve MV-07-2A/B	Fails to open on RAS	Loss of one 100% capacity recirculation mode subsystem	Loss flow indication and low pressure alarm downstream of HX – valve position indicating lights. Failure to open alarm	CRI	One 100% capacity recirculation mode subsystem remains operable supplemented by the containment cooling system.
Refueling water tank suction line valve MV-07-1A,B (2)	Fails to close After RAS	No effect on spray subsystem-recirculating mode will be operating	Valve position indicating lights Failure to close alarm	CRI	Check valves V07119 & V07120 prevent spray water flow from containment sump back to refueling water tank
Component Cooling Water pump (2)	Loss of cooling water	Loss of two fan coolers and one shutdown heat exchanger	Containment airflow high temperature alarm and CCW low flow alarm	CRI	Two containment fan coolers and one containment spray substation remain operable.
Containment cooling Fan (4)	Fails to start	Loss of one containment fan cooler	Containment fan cooler low airflow alarm	CRI	Three remaining fan coolers and two containment spray subsystem are operable
Diesel generator set (2)	Fails to start	Loss of half of the containment heat removal system	Diesel generator malfunction alarm	CRI	One diesel generator set remains operable supplying power to one full capacity spray substation and two containment fan coolers.

CRI – Control room Indication
HX – Heat exchanger
CCW – Component cooling water
RAS– Recirculation actuation signal

TABLE 6.2-42

CONTAINMENT SPRAY-PUMP NPSH DATA

	Injection Phase (Suction From Refueling Water Tank)	Recirc Phase (Suction From Sump)
Elevation of Pump Suction, Ft	-4.82	-4.82
Elevation of Source, Ft	21.06	23.38
Fluid Temperature, F	124	192
Fluid Vapor Pressure, Ft	4.4	23.3
Head Loss Due to Friction, Ft	3.6	2.12
NPSH Available, Ft	48.5	32.42
NPSH Required at Pump Runout, Ft	31.8	23.9
Sump Loss (P_b), Ft		2.79
Pump Flow, gpm	3642	4350

The following formula is used:

$$NPSH\ (available) = Pt - Pv = Pa + Ps + Pe - Pi - Pv - P_b$$

where:
- Pt = pressure at pump suction centerline
- Pv = vapor pressure of pumped water
- Pa = Air pressure
- Ps = Steam pressure
- Pe = Elevation pressure
- Pi = Head loss due to friction in the suction piping
- P_b = head loss due to sump

TABLE 6.2-43

SPRAY NOZZLE DESIGN DATA

Flow Rate gpm	vs	Pressure Drop Across Nozzle psi
7.71		10
10.95		20
13.54		50
15.20		40 Design Point
17.30		50

Spray Angle = 60 Degrees

Droplet Size at Design Pressure Drop of 40 psi [(1)], microns	
Maximum Diameter	1629
Minimum Diameter	70
Area Mean Diameter	46
Sauter Mean Diameter	690
Median Diameter	346
Area Length Mean Diameter	634
Volume Length Mean Diameter	610

(1) See Subsection 6.5.2

TABLE 6.2-44

PERFORMANCE OF CONTAINMENT COOLING SYSTEM
LOCA CONDITION

	Air/Steam Temperature (F)	Temperature of Component Cooling Water (F) 60	80	90	100	105	110
Heat Removal in BTU/hr x 10^6	80	2.476	1.455				
	120	10.245	7.288	5.681	3.915	-	2.015
	200	42.81	37.62	34.84	32.04	-	28.943
	280	82.67	76.15	72.59	69.3	-	65.622
Condensate Temperature in F	80	63.42	82.287	-	-	-	-
	120	76.876	92.647	99.743	106.85	110.26	-
	200	157.924	163.928	167.22	170.27	171.81	-
	280	255.55	257.26	258.11	259.3	259.67	-
Condensate Removal Rate in lb/hr	80	1483	1001.44	-	-	-	-
	120	7639	5685	4498	3157	2426	-
	200	40652	35875	33210	30740	29437	-
	280	87150	80271	76424	72954	70958	-

TABLE 6.2-45
INSTRUMENTATION APPLICATION CONTAINMENT HEAT REMOVAL SYSTEMS

Parameters Associated With the System	Indication: Local	Indication: Control Room	Alarm: Local	Alarm: Control Room	Recorder[1]	Range[2]	Instrument Normal Operation	Accuracy[2]
CCS Air								
1. Temperature entering					TR-25-1A TR-25-1B		105-120°F	
2. Temperature leaving				HI	TR-25-1A TR-25-1B		105-120°F	
3. Flow-Fan Discharge				LO				
4. Fan Vibration (excessive)				HI				
Cooling Unit CCW								
1. Temperature downstream					TR-09-5A		60-100°F	
2. Flow - downstream	FIS-14-12A, 12B, 12C, 12D			LO			›1,200 gpm	
CS Pump 2A & 2B								
1. Motor current		AM/287 AM/290					66 Amp 66 Amp	
2. Pressure-Pump Suction		PI-07-4A/5A			UR-07-1B		10-35 psig	
3. Pressure-Pump Discharge	PI-07-1A PI-07-1B	PIS-07-3A PIS-07-3B		LO			180-270 psig 180-270 psig	
4. Flow-Pump Discharge		FI-07-1A FI-07-1B			FR-07-1A FR-07-1B		2700-3800 gpm 2700-3800 gpm	

(1) All recordings are in the control room unless otherwise indicated.
(2) Instrument ranges are selected in accordance with standard engineering practices. Instrument accuracies are selected such that existing instrument loop performance and safety analysis assumptions remain valid. Where applicable, instrument accuracies are also evaluated for their impact on setpoints in accordance with the FPL Setpoint Methodology.

TABLE 6.2-46

CONTAINMENT HEAT REMOVAL SYSTEM ALARMS[1]

CS PUMP 2A & 2B

Motor overload (Pump trip)

Start Failure
CSAS override

FCV-07-1A & 1B
Fail to open

REFUELING WATER TANK

MV-07-1A & 1B closed
Motor overload,
Fail to close on RAS

CONTAINMENT SUMP

MV-07-2A & 2B open
Motor overload,
Fail to open on RAS

CONTAINMENT FAN COOLERS

Temperature Leaving - High
Fan Discharge Flow - Low
Fan Vibration - High (Excessive)
CCW Flow - Low
Motor Overload/Trip
SIAS Override/Control Switch Stop/SS Isolation

[1]ALL alarms are in control room unless otherwise indicated.

TABLE 6.2-47

SHIELD BUILDING DESIGN AND PERFORMANCE DATA

I. Annulus Free Volume, 5.47×10^5 ft^3

II. Pressure

1. Normal Operation: Atmospheric

2. Post accident: 1.75 in. wg. negative to 2.0 in. wg. negative

III. Leak Rate at Post Accident Pressure

1. From containment - 0.5% for the first 24 hrs
   0.25%/day thereafter

2. From outside environs - 5% per day at 0.25 in. wg.
   pressure differential

IV. Exhaust Fans

1 Number - 2

2. Type - Centrifugal type, direct-driven with airfoil blades.

V. Filter Trains

1.	First electric heater each train	1.5 kw
2.	Demister each train	99% efficiency
3.	Second electric heater each train	30 kw
4.	HEPA prefilters each train	99.97% efficiency
5.	Charcoal adsorber	99% efficiency gasketless
6.	HEPA after- filters each train	99.97% efficiency

TABLE 6.2-48

DESIGN DATA FOR SHIELD BUILDING VENTILATION SYSTEM COMPONENTS (HVE-6A & 6B)

1. SBVS Fans

Quantity	2	
Type	Centrifugal type, direct-driven with airfoil blades	
Material	Carbon steel	
Actual air flow at inlet, per fan, cfm	6000 nominal	
Air density, lb/ft^3	0.075	
Static pressure, inches wg.	21	
Fan performance, each (see Figure 6.2-52)	Capacity (cfm)	Static Pressure (in.wg)
	6,000	21
	6,500	19
	6,750	17.6
	7,500	14
Code	Air Moving and Conditioning Association (AMCA), Anti-Friction Bearing Manufacturer's Association (AFBMA) Welding per AWS D1.1 and ASME IX	

2. Motors

Quantity	2, one per fan
Type	60 hp, 480 volt, 60 Hz, 3 phase horizontal induction type
Insulation	Class H
Enclosure & Ventilation	Drip-proof
Code	NEMA

3. 1st Electric Heater

Quantity	2, one per train
Size, KW	1.5 (480V, 3 phase, 60 Hz)

4. Demister Cells

Quantity	12, 6 per filter train

TABLE 6.2-48 (Cont'd)

4. Demister Cells (Cont'd)	
Air Flow, cfm	6,000 per filter train
Face velocity, fpm	284.4
Cell size	24 in. high, 24 in. wide, 6 in. deep
Cell arrangement	2 wide x 3 high
Maximum pressure drop dry, in. wg.	1.0
Maximum pressure drop wet, in. wg.	2.0
Design dry bulb/rel. humidity, F/percent	260/100
Efficiency	99 percent when exposed to entrained water particles of 1 to 5 microns in size
Material	Stainless steel casing, fiberglass pads
5. 2nd Electric Heater	
Quantity	2, one per filter train
Size, kW	30 (480V, 3 phase, 60 Hz)
6. HEPA Filters	
Quantity	24, 12 per filter train
Location	6 pre-HEPA and 6 after-HEPA
Air flow, cfm	6000 per filter train
Cell size	24 in. high, 24 in. wide, 11 1/2 in. deep
Cell arrangement	2 wide x 3 high
Maximum pressure drop clean, in. wg.	1.0
Maximum pressure drop loaded, in. wg.	3.0
Efficiency	99.97 percent when tested with 0.3 micron DOP

6.	HEPA Filters (Cont'd)	
	Material	Pleated glass paper without separator, enclosed in stainless steel frame ASTM A268, alternate design includes Aluminum Separators
	Make	Flanders 7023-NU-GGF or equivalent
	Construction Standard	MIL-F-51068D and MIL-F-51079A ASME AG-1
7.	Charcoal Adsorbers	
	Make and type	CVI HECA Module
	Cell construction	Vertical
	Material	Adsorber, activated coconut shell; charcoal enclosure, stainless steel, ASTM A240, Type 304
	Bulk Density (bone dry),	.38 g/ml minimum tested per ASTM D2854
	Loading capacity	2.5 mg of iodine per gram of charcoal
	Particle size distribution	10 through 14 mesh
	Plenum size	60 in. deep x 96 in. high x 66 in. wide
	Bed arrangement	3 wide x 1 high
	Bed thickness, in. nominal	2
	Amount of charcoal, lb	2362, 1181 lb per filter train
	Air flow, cfm	6000 per filter train
	Efficiency	90 % for radioactive methyl when tested In accordance with ASTM D3803-1989
	Maximum pressure drop in. wg.	1.35
	Nondestructive test	USNRC Bulletin No. 120 and ANSI N510-1980
8.	Ducts	
	Material	Galvanized sheet metal, ASTM A526; Structural shapes ASTM A36

TABLE 6.2-49

DESCRIPTION OF ASSUMPTIONS USED IN SHIELD BUILDING ANNULUS TRANSIENT ANALYSES

I. Secondary Containment Design

Refer to Table 6.2-47 for analysis data regarding the Shield Building.

		Primary	Secondary
II.	Transient Analysis		
	A. Initial Conditions		
	1. Pressure, psia	14.7	14.7
	2. Temperature, F	120	120
	3. Humidity (percent)	45	100
	4. Constant outside air pressure		14.7
	5. Constant outside air temperature, F		101
	6. Constant outside humidity (percent)		100
	7. Steel cylinder and dome internal radii, ft		70
	8. Steel building and concrete shield building		
	Cylinder heights:		
	From annulus floor to midpoint (effective height of secondary shield wall), ft.		53
	From midpoint (effective height secondary shield wall) to springline, ft		74.1
	9. Thickness of steel cylinder, in.		1.92
	10. Thickness of steel dome, in.		.96
	11. Thickness of concrete shield building cylinder , ft		3.0
	Thickness of concrete shield building top (for heat transfer calculation to 59.644% of a sphere with a 74 ft radius) (ft)		2.5

TABLE 6.2-49 (Cont'd)

			Secondary
B.	Thermal Characteristics		
	1.	Primary Containment Wall	
	a.	Coefficient of temperature Linear Expansion in/in-F	0.65×10^{-5}
	b.	Modulus of Elasticity, psi	$30. \times 10^{6}$
	c.	Poissons Ratio	0.3
	d.	Thermal Conductivity, BTU/hr-ft-F	25.9
	e.	Thermal Capacitance, BTU/ft^3-F	53.57
	2.	Secondary Containment Wall	
	a.	Thermal Conductivity, BTU/hr – ft- F	1.0
	b.	Thermal Capacitance, BTU/ft^3-F	31.9
	3.	Heat Transfer Coefficients	
	a.	Containment vapor to steel Vessel (BTU/hr – ft^2-F)	4 x Tagami heat transfer co-efficient
	b.	Steel Cylinder 2 to annulus air (BTU/hr-ft^2-F)	GOTHIC Heat Transfer Option
	c.	Steel dome to annulus air (BTU/hr-ft^2-F)	GOTHIC Heat Transfer Option
	d.	Annulus air to concrete cylinde (BTU/hr-ft^2-F)	GOTHIC Heat Transfer Option
	e.	Annulus air to concrete dome (BTU/hr-ft^2-F)	GOTHIC Heat Transfer Option
	f.	Primary Containment Emissivit	0.9
	g.	Secondary Containment Emissivity	0.9
	h.	Concrete cylinder to outside air (BTU/hr-ft^2-F)	0.0
	i.	Concrete dome to outside air (BTU/hr-ft^2-F)	0.0
C.		Additional Heat Sources in Shield Building Annulus	
	1.	SBVS filter heaters and pump motor (kW)	76.26

TABLE 6.2-50

FAILURE MODE AND EFFECT ANALYSIS - SHIELD BUILDING VENTILATION SYSTEM

Component Identification and Quality	Failure Mode	Effect on System	Method of Detection	Monitor	Remarks
1. Damper downstream of fan (2) (D-23 or D-24)	Fails to modulate	None	Air flow indicator downstream of fan	CRI*	No impact; this damper is normally open and is designed to open upon failure of control power.
2. Filter train (2) or component	Clogs	Loss of one fan and filter subsystem	Low air flow alarm downstream of fan and differential pressure on 1st HEPA	CRI	One 100% capacity fan and filter subsystem operable.
3. Outside cooling air valve (2)	Fails to open	Subsystem will operate satisfactorily until charcoal adsorber temperature reached 200°F.	High temperature alarm on charcoal adsorbers and valve position indicator	CRI	One 100% capacity fan and filter subsystem operable. Cross connect will supply required cooling air from annulus space to inoperable train.
4. Fan (2) (HVE-6A or 6B)	Fails to start	Loss of one fan and filter subsystem	Low air flow alarm and indicator downstream of fan	CRI	One 100% capacity fan and filter subsystem operable.
5. Diesel generator set (2)	Fails to start	Loss of one fan and filter subsystem	Diesel generator malfunction alarm	CRI	One 100% capacity fan and filter subsystem operable.
6. Fuel Handling Building isolation valve (2) in the SBVS suction header (FCV-25-30 or 31)	Spurious signal opens or failure to close on CIS after opening for fuel handling high radiation initiation	Partial loss of suction in the annulus	Valve position indicator and annulus high pressure indicator and alarm	CRI	One 100% capacity, fan and filter subsystem operable.
7. Hydrogen purge isolation valve (2) in the SBVS suction header (FCV-25-29 or 34)	Spurious signal opens or failure to close after purge	None	Valve position indicator	CRI	One 100% capacity and filter subsystem operable.
8. SBVS suction valve (2) (FCV-25-32 or 33)	Spurious signal closes or failure to open on CIS after fuel handling high radiation initiation	Loss of suction in one fan and filter subsystem	Low air flow alarm and indicator downstream of fan valve position indicator	CRI	One 100% capacity fan and filter subsystem operable.
9. Loss of 30 kw heating coil (2) (EHC-HVE-6A1 or 6B1)	Will not operate	Reduction of methyl iodide removal efficiency	High humidity alarm	CRI	One 100% capacity fan and filter subsystem available.

*CRI - Control Room Indication

TABLE 6.2-50 (Cont'd)

Component Identification and Quality	Failure Mode	Effect on System	Method of Detection	Monitor	Remarks
10. Decay heat cooling valve (1) on interconnecting pipe (FCV-25-13)	Closed and cannot be reopened	Loss of decay heat cooling air	High charcoal temperature alarm, valve position indicator	CRI	This is unlikely to happen because the valve is normally open and fails as-is. Both fans are available to supply cooling air.
11 Loss of operating filter train	Will not operate	Loss of one fan and filter subsystem	Low air flow alarm and indication downstream of fan	CRI	One 100% capacity fan and filter subsystem are available and a 1.5 kW heating coil powered from the opposite bus precludes desorption due to humidity on the inoperable train.

TABLE 6.2-51

SHIELD BUILDING VENTILATION SYSTEM INSTRUMENTATION APPLICATION

	System Parameter & Location	Indication Local	Indication Control Room	Alarm Control Room	Control Room Recording	Automatic Control Function	Instrument[1] Range	Normal Operating Range	Instrument[1] Accuracy
1.	Annulus/atmosphere pressure differential		*	Hi-Lo		Energizes fan discharge damper motor and regulates flow to preset value and energize outside cooling air valve		-1 to -3 in. H_2O	
2.	Fuel pool area/atmosphere pressure differential		*	Hi-Lo		Energizes fan discharge damper motor and regulates flow to preset value and energize outside cooling air valve		-1 to -3 in. H_2O	
3.	Air flow temperature downstream of demister	*						40-177 F	
4.	Inlet temperature upstream of filter train	*						40-177 F	
5.	Demister & Electric Heaters differential pressure	*						0.15-2 in. H_2O	
6.	Air flow temperature downstream of 30 Kw heating coil				*			40-177 F	
7.	Pre-HEPA filter differential pressure		*	Hi	*			1 to 3 in. H_2O	
		*						1 to 3 in. H_2O	
8.	After-HEPA filter differential pressure				*			1 to 3 in. H_2O	
		*						1 to 3 in. H_2O	
9.	Air flow moisture downstream of pre-HEPA filter			Hi				50-70%	
10.	Charcoal adsorber differential pressure indicator				*			1 to 1.35 in. H_2O	
		*						1 to 1.35 in. H_2O	
11.	Charcoal adsorber temperature			Hi	*			40-177 F	
12.	Air flow temperature downstream of charcoal adsorbers	*			*			40-177 F	

TABLE 6.2-51 (Cont'd)

	System Parameter & Location	Indication Local	Indication Control Room	Alarm Control Room	Control Room Recording	Automatic Control Function	Instrument[1] Range	Normal Operating Range	Instrument[1] Accuracy
13.	Air flow downstream of fan or Air flow of filter train	*	*	*	*	Energize, idle fan to start, at low flow and alarms		6000 cfm	
14.	Filter train differential pressure				*			4 to 8 in. H_2O	
15.	Cross-connect flow control valve position		*					-	
16.	Outside cooling air flow control valve position		*					-	
17.	Shield building suction valve position		*					-	
18.	Fuel handling building suction valve position		*					-	
19.	Purge discharge valve position		*					-	

(1) Instrument ranges are selected in accordance with standard engineering practices. Instrument accuracies are selected such that existing instrument Loop performance and safety analysis assumptions remain valid. Where applicable, instrument accuracies are also evaluated for their impact on setpoints in accordance with the FPL Setpoint Methodology.

TABLE 6.2-52
CONTAINMENT PENETRATION AND ISOLATION VALVE INFORMATION

Penetration No.	System	GDC (UFSAR) Classification	Fluid	Line Size (In.)	ESF	Leak[1][2][5] Testing	UFSAR Figure	Valve Number	Location	Pipe Length [3] to Containment (Ft)	Valve Type	Valve Operator
1, 2	Main Steam	GDC-57 (C)	Steam	34 3	No No	-	10.1-1	HCV-08-1A, B MV-08-1A, B	Out Out	52 52	Stop Globe	Piston Motor
1, 2	AFW PP 2C Warm-Up Line		Steam	3/4	No			SE-08-1, 2	Out		Globe	Solenoid
1, 2	Main Steam/Aux. FW	GDC-57 (C)	Steam	4	Yes	-		MV-08-12, 13	Out		Gate	Motor
3, 4	Feedwater	GDC-57 (C)	Water	20	No	-	10.1-2	HCV-09-1A, 1B, 2A, 2B	Out	27	Gate	Electro-hydraulic
3,4	Auxiliary Feedwater	GDC-57 (C)	Water	4	Yes	-	10.1-2	MV-09-9, 10, 11, 12	Out	44	Globe	Motor
5, 6	Steam Generator Blowdown	GDC-57 (C)	Water	2	No	-	10.4-6	FCV-23-3, 5	Out	11.0	Globe	Diaphragm
7[6]	Primary Makeup	GDC-56 (A1)	Water	2	No	BL-C BL-C	9.2-4	HCV-15-1[7] V15328	Out In	17	Globe Check	Diaphragm
8	Station Air	GDC-56 (A1)	Air	2	No	BL-C BL-C BL-C	9.3-1	HCV-18-2[7] V181270 SH18797[7]	Out In Annulus	11.0 2.0	Globe Check Ball	Diaphragm
9	Instrument Air	GDC-56 (A1)	Air	2	No	BL-C BL-C	9.3-2	HCV-18-1[7] V18195	Out In	9.0 1.5	Globe Check	Diaphragm
10	Cont Purge Exhaust	GDC-56 (A2)	Air	48 48	No No	-C -C	9.4-9	FCV-25-5[7] FCV-25-4[7]	Annulus In	1.0 3.0	B'fly B'fly	Piston Piston
11	Cont Purge Supply	GDC-56 (A2)	Air	48 48	No No	-C -C	9.4-9	FCV-25-2[7] FCV-25-3[7]	Annulus In	1.0 3.0	B'fly B'fly	Piston Piston
12, 13	Spare											
14	N2 Supply to Reactor Containment Building	GDC-56 (A1)	N_2	1	No	BL-C BL-C	11.3-1	V6741[7] V6792	Out In	16	Globe Check	Diaphragm
15, 17, 19, 21	Cont Fan Coolers Cooling Water	GDC-57 (C)	Water	8	Yes	-	9.2-2	MV-14-10, 12, 14, 16	Out	13	B'fly	Motor
16, 18, 20, 22	Cont Fan Coolers Cooling Water	GDC-57 (C)	Water	8	Yes	-	9.2-2	MV-14-9, 11, 13, 15	Out	13	B'fly	Motor
23[6]	RC Pump Cooling Water Supply	GDC-56 (A1)	Water	8	No	BL-C BL-C(R)	9.2-2	HCV-14-7[7] HCV-14-1[7]	Out In	12.5 7.0	B'fly B'fly	Piston Piston
24[6]	RC Pump Cooling Water Return	GDC-56 (A1)	Water	8	No	BL-C BL-C(R)	9.2-2	HCV-14-6[7] HCV-14-2[7]	Out In	12.5 7.0	B'fly B'fly	Piston Piston

TABLE 6.2-52 (Cont'd)

Penetration No.	System	GDC (UFSAR) Classification	Fluid	Line Size (In.)	ESF	Leak(1)(2)(5) Testing	UFSAR Figure	Valve Number	Location	Pipe Length (3) to Containment (Ft)	Valve Type	Valve Operator
25	Fuel Transfer Tube	--	Water	36	No	BL-B		--	In			Double Gasketed Flange
26	CVCS (letdown)	GDC-55 (B1)	Water	2	No	BL-C BL-C	9.3-5c 9.3-5a	V2516 V2522	In Out	- 12	Globe Globe	Diaphragm Diaphragm
27	CVCS (charging)	GDC-55 (E)	Water	2	Yes	-	9.3-5c	V2462 V2523	In Out	2 11	Check Globe	- Diaphragm
28A	Safety Injection Tank Sample	GDC-55 (B1)	Water	3/8	No	BL-C BL-C	9.3-3a	SE-05-1A(7), 1B(7), 1C(7), 1D(7) SE-05-1E(7)	In Out	10 5	Globe Globe	Solenoid Solenoid
28B	RCS Hot Leg Sample	GDC-55 (B1)	Water	3/8	No	BL-C BL-C	9.3-3a	V5200(7) V5203(7)	In Out	10 15	Globe Globe	Solenoid Diaphragm
29A	Press. Surge Sample	GDC-55 (B1)	Water	3/8	No	BL-C BL-C	9.3-3a	V5204(7) V5201(7)	Out In	5 2	Globe Globe	Diaphragm Solenoid
29B	Press. Steam Sample	GDC-55 (B1)	Water	3/8	No	BL-C BL-C	9.3-3a	V5205(7) V5202(7)	Out In	15 10	Globe Globe	Diaphragm Solenoid
30, 49	Steam Generator Blow-down Sampling	GDC-57 (C)	Water	1/2	No	-	10.4-6, 7	FCV-23-7, 9	Out	15	Globe	Diaphragm
31	Containment Vent Header	GDC-56 (A1)(4)	N_2	1	No	BL-C BL-C	11.3-1	V6718(7) V6750(7)	In Out	10 15	Diaph Diaph	Diaphragm Diaphragm
32, 33	Containment Sump Suction	GDC-56 (E)	Water	24	Yes	-	6.2-41	MV-07-2A, 2B	Out	12	B'fly	Motor
34A & B, 35A & B	Containment Spray	GDC-56 (E)	Water	12 10	Yes Yes	- -	6.2-41	FCV-07-1A, B V07193, V07192	Out In	16, 20 11, 7	B'fly Check	Diaphragm 36-49
36-39	Safety Injection	GDC-55 (E)	Water	6	Yes	-	6.3-1	HCV-3615, 3625, 3635, 3645, HCV-3616, 3626, 3636, 3646, HCV-3617, 3627, 3637, 3647 V3258, 3259, 3260, 3261	Out In	- -	Globe Check	Motor
40, 64	Shutdown Cooling	GDC-55 (B2)	Water	10	Yes	-	6.3-1b	V3665, 3664 V3651, 3481 V3652, 3480	Out In In	14.0 80	Gate Gate Gate	Motor Motor Motor
41	SIT Test Line	GDC-55 (B2)	Water	2	No	BL-C BL-C	6.3-1a	SE-03-2A(7), 2B(7) V3463(7)	In Out	10.0, 6x4 11.0, 12	Globe Gate	Solenoid -
42(6)	Reactor Cavity Sump Pump Discharge	GDC-56 (A1)(4)	Water	2	No	BL-C BL-C	6.2-41	LCV-07-11A(7) LCV-07-11B(7)	In Out	4.0 15	Globe Globe	Diaphragm Diaphragm

TABLE 6.2-52 (Cont'd)

Penetra-tion No.	System	GDC (UFSAR) Classification	Fluid	Line Size (In.)	ESF	Leak(1)(2)(5) Testing	UFSAR Figure	Valve Number	Location	Pipe Length (3) to Containment (Ft)	Valve Type	Valve Operator
43	RCDT Pump Suction	GDC-56 (A1)	Water	3	No	BL-C BL-C	11.2-1	V6341(7) V6342(7)	In Out	80.0 9.0	Diaph Diaph	Diaphragm Diaphragm
44(6)	RCP Controlled Bleed-off	GDC-55 (B1)	Water	3/4	No	BL-C BL-C	9.3-5b	V2524(7) V2505(7)	In Out	4.0 15.0	Globe Globe	Diaphragm Diaphragm
45, 53	Fluid Inst Line	RG 1.11(D)	Air	3/8	No	-	6.2-41	Excess Flow	Out	15.0	Check	-
45B, 53B	Fluid Inst Line (Does not penetrate containment)											
46(6)	Fuel Pool Cleanup (Return)	GDC-56 (A2)	Water	3	No	BL-C BL-C	6.2-41	V07206(7) V07189(7)	Out In	11.0 10.0	Gate Gate	- -
47(6)	Fuel Pool Cleanup (Supply)	GDC-56 (A2)	Water	3	No	BL-C BL-C	6.2-41	V07170(7) V07188(7)	Out In	11.0 10.0	Gate Gate	- -
48A, 51A	Hydrogen Sample Lines (Outlet)	GDC-56 (E)	Air	3/8	Yes	BL-C BL-C	12.3-14	FSE-27-8(7) to 14(7) FSE-27-15(7), 18(7)	In Out	10.0 15	Globe Globe	Solenoid Solenoid
48, 51	Fluid Instrument Line	RG 1.11 (D)	Air	3/8	No	-	6.2-41	Excess flow	Out	-	Check	
48B, 51B	Hydrogen Sample Lines (Inlet)	GDC-56 (E)	Air	3/8	Yes	BL-C BL-C	12.3-14	V27101 & V27102 FSE-27-16(7), 17(7)	In Out	10.0 15.0	Check Globe	- Solenoid
48, 51	Containment Pressure	RG 1.11 (D)	Air	3/8	No	-	6.2-41	SE-07-5E, 5F	Out	-	Globe	Solenoid
49	Refer to Penetration 30											
50	Spare											
51	Refer to Penetration 48											
52A	Containment Radiation Monitoring	GDC-56 (A1)	Air	1	No	BL-C BL-C	12.3-14	FCV-26-1(7) FCV-26-2(7)	In Out	10.0 15.0	Globe Globe	Diaphragm Diaphragm
52B	Containment Radiation Monitoring	GDC-56 (A1)	Air	1	No	BL-C BL-C	12.3-14	FCV-26-3(7) FCV-26-4(7)	In Out	10.0 15.0	Globe Globe	Diaphragm Diaphragm
52C	Containment Radiation Monitoring	GDC-56 (A1)	Air	1	No	BL-C BL-C	12.3-14	FCV-26-5(7) FCV-26-6(7)	In Out	10.0 15.0	Globe Globe	Diaphragm Diaphragm
52D	ILRT	GDC-56 (A2)	Air	1	No	BL-C BL-C		V00140(7) V00143(7)	In Out	10.0 15.0	Globe Globe	- -
52E	ILRT	GDC-56 (A2)	Air	3/8	No	BL-C BL-C		V00139(7) V00144(7)	In Out	10.0 15.0	Globe Globe	- -

TABLE 6.2-52 (Cont'd)

Penetration No.	System	GDC (UFSAR) Classification	Fluid	Line Size (In.)	ESF	Leak[(1)(2)(5)] Testing	UFSAR Figure	Valve Number	Location	Pipe Length [(3)] to Containment (Ft)	Valve Type	Valve Operator
53	Refer to Penetration 45											
54	ILRT	GDC-56 (A2)	Air	8	No	BL-C BL-C(R)		V00101[(7)]	Out In	11.0	Gate Flange	-
55	Containment Pressure	RG 1.11 (D)	Air	3/8	No	-	6.2-41	SE-07-5A	Out	-	Globe	Solenoid
56	Cont Containment/ H_2 Purge - Makeup Inlet	GDC-56 (A1)	Air Air	8 8	No No	BL-C BL-C	9.4-11 9.4-11	FCV-25-26[(7)] FCV-25-36[(7)]	Out In	3 2	B'fly B'fly	Piston Piston
57	Cont Containment/ H_2 Purge - Exhaust	GDC-56 (A1)	Air	8	No	BL-C	9.4-11	FCV-25-20[(7)] FCV-25-21[(7)]	In Out	2 9	B'fly B'fly	Piston Piston
58	Containment Press (Inst)	RG 1.11 (D)	Air	3/8	No	-	6.2-41	SE-07-5C	Out	-	Globe	Solenoid
59, 60	Shield Building Vent (Does not penetrate containment vessel)											
61	Spare											
62	HVAC Instr - (Does not pentrate containment)											
63	Spare											
64	Refer to Penetration 40											
65, 66	HVAC Instr - (Does not penetrate containment)											
67, 68	Containment Vacuum Relief	GDC-56 (A1)	Air	24	Yes	-C -C	9.4-9	V-25-20, 21 FCV-25-7, 8[(7)]	In Out	1 1	Check B'fly	- Piston
69	Hot Leg Injection	GDC-55 (B2)	Water	3	Yes	-	6.3-1a & b	V3540 V3524	Out In	18 150	Globe Check	Motor -
70	Hot Leg Injection	GDC-55 (B2)	Water	3	Yes	-	6.3-1a & b	V3523 V3526	Out In	36 86	Globe Check	Motor -
71, 72	Cont Press Inst	RG 1.11 (D)	Air	3/8	No	-	6.2-41	SE-07-5D, 5B	Out	-	Globe	Solenoid

TABLE 6.2-52 (Cont'd)

Footnotes for Table

(1) Type C testing is defined in 10 CFR 50, Appendix J and/or Appendix J, Option B.

(2) BL - Bypass leakage.

(3) Length of pipe is measured from steel containment to outermost isolation valve outside containment.

(4) Although this is a class A1 penetration, the piping is seismic Category I on both sides of containment.

(5) (R) - Tested with pressure applied in reverse direction to that experienced under DBA.

(6) Thermal relief valve(s) provided to address thermal overpressurization concerns of NRC Generic Letter 96-06.

(7) Valve covered by Technical Specification 3/4.6.3.

TABLE 6.2-53
CONTAINMENT ISOLATION ANALYSIS *

Penetra-tion No.	System	Valve Type	Containment Isolation Actuation Signal (1)	Primary Actuation Mode(2)	Secondary Actuation Mode (2)	Normal Valve Position(3)	Shutdown Valve Position(3)	Post-Accident Valve Position(3)	Power Failure Valve Position	Time to Close (sec)	Power Source
1, 2	Main Steam	Stop Globe	MSIS MSIS	P E	- M	Open Closed	Closed Closed	Closed Closed	As is As is	5.6 30	125 DC 480 AC
1, 2	AFW PP 2C Warm-Up Line	Gate	-	-	M	Open	Open	Open	Open	-	125 DC
1, 2	Main Steam/Aux FW	Gate	-	E	M	Closed	Open	Open	As is	30	125 DC
3, 4	Feedwater	Gate	MSIS AFAS	H	-	Open	Closed	Closed	As is	4.0	125 VDC
3, 4	Auxiliary Feedwater	Globe	AFAS	E	M	Closed	Open	Open	As is	-	480 AC
5, 6	Steam Generator Blowdown	Globe	CIS/A	P	M	Open	Closed	Closed	Closed	5	125 DC-SA
7	Primary Makeup Water	Globe Check	CIS/B -	P Self	M -	Open -	Closed	Closed	Closed	5 -	125 DC-SB
8	Station Air	Globe Check Ball	CIS/A - -	P Self M	M - -	Closed - Closed	Open - Closed	Closed - Closed	Closed - -	5 - -	125 DC-SA
9	Instrument Air	Globe Check	CIS/A -	P Self	M -	Open -	Closed -	Closed -	Closed -	5 -	125 DC-SA
10	Cont Purge Exhaust	B'fly B'fly	CIS/B CIS/A	P P	M M	Closed Closed	Closed Closed	Closed Closed	Closed Closed	5 5	125 DC-SB 125 DC-SA
11	Cont Purge Supply	B'fly B'fly	CIS/A CIS/B	P P	M M	Closed Closed	Closed Closed	Closed Closed	Closed Closed	5 5	125 DC-SA 125 DC-SB
12, 13	Spare										
14	N2 Supply to Reactor Containment Bldg.	Globe Check	CIS/B -	P Self	M -	Closed -	Open -	Closed -	Closed -	5 -	125 DC-SB -
15,17 19, 21	Cont Fan Coolers Cool. Water Return	B'fly	-	E	M	Open	Open	Open	As is	30	480 AC
16, 18 20, 22	Cont Fan Coolers Cooling Water Supply	B'fly	-	E	M	Open	Open	Open	As is	30	480 AC
23	RC Pump Cooling Water Supply	B'fly B'fly	SIAS/B SIAS/A	P P	M M	Open Open	Closed Closed	Closed Closed	Closed Closed	5 5	125 DC-SB 125 DC-SA
24	RC Pump Cooling Water Return	B'fly B'fly	SIAS/B SIAS/A	P P	M M	Open Open	Closed Closed	Closed Closed	Closed Closed	5 5	125 DC-SB 125 DC-SA
25	Fuel Transfer Tube	-	-	-	-	Closed	-	Closed	-	-	-

*Actual Components may be procured and tested to more conservative stroke times than those listed.

TABLE 6.2-53 (Cont'd)

Penetration No.	System	Valve Type	Containment Isolation Actuation Signal(1)	Primary Actuation Mode(2)	Secondary Actuation Mode (2)	Normal Valve Position	Shutdown Valve Position(3)	Post-Accident Valve Position(3)	Power Failure Valve Position	Time to Close (sec)	Power Source
26	CVCS (letdown)	Globe Globe	SIAS/CIS-A CIS/B	P P	M M	Open Open	Closed Closed	Closed Closed	Closed Closed	5 5	125 DC-SA 125 DC-SB
27	CVCS (charging)	Check Globe	- -	Self P	- -	- LO	- LO	- LO	- LO	- 10	- 125 DC-SA
28A	Safety Injection Tank Sample	Globe Globe	CIS/B CIS/A	S S	M M	Closed Closed	Closed Closed	Closed Closed	Closed Closed	5 5	125 DC-SB 125 DC-SA
28B	RCS Hot Leg Sample	Globe Globe	CIS/A CIS/B	S P	M M	Closed Closed	Closed Closed	Closed Closed	Closed Closed	5 5	125 DC-SA 125 DC-SB
29A	Press. Surge Sample	Globe Globe	CIS/B CIS/A	P S	M M	Closed Closed	Closed Closed	Closed Closed	Closed Closed	5 5	125 DC-SB 125 DC-SA
29B	Press. Steam Sample	Globe Globe	CIS/B CIS/A	P S	M M	Closed Closed	Closed Closed	Closed Closed	Closed Closed	5 5	125 DC-SB 125 DC-SA
30, 49	Steam Generator Blowdown Sampling	Globe	CIS/A	P	M	Open	Closed	Closed	Closed	10	125 DC-SA
31	Containment Vent Header	Diaph Diaph	CIS/B CIS/A	P P	M M	Open Open	Open Open	Closed Closed	Closed Closed	5 5	125 DC-SB 125 DC-SA
32, 33	Containment Sump Suction	B'fly	RAS	E	M	Closed	Closed	Open on RAS	As is	30	480 AC
34A & B 35A & B	Containment Spray	B'fly Check	CSAS -	P Self	M -	Closed -	Closed -	CSAS Opens	Open -	10 -	125 DC-SA/SB
36-39	Safety Injection	Globe Check	SIAS Opens	E Self	M -	Closed -	Open -	Open -	As is -	- -	480 AC -
40, 64	Shutdown Cooling	Gate Gate	- -	- -	- -	LC LC	Open Open	Closed Closed	As is As is	- -	480 AC 480 AC
41	SIT Test Line	Globe Gate	SIAS/CIS -	S -	M	Closed LC	Closed Closed	Closed LC	FC LC	10 -	125 DC-SA/SB -
42	Reactor Cavity Sump Pump Discharge	Globe Globe	CIS/SIAS-A CIS/SIAS-B	P P	M M	Closed Closed	Closed Closed	Closed Closed	Closed Closed	5 5	125 DC-SA 125 DC-SB
43	RCDT Pump Suction	Diaph Diaph	CIS/A CIS/B	P P	M M	Open Open	Closed Closed	FC FC	FC FC	5 5	125 DC-SA 125 DC-SB

TABLE 6.2-53 (Cont'd)

Penetration No.	System	Valve Type	Containment Isolation Actuation Signal(1)	Actuation Mode(2) Primary	Actuation Mode(2) Secondary	Normal Valve Position(3)	Shutdown Valve Position	Post-Accident Valve Position	Power Failure Valve Position	Closure Time (sec)	Power Source
44	RCP Controlled Bleedoff	Globe	CIS/B	P	M	Open	Closed	FC	FC	5	125DC/SB
		Globe	CIS/A	P	M	Open	Closed	FC	FC	5	125DC/SA
45,53	Fluid Inst Line	Check	-	Self	-	-	-	-	-	-	-
45B,53B	Fluid Inst Line	Does not penetrate containment									
46	Fuel Pool Cleanup (Return)	Gate	-	M	-	LC	Closed	Closed	Closed	-	-
		Gate	-	M	-	LC	Closed	Closed	Closed	-	-
47	Fuel Pool Cleanup (Supply)	Gate	-	M	-	LC	Closed	Closed	Closed	-	-
		Gate	-	M	-	LC	Closed	Closed	Closed	-	-
48A,51A	Hydrogen Sample Lines (Outlet)	Globe	-	S	M	Closed	Closed/Open	Closed/Open	Closed	5	120 AC
		Globe	-	S	M	Closed	Closed/Open	Closed/Open	Closed	5	120 AC
48B,51B	Hydrogen Sample Lines (inlet)	Check	-	Self	-	-	Closed	-	-	-	-
		Globe	-	S	M	Closed	Closed	Closed	Closed	5	120 AC
48,51	Fluid Inst Line	Check	-	Self	-	Open	Open	-	-	-	-
48,51	Cont Press	Globe	-	S	-	Open	Open	Open	Closed	-	125 DC
49	Refer to Penetration 30										
50	Spare										
51	Refer to Penetration 48										
52A	Containment Radiation	Globe	CIS/B	P	M	Open	Closed	Closed	Closed	10	125 DC-SB
	Monitoring	Globe	CIS/A	P	M	Open	Closed	Closed	Closed	10	125 DC-SA
52B	Containment Radiation	Globe	CIS/B	P	M	Open	Closed	Closed	Closed	10	125 DC-SB
	Monitoring	Globe	CIS/A	P	M	Open	Closed	Closed	Closed	10	125 DC-SA
52C	Containment Radiation Monitoring	Globe	CIS/B	P	M	Open	Closed	Closed	Closed	10	125 DC-SB
		Globe	CIS/A	P	M	Open	Closed	Closed	Closed	10	125 DC-SA
52D	ILRT	Globe	-	M	-	LC	Closed	Closed	Closed	-	-
		Globe	-	M	-	LC	Closed	Closed	Closed	-	-
52E	ILRT	Globe	-	M	-	LC	Closed	Closed	Closed	-	-
		Globe	-	M	-	LC	Closed	Closed	Closed	-	-
53	Refer to Penetration 45										

TABLE 6.2-53 (Cont'd)

Penetration No.	System	Valve Type	Containment Isolation Actuation Signal (1)	Primary Actuation Mode (2)	Secondary Actuation Mode (2)	Normal Valve Position (3)	Shutdown Valve Position (3)	Post Accident Valve Position	Power Failure Valve Position	Time to Close (sec)	Power Source
54	ILRT	Gate	-	M	LC -	Closed -	Closed -	Closed -	- -	- -	
55	Conainment Press (Inst)	Globe	-	S	-	Open	Open	Open	Open	-	125V DC
56	Cont Containment/H_2 Purge - Makeup Inlet	B'fly B'fly	CIS/A CIS/B	P P	M M	Closed Closed	Closed Closed	Closed Closed	Closed Closed	5 5	125V DC-SA 125V DC-SB
57	Cont Containment/H_2 Purge Exhaust	B'fly B'fly	CIS/A CIS/B	P P	M M	Closed Closed	Closed Closed	Closed Closed	Closed Closed	3 3	125V DC-SA 125V DC-SB
58	Cont Pressure	Globe	-	S	-	Open	Open	Open	Open	-	125V DC
59, 60	Shield Building Vent System (Does not penetrate containment)										
61	Spare										
62	HVAC Instr - (Does not penetrate containment)										
63	Spare										
64	Refer to Penetration 40										
65, 66	HVAC Instr - (Does not penetrate containment)										
67, 68	Containment Vacuum Relief	Check B'fly	- HDP	Self P	- M	Closed Closed	Closed Closed	Closed Closed	Closed Closed	- -	- 125V DC-SA/SB
69	Hot Leg Injection	Globe Check	- -	E Self	M -	LC -	Closed -	Closed -	FC -	- -	- -
70	Hot Leg Injection	Globe Check	- -	E Self	M -	LC -	Closed -	Closed -	FC -	- -	- -
71, 72	Containment Pressure (Inst)	Globe	-	S	-	Open	Open	Open	Open	-	125V DC

TABLE 6.2-53 (Cont'd)

Notes for Table 6.2-53

(1)	CIAS	-	Containment Isolation Actuation Signal
	SIAS	-	Safety Injection Actuation Signal
	HDP	-	High Differential Pressure Between Containment Vessel and Annulus
	RAS	-	Recirculation Actuation Signal
	CSAS	-	Containment Spray Actuation Signal
	MSIS	-	Main Steam Isolation Signal
	AFAS	-	Auxiliary Feedwater Actuation Signal (Auto Initiation)
(2)	E	-	Electric motor operated
	P	-	Pneumatic cylinder or diaphragm or piston operated
	S	-	Solenoid operated
	M	-	Manually operated
	Self	-	Actuated by the fluid pressure
	H	-	Hydraulic cylinder operated
(3)	FC	-	Fail closed
	FO	-	Fail open
	AI	-	As is
	LC	-	Locked Close
	LO	-	Locked Open
	NA	-	Not Applicable

(4) Valves actuated on SIAS are listed on Table 7.3-2

(5) Valves actuated on RAS are listed on Table 7.3-3

(6) Valves actuated on a CSAS are listed on Table 7.3-4

(7) Valves actuated on a CIAS are listed on Table 7.3-5

(8) Valves actuated on a MSIS are listed on Table 7.3-6

TABLE 6.2-54

DESIGN AND PERFORMANCE DATA SPECIFIED FOR HYDROGEN ANALYZER SYSTEM

Analyzing Panel	
Approximate Dimensions;	
Height, inch	72
Width, inch	30
Depth, inch	30
Weight, lbm	750
Thermal Conductivity Analyzer:	
Sensitivity	0.2 percent H_2 by volume
Range	0-10 percent H_2 by volume
Accuracy	±5.0 percent of full scale
Calibration	Passing known gaseous mixture through analyzer
Control and Power Supply	
Electric Power Supply	115 Volt, 1 phase, 60 Hz and 460 Volt, 3 phase, 60 Hz
Control valves (solenoids)	Sample points in the containment
Tubing (External to Analyzer Package)	
Size	3/8" OD x .065 in
Material	ASME-SA-213 Type 316 GR
Design Temperature °F	340
Design Pressure psig	60
Code	ASME SECTION III, 1974 Edition,
Valves (Containment Sampling Isol. Valves FSE-27-15, 16, 17, 18)	
Size	3/8"
Material	ASME SA-240 Type F316
Design Temperature°F	350

TABLE 6.2-54 (Cont'd)

Design Pressure, psig	150
Code	ASME Section III Class 2, 1974 Edition

<u>Pumps</u>

Type	Diaphragm
Pump material	Type 316 SS
Required H.P.	1.0
Design Flowrate scfm	0.3

<u>Cooler</u>

Type	Natural Convection
Design	40 foot coil of sample tubing
Temp., F in/out	300/175
Pressure	10 inches of mercury vacuum
Material	ASME-SA-213 Type 316

<u>Water Separator</u>

Body Material	Type 304 SE
Design Pressure psig	psig 100
Design Temperature F	400

TABLE 6.2-55

DESIGN AND PERFORMANCE DATA FOR HYDROGEN RECOMBINER SYSTEMS

Recombiner Unit	
Approximate Dimensions:	
Height ft.	8.0
Width ft. (skid)	3.9
Depth ft. (skid)	4.6
Weight lb.	4500
Capacity scfm	100
Gas Temperatures: F	
Inlet	80 to 155
Outlet of heater section	1150 to 1400
Exhaust, ΔT above ambient	Approximately 100
Materials:	
Outer structure	Type 300 SS
Inner structure	Inconel 600
Heater element sheath	Incoloy 800
Power:, kW	
Maximum	75
Nominal	50
Recombination %	99-100
Control and Power Supply	
Electric Power Supply	480 Volt, 3 phase, 60 Hz
Thermocouple leads	3 pair (Chromel-Alumel)

TABLE 6.2-56

DESIGN DATA AND MATERIALS FOR HYDROGEN PURGE SYSTEM

Fan Section	
Quantity	2
Fan Performance, each	
H_2 Purging Capacity, cfm	100
Type	Centrifugal Fan, direct-driven with radial blades
Material	Carbon Steel
Bearings	131,000 hour B-10 rating life
Code	AWS D1.1, AMCA 210
Motor	
Quantity	One per fan
Type	Horizontal squirrel cage motor, totally enclosed fan cooled
Electrical Characteristics	480v, 3ph, 60Hz
Rating Hp	40
Insulation	Type H
Bearings	AF
Finish	Epoxy
Connections	Burndy connectors, grounding pad
Filter Train	
Demister Cells	2; 99% efficiency for 1 to 5 micron water particle size; 2.0 in. wg. max pressure drop, fiberglass pads in stainless steel casing.
Electric Heater	12 kW; 480V/3ph/60Hz
Medium Efficiency Filters	2; 90% efficiency per ASHRAE 52 test; glass fiber media in galvanized frame; 1.0 in. wg. max. pressure drop
High Efficiency Filters	90-95% efficiency per ASHRAE 52-76 test

TABLE 6.2-56 (Cont'd)

HEPA Filters	4: 2 Pre-HEPA, 2 After HEPA: 99.97% efficiency when tested with 0.3 micron DOP; 3.0 in. wg. max pressure drop; pleated glass paper enclosed in stainless steel frame
Charcoal Absorbers	99% efficiency in accordance with ANSI N509-1976; 1.37 in. wg. max pressure drop; activated coconut shell charcoal in stainless steel enclosure.
V-Bank Carbon Adsorber Cells	>99.9% efficiency in accordance with AACC-CS-8 (ANSI N509-1980)

TABLE 6.2-57

COMBUSTIBLE GAS CONTROL SYSTEM - FAILURE MODES AND EFFECTS ANALYSIS

Component Identification and Quantity	Failure Mode	Effects of System	Method of Detection	Monitor	Remarks
One Diesel	Fails	Loss of one train H_2 analyzer, one recombiner	Indication of the Diesel Generator voltage, and/or current.	CRI	Redundant diesel available redundant train of analyzer and recombiner available.
One Hydrogen Recombiner	Fails to operate	Loss of one H_2 recombiner	Low Power from heating coils, High H_2 alarm.	CRI*	Second recombiner available
One Hydrogen Analyzer	Fails internally	Loss of use of one H_2 analyzer; other analyzer available.	Low flow or sudden change on recorder	CRI	Use of redundant H_2 analyzer or Grab Samples.
One valve failure (normally open post accident; solenoid valves to and from analyzer) I-FSE-27-15, 16, 17 or 18	Fails closed and will not open	Loss of use of one analyzer	Low flow	CRI	Use of redundant H_2 analyzer or Grab Samples.
One Catalytic recombiner in analyzer panel	Poisoned	Temporary loss of use of one H_2 analyzer	Calibration gas (4% H_2) will read 0% H_2.	CRI	Replacement of catalyst; use redundant H_2 analyzer or Grab Samples.
Containment isolation valve I-FSE-27-15 or 18	Valve fail to open	Loss of up to 4 sample points	Position indication	CRI	Three sample points remain, giving adequate sample points.
One sampling pump	Fails	Loss of use of one analyzer	Low flow	CRI	Use of redundant H_2 analyzer

* Control Room Indication

TABLE 6.2-58

PARAMETERS FOR ANALYSIS OF HYDROGEN GENERATION AND CONTROL

Hydrogen Dissolved in Reactor Coolant	423 SCF
Release Rate for Dissolved Hydrogen	Instantaneous
Amount of Zircaloy in core	48019 lbm
Fraction of Zirconium Oxidized During Design Basis Accident	1% (maximum)
Fraction of Zirconium Assumed to Oxidize for Purposes of Hydrogen Generation Analysis	5%
Release Rate from Zirconium-Water Reaction	Instantaneous
Fission Product Distribution Model	50% of the halogens and 1% of the solids present in the core are intimately mixed with the coolant water. All noble gases are released to the containment. All other fission products remain in fuel rods.
Fraction of Fission Product Radiation Energy Absorbed by the Coolant	(a) Beta (1) Betas from fission products in the fuel rods: 0.0 (2) Betas from fission products intimately mixed with coolant:1.0 (b) Gamma (1) Gammas from fission products in the fuel rods, coolant in core region: 0.1

TABLE 6.2-58 (Cont'd)

	(2) Gammas from fission products intimately mixed with coolant, all coolant: 1.0
Hydrogen Yield Rate $G(H_2)$	0.5 molecule per 100 Ev
Oxygen Yield Rate $G(O_2)$	0.25 molecule per 100 Ev
Reactor Thermal Power, mwt	2700
Inventory of Corrosible Metal	Table 6.2-59
Assumed Aluminum Corrosion Rate	Figure 6.2-65
Assumed Zinc Corrosion Rate	Figure 6.2-66
Assumed Hydrogen Generation Rate Due to Zinc-Base Paint	Figure 6.2-67
Containment Net Free Volume, ft^3	2,500,000
Initial Containment Temperature, F	120
Initial Containment Pressure, psia	14.7

TABLE 6.2-59

INVENTORY OF CORROSIBLE MATERIALS

I. Aluminum:

1 -	Overhead Cranes	125.5 lbm (reacts instantaneously)
2 -	Stair Treads	5342 lbm (1703 ft^2)
3 -	Transmitters	32 lbm (9 ft^2)
4 -	NSSS Equipment	59 lbm (17 ft^2)
5 -	Containment Piping Penetration Insulation Lagging	265 lbm (843 ft^2)
6 -	Cable Shielding	141 lbm (10,681 ft^2)
7 -	Boxes	100 lbm (5.2 ft^2)
8 -	Excore Nuclear Instrumentation	229.8 lbm (128.5 ft^2)
9 -	Ladders	75.0 lbm (232 ft^2)
10 -	Excore Dust Covers and Clamps	175 lbm (13.02 ft^2) 3 lbm (2.0 ft^2)
	Total Aluminum	6422.0 lbm (13,634 ft^2)

II. Zinc:

1 - Galvanized Steel: Ductwork, conduits, cable trays, tube supports/hangers, instrument racks, platforms and polar cranes

a. 41,417 ft^2 at 0.8 mils

b. 50,926 ft^2 at 1.53 mils

c. 91,047 ft^2 at 3.0 mils

d. 8,625 ft^2 at .0034 mils

e. 852 ft^2 at .0034 mils

f. 43,054 ft^2 at 3.91 mils

g. 4,720 ft^2 at 3.4 mils

TABLE 6.2-59 (Cont'd)

h. 10 ft^2 at 2.0 mils

i. 3,835 ft^2 at 4.62 mils

Total galvanized steel:	30,235 lbm (244,559 ft^2)
2 - Speltering	1600 lbm (10 ft^2)

III - Zinc - Base Paint

Total Area	245,443 ft^2 (see revised Table 6.1-2)

TABLE 6.2-60

COMBUSTIBLE GAS CONTROL SYSTEM INSTRUMENTATION

System Parameter and Location	Indication		Alarm [a]				Normal	
	Local	Control Room	Local	Control Room	Recording [a]	Inst. [c] Range	Operation Range	Inst. [c] Accuracy
Hydrogen Recombiner								
1) Power to recombiner		*					50-75 KW	
2) Coil temperature		*					1150-1400 F	
Containment Hydrogen Analyzers								
1) Sample hydrogen	*	*		Hi	*		--	
2) Containment isolation valve position indication		*				--	--	--
3) Containment hydrogen analyzer system failure				Failure		--	--	--

TABLE 6.2-60 (Cont'd)

System Parameter and Location	Indication Local	Indication Control Room	Alarm[(a)] Local	Alarm[(a)] Control Room	Recording[(a)]	Inst.[(c)] Range	Normal Operation Range	Inst.[(c)] Accuracy
Hydrogen Purge System		*					0-2000 cfm	
1) Flow rate[(b)]		*		*	*		0-100 cfm	
		*					0-2500 cfm	
2) Containment isolation valves position indication		*				--	--	--
3) SBVS/Purge isolation valves position indication		*				--	--	--
4) Modulating valves position indication		*					--	--
5) Makeup air containment isolation valves position indication		*				--	--	--
6) Continuous containment Purge/H_2 Purge air clean-up unit								
a) Demister differential pressure	*						1-3 in. wg.	
b) Inlet air temperature	*						60-270 F	
c) Air temperature after demister	*						60-270 F	

TABLE 6.2-60 (Cont'd)

System Parameter & Location			Indication Local	Indication Control Room	Alarm [(a)] Local	Alarm [(a)] Control Room	Recording [(a)]	Instrument[(c)] Range	Normal Operation Range	Instrument[(c)] Accuracy
Hydrogen Purge System (Cont'd)										
6)	d)	Electric heater diff. pressure	*						0-0.30 in. wg	
	e)	Air temperature downstream of electric heater					*		0-270 F	
	f)	Pre HEPA differential pressure	 *	*		Hi	*		1-3 in. wg 1-3 in. wg	
	g)	Relative humidity after pre HEPA filter	*			Hi			0-100%	
	h)	Charcoal Adsorber temperature					*		0-270 F	
	i)	Charcoal Adsorber diff. pressure	 *				*		1-1.15 in. wg 1-1.15 in. wg	
	j)	Air temperature after charcoal adsorber	 *				*		0-270 °F 0-270 °F	
	k)	After HEPA filter diff. pressure	 *				*		1-3 in. wg 1-3 in. wg	
	l)	Filter train differential pressure					*		1-10 in. wg	

(a) All alarms and recordings are in the control room unless otherwise indicated.

(b) Alarm on 0 flow -10 second time delay.

(c) Instrument ranges are selected in accordance with standard engineering practices. Instrument accuracies are selected such that existing instrument loop performance and safety analysis assumptions remain valid. Where applicable, instrument accuracies are also evaluated for their impact on setpoints in accordance with the FPL Setpoint Methodology.

TABLE 6.2-61

This table has been deleted.

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

TABLE 6.2-62

CONTAINMENT ISOLATION VALVE TESTING*

Penetration No.	Isolation Valve Tag No.	Applicable Test: Type C	Applicable Test: Bypass Leakage	Drawing Reference
7	V15328 (inside)	X	X	2998-G-084,Sh 1
	HCV-15-1 (outside)	X	X	
8	V181270 (inside)	X	X	2998-085, Sh 1
	HCV-18-2 (outside)	X	X	
9	V18195 (inside)	X	X	2998-G-085, Sh 2A
	HCV-18-1 (outside)	X	X	
10	FCV-25-4 (inside)	X	-	2998-G-878
	FCV-25-5 (outside)	X	-	

TABLE 6.2-62 (Cont'd)

Penetration No.	Isolation Valve Tag No.	Applicable Test: Type C	Applicable Test: Bypass Leakage	Drawing Reference
11	FCV-25-3 (inside)	X	-	2998-G-878
	FCV-25-2 (outside)	X	-	
14	V6792 (inside)	X	X	2998-G-087, Sh 163B
	V6741 (outside)	X	X	
23	HCV-14-1 (inside)	X	X	2998-G-083, Sh 2
	HCV-14-7 (outside)	X	X	
24	HCV-14-2 (inside)	X	X	2998-G-083, Sh 2
	HCV-14-6 (outside)	X	X	
26	V2516 (inside)	X	x	2998-G-078, Sh 120 & 122

TABLE 6.2-62 (Cont'd)

Penetration No.	Isolation Valve Tag No.	Applicable Test: Type C	Applicable Test: Bypass Leakage	Drawing Reference
	V2522 (outside)	x	x	
28A	SE-05-1A, 1B,1C,1D (inside)	x	x	2998-G-078, Sh 153
	SE-05-1E (outside)	x	x	
28B	V5200 (inside)	x	x	2998-G-078, Sh 153
	V5203 (outside)			
29A	V5201 (inside)	x	x	2998-G-078, Sh 153
	V5204 (outside)			
29B	V5202 (inside)	x	x	2998-G-078, Sh 153

TABLE 6.2-62 (Cont'd)

Penetration No.	Isolation Valve Tag No.	Applicable Test Type C	Bypass Leakage	Drawing Reference
	V5205 (outside)	x	x	
31	V6718 (inside)	x	x	2998-G-078, Sh 163A
	V6750 (outside)	x	x	
41	SE-03-2B SE-03-2A (inside)	x	x	2998-G-078, Sh 130B
	V3463 (outside)	x	x	
42	LCV-07-11A (inside)	x	x	2998-G-088, Sh 2
	LCV-07-11B (outside)	x	x	
43	V6341 (inside)	x	x	2998-G-078, Sh 160A

TABLE 6.2-62 (Cont'd)

Penetration No.	Isolation Valve Tag No.	Applicable Test: Type C	Applicable Test: Bypass Leakage	Drawing Reference
	V6342 (outside)	x	x	
44	V2524 (inside)	x	x	2998-G-078, Sh 121A
	V2505 (outside)	x	x	
46	V07189 (inside)	x	x	2998-G-088, Sh 2
	V07206 (outside)	x	x	
47	V07188 (inside)	x	x	2998-G-088, Sh 2
	V07170 (outside)	x	x	

TABLE 6.2-62 (Cont'd)

Penetration No.	Isolation Valve Tag No.	Applicable Test Type C	Applicable Test Bypass Leakage	Drawing Reference
48A	FSE-27-8 through FSE-27-11 (inside)	x	-	2998-G-092, Sh 1
	FSE-27-15 (outside)	x	-	
48B	V27101 (inside)	x	-	2998-G-092, Sh 1
	FSE-27-16 (outside)	x	-	
51A	FSE-27-12 through FSE-27-14 (inside)	x	-	2998-G-092, Sh 1
	FSE-27-18 (outside)	x	-	

TABLE 6.2-62 (Cont'd)

Penetration No.	Isolation Valve Tag. No.	Applicable Test Type C	Applicable Test Bypass Leakage	Drawing Reference
51B	V27102 (inside)	x	-	2998-G-092, Sh 1
	FSE-27-17 (outside)	x	-	
52A,B	FCV-26-1,3 (inside)	x	x	2998-G-092, Sh 1
	FCV-26-2,4 (outside)	x	x	
52C	FCV-26-5 (inside)	x	x	2998-G-092, Sh 1
	FCV-26-6 (outside)	x	x	
52E	V00139 (inside)	x	x	2998-G-091, Sh 1

TABLE 6.2-62 (Cont'd)

Penetration No.	Isolation Valve Tag. No.	Applicable Test Type C	Applicable Test Bypass Leakage	Drawing Reference
	V00144 (outside)	x	x	
52D	V00140 (inside)	x	x	2998-G-091, Sh 1
	V00143 (outside)	x	x	
54	V00101 (outside)	x	x	2998-G-091, Sh 1
56	FCV-25-36 (inside)	x	x	2998-G-879, Sh 3
	FCV-25-26 (outside)	x	x	
57	FCV-25-20 (inside)	x	x	2998-G-879, Sh 3
	FCV-25-21 (outside)	x	x	

TABLE 6.2-62 (Cont'd)

Penetration No.	Isolation Valve Tag. No.	Applicable Test: Type C	Applicable Test: Bypass Leakage	Drawing Reference
67	V-25-20 (inside)	x	-	2998-G-878
	FCV-25-7 (outside)	x	-	
68	V-25-21 (inside)	x	-	2998-G-878
	FCV-25-8 (outside)	x	-	

* Refer to the Containment Leakage Rate Testing Program (10 CFR 50 Appendix J and/or Appendix J, Option B)

TABLE 6.2-63A

POSTULATED ACCIDENTS FOR CONTAINMENT DESIGN (STRETCH POWER), HISTORICAL

Containment Design Parameter	Postulated Accidents	Mass and Energy Release Data
Containment Peak Pressure/Temperature	Limiting Updated Loss of Coolant Accidents (LOCA)*	
	Double ended suction leg slot, 9.82 ft^2 area (minimum SI)	*
	Main Steam Line Breaks (MSLB)	
	7.6257 ft^2 MSLB, 102% power containment cooling train failure, increased core inlet temperature, decreased RCS flow rate	Table 6.2-67A (Deleted)
	7.6257 ft^2 MSLB, 102% power containment cooling train failure, increased core inlet temperature	Table 6.2-67B (Historical)
	7.6257 ft^2 MSLB, 102% power containment cooling train failure, decreased RCS flow rate	Table 6.2-67C (Historical)
	7.6257 ft^2 MSLB, 102% power MSIV failure, increased core inlet temperature, decreased RCS flow rate	Table 6.2-67D (Historical)
	7.6257 ft^2 MSLB, 102% power MSIV failure, increased core inlet temperature	Table 6.2-67E (Deleted)
	7.6257 ft^2 MSLB, 102% power MSIV failure, decreased RCS flow rate	Table 6.2-67F (Deleted)

* See Ref. (27) for break spectrum analysis.

TABLE 6.2-63B

POSTULATED ACCIDENTS FOR CONTAINMENT DESIGN (EPU)

Containment Design Parameter	Postulated Accidents	Mass and Energy Release Data
Containment Peak Pressure/Temperature	Limiting Updated Loss of Coolant Accidents (LOCA)	
	Double ended hot leg slot, 19.24 ft^2 area (min. SI)	
	Main Steam Line Breaks (MSLB)	
	100.3% power MSLB with the failure of one Containment Cooling Train	
	75% power MSLB with the failure of one Containment Cooling Train	
	50% power MSLB with the failure of one Containment Cooling Train	
	25% power MSLB with the failure of one Containment Cooling Train	
	Limiting EPU Peak Pressure	Table 6.2-12F
	0% power MSLB with the failure of one Containment Cooling Train	
	100.3% power MSLB with the failure of a MFIV to close	
	75% power MSLB with the failure of a MFIV to close	
	50% power MSLB with the failure of a MFIV to close	
	25% power MSLB with the failure of a MFIV to close	
	0% power MSLB with the failure of a MFIV to close	
	100.3% power MSLB with the failure of a MSIV to close	
	75% power MSLB with the failure of MSIV to close	
	50% power MSLB with the failure of MSIV to close	
	25% power MSLB with the failure of MSIV to close	
	0% power MSLB with the failure of MSIV to close	
	100.3% power EQ-MSLB with the failure of a MSIV to close	Table 6.2-12G

Table 6.2-64
This table has been deleted.

PAGE LEFT INTENTIONALLY BLANK

PAGE LEFT INTENTIONALLY BLANK

TABLE 6.2-65

VALVE CLOSURE TIMES AND ASSUMPTIONS USED IN MSLB ANALYSIS

EPU

Valve Closure Times (from receipt of signal)

1.	MSIV (I-HCV-08-1A and 1-HCV-08-1B) closure time, sec	5.6
2.	MFIV (1-HCV-09-1A and I-HCV-09-2A) closure time, sec	4.0
3.	Backup feedwater isolation valve (I-HCV-09-1B, 2B) closure time, sec	4.0
4.	Reactor trip on low steam generator pressure psia	546 psia
5.	Reactor trip on high containment pressure, psig	6.4 psig
6.	Instrumentation delay time, sec	1.15

Assumptions

1. Credit is not taken for the coastdown of the main feedwater pumps or condensate pumps in the MSLB analysis. Therefore, no degradation of the feedwater flow occurs until the closure of the main feedwater isolation or backup valves.

2. Offsite power is assumed to be available for the analysis. Availability of offsite power allows the continuation of reactor coolant pump and feedwater pump flows. Maintaining reactor coolant and feedwater pump flow maximizes the rate of primary to secondary heat transfer which maximizes the rate of mass/energy release.

TABLE 6.2-66A

MASS/ENERGY RELEASE RATES FOR 1000 SQ IN
HOT LEG GUILL. BREAK AT SG NOZZLE
FOR SUBCOMPARTMENT ANALYSIS OF
ST LUCIE II STRETCH POWER(2700 MWT)
(FLOW FROM RCS SIDE)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)	
0.00000	0.0	615.40	0.	2256.0	43.0	
.00100	6457.0	615.20	3972346.	2211.0	43.0	
.00200	11580.0	614.60	7362908.	2067.0	42.8	
.00300	15720.0	613.70	9647364.	1857.0	42.8	
.00400	18230.0	612.80	11171344.	1652.0	42.7	
.00500	20050.0	612.20	12274610.	1506.0	42.6	
.00600	24040.0	612.20	14717288.	1505.0	42.5	
.00700	28020.0	612.20	17153844.	1504.0	42.4	
.00800	31980.0	612.20	19578156.	1503.0	42.3	
.00900	35920.0	612.20	21990224.	1502.0	42.2	
.01000	39830.0	612.20	24383926.	1501.0	42.1	
.01200	39690.0	612.30	24302187.	1498.0	41.8	
.01400	39570.0	612.30	24228711.	1496.0	41.6	1
.01600	39480.0	612.30	24173604.	1495.0	41.5	
.01800	39400.0	612.30	24124620.	1494.0	41.3	
.02000	39350.0	612.40	24097940.	1493.0	41.2	
.02200	39320.0	612.40	24079568.	1493.0	41.2	
.02400	39300.0	612.40	24067320.	1493.0	41.1	
.02600	39300.0	612.50	24071250.	1493.0	41.1	
.02800	39320.0	612.50	24083500.	1493.0	41.2	
.03000	39350.0	612.50	24101875.	1494.0	41.2	
.03200	39390.0	612.60	24130314.	1495.0	41.3	
.03400	39450.0	612.60	24167070.	1496.0	41.4	
.03600	39520.0	612.60	24209952.	1498.0	41.5	
.03800	39610.0	612.70	24269047.	1499.0	41.6	
.04000	39700.0	612.70	24324190.	1501.0	41.8	
.04200	39800.0	612.70	24385460.	1503.0	41.9	
.04400	39920.0	612.70	24458984.	1505.0	42.1	
.04600	40030.0	612.80	24530384.	1508.0	42.3	
.04800	40130.0	612.80	24591664.	1510.0	42.5	
.05000	46440.0	613.50	28490940.	1686.0	42.7	
.05500	41240.0	612.90	25275996.	1537.0	42.6	
.06000	40120.0	612.70	24581524.	1509.0	42.5	
.06500	40100.0	612.70	24569270.	1509.0	42.5	
.07000	40070.0	612.60	24546882.	1507.0	42.4	
.07500	39980.0	612.60	24491748.	1506.0	42.3	
.08000	39820.0	612.50	24389750.	1503.0	42.0	
.08500	39590.0	612.50	24248875.	1498.0	41.6	
.09000	39280.0	612.40	24055072.	1492.0	41.1	
.09500	38930.0	612.40	23840732.	1486.0	40.6	
.10000	38550.0	612.30	23604165.	1479.0	40.0	
.11000	37800.0	612.20	23141160.	1464.0	38.8	
.12000	37100.0	612.10	22708910.	1451.0	37.8	
.13000	36510.0	612.00	22344120.	1440.0	36.9	
.14000	36080.0	612.00	22080960.	1432.0	36.3	

TABLE 6.2-66A (CONT'D)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)	
.15000	35880.0	612.00	21958560.	1428.0	36.0	
.16000	35900.0	612.00	21970800.	1429.0	36.0	
.17000	36050.0	612.00	22062600.	1431.0	36.2	
.18000	36230.0	612.00	22172760.	1435.0	36.5	
.19000	36350.0	612.00	22246200.	1437.0	36.7	
.20000	36380.0	612.00	22264560.	1438.0	36.7	
.22000	36150.0	612.00	22123800.	1433.0	36.4	
.24000	35490.0	612.00	21719880.	1422.0	35.5	
.26000	34570.0	611.90	21153383.	1405.0	34.3	
.28000	33660.0	611.90	20584316.	1390.0	33.2	
.30000	33060.0	612.00	20232720.	1381.0	32.5	
.32000	32960.0	612.10	20174816.	1380.0	32.4	
.34000	33040.0	612.20	20227088.	1382.0	32.5	
.36000	32940.0	612.20	20165868.	1380.0	32.4	
.38000	32550.0	612.30	19930365.	1375.0	31.9	
.40000	32010.0	612.30	19599723.	1367.0	31.4	
.42000	31550.0	612.40	19321220.	1361.0	30.9	
.44000	31290.0	612.50	19165125.	1357.0	30.7	
.46000	31140.0	612.60	19076364.	1356.0	30.5	1
.48000	30960.0	612.70	18969192.	1353.0	30.3	
.50000	30650.0	612.80	18782320.	1349.0	30.0	
.55000	29750.0	612.90	18233775.	1337.0	29.2	
.60000	29080.0	613.20	17831856.	1329.0	28.6	
.65000	28520.0	613.60	17499872.	1320.0	27.9	
.70000	28330.0	614.10	17397453.	1309.0	27.1	
.75000	28150.0	614.50	17298175.	1299.0	26.4	
.80000	27950.0	615.00	17189250.	1289.0	25.7	
.85000	27740.0	615.30	17068422.	1277.0	25.0	
.90000	27510.0	615.60	16935156.	1266.0	24.4	
.95000	27280.0	615.80	16799024.	1256.0	23.7	
1.00000	27070.0	616.20	16680534.	1244.0	23.1	
1.10000	26730.0	617.20	16497756.	1231.0	22.3	
1.20000	26330.0	618.30	16279839.	1215.0	21.4	
1.30000	25920.0	618.80	16039296.	1198.0	20.6	
1.40000	25490.0	619.00	15778310.	1182.0	19.5	
1.50000	25050.0	619.20	15510960.	1165.0	19.2	
1.60000	24650.0	619.70	15275605.	1151.0	18.5	
1.70000	24360.0	620.20	15108072.	1140.0	18.1	
1.80000	24030.0	620.40	14908212.	1127.0	17.6	
1.90000	23680.0	620.30	14688704.	1112.0	17.1	
2.00000	23390.0	620.40	14511156.	1099.0	16.7	
2.10000	23140.0	620.70	14362998.	1088.0	16.3	
2.20000	22870.0	621.00	14202270.	1075.0	15.8	
2.30000	22670.0	621.20	14082604.	1066.0	15.5	
2.40000	22470.0	621.20	13958364.	1056.0	15.2	

TABLE 6.2-66A (CONT'D)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.H. PRESS (PSIA)	DENSITY (LB/CU.FT)
2.50000	22260.0	621.40	13832364.	1046.0	14.9
2.60000	22070.0	621.60	13718712.	1036.0	14.6
2.70000	21890.0	621.70	13609013.	1028.0	14.4
2.80000	21730.0	621.80	13511714.	1020.0	14.2
2.90000	21620.0	621.80	13443316.	1014.0	14.0
3.00000	21550.0	621.50	13393325.	1010.0	13.9
3.50000	21200.0	618.70	13116440.	983.4	13.5
4.00000	20510.0	615.10	12861741.	958.7	13.3

1

TABLE 6.2-66B

MASS/ENERGY RELEASE RATES FOR 1000 SQ IN
HOT LEG GUILL. BREAK AT SG NOZZLE
FOR SUBCOMPARTMENT ANALYSIS OF
ST LUCIE II STRETCH POWER (2700 MWT)
(FLOW FROM SG SIDE)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)	
0.00000	0.0	615.40	0.	2255.0	43.0	
.00100	6072.0	614.70	3732458.	2091.0	42.9	
.00200	11360.0	614.20	6977312.	1977.0	42.9	
.00300	17170.0	614.30	10547531.	1990.0	42.9	
.00400	20790.0	613.70	12758823.	1846.0	42.8	
.00500	23080.0	612.90	14145732.	1670.0	42.7	
.00600	25990.0	612.50	15918875.	1590.0	42.6	
.00700	28930.0	612.30	17713839.	1537.0	42.6	
.00800	32070.0	612.20	19633254.	1505.0	42.6	
.00900	35990.0	612.20	22033078.	1503.0	42.4	1
.01000	39790.0	612.10	24355459.	1499.0	42.0	
.01200	39320.0	612.10	24067772.	1491.0	41.3	
.01400	38920.0	612.10	23822932.	1484.0	40.6	
.01600	38600.0	612.10	23627060.	1478.0	40.1	
.01800	38370.0	612.10	23486277.	1473.0	39.7	
.02000	38240.0	612.10	23406704.	1471.0	39.5	
.02200	38210.0	612.10	23388341.	1470.0	39.4	
.02400	38260.0	612.10	23418946.	1471.0	39.5	
.02600	38390.0	612.10	23498519.	1474.0	39.7	
.02800	38560.0	612.10	23602576.	1477.0	40.0	
.03000	38760.0	612.20	23728872.	1481.0	40.3	
.03200	38960.0	612.20	23851312.	1485.0	40.6	
.03400	39140.0	612.20	23961508.	1488.0	40.9	
.03600	39290.0	612.20	24053338.	1491.0	41.2	
.03800	39400.0	612.20	24120680.	1493.0	41.3	
.04000	39460.0	612.20	24157412.	1494.0	41.4	
.04200	39470.0	612.30	24167481.	1494.0	41.5	
.04400	39440.0	612.30	24149112.	1494.0	41.4	
.04600	39360.0	612.30	24100128.	1492.0	41.3	
.04800	39240.0	612.30	24026652.	1490.0	41.1	
.05000	39150.0	612.30	23971545.	1489.0	40.9	
.05500	39790.0	612.40	24367396.	1501.0	41.9	
.06000	40300.0	612.50	24683750.	1511.0	42.6	
.06500	39730.0	612.50	24334625.	1501.0	41.9	
.07000	38800.0	612.40	23761120.	1484.0	40.4	
.07500	37780.0	612.40	23136472.	1465.0	38.8	
.08000	37080.0	612.40	22707792.	1452.0	37.7	
.08500	36880.0	612.40	22585312.	1449.0	37.4	
.09000	37050.0	612.40	22689420.	1452.0	37.6	
.09500	37310.0	612.40	22848644.	1457.0	38.0	
.10000	37480.0	612.40	22952752.	1460.0	38.3	
.11000	37290.0	612.30	22832667.	1456.0	38.0	
.12000	36780.0	612.20	22516716.	1446.0	37.3	
.13000	36560.0	612.20	22382032.	1442.0	37.0	
.14000	36610.0	612.10	22408981.	1442.0	37.0	

TABLE 6.2-66B (CONT'D)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	R.N. PRESS (PSIA)	DENSITY (LB/CU.FT)
.15000	36580.0	612.10	22390619.	1441.0	37.0
.16000	36330.0	612.00	22233960.	1437.0	36.7
.17000	35970.0	612.00	22013640.	1430.0	36.2
.18000	35630.0	612.00	21805560.	1424.0	35.7
.19000	35510.0	612.00	21732120.	1421.0	35.5
.20000	35460.0	612.00	21713760.	1421.0	35.5
.22000	35010.0	612.00	21426120.	1413.0	34.9
.24000	34370.0	611.90	21031003.	1401.0	34.0
.26000	33630.0	611.90	20578197.	1389.0	33.2
.28000	33110.0	611.90	20260009.	1381.0	32.6
.30000	32900.0	611.90	20131510.	1378.0	32.4
.32000	32570.0	612.00	19932840.	1373.0	32.0
.34000	32210.0	612.00	19712520.	1369.0	31.6
.36000	31790.0	612.10	19458659.	1362.0	31.2
.38000	31440.0	612.20	19247568.	1358.0	30.8
.40000	31170.0	612.20	19082274.	1354.0	30.6
.42000	30920.0	612.30	18932316.	1351.0	30.3
.44000	30660.0	612.40	18776184.	1347.0	30.1
.46000	30360.0	612.50	18595500.	1343.0	29.8
.48000	30040.0	612.60	18402504.	1339.0	29.5
.50000	29700.0	612.60	18194220.	1335.0	29.2
.55000	29000.0	612.80	17771200.	1326.0	28.6
.60000	28510.0	613.10	17479481.	1317.0	27.9
.65000	28360.0	613.40	17396024.	1307.0	27.3
.70000	28170.0	613.90	17293563.	1297.0	26.5
.75000	27970.0	614.30	17181971.	1287.0	25.8
.80000	27760.0	614.80	17066848.	1276.0	25.1
.85000	27540.0	615.10	16939854.	1265.0	24.5
.90000	27310.0	615.40	16806574.	1253.0	23.8
.95000	27070.0	615.70	16666999.	1241.0	23.2
1.00000	26850.0	616.00	16539600.	1231.0	22.7
1.10000	26500.0	617.00	16350500.	1218.0	21.8
1.20000	26070.0	618.10	16113867.	1201.0	20.9
1.30000	25600.0	618.70	15838720.	1185.0	20.1
1.40000	25180.0	619.00	15586420.	1170.0	19.4
1.50000	24740.0	619.10	15316534.	1153.0	18.7
1.60000	24340.0	619.60	15081064.	1138.0	18.1
1.70000	24070.0	620.10	14925807.	1128.0	17.7
1.80000	23750.0	620.30	14732125.	1115.0	17.2
1.90000	23410.0	620.30	14521223.	1100.0	16.7
2.00000	23150.0	620.40	14362260.	1087.0	16.3
2.10000	22890.0	620.70	14207823.	1075.0	15.9
2.20000	22630.0	620.90	14050967.	1063.0	15.5
2.30000	22440.0	621.10	13937484.	1054.0	15.2
2.40000	22250.0	621.10	13819475.	1044.0	14.9

TABLE 6.2-66B (CONT'D)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)
2.50000	22040.0	621.20	13691248.	1034.0	14.6
2.60000	21840.0	621.40	13571376.	1024.0	14.3
2.70000	21660.0	621.50	13461690.	1016.0	14.1
2.80000	21500.0	621.70	13366550.	1008.0	13.8
2.90000	21390.0	621.60	13296024.	1002.0	13.7
3.00000	21350.0	621.30	13264755.	998.7	13.6
3.50000	21090.0	617.40	13020966.	974.0	13.4
4.00000	20850.0	612.90	12778965.	949.3	13.3

1

TABLE 6.2-66C

MASS/ENERGY RELEASE RATES FOR 1400 SQ IN
DISCH. LEG GUILL. BREAK AT PUMP NOZZLE
FOR SUBCOMPARTMENT ANALYSIS OF
ST LUCIE II STRETCH POWER (2700 MWT)
(FLOW FROM PUMP SIDE)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)	
0.00000	0.0	547.90	0.	2291.0	46.8	
.00100	10580.0	546.90	5786202.	2042.0	46.7	
.00200	15700.0	544.80	8553360.	1506.0	46.5	
.00300	15160.0	542.90	8230364.	1041.0	46.3	
.00400	19120.0	542.80	10378336.	1002.0	46.2	
.00500	23860.0	542.80	12951208.	1001.0	46.0	
.00600	28550.0	542.80	15486940.	999.9	45.7	
.00700	33180.0	542.90	18013422.	998.6	45.4	
.00800	37730.0	542.90	20483617.	996.8	45.0	1
.00900	42200.0	542.90	22910380.	994.7	44.5	
.01000	46550.0	542.90	25271995.	992.1	44.0	
.01200	45890.0	542.90	24913681.	987.1	43.0	
.01400	45250.0	542.90	24566225.	982.1	42.0	
.01600	44600.0	542.90	24213340.	977.0	41.1	
.01800	43940.0	542.90	23855026.	971.9	40.2	
.02000	43260.0	542.90	23485654.	966.8	39.3	
.02200	42600.0	543.00	23131800.	961.8	38.4	
.02400	41940.0	543.00	22773420.	957.0	37.6	
.02600	41300.0	543.00	22425900.	952.4	36.9	
.02800	40700.0	543.10	22104170.	948.2	36.2	
.03000	40120.0	543.20	21793184.	944.2	35.5	
.03200	39580.0	543.20	21499856.	940.6	35.0	
.03400	39060.0	543.30	21221298.	937.1	34.4	
.03600	38560.0	543.40	20953504.	933.8	33.9	
.03800	38060.0	543.40	20681604.	930.5	33.4	
.04000	37570.0	543.40	20415538.	927.1	33.0	
.04200	37070.0	543.50	20147545.	923.6	32.5	
.04400	36560.0	543.50	19870360.	920.4	32.1	
.04600	36150.0	543.50	19647525.	916.9	31.6	
.04800	36060.0	543.60	19602216.	913.3	31.2	
.05000	35950.0	543.60	19542420.	909.7	30.7	
.05500	35690.0	543.70	19404653.	900.6	29.7	
.06000	35360.0	543.80	19228768.	892.0	28.7	
.06500	35010.0	543.90	19041939.	883.7	27.7	
.07000	34580.0	544.00	18811520.	875.1	26.9	
.07500	34030.0	544.00	18512320.	865.5	26.0	
.08000	33260.0	544.00	18093440.	854.9	25.0	
.08500	32530.0	544.00	17696320.	843.6	24.1	
.09000	32150.0	544.00	17489600.	832.8	23.2	
.09500	31750.0	544.00	17272000.	821.6	22.4	
.10000	31320.0	543.90	17034948.	809.8	21.5	
.11000	30230.0	543.70	16436051.	784.7	19.8	
.12000	29310.0	543.70	15935847.	759.8	18.1	
.13000	28520.0	543.70	15506324.	739.0	16.8	
.14000	27840.0	543.80	15139392.	721.9	15.9	

TABLE 6.2-66C (CONT'D)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)
.15000	27280.0	543.90	14837592.	708.2	15.1
.16000	26910.0	544.10	14641731.	698.9	14.6
.17000	26670.0	544.20	14513814.	693.3	14.3
.18000	26390.0	544.30	14364077.	687.0	14.0
.19000	26100.0	544.40	14208840.	680.1	13.7
.20000	25850.0	544.50	14075325.	674.4	13.4
.22000	25470.0	544.60	13870962.	665.6	13.0
.24000	25280.0	544.80	13772544.	661.4	12.8
.26000	25180.0	545.00	13723100.	659.5	12.7
.28000	24920.0	545.00	13581400.	653.8	12.5
.30000	24650.0	545.10	13436715.	647.7	12.2
.32000	24300.0	545.10	13245930.	639.8	11.9
.34000	24130.0	545.30	13158089.	635.6	11.7
.36000	23840.0	545.40	13002336.	629.9	11.5
.38000	23440.0	545.40	12784176.	620.9	11.2
.40000	23100.0	545.50	12601050.	613.2	10.9
.42000	22810.0	545.60	12445136.	606.9	10.7
.44000	22490.0	545.70	12272793.	599.1	10.4
.46000	22140.0	545.70	12081798.	589.9	10.0
.48000	21790.0	545.90	11895161.	581.2	9.7
.50000	21550.0	546.00	11766300.	575.7	9.5
.55000	20800.0	546.30	11417670.	560.8	9.0
.60000	20370.0	546.70	11136279.	549.8	8.6
.65000	20010.0	547.10	10947471.	542.9	8.4
.70000	19680.0	547.40	10772832.	536.8	8.2
.75000	19360.0	547.80	10605408.	530.9	8.0
.80000	19040.0	548.10	10435824.	525.3	7.8
.85000	18760.0	548.50	10289860.	520.5	7.7
.90000	18500.0	548.90	10154650.	516.2	7.5
.95000	18200.0	549.20	9995440.	511.3	7.4
1.00000	17890.0	549.60	9832344.	506.2	7.2
1.10000	17350.0	550.30	9547705.	497.8	7.0
1.20000	16890.0	551.00	9306390.	491.1	6.8
1.30000	16410.0	551.70	9053397.	482.9	6.5
1.40000	16000.0	552.40	8838400.	472.1	6.3
1.50000	15640.0	553.00	8648920.	462.7	6.0
1.60000	15380.0	553.70	8515906.	456.0	5.9
1.70000	15090.0	554.30	8364387.	448.6	5.7
1.80000	14870.0	554.80	8249876.	443.3	5.5
1.90000	14650.0	555.40	8136610.	439.0	5.4
2.00000	14420.0	555.90	8016078.	432.2	5.3
2.10000	14280.0	556.40	7945392.	429.0	5.2
2.20000	14100.0	556.80	7850880.	424.7	5.1
2.30000	14000.0	557.30	7802200.	422.5	5.1
2.40000	13850.0	557.70	7724145.	419.2	5.0

TABLE 6.2-66C (CONT'D)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)	
2.50000	13940.0	556.30	7782702.	427.0	5.0	
2.60000	14330.0	558.70	8006171.	433.5	5.2	
2.70000	14560.0	559.10	8140496.	440.6	5.3	1
2.80000	14660.0	559.40	8200804.	443.8	5.4	
2.90000	14740.0	559.80	8251452.	446.5	5.4	
3.00000	14800.0	560.30	8292440.	448.8	5.4	
3.50000	15360.0	563.40	8653824.	468.9	5.7	
4.00000	15910.0	568.60	9046426.	492.1	5.9	

TABLE 6.2-66D

MASS/ENERGY RELEASE RATES FOR 1400 SQ IN DISCH. LEG GUILL. BREAK AT PUMP NOZZLE FOR SUBCOMPARTMENT ANALYSIS OF ST LUCIE II STRETCH POWER (2700 MWT) (FLOW FROM RV SIDE)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)
0.00000	0.0	547.90	0.	2291.0	46.8
.00100	10990.0	547.30	6014827.	2144.0	46.8
.00200	17930.0	545.70	9784401.	1739.0	46.6
.00300	18220.0	543.50	9902570.	1197.0	46.4
.00400	19110.0	542.70	10370997.	1002.0	46.2
.00500	23850.0	542.70	12943395.	1000.0	46.0
.00600	28530.0	542.70	15483231.	999.1	45.7
.00700	33160.0	542.70	17995932.	997.7	45.4
.00800	37740.0	542.70	20481498.	996.1	45.0
.00900	42250.0	542.70	22929075.	994.2	44.7
.01000	46700.0	542.70	25344090.	992.2	44.3
.01200	46130.0	542.70	25034751.	987.8	43.4
.01400	45590.0	542.70	24741693.	983.5	42.6
.01600	45060.0	542.70	24454062.	979.3	41.8
.01800	44550.0	542.70	24177285.	975.3	41.1
.02000	44050.0	542.70	23805935.	971.5	40.4
.02200	43580.0	542.70	23650866.	967.9	39.7
.02400	43130.0	542.70	23406651.	964.5	39.2
.02600	42700.0	542.70	23173290.	961.3	38.6
.02800	42300.0	542.70	22956210.	958.4	38.1
.03000	41930.0	542.80	22759604.	955.7	37.7
.03200	41590.0	542.80	22575052.	953.3	37.2
.03400	41270.0	542.80	22401356.	951.1	36.9
.03600	40990.0	542.90	22253471.	949.2	36.5
.03800	40740.0	542.90	22117746.	947.5	36.2
.04000	40510.0	542.90	21992879.	946.1	36.0
.04200	40320.0	543.00	21893760.	944.9	35.8
.04400	40160.0	543.00	21806880.	944.0	35.6
.04600	40030.0	543.10	21740293.	943.3	35.5
.04800	39930.0	543.20	21689976.	942.8	35.3
.05000	39860.0	543.20	21651952.	942.5	35.2
.05500	39790.0	543.30	21617907.	942.7	35.2
.06000	39860.0	543.50	21663910.	943.9	35.2
.06500	40070.0	543.60	21782052.	946.0	35.4
.07000	40370.0	543.70	21949169.	948.7	35.7
.07500	40740.0	543.80	22154412.	951.8	36.1
.08000	41140.0	543.80	22371932.	955.0	36.5
.08500	41550.0	543.90	22599045.	958.2	37.0
.09000	41960.0	543.90	22822044.	961.3	37.5
.09500	42380.0	543.90	23050482.	964.6	37.9
.10000	42800.0	543.90	23278920.	968.0	38.4
.11000	43710.0	544.00	23778240.	975.3	39.6
.12000	44670.0	544.10	24304947.	983.2	40.8
.13000	45660.0	544.20	24848172.	991.3	42.2
.14000	46650.0	544.30	25391595.	999.4	43.7

TABLE 6.2-66D (CONT'D)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)
.15000	47650.0	544.40	25940660.	1009.0	45.3
.16000	65260.0	545.50	35599330.	1287.0	46.3
.17000	64560.0	545.30	35204568.	1274.0	46.3
.18000	63720.0	545.20	34740144.	1259.0	46.3
.19000	63030.0	545.00	34351350.	1247.0	46.3
.20000	62180.0	544.90	33881882.	1232.0	46.3
.22000	62170.0	544.80	33870216.	1232.0	46.3
.24000	62070.0	544.80	33815736.	1230.0	46.3
.26000	62060.0	544.80	33810288.	1230.0	46.3
.28000	61210.0	544.60	33334966.	1215.0	46.3
.30000	60860.0	544.60	33144356.	1209.0	46.3
.32000	60980.0	544.60	33209708.	1211.0	46.3
.34000	60920.0	544.60	33177032.	1210.0	46.3
.36000	60330.0	544.60	32855718.	1200.0	46.3
.38000	60050.0	544.60	32703230.	1196.0	46.3
.40000	60230.0	544.70	32807281.	1199.0	46.3
.42000	60160.0	544.70	32769152.	1198.0	46.3
.44000	59580.0	544.70	32453226.	1189.0	46.3
.46000	59120.0	544.70	32202664.	1182.0	46.3
.48000	59250.0	544.70	32273475.	1184.0	46.3
.50000	59240.0	544.80	32273952.	1185.0	46.3
.55000	58380.0	544.90	31811262.	1172.0	46.3
.60000	58010.0	545.00	31615450.	1167.0	46.3
.65000	57670.0	545.20	31441684.	1163.0	46.3
.70000	56910.0	545.30	31033023.	1152.0	46.3
.75000	56510.0	545.50	30826205.	1147.0	46.2
.80000	56020.0	545.70	30570114.	1140.0	46.2
.85000	55490.0	545.90	30291991.	1133.0	46.2
.90000	55030.0	546.10	30051883.	1128.0	46.2
.95000	54580.0	546.40	29822512.	1122.0	46.2
1.00000	54160.0	546.70	29609272.	1117.0	46.2
1.10000	53350.0	547.20	29193120.	1108.0	46.1
1.20000	52440.0	547.80	28726632.	1097.0	46.1
1.30000	51440.0	548.50	28214840.	1085.0	46.0
1.40000	50920.0	549.20	27965264.	1080.0	46.0
1.50000	50330.0	549.90	27676467.	1075.0	46.0
1.60000	49700.0	550.60	27364820.	1069.0	45.9
1.70000	49000.0	551.20	27008800.	1061.0	45.9
1.80000	48840.0	551.80	26949912.	1062.0	45.5
1.90000	48200.0	552.30	26620640.	1059.0	44.3
2.00000	47340.0	552.80	26169552.	1054.0	42.9
2.10000	46470.0	553.30	25711851.	1049.0	41.6
2.20000	45680.0	553.70	25293016.	1044.0	40.4
2.30000	44970.0	554.10	24917877.	1040.0	39.4
2.40000	44310.0	554.40	24565464.	1036.0	38.5

TABLE 6.2-66D (CONT'D)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)
2.50000	43690.0	554.70	24234843.	1032.0	37.7
2.60000	43080.0	555.00	23909400.	1029.0	36.9
2.70000	42450.0	555.30	23572485.	1024.0	36.2
2.80000	41810.0	555.50	23225455.	1020.0	35.4
2.90000	41160.0	555.70	22872612.	1015.0	34.7
3.00000	40490.0	555.90	22508391.	1010.0	34.0
3.50000	37360.0	557.60	20831936.	992.4	30.9
4.00000	36130.0	559.10	20200283.	990.3	29.8

TABLE 6.2-66E

MASS/ENERGY RELEASE RATES FOR 1400 SQ IN
SUCTION LEG GUILL. BREAK AT PUMP NOZZLE
FOR SUBCOMPARTMENT ANALYSIS OF
ST LUCIE II STRETCH POWER(2700 MWT)
(FLOW FROM SG SIDE)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)
0.00000	0.0	547.90	0.	2214.0	46.8
.00100	10780.0	547.40	5900972.	2087.0	46.7
.00200	17850.0	546.00	9746100.	1730.0	46.6
.00300	18850.0	544.00	10254400.	1234.0	46.4
.00400	19180.0	543.10	10416658.	1004.0	46.2
.00500	23930.0	543.10	12996383.	1003.0	46.0
.00600	28670.0	543.10	15570677.	1002.0	45.7
.00700	33350.0	543.10	18112385.	1000.0	45.5
.00800	37980.0	543.10	20626938.	999.0	45.1
.00900	42560.0	543.10	23114336.	997.4	44.8
.01000	47090.0	543.10	25574579.	995.9	44.5
.01200	46780.0	543.10	25406218.	993.5	44.0
.01400	46490.0	543.20	25253368.	991.4	43.5
.01600	46230.0	543.20	25112136.	989.5	43.1
.01800	45980.0	543.20	24976336.	987.7	42.7
.02000	45750.0	543.20	24851400.	986.0	42.4
.02200	45540.0	543.30	24741882.	984.4	42.1
.02400	45340.0	543.30	24633222.	983.0	41.8
.02600	45160.0	543.30	24535428.	981.7	41.5
.02800	44990.0	543.30	24443067.	980.6	41.3
.03000	44840.0	543.40	24366056.	979.5	41.0
.03200	44700.0	543.40	24289980.	978.6	40.8
.03400	44570.0	543.40	24219338.	977.7	40.7
.03600	44450.0	543.40	24154130.	976.9	40.5
.03800	44350.0	543.50	24104225.	976.2	40.3
.04000	44260.0	543.50	24055310.	975.6	40.2
.04200	44180.0	543.50	24011830.	975.1	40.1
.04400	44100.0	543.50	23968350.	974.6	40.0
.04600	44040.0	543.50	23935740.	974.3	39.9
.04800	43990.0	543.60	23912964.	973.9	39.8
.05000	43940.0	543.60	23885784.	973.7	39.8
.05500	43860.0	543.60	23842296.	973.3	39.7
.06000	43820.0	543.70	23824934.	973.2	39.6
.06500	43810.0	543.70	23819497.	973.3	39.6
.07000	43830.0	543.70	23830371.	973.6	39.6
.07500	43870.0	543.80	23856506.	974.1	39.6
.08000	43940.0	543.80	23894572.	974.8	39.7
.08500	44030.0	543.90	23947917.	975.8	39.8
.09000	44150.0	543.90	24013185.	977.0	40.0
.09500	44310.0	544.00	24104640.	978.6	40.1
.10000	44500.0	544.10	24212450.	980.4	40.4
.11000	44970.0	544.30	24477171.	984.8	41.0
.12000	45520.0	544.40	24781088.	989.9	41.7
.13000	46150.0	544.60	25133290.	995.6	42.6
.14000	46770.0	544.70	25475619.	1002.0	43.6

1

TABLE 6.2-66E (CONT'D)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	R.N. PRESS (PSIA)	DENSITY (LB/CU.FT)	
.15000	47410.0	544.90	25833709.	1008.0	44.5	
.16000	48050.0	545.10	26192055.	1015.0	45.6	
.17000	60830.0	546.00	33213180.	1214.0	46.3	
.18000	61070.0	546.10	33350327.	1219.0	46.3	
.19000	61360.0	546.30	33531894.	1225.0	46.3	
.20000	61160.0	546.40	33417824.	1222.0	46.2	
.22000	60500.0	546.60	33069300.	1212.0	46.2	
.24000	60680.0	547.00	33191960.	1217.0	46.2	
.26000	61170.0	547.40	33484458.	1227.0	46.2	
.28000	60260.0	547.70	33004602.	1214.0	46.2	
.30000	59850.0	548.10	32803785.	1209.0	46.1	
.32000	59550.0	548.50	32663175.	1206.0	46.1	
.34000	59770.0	549.00	32813730.	1212.0	46.1	
.36000	59050.0	549.30	32436165.	1202.0	46.1	
.38000	58870.0	549.80	32366726.	1202.0	46.0	
.40000	58700.0	550.30	32302610.	1202.0	46.0	
.42000	58290.0	550.70	32100303.	1197.0	46.0	
.44000	58030.0	551.20	31986136.	1196.0	46.0	
.46000	57410.0	551.60	31667356.	1188.0	45.9	
.48000	57430.0	552.10	31707103.	1191.0	45.9	
.50000	57110.0	552.50	31553275.	1188.0	45.9	
.55000	56340.0	553.60	31189824.	1182.0	45.8	
.60000	55590.0	554.70	30835773.	1176.0	45.7	
.65000	54900.0	555.80	30513420.	1171.0	45.7	1
.70000	54390.0	556.90	30289791.	1168.0	45.6	
.75000	53830.0	557.90	30031757.	1165.0	45.6	
.80000	53140.0	558.80	29694632.	1159.0	45.5	
.85000	52510.0	559.60	29384596.	1154.0	45.5	
.90000	52060.0	560.50	29179630.	1151.0	45.4	
.95000	51590.0	561.20	28952308.	1147.0	45.4	
1.00000	51060.0	561.90	28690614.	1143.0	45.3	
1.10000	50410.0	563.20	28390912.	1139.0	44.9	
1.20000	49760.0	564.20	28074592.	1139.0	43.6	
1.30000	48680.0	565.00	27504200.	1134.0	42.0	
1.40000	47330.0	565.60	26769848.	1125.0	40.2	
1.50000	45950.0	566.30	26021485.	1117.0	38.4	
1.60000	44720.0	567.00	25356240.	1110.0	36.9	
1.70000	43650.0	567.80	24784470.	1104.0	35.7	
1.80000	42610.0	568.60	24228046.	1099.0	34.6	
1.90000	41490.0	569.30	23620257.	1093.0	33.5	
2.00000	40320.0	570.10	22986432.	1087.0	32.4	
2.10000	39180.0	570.90	22367862.	1082.0	31.3	
2.20000	38130.0	571.70	21798921.	1077.0	30.4	
2.30000	37180.0	572.70	21292986.	1073.0	29.6	
2.40000	36830.0	573.60	21125688.	1067.0	28.6	

TABLE 6.2-66E (CONT'D)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)
2.50000	36430.0	574.70	20936321.	1060.0	27.5
2.60000	36100.0	575.80	20786380.	1054.0	26.6
2.70000	35820.0	577.10	20671722.	1050.0	25.7
2.80000	35530.0	578.40	20550552.	1046.0	24.9
2.90000	35230.0	580.00	20433400.	1042.0	24.1
3.00000	34890.0	581.70	20295513.	1038.0	23.3
3.50000	32910.0	594.00	18548540.	1016.0	18.8
4.00000	29720.0	612.30	18197556.	964.0	13.8

1

TABLE 6.2-66F

MASS/ENERGY RELEASE RATES FOR 1400 SQ IN
SUCTION LEG GUILL. BREAK AT PUMP NOZZLE
FOR SUBCOMPARTMENT ANALYSIS OF
ST LUCIE II STRETCH POWER(2700 MWT)
(FLOW FROM PUMP SIDE)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)
0.00000	0.0	547.90	0.	2213.0	46.8
.00100	10130.0	546.90	5540097.	1973.0	46.7
.00200	15250.0	544.90	8309725.	1466.0	46.5
.00300	15230.0	543.20	8272936.	1045.0	46.3
.00400	19170.0	543.10	10411227.	1004.0	46.2
.00500	23940.0	543.10	13001814.	1003.0	46.0
.00600	28670.0	543.10	15570677.	1002.0	45.8
.00700	33350.0	543.10	18112385.	1000.0	45.5
.00800	37960.0	543.00	20612280.	998.5	45.0
.00900	42480.0	543.00	23066640.	996.4	44.7
.01000	46870.0	543.00	25450410.	993.8	44.2
.01200	45890.0	543.00	24918270.	986.2	42.7
.01400	44900.0	543.00	24380700.	978.2	41.3
.01600	43870.0	542.90	23817023.	970.1	39.9
.01800	42800.0	542.90	23236120.	961.7	38.5
.02000	41690.0	542.80	22629332.	953.2	37.2
.02200	40540.0	542.80	22005112.	944.4	35.9
.02400	39380.0	542.80	21375464.	935.7	34.7
.02600	38220.0	542.70	20741994.	927.2	33.5
.02800	37110.0	542.70	20139597.	919.2	32.5
.03000	36190.0	542.70	19640313.	912.2	31.6
.03200	36020.0	542.70	19548054.	905.9	30.9
.03400	35850.0	542.80	19459380.	900.4	30.2
.03600	35680.0	542.80	19367104.	895.6	29.6
.03800	35510.0	542.90	19278379.	891.1	29.1
.04000	35340.0	542.90	19186086.	886.7	28.6
.04200	35150.0	543.00	19086450.	882.2	28.1
.04400	34940.0	543.00	18972420.	877.6	27.7
.04600	34690.0	543.00	18836670.	872.8	27.2
.04800	34420.0	543.00	18690060.	867.9	26.7
.05000	34120.0	543.00	18527160.	862.9	26.2
.05500	33270.0	543.10	18068937.	851.4	25.2
.06000	32580.0	543.30	17700714.	842.5	24.3
.06500	32320.0	543.50	17565920.	835.5	23.6
.07000	32060.0	543.60	17427816.	828.4	23.0
.07500	31750.0	543.60	17259300.	820.1	22.4
.08000	31390.0	543.70	17066743.	810.5	21.7
.08500	30990.0	543.70	16849263.	800.2	20.8
.09000	30510.0	543.70	16588287.	790.7	20.2
.09500	30120.0	543.70	16376244.	780.4	19.5
.10000	29690.0	543.60	16139484.	769.0	18.7
.11000	28750.0	543.50	15625625.	744.4	17.2
.12000	27930.0	543.50	15179955.	723.3	16.0
.13000	27240.0	543.50	14804940.	706.3	15.1
.14000	26580.0	543.60	14448888.	690.3	14.3

TABLE 6.2-66F (CONT'D)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)
.15000	26020.0	543.70	14147074.	676.8	13.7
.16000	25740.0	543.90	13999986.	670.1	13.3
.17000	25640.0	544.20	13953288.	668.1	13.2
.18000	25450.0	544.20	13849890.	664.1	13.0
.19000	25190.0	544.20	13708398.	658.8	12.8
.20000	24960.0	544.30	13585728.	652.7	12.5
.22000	24380.0	544.30	13270034.	639.5	12.0
.24000	24470.0	544.60	13326362.	641.3	12.0
.26000	24490.0	544.70	13339703.	642.1	12.1
.28000	24540.0	544.70	13366938.	643.5	12.1
.30000	24190.0	544.60	13173874.	635.7	11.8
.32000	24090.0	544.60	13119414.	633.0	11.7
.34000	24170.0	544.80	13167816.	634.6	11.8
.36000	24050.0	544.70	13100035.	632.7	11.7
.38000	23850.0	544.70	12991095.	628.0	11.5
.40000	23650.0	544.70	12882155.	623.4	11.4
.42000	23600.0	544.80	12857280.	622.2	11.3
.44000	23410.0	544.70	12751427.	618.5	11.2
.46000	23090.0	544.70	12577123.	611.2	10.9
.48000	22860.0	544.70	12451842.	605.9	10.7
.50000	22760.0	544.80	12399648.	603.6	10.6
.55000	22170.0	544.70	12075999.	588.1	10.1
.60000	21650.0	544.80	11794920.	575.3	9.6
.65000	21060.0	544.90	11475594.	561.6	9.1
.70000	20650.0	544.90	11252185.	552.5	8.9
.75000	20250.0	545.00	11036250.	544.0	8.6
.80000	19810.0	545.10	10798431.	535.1	8.3
.85000	19450.0	545.10	10602195.	527.9	8.1
.90000	19120.0	545.20	10424224.	521.5	7.9
.95000	18780.0	545.20	10238856.	515.2	7.7
1.00000	18550.0	545.30	10115315.	510.9	7.6
1.10000	18230.0	545.40	9942642.	505.3	7.5
1.20000	18000.0	545.50	9819000.	501.3	7.3
1.30000	17750.0	545.60	9684400.	497.0	7.2
1.40000	17450.0	545.60	9520720.	492.0	7.1
1.50000	17150.0	545.70	9358755.	486.9	7.0
1.60000	16900.0	545.80	9224020.	482.7	6.9
1.70000	16690.0	545.90	9111071.	479.5	6.8
1.80000	16510.0	546.00	9014460.	476.7	6.7
1.90000	16360.0	546.20	8935832.	473.2	6.6
2.00000	16180.0	546.30	8839134.	468.5	6.5
2.10000	16020.0	546.40	8753328.	463.9	6.4
2.20000	15870.0	546.60	8674542.	459.8	6.3
2.30000	15720.0	546.80	8595696.	455.9	6.2
2.40000	15580.0	546.90	8520702.	452.2	6.1

TABLE 6.2-66F (CONT'D)

TIME (SECONDS)	FLOW RATE (LB/SEC)	ENTHALPY (BTU/LB)	ENERGY RATE (BTU/SEC)	B.N. PRESS (PSIA)	DENSITY (LB/CU.FT)	
2.50000	15450.0	547.10	8452695.	448.9	6.0	
2.60000	15370.0	547.40	8413538.	446.7	5.9	
2.70000	15260.0	547.60	8356376.	443.9	5.9	
2.80000	15190.0	547.80	8321082.	442.2	5.8	1
2.90000	15140.0	548.10	8298234.	441.2	5.8	
3.00000	15020.0	548.40	8236968.	438.5	5.7	
3.50000	14450.0	550.20	7950390.	425.1	5.4	
4.00000	13350.0	552.70	7376545.	398.6	4.8	

Table 6.2-67A
This table has been deleted.

Table 6.2-67A
This table has been deleted.

TABLE 6.2-67B

MSLB MASS/ENERGY RELEASE RATE FOR ST. LUCIE 2 STRETCH POWER SLB
(Loss of Containment Cooling)
Increased Core Inlet Temp.
(Historical)

Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)	Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)
0.00	14045.1	1.680E+07	11.00	5926.2	7.138E+06
.20	13543.3	1.622E+07	11.50	5722.1	6.892E+06
.40	13088.1	1.568E+07	12.00	5525.8	6.655E+06
.60	12695.1	1.522E+07	12.50	5337.0	6.427E+06
.80	12343.6	1.481E+07	13.00	5168.6	6.223E+06
1.00	12022.4	1.443E+07	13.50	5009.5	6.031E+06
1.50	11144.7	1.339E+07	14.00	4857.5	5.847E+06
2.00	10412.1	1.252E+07	14.50	4714.5	5.674E+06
2.50	9818.0	1.181E+07	15.00	4581.7	5.513E+06
3.00	9331.0	1.123E+07	15.50	4459.9	5.365E+06
3.50	8928.9	1.075E+07	16.00	4349.3	5.231E+06
4.00	8633.9	1.039E+07	16.50	4250.0	5.111E+06
4.50	8460.0	1.018E+07	17.00	4161.5	5.003E+06
5.00	8341.3	1.004E+07	17.50	4083.2	4.909E+06
5.50	8262.8	9.947E+06	18.00	4014.3	4.825E+06
6.00	8214.2	9.888E+06	18.50	3953.8	4.752E+06
6.50	8190.3	9.860E+06	19.00	3899.9	4.686E+06
7.00	8182.7	9.851E+06	19.50	3852.3	4.629E+06
7.50	8190.8	9.859E+06	20.00	3808.9	4.576E+06
8.00	8203.9	9.876E+06	20.50	3767.9	4.526E+06
8.50	8213.9	9.888E+06	21.00	3729.2	4.479E+06
9.00	8216.6	9.891E+06	21.50	3692.0	4.434E+06
9.50	6631.4	7.988E+06	22.00	3655.8	4.390E+06
10.00	6370.7	7.674E+06	22.50	3620.0	4.347E+06
10.50	6140.3	7.396E+06	23.00	3583.9	4.303E+06

TABLE 6.2-67B (CONT'D)
(Historical)

Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)	Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)
23.50	3547.4	4.259E+06	36.00	2823.7	3.381E+06
24.00	3510.4	4.214E+06	36.50	2803.1	3.356E+06
24.50	3472.7	4.168E+06	37.00	2782.3	3.330E+06
25.00	3434.5	4.122E+06	37.50	2761.1	3.305E+06
25.50	3396.2	4.075E+06	38.00	2739.8	3.279E+06
26.00	3357.8	4.029E+06	38.50	2718.4	3.253E+06
26.50	3319.8	3.983E+06	39.00	2194.9	2.619E+06
27.00	3282.5	3.937E+06	39.50	2006.0	2.383E+06
27.50	3246.2	3.893E+06	40.00	1491.8	1.833E+06
28.00	3211.2	3.851E+06	40.50	549.7	6.480E+05
28.50	3177.6	3.810E+06	41.00	266.4	3.244E+05
29.00	3145.8	3.771E+06	41.50	127.8	1.586E+05
29.50	3115.6	3.735E+06	42.00	54.0	6.769E+04
30.00	3087.3	3.700E+06	42.50	26.3	3.322E+04
30.50	3060.6	3.668E+06	43.00	15.9	2.010E+04
31.00	3035.3	3.637E+06	43.50	11.0	1.393E+04
31.50	3010.7	3.607E+06	44.00	8.0	1.015E+04
32.00	2987.6	3.579E+06	44.50	6.0	7.563E+03
32.50	2965.6	3.553E+06	45.00	4.6	5.863E+03
33.00	2944.5	3.527E+06	45.50	3.9	4.947E+03
33.50	2924.0	3.502E+06	46.00	3.8	4.748E+03
34.00	2903.9	3.478E+06	46.50	4.1	5.180E+03
34.50	2884.0	3.454E+06	47.00	4.9	6.140E+03
35.00	2864.1	3.430E+06	47.50	6.0	7.507E+03
35.50	2844.0	3.405E+06	48.00	7.3	9.147E+03

TABLE 6.2-67C

MSLB MASS/ENERGY RELEASE RATE FOR ST. LUCIE 2 STRETCH POWER SLB
(Loss of Containment Cooling)
Decreased RCS Flow Rate
(Historical)

Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)	Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)
0.00	14045.1	1.680E+07	11.00	5918.6	7.129E+06
.20	13541.4	1.621E+07	11.50	5720.1	6.889E+06
.40	13084.7	1.568E+07	12.00	5528.5	6.658E+06
.60	12690.4	1.522E+07	12.50	5343.1	6.434E+06
.80	12337.7	1.481E+07	13.00	5176.3	6.232E+06
1.00	12015.6	1.443E+07	13.50	5018.2	6.041E+06
1.50	11135.9	1.338E+07	14.00	4866.0	5.857E+06
2.00	10401.8	1.251E+07	14.50	4721.7	5.682E+06
2.50	9807.8	1.180E+07	15.00	4586.6	5.519E+06
3.00	9318.5	1.122E+07	15.50	4461.8	5.367E+06
3.50	8914.4	1.073E+07	16.00	4347.6	5.229E+06
4.00	8616.7	1.037E+07	16.50	4244.4	5.104E+06
4.50	8438.9	1.016E+07	17.00	4151.8	4.992E+06
5.00	8318.9	1.001E+07	17.50	4069.6	4.892E+06
5.50	8237.3	9.916E+06	18.00	3997.0	4.804E+06
6.00	8183.3	9.851E+06	18.50	3933.1	4.727E+06
6.50	8154.6	9.817E+06	19.00	3876.8	4.658E+06
7.00	8146.5	9.807E+06	19.50	3826.6	4.597E+06
7.50	8150.8	9.812E+06	20.00	3782.1	4.543E+06
8.00	8164.1	9.828E+06	20.50	3740.5	4.493E+06
8.50	8176.0	9.843E+06	21.00	3701.9	4.446E+06
9.00	8181.3	9.849E+06	21.50	3665.5	4.402E+06
9.50	6605.8	7.957E+06	22.00	3630.6	4.360 E+06
10.00	6350.9	7.650E+06	22.50	3596.5	4.318E+06
10.50	6126.7	7.380E+06	23.00	3562.6	4.277E+06

TABLE 6.2-67C (CONT'D)
(Historical)

Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)	Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)
23.50	3528.4	4.236E+06	36.00	2810.0	3.364E+06
24.00	3493.8	4.194E+06	36.50	2790.6	3.340E+06
24.50	3458.5	4.151E+06	37.00	2770.8	3.317E+06
25.00	3422.5	4.107E+06	37.50	2750.8	3.292E+06
25.50	3386.0	4.063E+06	38.00	2730.6	3.268E+06
26.00	3349.2	4.018E+06	38.50	2710.0	3.243E+06
26.50	3312.4	3.973E+06	39.00	2502.9	2.992E+06
27.00	3275.7	3.929E+06	39.50	1817.6	2.062E+06
27.50	3239.6	3.885E+06	40.00	1737.3	2.061E+06
28.00	3204.4	3.842E+06	40.50	1045.1	1.275E+06
28.50	3170.3	3.801E+06	41.00	463.9	5.649E+05
29.00	3137.5	3.761E+06	41.50	180.6	2.227E+05
29.50	3106.3	3.723E+06	42.00	75.2	9.404E+04
30.00	3076.7	3.688E+06	42.50	34.6	4.357E+04
30.50	3048.8	3.654E+06	43.00	19.3	2.435E+04
31.00	3021.6	3.621E+06	43.50	12.9	1.634E+04
31.50	2995.9	3.590E+06	44.00	9.4	1.182E+04
32.00	2971.9	3.560E+06	44.50	6.9	8.761E+03
32.50	2949.2	3.533E+06	45.00	5.3	6.641E+03
33.00	2927.7	3.507E+06	45.50	4.2	5.314E+03
33.50	2907.1	3.482E+06	46.00	3.7	4.715E+03
34.00	2887.1	3.458E+06	46.50	3.8	4.777E+03
34.50	2807.7	3.434E+06	47.00	4.3	5.413E+03
35.00	2848.4	3.411E+06	47.50	5.2	6.518E+03
35.50	2829.3	3.387E+06	48.00	6.3	7.972E+03

TABLE 6.2-67D

MSLB MASS/ENERGY RELEASE RATE FOR ST. LUCIE 2 STRETCH POWER SLB
(MSIV Failure)
Increased Core Inlet Temp.
Decreased RCS Flow Rate
(Historical)

Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)	Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)
0.00	14045.1	1.680E+07	11.00	7162.6	8.628E+06
.20	13544.8	1.622E+07	11.50	6754.4	8.136E+06
.40	13091.0	1.569E+07	12.00	6400.3	7.708E+06
.60	12699.2	1.523E+07	12.50	6089.3	7.333E+06
.80	12348.7	1.482E+07	13.00	5813.9	7.000E+06
1.00	12028.3	1.444E+07	13.50	5573.0	6.709E+06
1.50	11152.9	1.340E+07	14.00	5356.2	6.447E+06
2.00	10419.5	1.253E+07	14.50	5159.6	6.209E+06
2.50	9824.7	1.182E+07	15.00	4980.2	5.992E+06
3.00	9336.6	1.124E+07	15.50	4816.7	5.794E+06
3.50	8933.6	1.076E+07	16.00	4867.5	5.614E+06
4.00	8635.2	1.040E+07	16.50	4532.1	5.450E+06
4.50	8458.6	1.016E+07	17.00	4409.8	5.302E+06
5.00	8338.8	1.004E+07	17.50	4300.8	5.169E+06
5.50	8257.8	9.941E+06	18.00	4202.2	5.050E+06
6.00	8207.3	9.880E+06	18.50	4115.4	4.945E+06
6.50	8180.8	9.848E+06	19.00	4038.9	4.856E+06
7.00	8173.5	9.840E+06	19.50	3971.4	4.771E+06
7.50	8181.9	9.850E+06	20.00	3912.0	4.699E+06
8.00	8196.7	9.867E+06	20.50	3859.1	4.635E+06
8.50	8211.6	9.885E+06	21.00	3811.7	4.578E+06
9.00	8218.5	9.893E+06	21.50	3768.4	4.526E+06
9.50	8177.3	9.847E+06	22.00	3728.5	4.477E+06
10.00	7953.3	9.580E+06	22.50	3690.0	4.431E+06
10.50	7638.8	9.202E+06	23.00	3653.7	4.387E+06

TABLE 6.2-67D (CONT'D)
(Historical)

Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)	Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)
23.50	3618.2	4.344E+06	36.00	2860.0	3.424E+06
24.00	3583.2	4.301E+06	36.50	2840.4	3.400E+06
24.50	3548.2	4.259E+06	37.00	2821.5	3.377E+06
25.00	3512.7	4.215E+06	37.50	2801.5	3.353E+06
25.50	3476.7	4.172E+06	38.00	2781.7	3.329E+06
26.00	3440.2	4.127E+06	38.50	2761.9	3.305E+06
26.50	3402.7	4.082E+06	39.00	2598.4	3.101E+06
27.00	3365.1	4.036E+06	39.50	2365.4	2.813E+06
27.50	3327.6	3.991E+06	40.00	2165.7	2.571E+06
28.00	3290.2	3.945E+06	40.50	1349.8	1.625E+06
28.50	3253.3	3.901E+06	41.00	827.8	9.892E+05
29.00	3217.8	3.858E+06	41.50	574.1	6.903E+05
29.50	3183.1	3.816E+06	42.00	373.0	4.499E+05
30.00	3149.6	3.775E+06	42.50	216.2	2.611E+05
30.50	3118.3	3.737E+06	43.00	104.5	1.271E+05
31.00	3088.4	3.701E+06	43.50	36.7	4.532E+04
31.50	2059.7	3.666E+06	44.00	27.3	3.351E+04
32.00	3033.0	3.634E+06	44.50	24.3	2.965E+04
32.50	3007.2	3.602E+06	45.00	22.3	2.711E+04
33.00	2983.4	3.573E+06	45.50	21.2	2.574E+04
33.50	2960.5	3.546E+06	46.00	19.9	2.414E+04
34.00	2938.9	3.520E+06	46.50	20.0	2.425E+04
34.50	2918.4	3.495E+06	47.00	20.5	2.490E+04
35.00	2898.4	3.471E+06	47.50	21.4	2.597E+04
35.50	2879.4	3.448E+06	48.00	23.3	2.837E+04

TABLE 6.2-67E

MSLB MASS/ENERGY RELEASE RATE FOR ST. LUCIE 2 STRETCH POWER SLB
(MSIV Failure)

increased core inlet temp.

Time (sec.)	Mass flow (lbm/sec)	Energy release rate (Btu/sec)	Time (sec)	Mass flow (lbm/sec)	Energy release rate (Btu/sec)
~~0.00~~	~~14045.1~~	~~1.680E+07~~	~~11.00~~	~~7146.4~~	~~8.698E+06~~
~~.20~~	~~13543.3~~	~~1.622E+07~~	~~11.50~~	~~6735.0~~	~~8.112E+06~~
~~.40~~	~~13048.1~~	~~1.568E+07~~	~~12.00~~	~~6370.1~~	~~7.642E+06~~
~~.60~~	~~12693.1~~	~~1.522E+07~~	~~12.50~~	~~6064.7~~	~~7.303E+06~~
~~.80~~	~~12343.6~~	~~1.481E+07~~	~~13.00~~	~~5788.7~~	~~6.970E+06~~
~~1.00~~	~~12022.4~~	~~1.443E+07~~	13.50	5547.1	6.678E+06
~~1.50~~	~~11144.7~~	~~1.339E+07~~	~~14.00~~	~~5330.4~~	~~6.416E+06~~
~~2.00~~	~~10412.1~~	~~1.292E+07~~	~~14.50~~	~~5134.7~~	~~6.179E+06~~
~~2.50~~	~~9419.0~~	~~1.191E+07~~	~~15.00~~	~~4957.2~~	~~5.964E+06~~
~~3.00~~	~~9331.[illegible]~~	~~1.123E+07~~	~~15.50~~	~~4796.3~~	~~5.769E+06~~
~~3.50~~	~~8926.4~~	~~1.075E+07~~	~~16.00~~	~~4649.8~~	~~5.592E+06~~
~~4.00~~	~~8633.4~~	~~1.039E+07~~	~~16.50~~	~~4518.1~~	~~5.433E+06~~
~~4.50~~	[illegible]	[illegible]	~~17.00~~	~~4399.0~~	~~5.291E+06~~
~~5.00~~	~~8341.3~~	~~1.004E+07~~	~~17.50~~	~~4293.9~~	~~5.162E+06~~
~~5.50~~	~~8262.8~~	~~9.947E+06~~	~~18.00~~	~~4199.9~~	[illegible]
~~6.00~~	~~8214.2~~	[illegible]	~~18.50~~	~~4116.8~~	~~4.947E+06~~
~~6.50~~	~~8192.5~~	~~9.863E+06~~	~~19.00~~	~~4043.6~~	[illegible]
~~7.00~~	~~8192.7~~	~~9.851E+06~~	~~19.50~~	~~3979.2~~	[illegible]
~~7.50~~	~~8190.0~~	[illegible]	~~20.00~~	~~3922.6~~	~~4.712E+06~~
~~8.00~~	~~8203.9~~	[illegible]	~~20.50~~	~~3871.5~~	~~4.650E+06~~
~~8.50~~	~~8213.9~~	~~9.880E+06~~	~~21.00~~	~~3825.6~~	~~4.595E+06~~
~~9.00~~	~~8216.6~~	~~9.891E+06~~	~~21.50~~	~~3783.9~~	~~4.543E+06~~
~~9.50~~	~~8169.8~~	~~9.834E+06~~	~~22.00~~	~~3743.2~~	~~4.495E+06~~
~~10.00~~	~~7743.4~~	~~9.505E+06~~	~~22.50~~	~~3703.9~~	~~4.447E+06~~
~~10.50~~	~~7626.3~~	~~9.179E+06~~	~~23.00~~	~~3666.1~~	~~4.402E+06~~

TABLE 6.2-67E (CONT'D)

Time (sec.)	Mass flow (lbm/sec)	Energy release rate (BTU/sec)
~~23.50~~	~~3629.7~~	~~4.357E+06~~
~~24.00~~	~~3592.7~~	~~4.312E+06~~
~~24.50~~	~~3555.5~~	~~4.267E+06~~
~~25.00~~	~~3517.7~~	~~4.222E+06~~
~~25.50~~	~~3479.6~~	~~4.175E+06~~
~~26.00~~	~~3441.2~~	~~4.129E+06~~
~~26.50~~	~~3402.7~~	~~4.082E+06~~
~~27.00~~	~~3364.1~~	~~4.035E+06~~
~~27.50~~	~~3325.7~~	~~3.9[illegible]E+06~~
~~28.00~~	~~3288.4~~	~~3.943E+06~~
~~28.50~~	~~3251.8~~	~~3.894E+06~~
~~29.00~~	~~3216.5~~	~~3.856E+06~~
~~29.50~~	~~3182.8~~	~~3.815E+06~~
~~30.00~~	~~3150.4~~	~~3.776E+06~~
~~30.50~~	~~3120.3~~	~~3.734E+06~~
~~31.00~~	~~3091.7~~	~~3.7[illegible]E+06~~
~~31.50~~	~~3064.6~~	~~3.672E+06~~
~~32.00~~	~~3038.7~~	~~3.641E+06~~
~~32.50~~	~~3014.0~~	~~3.613E+06~~
~~33.00~~	~~2991.3~~	~~3.583E+06~~
~~33.50~~	~~2969.1~~	~~3.550E+06~~
~~34.00~~	~~2947.4~~	~~3.531E+06~~
~~34.50~~	~~2927.3~~	~~3.506E+06~~
~~35.00~~	~~2907.2~~	~~3.481E+06~~
~~35.50~~	~~2887.7~~	~~3.456E+06~~
~~36.00~~	~~2867.4~~	~~3.433E+06~~
~~36.50~~	~~2847.6~~	~~3.404E+06~~
~~37.00~~	~~2828.8~~	~~3.384E+06~~
~~37.50~~	~~2810.3~~	~~3.359E+06~~
~~38.00~~	~~2785.3~~	~~3.333E+06~~
38.50	2764.6	3.308E+06
~~39.00~~	~~2630.1~~	~~3.113E+06~~
~~39.50~~	~~2366.8~~	~~2.815E+06~~
~~40.00~~	~~2165.9~~	~~2.572E+06~~
~~40.50~~	~~1349.8~~	~~1.625E+06~~
~~41.00~~	~~826.2~~	~~9.896E+05~~
~~41.50~~	~~571.2~~	~~6.867E+05~~
~~42.00~~	~~373.3~~	~~4.521E+05~~
~~42.50~~	~~219.7~~	~~2.655E+05~~
~~43.00~~	~~102.4~~	~~1.245E+05~~
~~43.50~~	~~36.8~~	~~4.535E+04~~
~~44.00~~	~~27.2~~	~~3.3[illegible]E+04~~
~~44.50~~	~~23.3~~	~~2.849E+04~~
~~45.00~~	~~22.2~~	~~2.702E+04~~
~~45.50~~	~~21.2~~	~~2.574E+04~~
~~46.00~~	~~21.0~~	~~2.[illegible]E+04~~
~~46.50~~	~~21.0~~	~~2.552E+04~~
~~47.00~~	~~20.8~~	~~2.520E+04~~
~~47.50~~	~~21.7~~	~~2.632E+04~~
~~48.00~~	~~22.9~~	~~2.707E+04~~

1

TABLE 6.2-67F

MSLB MASS/ENERGY RELEASE RATE FOR ST. LUCIE 2 STRETCH POWER SLB
(MSIV Failure)

decreased RCS flow rate

Time (sec.)	Mass flow (lbm/sec)	Energy release rate (Btu/sec)
~~0.00~~	~~14049.1~~	~~1.649E+07~~
~~.20~~	~~13541.4~~	~~1.621E+07~~
~~.40~~	~~13044.7~~	~~1.568E+07~~
~~.60~~	~~12649.4~~	~~1.522E+07~~
~~.80~~	~~12337.7~~	~~1.481E+07~~
~~1.00~~	~~12015.6~~	~~1.443E+07~~
~~1.50~~	~~11139.9~~	~~1.336E+07~~
~~2.00~~	~~10401.8~~	~~1.251E+07~~
~~2.50~~	~~9067.8~~	~~1.1[illegible]E+07~~
~~3.00~~	~~9319.9~~	~~1.122E+07~~
~~3.50~~	~~8914.4~~	~~1.[illegible]75E+07~~
~~4.00~~	~~8618.7~~	~~1.[illegible]37E+07~~
~~4.50~~	~~[illegible]~~	~~1.[illegible]10E+07~~
~~5.00~~	~~8314.4~~	~~1.[illegible]E+07~~
~~5.50~~	~~8237.3~~	~~9.916E+06~~
~~6.00~~	~~8193.1~~	~~9.[illegible]31E+06~~
~~6.50~~	~~8154.6~~	~~9.817E+06~~
~~7.00~~	~~8144.5~~	~~9.[illegible]E+06~~
~~7.50~~	~~8150.4~~	~~9.[illegible]12E+06~~
~~8.00~~	~~8104.1~~	~~9.[illegible]24E+06~~
~~8.50~~	~~8170.0~~	~~9.[illegible]43E+06~~
~~9.00~~	~~8144.5~~	~~9.[illegible]E+06~~
~~9.50~~	~~8137.5~~	~~9.[illegible]E+06~~
~~10.00~~	~~7414.3~~	~~9.533E+06~~
~~10.50~~	~~7001.0~~	~~9.190E+06~~
~~11.00~~	~~7128.0~~	~~8.580E+06~~
~~11.50~~	~~6721.9~~	~~8.190E+06~~
~~12.00~~	~~6309.9~~	~~7.672E+06~~
~~12.50~~	~~6000.8~~	~~7.299E+06~~
~~13.00~~	~~5700.3~~	~~6.969E+06~~
13.50	5594.5	6.681E+06
~~14.00~~	~~5334.9~~	~~6.421E+06~~
~~14.50~~	~~5139.7~~	~~6.165E+06~~
~~15.00~~	~~4902.0~~	~~5.9[illegible]E+06~~
~~15.50~~	~~4749.7~~	~~5.774E+06~~
~~16.00~~	~~4651.9~~	~~5.545E+06~~
~~16.50~~	~~4517.5~~	~~5.432E+06~~
~~17.00~~	~~4348.1~~	~~5.275E+06~~
~~17.50~~	~~4247.1~~	~~5.155E+06~~
~~18.00~~	~~4189.4~~	~~[illegible]~~
~~18.50~~	~~4103.2~~	~~4.911E+06~~
~~19.00~~	~~4020.8~~	~~4.838E+06~~
~~19.50~~	~~3959.4~~	~~4.797E+06~~
~~20.00~~	~~3900.0~~	~~4.655E+06~~
~~20.50~~	~~3847.2~~	~~4.621E+06~~
~~21.00~~	~~3794.0~~	~~4.563E+06~~
~~21.50~~	~~3756.5~~	~~4.511E+06~~
~~22.00~~	~~3715.9~~	~~4.462E+06~~
~~22.50~~	~~3677.9~~	~~4.416E+06~~
~~23.00~~	~~3641.5~~	~~4.372E+06~~

1