FLORIDA POWER & LIGHT COMPANY

ST LUCIE PLANT - UNIT NO. 1

SPENT FUEL STORAGE FACILITY MODIFICATION

SAFETY ANALYSIS REPORT

DOCKET NO. 50-335

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1.0 INTRODUCTION

1.1 LICENSE AMENDMENT REQUESTED

Florida Power & Light Company (FPL) has contracted for the design and manufacture of new spent fuel storage racks to be placed into the spent fuel pool of St Lucie Unit No. 1. The purpose of the new racks is to increase the amount of spent fuel that can be stored in the existing spent fuel pool. The racks are designed so that they can store spent fuel assemblies in a high density array. Therefore, FPL hereby requests that a License Amendment be issued to the St Lucie Unit No. 1 Facility Operating License DPR-67[1] to include installation and use of new storage racks that meet the criteria contained herein. This Safety Analysis Report (SAR) has been prepared to support this request for license amendment.

1.2 CURRENT STATUS

The existing racks in the spent fuel pool at St Lucie Unit No. 1 have 728 total storage cells. With the presently available storage cells, St Lucie Unit No. 1 lost the full-core reserve storage capability after the seventh refueling, which was completed in the spring of 1987. To correct this situation and provide sufficient capacity at St Lucie Unit No. 1 to store discharged fuel assemblies, FPL plans to replace the existing storage racks with new high density spent fuel storage racks. The design of the new racks will allow for more dense storage of spent fuel, thus enabling the existing pool to store more fuel in the spent fuel pool. The new high density racks have a usable storage capacity of 1706 cells, extending the full-core-reserve storage capability until the year 2009.

If a full core offload is required in the interim, prior to the installation of the new racks, FPL intends to transfer enough of the oldest spent fuel from St. Lucie Unit 1 to St. Lucie Unit 2 to allow full core offload. A proposed license amendment to allow spent fuel transfer was submitted in July 1986[2] and is being reviewed by the NRC.

1.3 INTERFACES WITH OTHER ORGANIZATIONS

FPL has overall responsibility for this modification. Holtec International has designed the new spent fuel storage racks. Joseph Oat (JO) is responsible for the fabrication of the new spent fuel storage racks and the evaluation of those racks under accident conditions. Ebasco Services, Inc. is responsible for the building structural analysis, the evaluation of the spent fuel cooling system and the related accident evaluations. The installer, who will be chosen later, is responsible for the installation of the new spent fuel pool racks.

1.4 SUMMARY OF REPORT

This Safety Analysis Report follows the guidance of the NRC position paper entitled, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978, as amended by the NRC letter dated January 18, 1979[3]. Sections 3.0 through 5.0 of this report are consistent with the section/subsection format and content of the NRC position paper, Sections III through V.

[illegible]

The nuclear and thermal-hydraulic aspects of the report (Section 3.0) address the neutron multiplication factor, considering normal storage and handling of spent fuel as well as postulated accidents with respect to criticality and the ability of the spent fuel pool cooling system to maintain sufficient cooling. Movement of spent fuel stored in the spent fuel pool during removal of the present racks and installation of the new racks is also addressed.

Section 4.0, which describes the mechanical, material and structural aspects of the new racks, contains information concerning the capability of the fuel assemblies, storage racks, and spent fuel pool system to withstand the effects of natural phenomena and other service loading conditions.

The environmental aspects of the report (Section 5.0) concern the thermal and radiological release from the facility under normal and accident conditions. This section also addresses the occupational radiation exposures, generation of radioactive waste, need for expansion, commitment of material and non-material resources, and a cost-benefit assessment.

1.5 CONCLUSIONS

On the basis of the evaluations and information presented in this report, plus operating experience with high density fuel storage at St Lucie Unit 2 and Turkey Point Unit 3, FPL concludes that the proposed modification of St Lucie Unit No. 1 spent fuel storage facilities provides safe spent fuel storage, and that the modification is consistent with the facility design and operating criteria as provided in the FSAR[4] and operating license.

1.6 REFERENCES

1. St Lucie Unit No. 1 Facility Operating Licenses DPR 67, Docket No. 50-335.

2. FPL letter L-86-250 dated July 2, 1986.

3. Nuclear Regulatory Commission, Letter to All Power Reactor Licensees, from B. K. Grimes, April 14, 1978, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," as amended by the NRC letter dated January 18, 1979.

4. St Lucie Plant Unit No. 1 Updated Final Safety Analysis Report, Docket No. 50-335.

2.0 SUMMARY OF RACK DESIGN

2.1 EXISTING RACKS

The spent fuel pool at St. Lucie Unit 1 presently contains spent fuel assembly storage racks which are designed to provide storage locations for up to 728 fuel assemblies. The racks are designed to maintain the stored fuel in a safe, coolable, and subcritical configuration during normal and abnormal conditions.

The present storage racks are a rectangular array composed of 14 modules. Each storage rack module is self supporting and rests on stainless steel pads. The present racks are free standing in that they are neither bolted nor welded to the floor, nor are they attached to the pool walls. The interface with the pool boundaries is designed to transfer normal and shear loads via the rack supports into the pool bottom slab.

Each fuel assembly storage module is composed of rectangular storage cavities fabricated from one-quarter inch thick stainless steel plate, with each cavity capable of accepting one fuel assembly. The fuel assembly storage cavities have lead-in surfaces at the top to provide guidance for insertion of fuel assemblies. The cavities are open at the top and bottom to provide a flow path for convective cooling of spent fuel assemblies through natural circulation. The fuel assembly storage cavities are connected by a chevron grid structure to form modules which limit structural deformations and maintain a nominal center-to-center spacing of 12.53 inches between adjacent storage cavities during design conditions including seismic.

For further information on the existing spent fuel storage racks see Section 9.1.2 in the St Lucie Unit No. 1 updated FSAR.

2.2 NEW HIGH DENSITY RACKS

The new high density spent fuel storage racks consist of individual cells with 8.65 inch by 8.65 inch (nominal) square cross-section, each of which accommodates a single Combustion Engineering or Exxon PWR fuel assembly or equivalent, from either St. Lucie Unit 1 or Unit 2. A total of 1706 cells are arranged in 17 distinct modules of varying sizes in two regions. Region 1 is designed for storage of new fuel assemblies with enrichments up to 4.5 weight percent U-235. Region 1 is also designed to store fuel assemblies with enrichments up to 4.5 weight percent U-235 that have not achieved adequate burnup for Region 2. The Region 2 cells are capable of accommodating fuel assemblies with various initial enrichments which have accumulated minimum burnups within an acceptable bound as discussed in this report. For example, corresponding to 4.5 and 4.0 percent initial enrichments, the minimum required burn-ups for safe storage in Region 2 are 36.5 and 30.9 MWD/KgU, respectively. Figure 2-1 shows the arrangement of the rack modules in the spent fuel pool.

The high density racks are engineered to achieve the dual objective of maximum protection against structural loadings (arising from ground motion, thermal stresses, etc.) and the maximization of available storage locations. In general, a greater width-to-height aspect ratio provides greater margin against rigid body tipping. Hence, the modules are made as large as possible within the constraints of transportation and site handling capabilities.

As shown in Figure 2-1, there are 17 discrete modules arranged in the fuel pool. Each rack module is equipped (see Figures 2-2 and 2-3) with girdle bars, 3/4-inch thick by 3-1/2 inches high. The nominal gap between adjacent module walls is 1-1/2 inches. The modules make surface contact between their contiguous walls at the girdle bar locations and thus maintain a specified gap between the cell walls. Table 2-1 gives the relevant design data on each region. The modules in the two regions are of eight different types. Tables 2-2 and 2-3 summarize the physical data for each module type.

The poison in Regions 1 and 2 is Boraflex. The use of this absorber material is to preclude inadvertent criticality.

TABLE 2-1

DESIGN DATA

Region	Cell Pitch (nominal inch)	Min. B-10 Loading (areal density)	Flux Trap Gap (nominal inch)
1	10.12	.020 gm/cm^2	1.12
2	8.86	.007 gm/cm^2	0.0

TABLE 2-2

TABLE OF MODULE DATA

MODULE I.D.	NO. OF MODULES	NO. OF CELLS IN N-S DIRECTION	NO. OF CELLS IN E-W DIRECTION	TOTAL NO. OF CELLS PER MODULE
Region 1 A1 and A2	2	9	9	81
Region 1 B1 and B2	2	9	10	90
Region 2 C1, C2, C3, C4	4	13	9	117
Region 2 D1, D2, D3	3	13	8	104
Region 2 E1 and E2	2	11	8	88
Region 2 F1	1	12	8	96
Region 2 G1 and G2	2	12	9	108
Region 2* H1	1	13	8	96

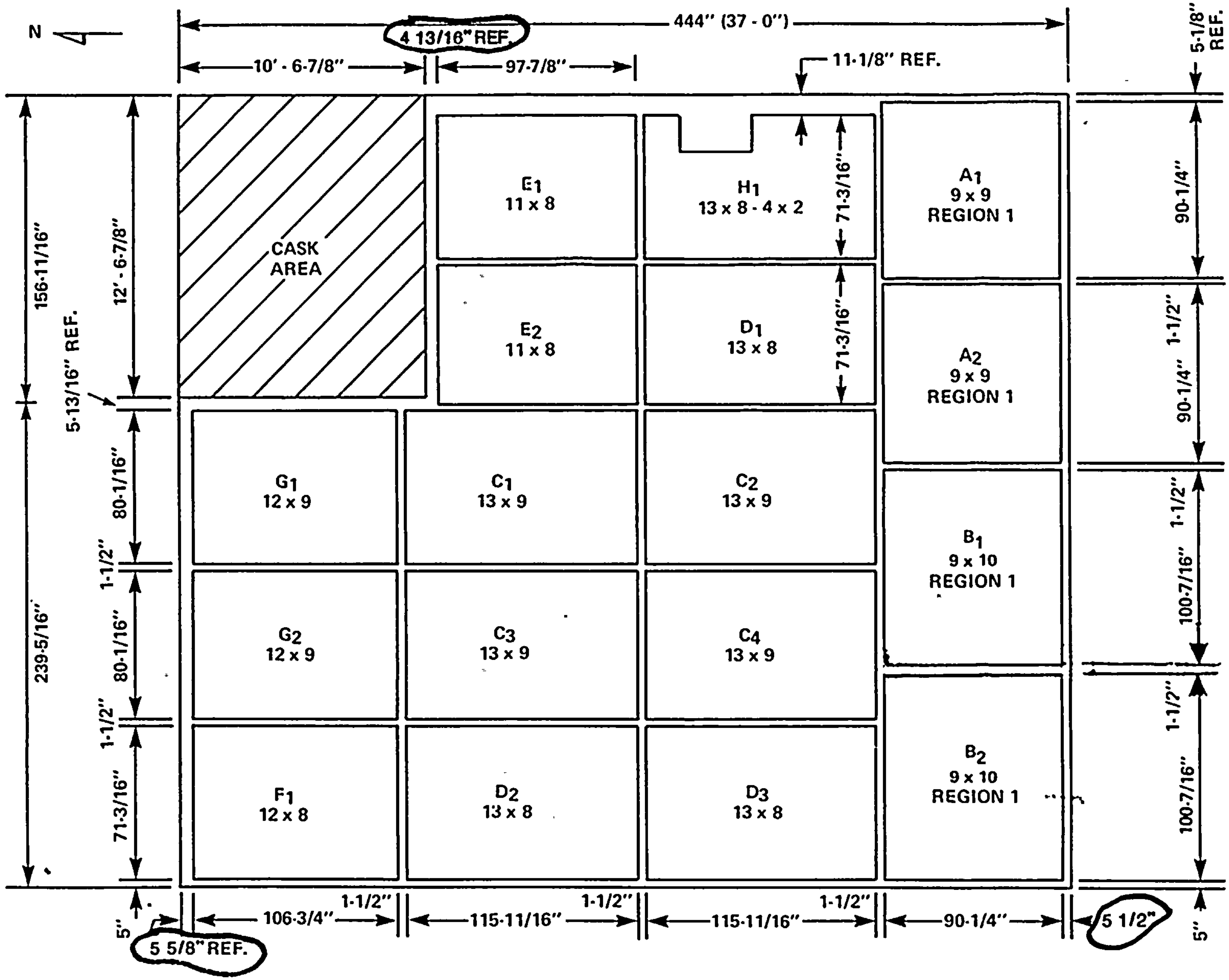
* Cells missing in this module due to sparger. Refer to Figure 2-1.

TABLE 2-3

MODULE DIMENSIONS AND WEIGHT

MODULE I.D.	NOMINAL CROSS-SECTION* DIMENSIONS N-S	E-W	ESTIMATED DRY WEIGHT (lbs) PER MODULE
Region 1 A1 and A2	90-1/4"	90-1/4"	26,700
Region 1 B1 and B2	90-1/4"	100-7/16"	29,800
Region 2 C1, C2, C3, C4	115-11/16"	80-1/6"	24,100
Region 2 D1, D2, D3	115-11/16"	71-3/16"	21,500
Region 2 E1 and E2	97-7/8"	71-3/16"	18,200
Region 2 F1	106-3/4"	71-3/16"	19,800
Region 2 G1 and G2	106-3/4"	80-1/16"	22,300
Region 2 H1	115-11/16"	71-3/16"	19,800

* Excluding girdle bars



REVISION 1

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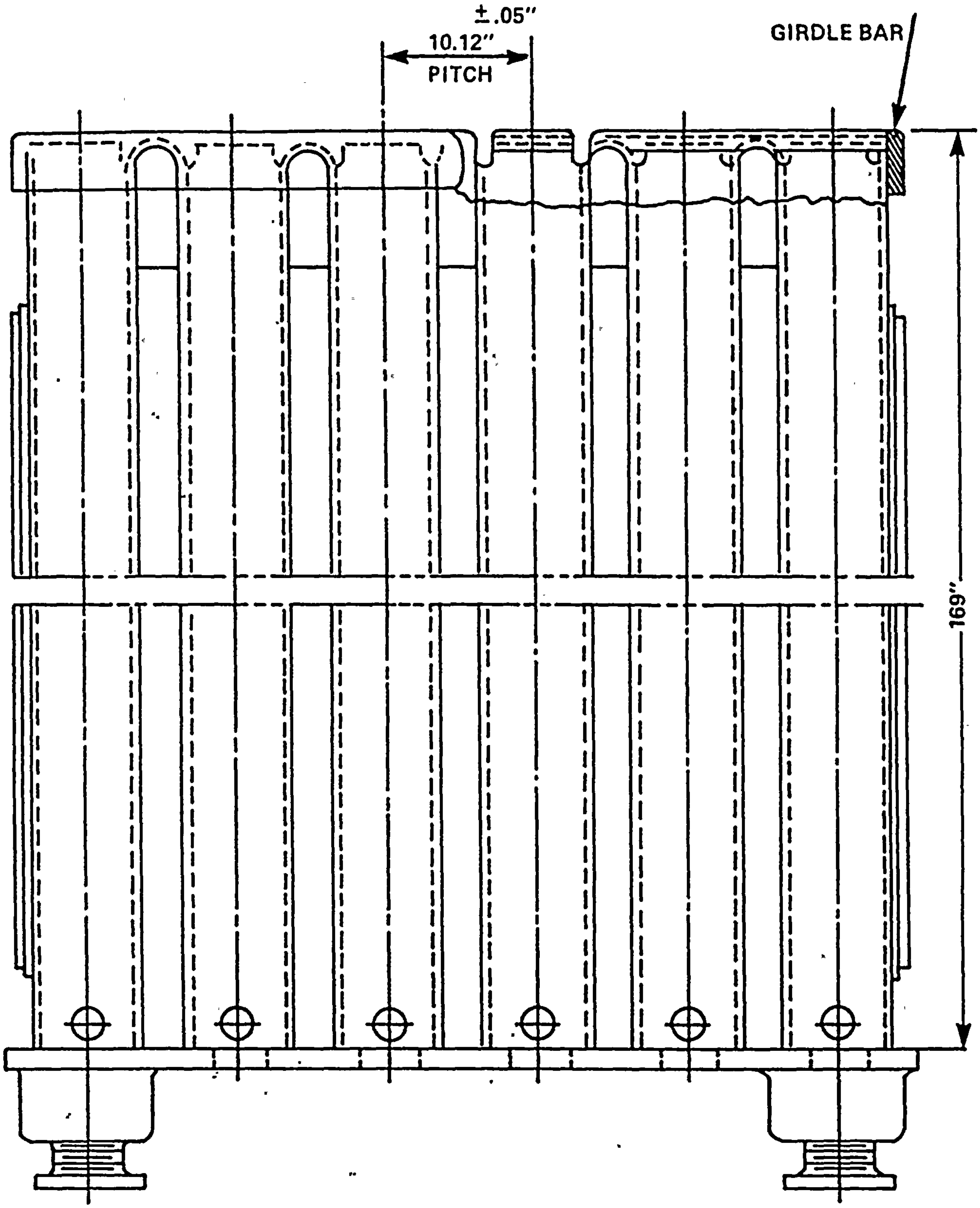
POOL LAYOUT
FIGURE 2-1



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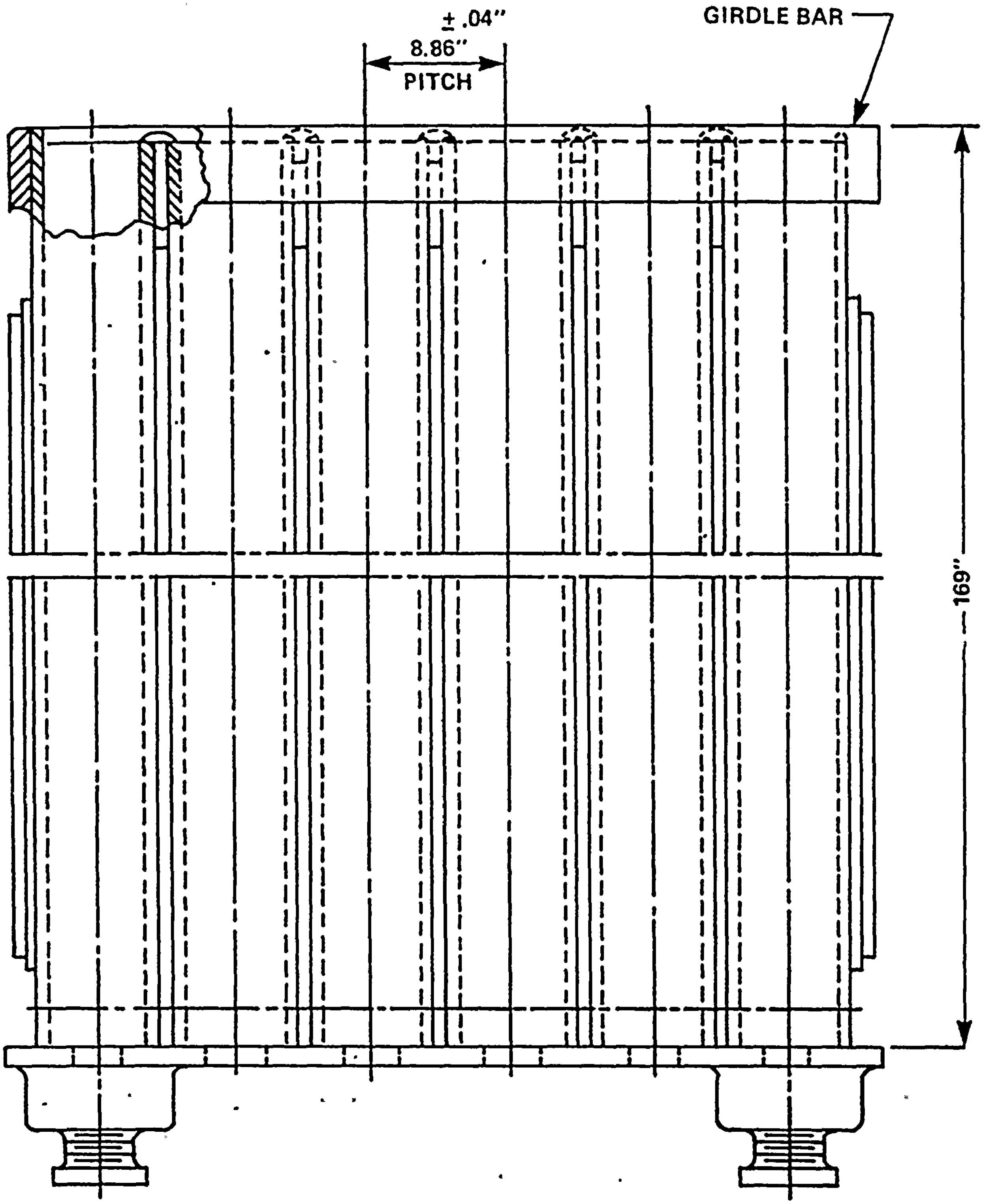
TYPICAL RACK ELEVATION
REGION 1
FIGURE 2-2



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TYPICAL RACK ELEVATION
REGION 2
FIGURE 2-3

3.0 NUCLEAR AND THERMAL-HYDRAULIC CONSIDERATIONS

3.1 NEUTRON MULTIPLICATION FACTOR

The following subsections describe the conditions in the spent fuel pool which are assumed in calculating the effective neutron multiplication factor (k_{eff}), the analysis methodology, and the analysis results.

3.1.1 Normal Storage

The criticality analyses of each of the two separate regions of the spent fuel storage pool are summarized in Table 3-1 for the anticipated normal storage conditions. The calculated maximum reactivity in Region 2 includes a burnup-dependent allowance for uncertainty in depletion calculations and, furthermore, provides an additional margin of 0.0065 Δk below the limiting effective multiplication factor (k_{eff}) of 0.95. As cooling time increases in long-term storage, decay of Pu-241 results in a significant decrease in reactivity, which will provide an increasing subcriticality margin and tends to further compensate for any uncertainty in depletion calculations. Spacing between two different rack modules is sufficient to preclude adverse nuclear interaction, since the minimum spacing between racks is greater than the design water gap spacing.

Region 2 can accommodate fuel of various initial enrichments and discharge fuel burnups, provided the combination falls within the acceptable domain illustrated in Figure 3-1. For convenience of reference, the minimum burnup values in Figure 3-1 have been fitted by linear tangents at various values and the results are tabulated in Table 3-2. Linear interpolation between the tabulated values will always yield values on or conservatively above the curve of limiting burnups.

These data will be implemented in appropriate administrative procedures to assure verified burnup as specified in draft Regulatory Guide 1.13, Revision 2. Administrative procedures will also be employed to confirm and assure the presence of soluble poison in the pool water at all times, providing a further margin of safety and assuring subcriticality in the event of fuel misplacement during fuel handling operations, as discussed in Section 3.1.2.

3.1.1.1 New Fuel Storage in Region 2

Criticality analyses confirm that a checkerboard pattern (fuel assemblies aligned diagonally) provides an acceptable k_∞ for the storage of fresh fuel assemblies of 4.5% enrichment in Region 2. These calculations indicate a nominal k_∞ of 0.819 $\pm$ 0.025 (95%/95%) when fully flooded with clean unborated water. This value is substantially less than the limiting k_{eff} of 0.95, even with the addition of a reasonable allowance for uncertainties.

With Boraflex absorber between assemblies, conditions do not exist for the appearance of a peak in reactivity at low moderator densities, and the fully flooded condition corresponds to the highest reactivity (optimum moderation). Thus, the checkerboard pattern of new 4.5% enriched fuel in Region 2 represents a safe configuration in conformance with both Standard Review Plan (SRP) 9.1.1 and 9.1.2.

3.1.2 Postulated Accidents

Although credit for the soluble poison normally present in the spent fuel pool water is permitted under abnormal or accident conditions*, most abnormal or accident conditions will not result in exceeding the limiting reactivity (k_{eff} of 0.95) even in the absence of soluble poison. The effects on reactivity of credible abnormal and accident conditions are summarized in Table 3-3. Of these abnormal/ accident conditions, only one has the potential for a more than negligible positive reactivity effect.

The inadvertent misplacement of a fresh fuel assembly (either into a Region 2 storage cell or outside and adjacent to a rack module) has the potential for exceeding the limiting reactivity should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. Administrative procedures assure the presence of soluble poison at all times and will preclude the possibility of the simultaneous occurrence of these two independent accident conditions. The largest reactivity increase occurs for accidentally placing a new fuel assembly into a Region 2 storage cell with all other cells fully loaded with fuel of the highest permissible reactivity. Under this condition, the presence of approximately 500 ppm soluble boron assures that the infinite multiplication factor would not exceed the design basis reactivity for Region 2. With the normal concentration of soluble poison present (1720 ppm boron), k_∞ is less than 0.80 and the storage racks would not be critical even if Region 2 were to be fully loaded with fresh fuel of 4.5% enrichment. This concentration of soluble boron also precludes the possibility of exceeding the criticality limit in the event of a dropped cask accident.

See Section 5.3 for discussions on Accident Evaluations.

3.1.3 Calculation Methods

3.1.3.1 Criticality Analysis for Region 1

3.1.3.1.1 Nominal Design Case

Under normal conditions, with nominal dimensions, the k_∞ values calculated by three different methods of analysis are as follows:

Analytical Method	Bias-corrected k_∞	Maximum k_∞ (95%/95%)
CASMO-2E	0.9313 + 0.0018	0.9331
AMPX-KENO (27-gp SCALE)	0.9210 + 0.0084	0.9294
Diffusion/blackness theory	0.9313	0.9313

The AMPX-KENO calculations include a one-sided tolerance factor[13] corresponding to 95% probability at a 95% confidence limit. For the nominal design case, the CASMO-2E calculation yields the highest reactivity and, therefore, the independent verification calculations substantiate CASMO-2E as the primary calculational method.

* Double contingency principle of ANSI N16.1-1975, as specified in the April 4, 1978 NRC letter (Section 1.2) and implied in the proposed revision (draft) to Reg. Guide 1.13 (Section 1.4, Appendix A).

3.1.3.1.2 Boron Loading Variation

The Boraflex absorber sheets used in Region 1 storage cells are nominally 0.075 inch thick, with a B-10 areal density of 0.0238 g/cm^2. Independent manufacturing tolerance limits are $\pm$ 0.007 inch in thickness and $\pm$ 0.009 g/cm^3 in B-10 content. This assures that at any point where the minimum boron concentration (0.1158 gram B-10/cm^3) and minimum Boraflex thickness (0.068 inch) may coincide, the boron-10 areal density will not be less than 0.020 g/cm^2. Differential CASMO-2E calculations indicate that these tolerance limits result in reactivity uncertainty of $\pm$ 0.0021 Δk for boron content and $\pm$ 0.0044 Δk for Boraflex thickness variations.

3.1.3.1.3 Storage Cell Lattice Pitch Variation

The design storage cell lattice spacing between fuel assemblies in Region 1 is 10.12 inches. A decrease in storage cell lattice spacing may or may not increase reactivity depending upon other dimensional changes that may be associated with the decrease in lattice spacing. Increasing the water thickness between the fuel and the inner stainless steel box results in a small increase in reactivity. The reactivity effect of the flux-trap water thickness, however, is more significant, and decreasing the flux-trap water thickness increases reactivity. Both of these effects have been evaluated for independent design tolerances.

The inner stainless steel box dimension, 8.650 $\pm$ 0.032 inches, defines the inner water thickness between the fuel and the inside box. For the tolerance limit, the uncertainty in reactivity is $\pm$ 0.0011 Δk as determined by differential CASMO-2E calculations, with k_∞ increasing as the inner stainless steel box dimension (and derivative lattice spacing) increases.

The design flux-trap water thickness is 1.120 $\pm$ 0.040 inches, which results in an uncertainty of $\pm$ 0.0043 Δk due to the tolerance in flux-trap water thickness, assuming the water thickness is simultaneously reduced on all four sides. Since the manufacturing tolerances on each of the four sides are statistically independent, then actual reactivity uncertainties would be less than $\pm$ 0.0043, although the more conservative value has been used in the criticality evaluation.

3.1.3.1.4 Boraflex Width Tolerance Variation

The reference storage cell design for Region 1 (Figure 3-2) uses a Boraflex blade width of 7.50 $\pm$ 0.0625 inches. A positive increment in reactivity occurs for a decrease in Boraflex absorber width. For a reduction in width of the maximum tolerance, 0.0625 inch, the calculated positive reactivity increment is +0.0017 Δk.

3.1.3.1.5 Stainless Steel Thickness Tolerances

The nominal stainless steel thickness in Region 1 is 0.080 $\pm$ 0.005 inch for the inner stainless steel box and 0.020 $\pm$ 0.003 inch for the Boraflex coverplate. The maximum positive reactivity effect of the expected stainless steel thickness tolerance variations, statistically combined, was calculated (CASMO-2E) to be + 0.0010 Δk.

3.1.3.1.6 Fuel Enrichment and Density Variation

The design maximum enrichment is 4.50 $\pm$ 0.05 wt% U-235. Calculations of the sensitivity to small enrichment variations by CASMO-2E yielded a coefficient of 0.0054 Δk per 0.1 wt% U-235 at the design enrichment. For a tolerance on U-235 enrichment of $\pm$ 0.05 in wt%, the uncertainty on k_∞ is $\pm$ 0.0027 Δk.

Calculations were also made with the UO_2 fuel density increased to the maximum expected value of 10.811 g/cm^3 (smeared density). For the reference design calculations, the uncertainty in reactivity is $\pm$ 0.0005 Δk over the maximum expected range of UO_2 densities.

3.1.3.1.7 Fuel Pin Pitch

Normally, the fuel pins in the lattice are arranged on a 0.577 inch lattice spacing. For the maximum expected tolerance of $\pm$ 0.0023 inch, the calculated uncertainty is $\pm$ 0.0024 Δk.

3.1.3.1.8 Eccentric Positioning of Fuel Assembly in Storage Rack

The Fuel Assembly is assumed to be normally located in the center of the storage rack cell. Calculations were also made with the fuel assemblies assumed to be the corner of the storage rack cell (four-assembly cluster at closest approach). These calculations indicated that the reactivity increases very slightly, as determined by differential PDQ07 calculations with diffusion coefficients* generated by NULIF and a blackness theory routine. This uncertainty is included in the evaluation of the highest possible reactivity of the Region 1 storage cells.

3.1.3.1.9 Summary of Region 1 Criticality Results

Table 3-1 demonstrates that the CASMO-2E calculated results for Region 1 storing fresh fuel at 4.50 w/o U-235 enrichment plus calculational bias and uncertainties exhibit a maximum k_∞ of 0.9409 which allows a margin of 0.0091 Δk below the limiting effective multiplication factor of 0.95.

3.1.3.2 Criticality Analysis for Region 2

3.1.3.2.1 Nominal Design Case

The principal method of analysis in Region 2 was the CASMO-2E code, using the restart option in CASMO to transfer fuel of a specified burnup into the storage rack configuration at a reference temperature of 4°C (maximum moderator density). Calculations were made for fuel of several different initial enrichments and, at each enrichment, a limiting k_∞ value was established which included an additional factor for uncertainty in the burnup analysis and for the axial burnup distribution. The restart CASMO-2E calculations (cold, clean, rack geometry) were then interpolated to define the burnup value yielding the limiting k_∞ value for each enrichment, as indicated in Table 3-4. These converged burnup values define the boundary of the acceptable domain shown in Figure 3-1.

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* This calculational approach was necessary since the reactivity effects are too small to be calculated by KENO, and CASMO-2E geometry is not readily amenable to eccentric positioning of a fuel assembly.

At a burnup of 36.5 Mwd/kgU, the sensitivity to burnup is calculated to be -0.0074 Δk per Mwd/kgU. During long-term storage, the k_∞ values of the Region 2 fuel rack will decrease continuously from decay of Pu-241, as indicated in Section 3.1.3.3.4.

Two independent calculational methods were used to provide additional confidence in the reference Region 2 criticality analyses. Fuel of 1.69% initial enrichment (approximately equivalent to the reference rack design for burned fuel) was analyzed by AMPX-KENO (27-group SCALE cross-section library) and by the CASMO-2E model used for the Region 2 rack analysis. For this case, the CASMO-2E k_∞ (0.9304) was within the statistical uncertainty of the bias-corrected value (0.9347 $\pm$ 0.0064) (95%/95%) obtained in the AMPX-KENO calculations. This agreement confirms the validity of the primary CASMO-2E calculations.

The second independent method of analysis used was the NULIF code for burnup analysis, and for generating diffusion theory constants (cold, clean) for the composition at 36.5 Mwd/kgU with fuel of 4.5% initial enrichment. These constants, together with blackness theory constants for the Boraflex absorber, were then used in a two-dimensional PDQ07 calculation for the storage rack configuration. The result of this calculation (k_∞ of 0.8959) was somewhat lower than the corresponding CASMO-2E calculation for the same conditions (k_∞ of 0.9114) and thus also tends to confirm the validity of the primary calculational method.

3.1.3.2.2 Boron Loading Variation

The Boraflex absorber sheets used in the Region 2 storage cells are nominally 0.031 inch thick with a B-10 areal density of 0.0097 g/cm^2. Independent manufacturing limits are $\pm$ 0.007 inch in thickness and $\pm$ 0.009 g/cm^3 in B-10 content. This assures that at any point where the minimum boron concentration (0.1158 g B-10/cm^3) and the minimum Boraflex thickness (0.024 inch) may coincide, the boron-10 areal density will not be less than 0.007 g/cm^2. Differential CASMO-2E calculations indicate that these tolerance limits result in an incremental reactivity uncertainty of $\pm$ 0.0036 Δk for boron content and $\pm$ 0.0111 Δk for Boraflex thickness.

3.1.3.2.3 Boraflex Width Tolerance

The reference storage cell design for Region 2 (Figure 3-3) uses a Boraflex absorber width of 7.25 $\pm$ 0.0625 inches. For a reduction in width of the maximum tolerance, the calculated positive reactivity increment is 0.0011 Δk.

3.1.3.2.4 Storage Cell Lattice Pitch Variations

The design storage cell lattice spacing between fuel assemblies in Region 2 is 8.86 $\pm$ 0.04 inches, corresponding to an uncertainty in reactivity of 0.0016 Δk.

3.1.3.2.5 Stainless Steel Thickness Tolerance

The nominal thickness of the stainless steel box wall is 0.080 inch with a tolerance limit of $\pm$ 0.005 inch, resulting in an uncertainty in reactivity of $\pm$ 0.0002 Δk.

3.1.3.2.6 Fuel Enrichment, Density and Pin Pitch Variation

Uncertainties in reactivity due to tolerances on fuel enrichment, UO_2 density, and pin pitch in Region 2 are assumed to be the same as those determined for Region 1.

3.1.3.2.7 Eccentric Positioning of Fuel Assembly in Storage Rack

The fuel assembly is assumed to be normally located in the center of the storage rack cell. Calculations were also made with the fuel assemblies assumed to be in the corner of the storage rack cell (four-assembly cluster at closest approach). These calculations indicated that the reactivity decreases very slightly, as determined by PDQ07 calculations with diffusion coefficients generated by NULIF and a blackness theory routine. The highest reactivity therefore corresponds to the reference design with the fuel assemblies positioned in the center of the storage cells.

3.1.3.3 Analytical Methodology

3.1.3.3.1 Reference Analytical Methods and Bias

The CASMO-2E computer code[1, 2, 3], a two-dimensional multigroup transport theory code for fuel assemblies, has been benchmarked and is used both as a primary method of analysis, and as a means of evaluating small reactivity increments associated with manufacturing tolerance. CASMO-2E benchmarking resulted in a calculational bias of 0.0013 ± 0.0018 (95%/95%).

In fuel rack analyses, for independent verification, criticality analyses of the high density spent fuel storage racks were also performed with the AMPX-KENO computer package[4, 5], using the 27-group SCALE* cross-section library[6] with the NITAWL subroutine for U-238 resonance shielding effects (Nordheim integral treatment). Benchmark calculations resulted in a bias of 0.0106 ± 0.0048 (95%/95%).

In the geometric model used in KENO, each fuel rod and its cladding were described explicitly. In Region 1 calculations, a reflecting boundary condition (zero neutron current) was used in the axial direction and at the centerline of the water gap between storage cells. These boundary conditions have the effect of creating an infinite array of storage cells in all directions. In Region 2, the zero current boundary condition was applied at the center of the Boraflex absorber sheets between storage cells. The AMPX-KENO Monte Carlo calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To minimize the statistical uncertainty of the KENO-calculated reactivity, a total of 50,000 neutron histories is normally accumulated for each calculation, in 100 generations of 500 neutrons each.

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* SCALE is an acronym for Standardized Computer Analysis for Licensing Evaluation, a standard cross-section set developed by ORNL for the USNRC.

CASMO-2E is also used for burnup calculations, with independent verification by EPRI-CELL and NULIF calculations. In tracking long-term (30-year) reactivity effects of spent fuel stored in Region 2 of the fuel storage rack, EPRI-CELL calculations indicate a continuous reduction in reactivity with time (after Xe decay) due primarily to Pu-241 decay and Am-241 growth.

A third independent method of criticality analysis, utilizing diffusion/blackness theory, was also used for additional confidence in results of the primary calculational methods, although no reliance for criticality safety is placed on the reactivity value from the diffusion/blackness theory technique. This technique, however, is used for auxiliary calculations of the small incremental reactivity effect of eccentric fuel positioning that would otherwise be lost in normal KENO statistical variations, or would be inconsistent with CASMO-2E geometry limitations.

Cross sections for the diffusion/blackness theory calculations were derived from the NULIF computer code(7), supplemented by a blackness theory routine that effectively imposes a transport theory boundary condition at the surface of the Boraflex neutron absorber. Two different spatial diffusion theory codes, PDQ07(8) in two dimensions and SNEID* in one dimension, were used to calculate reactivities.

3.1.3.3.2 Fuel Burnup Calculations

Fuel burnup calculations in the hot operating condition were performed primarily with the CASMO-2E code. However, to enhance the credibility of the burnup calculations, the CASMO-2E results were independently checked by calculations with the NULIF code(7) and with EPRI-CELL(9). Figure 3-4 compares results of these independent methods of burnup analysis under hot reactor operating conditions. The results agree with the CASMO calculation within 0.0054 Δk in the hot operating condition. An archive calculation with the CHEETAH-P code is also presented in Figure 3-4 for additional confidence. Similar comparisons were obtained in burnup calculations for other initial enrichments, as indicated in Figure 3-4.

In addition to depletion calculations under hot operating conditions, reactivity comparisons under conditions more representative of fuel to be stored in the racks (cold, xenon-free) are also significant in storage rack criticality analyses. Table 3-5 compares the cold, xenon-free reactivities calculated by CASMO-2E, EPRI-CELL, and diffusion/blackness theory. In the rack under cold conditions, the CASMO-2E calculations gave a slightly higher reactivity value for the Region 2 fuel storage cell, and the good agreement generally observed lends credibility to the calculations.

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* SNEID is a one-dimensional diffusion theory routine developed by Black & Veatch and verified by comparison with PDQ07 one-dimensional calculations.

No definitive method exists for determining the uncertainty in burnup-dependent reactivity calculations. All of the codes discussed above have been used to accurately follow reactivity loss rates in operating reactors. CASMO-2E has been extensively benchmarked[1, 2, 3, 10] against cold, clean, critical experiments (including plutonium-bearing fuel), Monte Carlo calculations, reactor operations, and heavy-element concentration in irradiated fuel. In particular, the analyses[10] of 11 critical experiments with plutonium-bearing fuel gave an average k_{eff} of 1.002 $\pm$ 0.011 (95%/95%), showing adequate treatment of the plutonium nuclides. In addition, Johansson[11] has obtained very good agreement in calculations of close-packed, high-plutonium-content, experimental configurations.

Since critical-experiment data with spent fuel is not available, it is necessary to assign an uncertainty in reactivity based on other considerations, supported by the close agreement between different calculational methods and the general industry experience in predicting reactivity loss rates in operating plants. Over a considerable portion of the burnup, the reactivity loss rate in PWRs is approximately 0.01 Δk for each Mwd/kgU burnup, becoming somewhat smaller at the higher burnups. By conservatively assuming an uncertainty in reactivity* of 0.0005 times the burnup in Mwd/kgU, a burnup-dependent uncertainty is defined that increases with increasing fuel burnup, as would be reasonably expected. This assumption provides an estimate of the burnup uncertainty that is more conservative and bounds estimates frequently employed in other fuel rack licensing applications (i.e., 5% of the total reactivity decrement). At the design basis burnup of 36.5 Mwd/kgU, the estimate of burnup uncertainty is 0.0183 Δk; Table 3-6 summarizes results of the burnup analyses and estimated uncertainties at other burnups. These uncertainties are appreciably larger, in general, than would be suggested by the industry experience in predicting reactivity loss rates and boron let-down curves over many cycles in operating plants. The increasing level of conservatism at the higher fuel burnups provides an adequate margin in the uncertainty estimate to accommodate the possible existence of a small positive reactivity increment from the axial distribution in burnup (see Section 3.1.3.3.3). In addition, although the burnup uncertainty may be either positive or negative, it is treated as an additive term rather than being combined statistically with other uncertainties. Thus, the allowance for uncertainty in burnup calculations is considered to be a conservative estimate, particularly in view of the substantial reactivity decrease with aged fuel, as discussed in Section 3.1.3.3.4.

* Only that portion of the uncertainty due to burnup. Other uncertainties are accounted for elsewhere.

3.1.3.3.3 Effect of Axial Burnup Distribution

Initially, fuel loaded into the reactor will burn with a slightly skewed cosine power distribution. As burnup progresses, the burnup distribution will tend to flatten, becoming more highly burned in the central regions than in the upper and lower ends. This effect may be clearly seen in the curves compiled in Reference 12. At high burnup, the more reactive fuel near the ends of the fuel assembly (less than average burned) occurs in regions of lower reactivity worth due to neutron leakage. Consequently, it is expected that distributed-burnup fuel assemblies would exhibit a slightly lower reactivity than that calculated for the average burnup. As burnup progresses, the distribution, to some extent, tends to be self-regulating as controlled by the axial power distribution, precluding the existence of large regions of significantly reduced burnup.

A number of one-dimensional diffusion theory analyses have been made based upon calculated and measured axial burnup distributions. These analyses confirm the minor, and generally negative, reactivity effect of the axially distributed burnup. The trends observed, however, suggest the possibility of a small positive reactivity effect at the high burnup values (estimated to be as much as 0.006 Δk at 36.5 Mwd/kgU); but the uncertainty in k_∞ due to burnup, assigned at the higher burnups (Section 3.1.3.3.2), is adequately conservative to encompass the potential for a small positive reactivity effect of axial burnup distributions. Furthermore, reactivity significantly decreases with time in storage (Section 3.1.3.3.4), and, in addition, there is a further margin in reactivity ($>0.006\ \Delta k$) since the maximum calculated value (0.9435) is below the limiting k_{eff} value (0.95). These factors would accommodate any reasonable reactivity effects that might be larger than expected.

3.1.3.3.4 Long-term Decay

Since the fuel racks in Region 2 are intended to contain spent fuel for long periods of time, calculations were made using EPRI-CELL (which incorporates the CINDER code) to follow the long-term changes in reactivity of spent fuel over a 30-year period. CINDER tracks the decay and burnup dependence of some 179 fission products. Early in the decay period, xenon grows from iodine decay (reducing reactivity) and subsequently decays, with the reactivity reaching a maximum at 100-200 hours. The decay of Pu-241 (13-year half-life) and growth of Am-241 substantially reduce reactivity during long term storage, as indicated in Table 3-7.

The reference design criticality calculations do not take credit for this long-term reduction in reactivity, other than to indicate an increasing subcriticality margin in Region 2 of the spent fuel storage pool.

3.1.4 Rack Modification

The design basis fuel assembly, illustrated in Figure 3-2, is a 14 x 14 array of fuel rods with 20 rods replaced by 5 control rod guide tubes. Table 3-8 summarizes the design specifications and the expected range of significant variations. Independent calculations, with other potential fuel assembly specifications, confirmed that the 14 x 14 CE design exhibited the highest reactivity and was therefore used as the design basis.

3.1.4.1 Region 1 Storage Cells

The nominal spent fuel storage cell used for the criticality analyses of Region 1 storage cells is shown in Figure 3-2. The rack is composed of Boraflex absorber material sandwiched between an 8.65-inch I.D., 0.080-inch thick inner stainless steel box, and a 0.020-inch outer stainless steel coverplate. The fuel assemblies are centrally located in each storage cell on a nominal lattice spacing of 10.120 $\pm$ 0.05 inches. Stainless steel gap channels connect one storage cell box to another in a rigid structure and define an outer water space between boxes. This outer water space constitutes a flux-trap between the two Boraflex absorber sheets that are essentially opaque (black) to thermal neutrons. The Boraflex absorber has a thickness of 0.075 $\pm$ 0.007 inch and a nominal B-10 areal density of 0.0238 g/cm^2.

3.1.4.2 Region 2 Storage Cells

Region 2 storage cells were designed for fuel of 4.5 wt% U-235 initial enrichment burned to 36.5 Mwd/kgU. In this region, the storage cells are composed of a single Boraflex absorber sandwiched between the 0.080-inch stainless steel walls of adjacent storage cells. These cells, shown in Figure 3-3, are located on a lattice spacing of 8.86 $\pm$ 0.040 inches. The Boraflex absorber has a thickness of 0.031 $\pm$ 0.007 inch and a nominal B-10 areal density of 0.0097 g/cm^2.

3.1.5 <u>Acceptance Criteria for Criticality</u>

Criticality is precluded by spacing of the fuel assemblies, which ensures that a subcritical array of k_{eff} less than or equal to 0.95 is maintained, assuming unborated pool water. The pool, however, will always contain boric acid at the refueling concentration of 1720 ppm whenever there is irradiated fuel in the pool.

The neutron multiplication factor in spent fuel pools shall be less than or equal to 0.95, including all uncertainties, under all conditions. Calculated maximum reactivity uncertainties for fuel stored in the racks are presented in Table 3-1.

Methods of initial and long-term verification of poison material stability and mechanical integrity are discussed in Section 4.8.

3.2 DECAY HEAT CALCULATIONS FOR THE SPENT FUEL POOL (BULK)

3.2.1 <u>Spent Fuel Pool Cooling System Design</u>

For normal refueling discharge conditions, one fuel pool pump and the fuel pool heat exchanger are in service. During abnormal refueling conditions, such as full core discharge, two fuel pool pumps and the heat exchanger are in service. The system is manually controlled and the operation monitored locally, except as follows. A pressure switch on the fuel pool pump discharge header annunciates low header pressure in the control room. The fuel pool high temperature alarm and low level alarms are annunciated in the control room. In the event the fuel pool pump breakers are opened, an alarm is annunciated in the control room. The component cooling water flow to the fuel pool heat exchanger is initially adjusted to the required flow. Further adjustments of the component cooling water are not required. The component cooling water discharge line has a flow indicator. High and low component cooling water flow alarms are annunciated in the control room.

The clarity and purity of the water in the fuel pool is maintained by the purification portion of the fuel pool system. The purification loop consists of the fuel pool purification pump, ion exchanger, filter, strainers and surface skimmers. Most of the purification flow is drawn through the surface skimmers to remove surface debris. A basket strainer is provided in the purification line to the pump suction to remove any relatively large particulate matter. The fuel pool water is circulated by the pump through a filter, which removes particulates larger than 5 micron size, and through an ion exchanger to remove ionic material. Connections are provided for purification of the refueling water tank and refueling water cavity. Fuel pool water chemistry is given in FSAR Table 9.1-2.

The fuel pool piping is arranged so that the pool cannot be inadvertently drained to uncover the fuel in the event of a supply or discharge pipe rupture. All fuel pool piping is arranged to prevent gravity draining the fuel pool. To prevent siphoning of the fuel pool, the fuel pool discharge and purification suction lines have 1/2" and 1/4" holes respectively 1 foot below the normal water level.

The only means of draining the pool below these siphon breaker holes is through an open line in the cooling loop while operating the pool cooling pumps. In such an event the fuel pool water level can be reduced by only 6 feet since the pump suction connection enters near the top of the pool. The remaining water in the Spent Fuel Pool will provide adequate shielding and heat removal capabilities at this point. The temperature and level alarms would warn the operator of such an event.

3.2.2 Decay Heat Analyses

3.2.2.1 Basis

The St. Lucie Plant Unit 1 reactor is rated at 2700 megawatts thermal (MWt). The core contains 217 fuel assemblies. Thus, the average operating power per fuel assembly, P_o, is 12.44 MW. The fuel discharge can be made in one of the following two modes:

- Normal refueling discharge
- Full core discharge

Tables 3-9 through 3-11 give the parameters for bulk and local pool temperature analyses.

3.2.2.2 Model Description

NUREG-0800 Branch Technical Position ASB 9-2, "Residual Decay Energy For Light Water Reactors For Long Term Cooling"(15) is utilized to compute the heat dissipation requirements in the pool.

With the long term uncertainty factor, K, as specified in SRP 9.1.3 (15), the operating power, P_o, is taken equal to the rated power, even though the reactor may be operating at less than its rated power during much of the exposure period for the batch of fuel assemblies. The computations and results reported here are based on the discharge taking place when the inventory of fuel in the pool will be at its maximum resulting in an upper bound on the decay heat rate.

Having determined the heat dissipation rate, the next task is to evaluate the time-dependent temperature of the pool water. Table 3-9 identifies the loading cases examined. This is a conservative representation of actual and future expected discharges such as those presented in Table 5-1. BULKTEM treats the generalized pool cooling problem shown in Figure 3-5.

A number of simplifying assumptions are made which render the analysis conservative, including:

- The heat exchanger is assumed to have maximum fouling. Thus, the temperature effectiveness, P, for the heat exchanger utilized in the analysis is the lowest postulated value calculated from heat exchanger technical data sheets.

- No credit is taken for the improvement in the film coefficients of the heat exchanger as the operating temperature rises due to monotonic reduction in the water kinematic viscosity with temperature rise. Thus, the film coefficient used in the computations are lower bounds.

- No credit is taken for heat loss by evaporation of the pool water.

- No credit is taken for heat loss to pool walls and pool floor slab.

The basic energy conservation relationship for the pool heat exchanger system yields:

$$C_t \frac{dt}{d\tau} = Q_1 - Q_2$$

where:

C_t	=	Thermal capacity of stored water in the pool
t	=	Temperature of pool water at time, τ
Q_1	=	Heat generation rate due to stored fuel assemblies in the pool
Q_2	=	Heat removed in the fuel pool heat exchanger

This equation is solved as an initial value problem by noting that the cooler heat removal rate must equal the heat generation rate from previously discharged assemblies. Hence,

$$W_{cool} \, P \, (T_{in} - t_{cool}) = PCONS$$

where:

PCONS:	Heat generation rate from previously stored assemblies
W_{cool}:	Coolant thermal flow rate
P:	Temperature effectiveness of the fuel pool cooler
T_{in}:	Coincident pool water temperature (initial value before beginning of discharge)
t_{cool}:	Coolant inlet temperature

The above equation yields

$$T_{in} = \frac{PCONS}{W_{cool}\ P} + t_{cool}$$

The value of T_{in} computed from the above formula is the initial value of the pool water temperature (at the start of fuel discharge).

BULKTEM automates the solution of the above equation using the theory presented in Reference 16. Tabulated results are presented in the next sub-section.

3.2.2.3 Bulk Pool Temperature Results

Table 3-12 gives the total dimensionless power generation ratio of all fuel assembly batches previously stored in the pool consisting of a total of 18 batches. The first column in Table 3-12 gives the batch number, and the last column gives the dimensionless power, defined as the heat generation rate of the batch divided by the nominal operating power of one fuel assembly. It is noted from Table 3-12 that the cumulative power is 0.14 times the operating power of one fuel assembly. Tables 3-13/3-14 and 3-16/3-17 give the bulk temperature vs. time data.

The following key output data is gleaned from these tables:

Maximum pool bulk temperature:

Normal discharge:	133.3°F	Table 3-14
Full core discharge:	150.8°F	Table 3-17

Tables 3-15 and 3-18 give time-to-boil data.

Time-to-boil (if coolant flow is lost upon completion of discharge and when the bulk pool temperature is maximum):

Normal discharge condition:	13.43 hours	Table 3-15
Full core discharge condition:	5.04 hours	Table 3-18

3.2.2.4 Spent Fuel Pool Cooling System Summary

The spent fuel decay heat calculations were performed in accordance with the method provided in NRC Branch Technical Position ASB 9-2, Residual Decay Energy for Light-Water Reactors for Long-Term Cooling[15].

The existing spent fuel pool cooling system is considered to be adequate. The spent fuel pool is designed to withstand stresses associated with a steady-state water temperature of 217° F. As shown in Table 3-17 the pool peak transient water temperature after full core discharge is less than 151° F.

In the event of a complete loss of cooling capability, there is sufficient time to provide an alternate means for cooling.

The total increase in heat load rejected to the environment through the cooling systems due to the increased spent fuel storage over the current heat load rejected is 1.7×10^6 Btu/hour. This represents an increase of approximately 0.03 percent of the total heat rejected to the environment. The increase in heat rejected will have negligible impact on the environment.

The increase in heat load does not alter in any way the existing facility design bases. Thus, the heat load increase is acceptable. This decay heat analysis is also bounding for the temporary fuel storage configuration (see Section 4.7.4) that will be utilized during rack installation.

3.2.2.4.1 Safety Evaluation

The calculations for the amount of thermal energy that may have to be removed by the spent fuel pool cooling system are made in accordance with Branch Technical Position ASB 9-2 (Reference 15). The resulting bulk spent fuel pool temperatures are acceptable.

3.2.3 Spent Fuel Pool Makeup

There are several sources of fresh water on the site that are available to the fuel handling building; namely, refueling water storage tank, city water storage tank via the fire main, city water storage tanks via the portable fire pump, and primary water tank. The concurrent loss of these sources and the fuel pool cooling system is remote. Due to the fuel pool's boil-off period, there is sufficient time to obtain makeup. It should be noted that a seismic Category I backup salt water supply is available from the intake cooling water intertie. A standpipe on the fuel handling building is provided from grade to the operating deck elevation and hose connections are provided at both ends of the standpipe. Thus, via fire hose, the fuel pool makeup can be readily supplied by the intake cooling water pumps. The head provided by these pumps is sufficient to provide the required fuel pool make up. The structural and leaktight integrity of the fuel pool will not be compromised by continuous fuel pool temperatures of up to 217° F. The results of the bulk decay heat analyses indicate that these temperatures are not exceeded. The intake cooling water system connection via the hose connections can provide 150 gpm of makeup. See FSAR Subsection 9.1.3.4.

[illegible]

3.3 THERMAL-HYDRAULIC ANALYSES FOR THE SPENT FUEL POOL (LOCALIZED)

The purpose of the thermal-hydraulic analyses is to determine the maximum fuel clad temperatures which may occur as a result of using the new high density spent fuel racks in the St Lucie Unit 1 spent fuel pool.

3.3.1 Bases

In order to determine an upper bound on the maximum fuel cladding temperature, a series of conservative assumptions are made. The most important assumptions are listed below:

- As stated above, the fuel pool will contain spent fuel with varying time-after-shutdown (τ_s). Since the heat emission falls off rapidly with increasing τ_s, it is obviously conservative to assume that all fuel assemblies are fresh and they all have had the maximum postulated years of operating time in the reactor. The heat emission rate of each fuel assembly is assumed to be equal and maximum.

- As shown in Figure 2-1, the modules occupy an irregular floor space in the pool. For the hydrothermal analysis, a circle circumscribing the actual rack floor space is drawn (Figure 3-6). It is further assumed that the cylinder with this circle as its base is packed with fuel assemblies at the nominal layout pitch.

- The actual downcomer space around the rack module group varies, as shown in Figure 2-1. The nominal downcomer gap available in the pool is assumed to be the total gap available around the idealized cylindrical rack; thus, the maximum resistance to downward flow is incorporated into the analysis (Figure 3-7).

- No downcomer flow is assumed to exist between the rack modules.

3.3.2 Model Description

Using the bases described above, a conservative idealized model for the rack assemblage is obtained. The water flow is axisymmetric about the vertical axis of the circular rack assemblage and, thus, the flow is two-dimensional (axisymmetric three-dimensional). Figure 3-7 shows a typical "flow chimney" rendering of the thermal hydraulics model. The governing equation to characterize the flow field in the pool is an integral equation that can be solved for the lower plenum velocity field (in the radial direction) and axial velocity (in-cell velocity field), by using the method of collocation. It should be added that the hydrodynamic loss coefficients which enter into the formulation of the integral equation are also taken from well-recognized sources[17] and wherever discrepancies in reported values exist, the conservative values are consistently used. Reference 18 gives the details of mathematical analysis used in this solution process.

After the axial velocity field is evaluated, the fuel assembly cladding temperature can be calculated. The knowledge of the overall flow field enables pinpointing of the storage location with the minimum axial flow (i.e., maximum water outlet temperatures). This is called the most "choked" location. In order to find an upper bound on the temperature in a typical cell, it is assumed that it is located at the most choked location. Knowing the global plenum velocity field, the revised axial flow through this choked cell can be calculated by solving the Bernoulli equation for the flow circuit through this cell. Thus, an absolute upper bound on the water exit temperature and maximum fuel cladding temperature is obtained. In view of the aforementioned assumptions, the temperatures calculated in this manner overestimate the temperature rise that will actually occur in the pool. THERPOOL, based on the theory of Reference 18, automates this calculation.

Finally, the maximum specific power of a fuel array q_A can be given by:

$$q_A = q F_{xy}$$

where:

q = average fuel assembly specific power
F_{xy} = radial peaking factor

The data on radial and axial peaking factors may be found in Table 3-10.

The maximum temperature rise of pool water in the most disadvantageously placed fuel assembly is computed for all loading cases. Table 3-19, third column, gives the outputs from THERPOOL in tabular form.

3.3.3. Cladding Temperature

Having determined the maximum local water temperature in the pool, it is now possible to determine the maximum fuel cladding temperature. A fuel rod can produce F_{Tot} times the average heat emission rate over a small length, where F_{Tot} is the total peaking factor. The axial heat dissipation in a rod is known to reach a maximum in the central region, and taper off at its two extremities. For added conservatism, it is assumed that the peak heat emission occurs at the top where the local water temperature also reaches its maximum. Futhermore, no credit is taken for axial conduction of heat along the rod. The highly conservative model thus constructed leads to simple algebraic equations which directly give the maximum local cladding temperature, t_c.

Table 3-19, fourth column, summarizes the key output data. It is found that the maximum value of the local water temperature is well below the nucleate boiling condition value. The incremental cladding temperature is too small to produce significant thermal stresses.

3.4 POTENTIAL FUEL AND RACK HANDLING ACCIDENTS

The method for moving the racks into and out of the spent fuel pool is briefly discussed in Section 4.7.4.2. The methods utilized ensure that postulated accidents do not result in a loss of cooling to either the spent fuel pool or the reactor, or result in a k_{eff} in the spent fuel pool exceeding 0.95.

3.4.1 Rack Module Mishandling

The potential for mishandling of rack modules during the rerack operation has been evaluated. At no time will the cask handling crane or the temporary construction crane carry a rack module directly over a rack containing spent fuel. The procedures and administrative controls governing the rerack operation will ensure the safe handling of rack modules. Both the temporary construction crane and the cask handling crane meet the design and operational requirements of Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"[19].

In the unlikely event that a rack should strike the side of another rack module containing fuel assemblies, the consequences of this postulated accident would be bounded by the cask drop evaluations described in Section 5.3.1.2.

3.4.2 Temporary Construction Crane Drop

During the rerack operation, a temporary construction crane will be installed in the Fuel Handling Building. This installation will be performed using lift rigs which meet the design and operational requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The consequences of a postulated accident during this installation are bounded by the cask drop evaluations described in Section 5.3.1.2.

3.4.3 Loss of Pool Cooling (Storage Rack Drop)

During the re-racking operation, it will be necessary to raise and maneuver the old racks out of the spent fuel pool in order to install the new spent fuel racks (See Section 4.7.4). The handling of these heavy loads will be accomplished by the use of a temporary construction crane and the cask handling crane. Both of these cranes meet the design and operational requirements of Section 5.1.1 of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

The consequences of dropping a rack in the Spent Fuel Pool were determined by reviewing the analysis in FSAR Subsection 9.1.4 for dropping of the spent fuel cask. The results of this cask drop analysis demonstrated that the pool floor would remain elastic during impact and that cracks would not develop. This cask weighs substantially more than a single rack assembly and has a smaller cross sectional area for load distribution. Therefore, the rack drop scenario is bounded by the previous analysis for a cask drop scenario, and loss of spent fuel cooling from loss of pool water inventory will not occur as a result of a rack drop.

3.5 TECHNICAL SPECIFICATION CHANGES

This proposed amendment permits replacement of the spent fuel pool racks to ensure that sufficient capacity exists for storage of spent fuel at St. Lucie Unit 1. The new racks increase the available storage to 1706 spent fuel assemblies and is expected to provide adequate storage space until the year 2009.

The proposed Technical Specification changes are described below:

1. Specification 3/4.9.14 Bases is revised to reflect the assumptions used in calculations of doses based on the Decay Times.

2. Specification 5.6.1.a.1 is revised to correspond to the Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors (NUREG-0212 Rev 2).

3. Specification 5.6.1.a.2 is revised to show the nominal center-to-center distance for the high capacity spent fuel storage racks.

4. Specification 5.6.1.a.3 is edited to discuss the boron concentration only.

5. Specification 5.6.1.a.4 is created to indicate the presence of Boraflex in the cells.

6. Specification 5.6.1.b and accompanying Figure 5.6-1 are created to define the fuel enrichment/burnup limits for storage in each region of the high capacity spent fuel storage racks.

7. Specification 5.6.1c is editorially changed from "b" to "c".

8. Specification 5.6.3 is changed to show the capacity of the high-capacity spent fuel storage racks.

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TABLE 3-1

SUMMARY OF CRITICALITY SAFETY ANALYSES

	Region 1	Region 2
Minimum acceptable burnup @ 4.5% initial enrichment	0	36.5 Mwd/kgU
Temperature assumed for analysis	4°C	4°C
Reference k_∞ (nominal)	0.9313	0.9114
Calculational bias	0.0013	0.0013
Uncertainties		
Bias	±0.0018	±0.0018
B-10 concentration	±0.0021	±0.0036
Boraflex thickness	±0.0044	±0.0111
Boraflex width	±0.0017	±0.0011
Inner box dimension	±0.0011	±0.0016
Water gap thickness	±0.0043	N/A
SS thickness	±0.0010	±0.0002
Fuel enrichment	±0.0027	±0.0027
Fuel density	±0.0005	±0.0005
Fuel element pitch	±0.0024	±0.0024
Statistical combination (1)	±0.0080	±0.0125
Eccentric assembly position	±0.0003	negative
Allowance for burnup uncertainty	N/A	+0.0183
Total	0.9329 ± 0.0080	0.9310 ± 0.0125
Maximum reactivity	0.9409	0.9435
(with 1720 ppm soluble boron)	(0.767)	(0.760)

(1) Square root of sum of squares.

N/A - Not Applicable

TABLE 3-2

MINIMUM BURNUP VALUES

Initial Enrichment, %	Minimum Burnup, Mwd/kgU
1.63	0
1.75	2.30
2.00	6.00
2.25	9.70
2.50	12.90
2.75	16.10
3.00	19.15
3.25	22.20
3.50	25.15
3.75	28.10
4.00	30.90
4.25	33.70
4.50	36.50

TABLE 3-3

REACTIVITY EFFECTS OF ABNORMAL AND ACCIDENT CONDITIONS

Accident/Abnormal Conditions	Reactivity Effect
Temperature increase	Negative in both regions
Void (boiling)	Negative in both regions
Assembly dropped on top of rack	Negligible
Lateral rack module movement	Negligible
Misplacement of a fuel assembly	Positive

TABLE 3-4

FUEL BURNUP VALUES FOR REQUIRED REACTIVITIES (k_∞)
WITH FUEL OF VARIOUS INITIAL ENRICHMENTS
(Reference k_∞ = 0.9297)

Initial Enrichment	Uncertainty[1] in Burnup, Δk	Design Limit k_∞	Calculated Burnup limit Mwd/kgU
1.6	0	0.9297	0
2.0	0.0030	0.9267	5.99
2.5	0.0064	0.9233	12.88
3.0	0.0096	0.9201	19.13
3.5	0.0126	0.9171	25.15
4.0	0.0154	0.9143	30.86
4.5	0.0183	0.9115	36.50

(1) See Subsection 3.1.3.3.2

TABLE 3-5

COMPARISON OF COLD, CLEAN REACTIVITIES CALCULATED
AT 36.5 Mwd/kgU BURNUP AND 4.5% ENRICHMENT

Calculational Method	k_∞ Xe-free, 4°C Infinite Array of Fuel Assemblies(1) in Reactor Spacing	k_∞ Xe-free, 4°C Assemblies in Region 2 Cell
CASMO-2E	1.1212	0.9114
DIFFUSION/BLACKNESS THEORY	1.1306	0.8972
EPRI-CELL	1.1281(2)	-

(1) Cold, clean condition in contrast to hot operating conditions of Figure 3-4.

(2) EPRI-CELL k_∞ at maximum value during long-term (30-year) storage.

TABLE 3-6

ESTIMATED UNCERTAINTIES IN REACTIVITY DUE TO
FUEL DEPLETION EFFECTS

Initial Enrichment	Design Burnup Mwd/kgU	0.0005 Times Burnup, Δk	Design k_∞	Reactivity Loss, $\Delta k^{(1)}$
1.6	0	0	0.9297	0
2.0	5.99	0.0030	0.9267	0.0579
2.5	12.88	0.0064	0.9233	0.1284
3.0	19.13	0.0096	0.9201	0.1828
3.5	25.15	0.0126	0.9171	0.2262
4.0	30.86	0.0154	0.9143	0.2620
4.5	36.50	0.0183	0.9115	0.2924

(1) Total reactivity decrease, calculated for the cold, Xe-free condition in the fuel storage rack, from the beginning-of-life to the design burnup.

TABLE 3-7

LONG-TERM CHANGES IN REACTIVITY IN STORAGE RACK

Storage Time, years	Δk from Shutdown (Xenon-free) at 4.5% E and 36.5 Mwd/kgU
0.5	-0.0047
1.0	-0.0088
10.0	-0.0470
20.0	-0.0673
30.0	-0.0788

TABLE 3-8

DESIGN BASIS (LIMITING)
FUEL ASSEMBLY SPECIFICATIONS
(CE 14 x 14)

Fuel Rod Data	
Cladding outside diameter, in.	0.440
Cladding thickness, in.	0.028
Cladding material	Zircaloy-4
Pellet diameter, in.	0.377
UO_2 stack density, g/cm^3	10.281 ± 0.031
Enrichment, wt% U-235	4.5 ± 0.05
Fuel Assembly Data	
Maximum number of fuel rods	176 (14 x 14 array)
Fuel rod pitch, in.	0.577 ± 0.0023
Control rod guide tube	
Number	5
Outside diameter, in.	1.115
Inside diameter, in.	1.035
Material	Zircaloy-4
U-235 Loading	
grams/axial cm of assembly	51.7 ± 0.7

TABLE 3-9

THERMAL/HYDRAULIC CASES TREATED*

1. Normal Batch Discharge:

   - Irradiation time: 54 months (1.42×10^8 secs)

   - Addition of the most recent batch : 150 hours after shutdown

   - Batch size: 80 assemblies

2. Full Core Discharge

   - Irradiation time:

Irradiation time:	73 assemblies	90 days
	72 assemblies	21 months
	72 assemblies	39 months

   - Fuel transfer begins 7 days after shutdown.

---

* The pool has total storage capacity of 1706 storage cells. It is conservatively assumed that 18 batches of 80 assemblies have been previously discharged at 18 month intervals. Each assembly in these previous discharges has had 54 months of exposure at full power (12.44 MWt).

TABLE 3-10

PEAKING FACTOR DATA

Fuel	Maximum Radial Peaking Factor	Maximum Axial Peaking Factor
St. Lucie Unit 1 CE 14 x 14 and Exxon 14 x 14	1.67	1.32
St Lucie Unit 2, CE 16 x 16	1.75	1.35

TABLE 3-11

ESSENTIAL HEAT TRANSFER DATA FOR THE FUEL POOL HEAT EXCHANGER

Number of heat exchangers:	one
Coolant flow rate:	3560 gpm
Temperature effectiveness:	0.36 (two pumps)* 0.263 (one pump)
Heat transfer surface area:	4380 sq. ft.
Overall heat transfer coefficient (fouled) (two pumps):	260 Btu/sq.ft.-hr-°F

---

* Temperature efficiency of the heat exchanger is calculated in the following manner, using the information provided in the FSAR:

$$P = \frac{\text{Cooling water outlet - inlet}}{\text{Pool water inlet - cooling water inlet}}$$

$$= \frac{118-100}{150-100}$$

$$= .36$$

TABLE 3-12

POWER GENERATION RATIO
PREVIOUSLY DISCHARGED BATCHES

Batch No.	Batch Size	Time After Shut Down in Days	Reactor Exposure Time in Days	Non Dimensional Power Gen. Ratio
1	80	9719.9	1643.5	.00487
2	80	9179.9	1643.5	.00505
3	80	8639.9	1643.5	.00523
4	80	8099.9	1643.5	.00542
5	80	7559.9	1643.5	.00562
6	80	7019.9	1643.5	.00582
7	80	6479.9	1643.5	.00603
8	80	5939.9	1643.5	.00624
9	80	5399.9	1643.5	.00647
10	80	4859.9	1643.5	.00670
11	80	4319.9	1643.5	.00694
12	80	3779.9	1643.5	.00720
13	80	3239.9	1643.5	.00746
14	80	2699.9	1643.5	.00776
15	80	2159.9	1643.5	.00815
16	80	1619.9	1643.5	.00888
17	80	1079.9	1643.5	.01097
18	80	540.0	1643.5	.01893

CUMULATIVE DIMENSIONLESS POWER = 1.3374E - 01

TABLE 3-13

BULK POOL TEMPERATURE VS. TIME
DURING NORMAL REFUELING DISCHARGE

Time (Hrs.)	Bulk Pool Temp. (^{o}F)	Heat Generation Rate (Btu/hr)
150.00*	106.0	.5689E + 07
151.00	108.8	.1643E + 08

* This table contains only two lines of output data. This is due to the fact that the discharge is assumed to take place instantaneously, simulated by one hour in this computer run.

TABLE 3-14

POOL BULK TEMPERATURE VS. TIME
SUBSEQUENT TO COMPLETION OF NORMAL REFUELING DISCHARGE

Time (Hrs.)	Bulk Pool Temp. (°F)	Heat Generation Rate (Btu/hr)
151.00	108.8	.1642E + 08
161.00	130.0	.1613E + 08
171.00	133.2	.1588E + 08
181.00	133.3	.1565E + 08
191.00	133.0	.1544E + 08
201.00	132.6	.1525E + 08
211.00	132.2	.1507E + 08
221.00	131.8	.1490E + 08
231.00	131.5	.1475E + 08
241.00	131.1	.1461E + 08
251.00	130.8	.1447E + 08
261.00	130.6	.1435E + 08
271.00	130.3	.1423E + 08
281.00	130.1	.1411E + 08
291.00	129.8	.1401E + 08
301.00	129.6	.1390E + 08
311.00	129.4	.1380E + 08
321.00	129.2	.1371E + 08
331.00	129.0	.1362E + 08
341.00	128.8	.1353E + 08
351.00	128.6	.1344E + 08
361.00	128.4	.1336E + 08
371.00	128.3	.1328E + 08
381.00	128.1	.1320E + 08
391.00	127.9	.1313E + 08

TABLE 3-15

LOSS OF COOLING AFTER COMPLETION
OF NORMAL REFUELING DISCHARGE

Case	Time to Boil (hrs)	Rate of Evaporation (lbm/hr)	Rate of Level Change (inch/hr)
When heat generation is maximum	16.79	16933.0	2.67
When the bulk pool temperature is maximum	13.43	16294.0	2.57

TABLE 3-16

BULK POOL TEMPERATURE VS TIME
DURING FULL CORE DISCHARGE

Time (Hrs.)	Bulk Pool Temp. (°F)	Heat Generation Rate (Btu/hr)
168.00*	113.6	.8690E + 07
169.00	117.8	.3371E + 08

* This table contains only two lines of output data. This is due to the fact that the discharge is assumed to take place instantaneously, simulated by one hour in this computer run.

TABLE 3-17

POOL BULK TEMPERATURE VS TIME SUBSEQUENT
TO COMPLETION OF FULL CORE DISCHARGE

Time (Hrs.)	Bulk Pool Temp. (°F)	Heat Generation Rate (Btu/hr)
169.00	117.8	.3370E + 08
179.00	148.8	.3307E + 08
189.00	150.8	.3249E + 08
199.00	150.2	.3197E + 08
209.00	149.4	.3149E + 08
219.00	148.7	.3104E + 08
229.00	148.1	.3062E + 08
239.00	147.4	.3024E + 08
249.00	146.9	.2987E + 08
259.00	146.3	.2953E + 08
269.00	145.8	.2921E + 08
279.00	145.3	.2991E + 08
289.00	144.8	.2862E + 08
299.00	144.4	.2834E + 08
309.00	144.0	.2807E + 08
319.00	143.6	.2782E + 08
329.00	143.2	.2758E + 08
339.00	142.8	.2734E + 08
349.00	142.5	.2712E + 08
359.00	142.1	.2690E + 08
369.00	141.8	.2668E + 08
379.00	141.5	.2648E + 08
389.00	141.1	.2628E + 08
399.00	140.8	.2608e + 08
409.00	140.5	.2589E + 08

TABLE 3-18

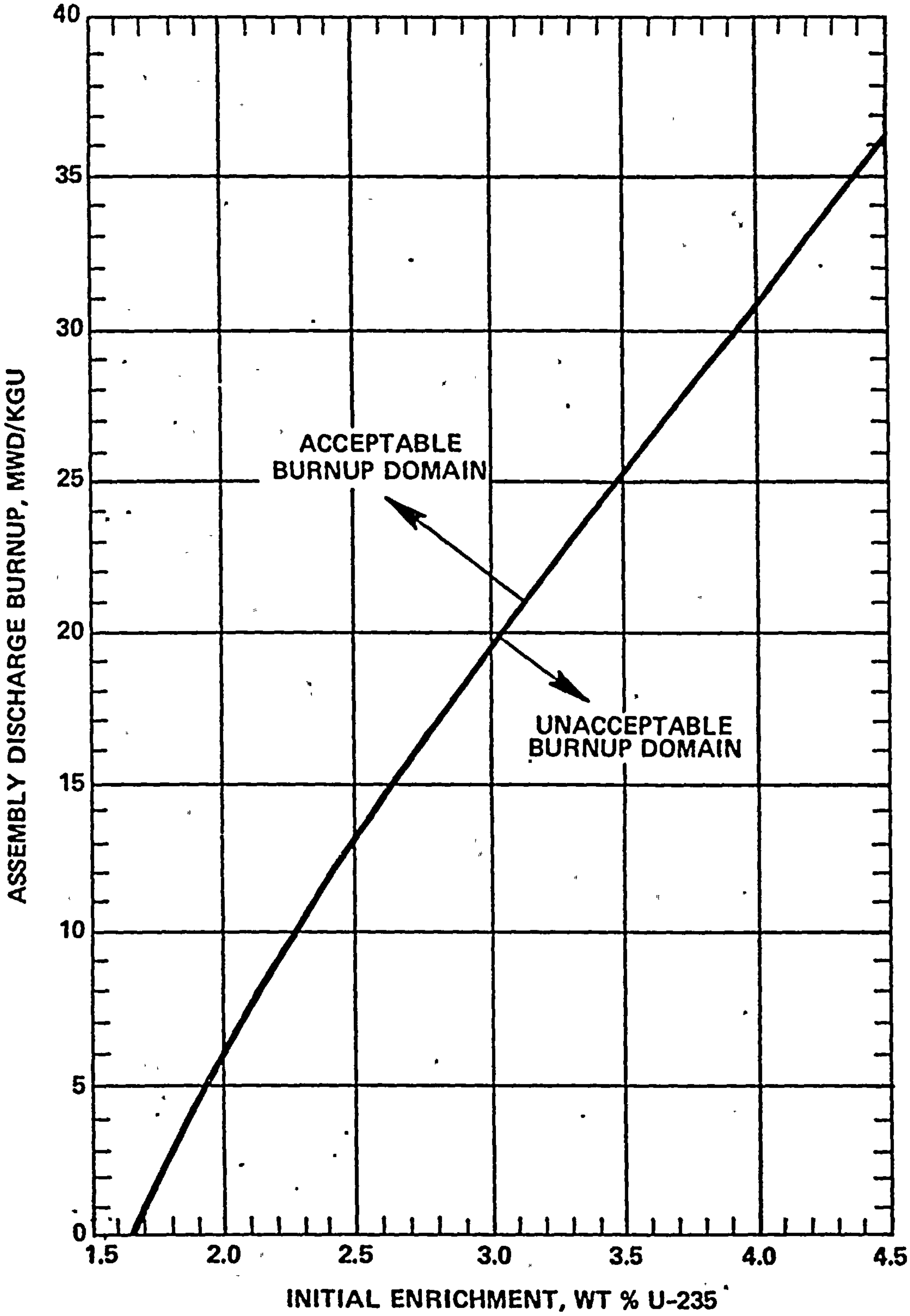
LOSS OF COOLING AFTER COMPLETION
OF FULL CORE DISCHARGE

Case	Time to Boil (hrs)	Rate of Evaporation (lbm/hr)	Rate of Level Change (inch/hr)
When heat generation is maximum	7.47	34742.2	5.47
When the bulk pool temperature is maximum	5.04	33660.0	5.3

TABLE 3-19

LOCAL AND CLADDING
TEMPERATURE DATA

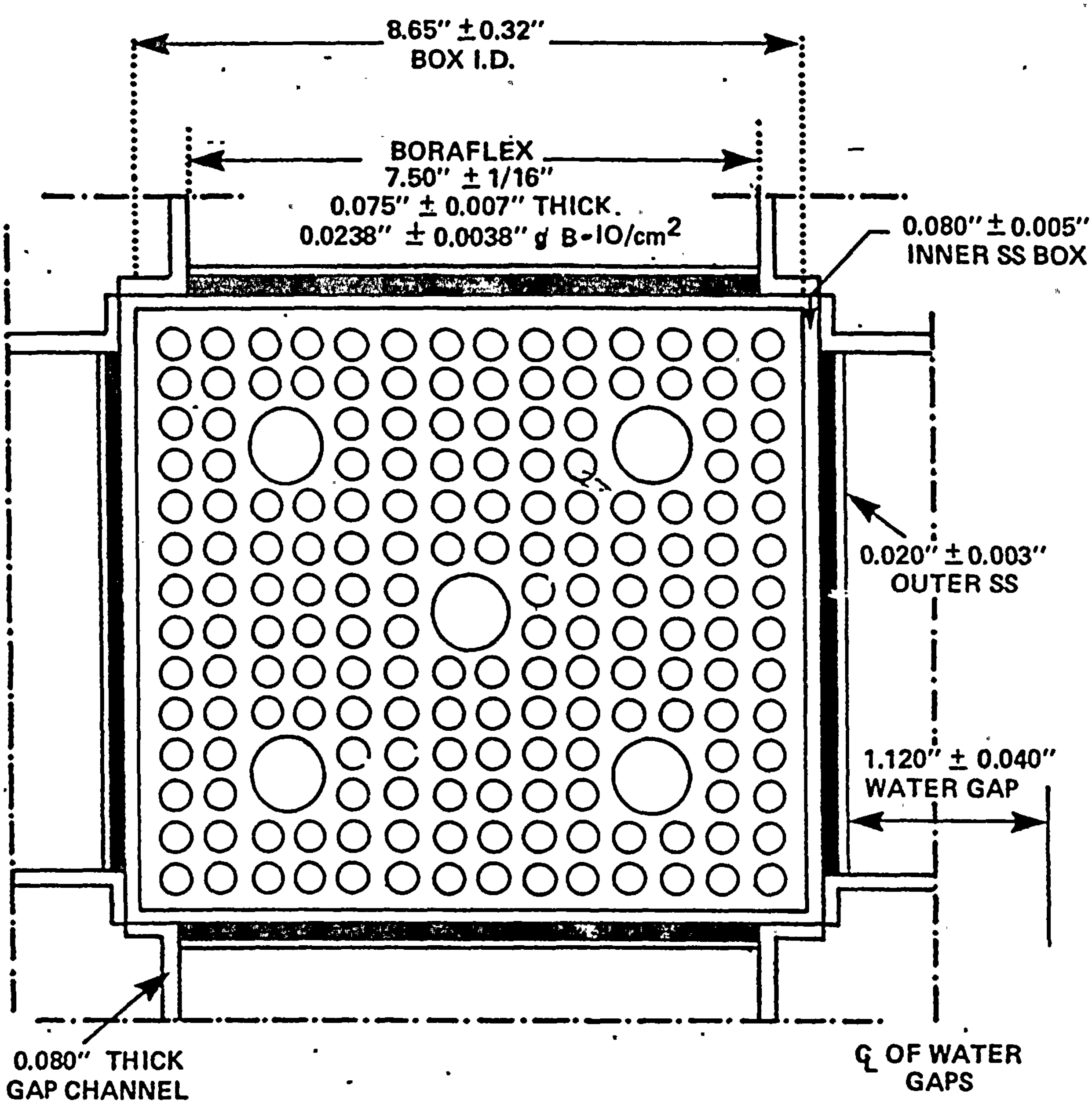
Case	Instant	Maximum Local Water Temp. °F	Maximum Cladding Temp. °F
Normal discharge	When the pool heat generation rate is at its peak value	155.9	198.8
Normal discharge	When the pool bulk temperature is at its peak value	179.2	219.4
Full core discharge	When the heat generation rate in the pool is at the peak value	162.8	209.4
Full core discharge	When the pool bulk temperature is at its peak value	188.0	222.8



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

ACCEPTABLE BURNUP DOMAIN IN REGION 2 OF THE ST. LUCIE PLANT SPENT FUEL STORAGE RACKS

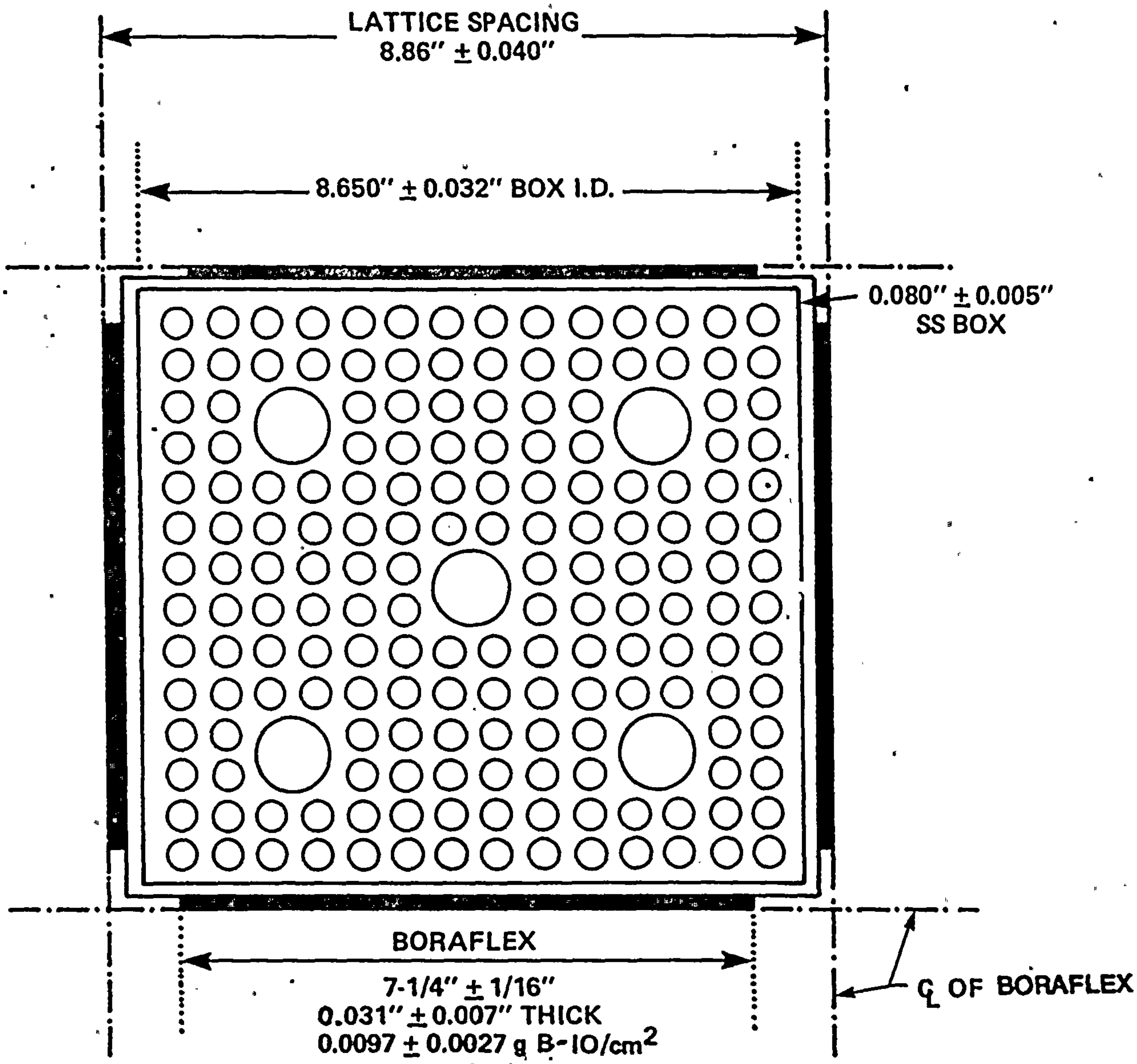
FIGURE 3-1



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

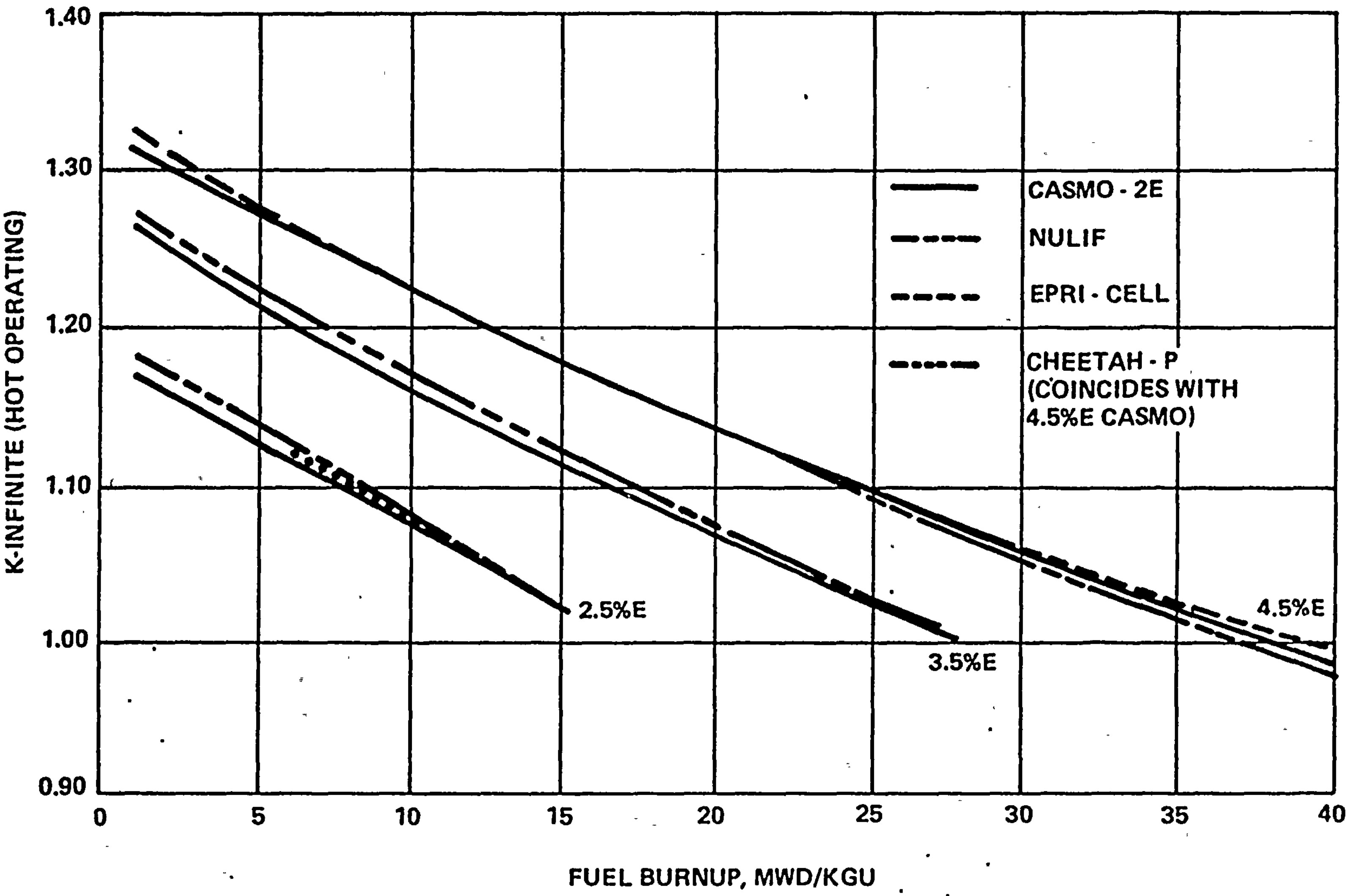
REGION 1 STORAGE CELL GEOMETRY

FIGURE 3-2



REGION 2
ST. LUCIE SPENT FUEL RACKS

FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 1
REGION 2 STORAGE CELL GEOMETRY FIGURE 3-3



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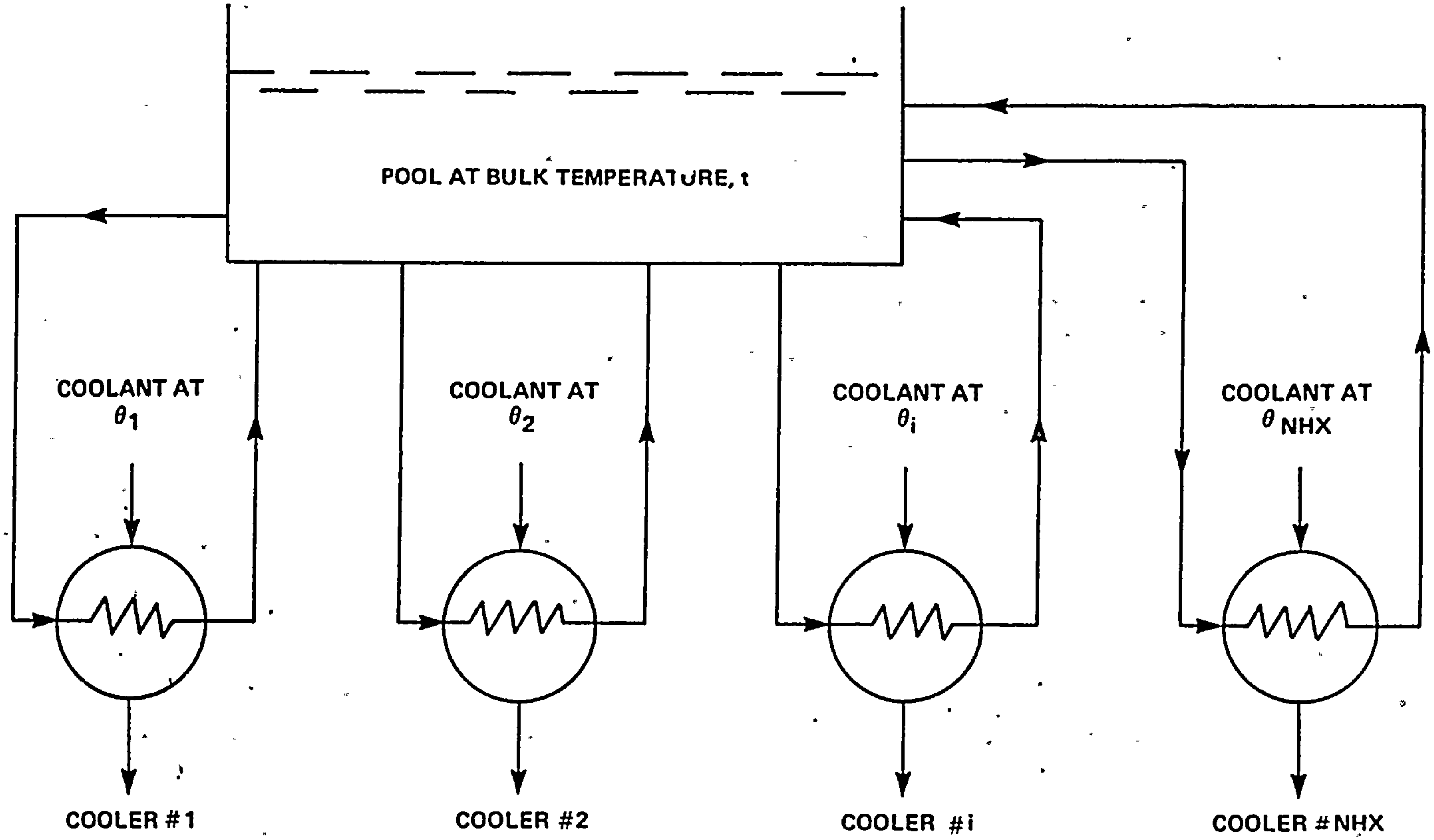
COMPARISON OF DEPLETION
CALCULATIONS FOR FUEL OF 4.5%
INITIAL ENRICHMENT
FIGURE 3-4



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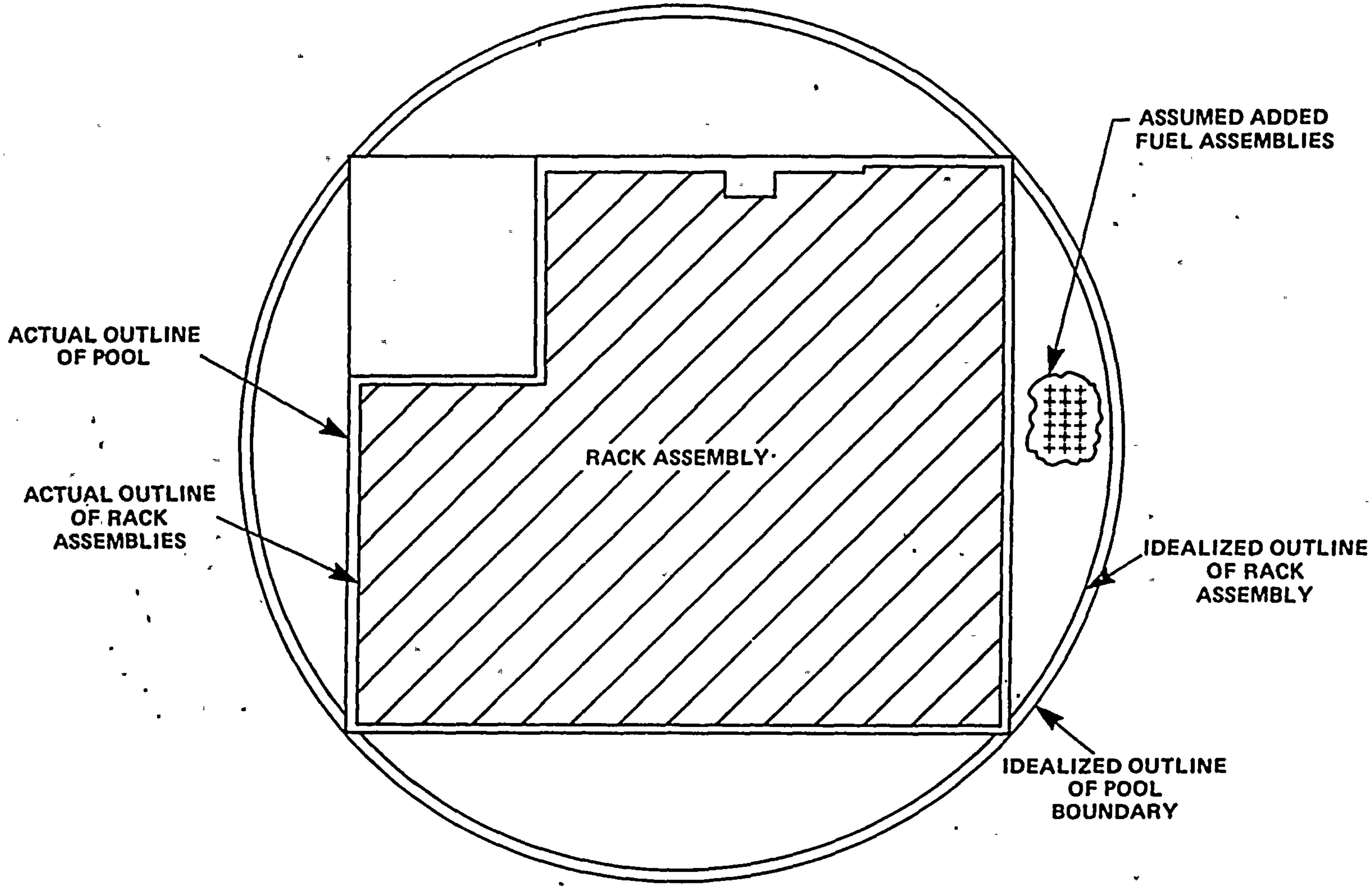
BULK POOL TEMPERATURE
MODEL FOR CODE BULKTEM
FIGURE 3-5



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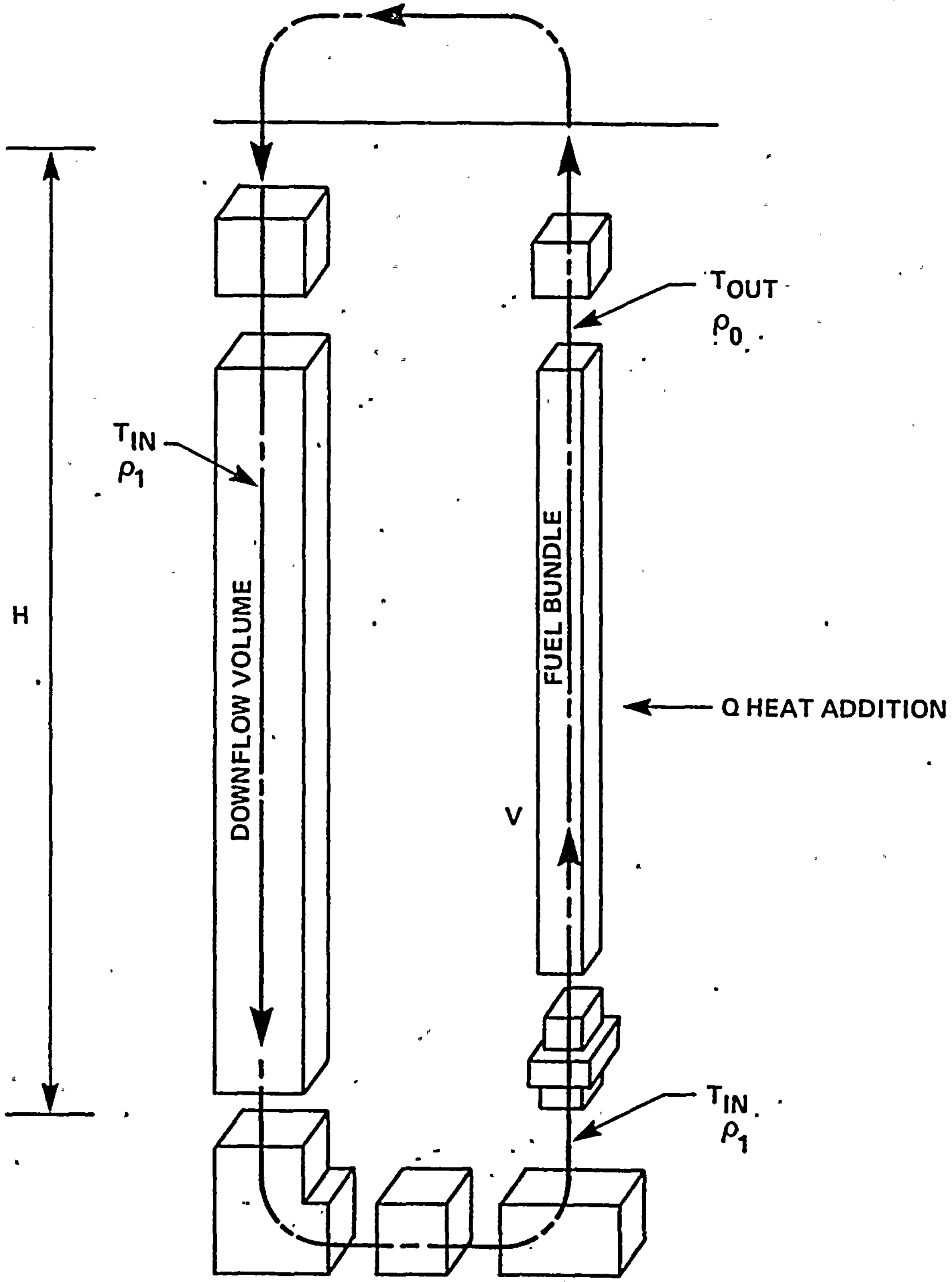
IDEALIZATION OF RACK ASSEMBLY

FIGURE 3-6



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ST. LUCIE PLANT UNIT 1

THERMAL CHIMNEY FLOW MODEL

FIGURE 3-7

4.0 MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

4.1 DESCRIPTION OF STRUCTURE

4.1.1 Description of the Fuel Handling Building

The Fuel Handling Building (FHB) consists of cast-in-place reinforced concrete interior and exterior walls. It is completely isolated from all other structures. The floors and roof are of beam and girder construction supported by columns. A complete description of the FHB is provided in Section 3.8.1.1.2 of the St Lucie Unit No. 1 updated FSAR. The FHB general arrangement is shown on FSAR Figures 1.2-18 and 1.2-19.

The FHB has been designed as a seismic Class I structure in accordance with the criteria outlined in Sections 3.8.1.1.2 and 3.8.1.4 through 3.8.1.7 of the updated FSAR. The building exterior walls, floors and interior partitions are designed to provide plant personnel with the necessary biological radiation shielding and protect the equipment inside from the effects of adverse environmental conditions including tornado and hurricane winds, temperature, external missiles and flooding.

The spent fuel pool is a cast-in-place steel lined reinforced concrete tank structure that provides space for storage of spent fuel assemblies, control element assemblies, new fuel during initial core loading and a spent fuel shipping cask. The fuel pool portion of the FHB including the walls and roof directly above the pool is designed to withstand, without penetration, the impact of high velocity external missiles that might occur during the passage of a tornado. The design missiles are further discussed in Section 3.5 of the St Lucie Unit No. 1 updated FSAR.

The spent fuel handling system includes interlocks, travel limits and other protective devices to minimize the probability of either mishandling or of equipment malfunction that could result in inadvertent damage to a fuel assembly and potential fission product release. The interlocks prevent movement into the walls while limit switches prevent the spent fuel handling machine from raising the fuel above a height where less than nine feet separates the surface of the water from the top of the active fuel length.

A leak detection system is provided in the spent fuel pool to monitor 100 percent of the pool liner plate weld seams. This system consists of a network of stainless steel angles attached to the outside of the pool liner walls and the underside of the pool liner floor by means of welds and sealed with epoxy material. In the event that one of the weld seams develops a leak, the liquid enters the monitor channel system and flows to one of 19 collection points at the base of the pool, from which the leak can be traced back to a specific pool area.

4.1.2 Description of Spent Fuel Racks

The function of the spent fuel storage racks is to provide for storage of spent fuel assemblies in a flooded pool, while maintaining a coolable geometry, preventing criticality, and protecting the fuel assemblies from excessive mechanical or thermal loadings.

A list of design criteria is given below:

1. The racks are designed in accordance with the NRC, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978 (as amended by the NRC letter dated January 18, 1979) and SRP Section 3.8.4 [1].

2. The racks are designed to meet the nuclear requirements of ANSI N210-1976. The effective multiplication factor, k_{eff}, in the spent fuel pool is less than or equal to 0.95, including all uncertainties and under all credible conditions.

3. The racks are designed to allow coolant flow such that boiling in the water channels between the fuel assemblies in the rack does not occur. Maximum fuel cladding temperatures are calculated for various pool cooling conditions as described in Section 3.3.

4. The racks are designed to seismic Category I requirements, and are classified as ANS Safety Class 3 and ASME Code Class 3 Component Support Structures. The structural evaluation and seismic analyses are performed using the specified loads and load combinations in Section 4.4.

5. The racks are designed to withstand loads without violating the criticality acceptance criteria which may result from fuel handling accidents and from the maximum uplift force of the spent fuel handling machine.

6. Each storage position in the racks is designed to support and guide the fuel assembly in a manner that will minimize the possibility of application of excessive lateral, axial and bending loads to fuel assemblies during fuel assembly handling and storage.

7. The racks are designed to preclude the insertion of a fuel assembly in other than design locations within the rack array.

8. The materials used in construction of the racks are compatible with the storage pool environment and will not contaminate the fuel assemblies.

4.1.2.1 Design of Spent Fuel Racks

4.1.2.1.1 Region 1

The rack module is fabricated from ASME SA-240-304L austenitic stainless steel sheet and plate material, and SA-351-CF3 casting material and SA-564-630 precipitation hardened stainless steel (to 1100°F) for supports only. The weld filler material utilized in body welds is ASME SFA-5.9, Classification ER 308L. Boraflex serves as the neutron absorber material. Additional information on Boraflex may be found in Section 3.1.3. The Boraflex experience list is given in Table 4-1.

A typical module contains storage cells which have an 8.65-inch nominal square cross-sectional opening. This dimension ensures that fuel assemblies with maximum expected axial bow can be inserted and removed from the storage cells without any damage to the fuel assemblies or the rack modules.

Figure 4-7 shows a horizontal cross-section of a 3 x 3 array. The cells provide a smooth and continuous surface for lateral contact with the fuel assembly. The anatomy of the rack modules is best explained by describing the components of the design, namely:

- Internal Square Tube
- Neutron Absorber material (Boraflex)
- Poison sheathing
- Gap element
- Baseplate
- Support assembly
- Top Lead-In

4.1.2.1.1.1 Internal Square Tube

This element provides the lateral bearing surface to the fuel assembly. It is fabricated by joining two formed channels (Figure 4-1) using a controlled seam welding operation. This element is an 8.65-inch square (nominal) cross-section by 169 inches long.

4.1.2.1.1.2 Neutron Absorber Material (Boraflex)

Boraflex is placed on all four sides of a square tube over a length of 143" (minimum), which provides the requisite B-10 screen for all stored assemblies including a four-inch shrinkage allowance.

4.1.2.1.1.3 Absorber Sheathing

The absorber sheathing (cover plate), shown in Figure 4-2, serves to position and retain the absorber material in its designated space. This is accomplished by spot welding the cover sheet to the square tube along the former's edges at numerous (at least 20) locations. This manner of attachment ensures that the absorber material will not sag or laterally displace during fabrication processes and under any subsequent loading condition.

4.1.2.1.1.4 Gap Element

Gap elements, illustrated in Figure 4-3, position two inner boxes at a predetermined distance to maintain the minimum flux trap gap required between two boxes. The gap element is welded to the inner box by fillet welds. An array of composite box assemblies welded as indicated in Figure 4-7 forms the honeycomb gridwork of cells which harnesses the structural strength of all sheet and plate type members in an efficient manner. The array of composite boxes has overall bending, torsional, and axial rigidities which are an order of magnitude greater than configurations utilizing grid bar type of construction.

4.1.2.1.1.5 Baseplate

The baseplate is a 3/4-inch thick plate type member which has 6-inch diameter holes concentrically located with respect to the internal square tube, except at support leg locations, where the hole size is 5 inches in diameter. These holes provide the primary path for coolant flow. Secondary flow paths are available between adjacent cells via the lateral flow holes (1 inch in diameter) near the root of the honeycomb (Figure 4-4) which preclude flow blockages. The honeycomb is welded to the baseplate with 3/32-inch fillet welds.

4.1.2.1.1.6 Support Assembly

Each module has at least four support legs. All supports are adjustable in length to enable leveling of the rack. The variable height support assembly consists of a flat-footed spindle which rides into an internally-threaded cylindrical member. The cylindrical member is attached to the underside of the baseplate through fillet and partial penetration welds. The base of the flat-footed spindle sits on the pool floor. Leveling of the rack modules is accomplished by turning the square sprocket in the spindle using a long arm (approximately 46 feet long) square head wrench. Figure 4-6 shows a vertical cross-section of the adjustable support assembly.

The supports elevate the module baseplate approximately 5-5/8 inches above the pool floor, thus creating the water plenum for coolant flow. The lateral holes in the cylindrical member provide the coolant entry path leading into the bottom of the storage locations.

4.1.2.1.1.7 Top Lead-In

Lead-ins are provided on each cell to facilitate fuel assembly insertion. Contiguous walls of adjacent cells are structurally connected at the lead-ins with a suitable vent opening. These lead-in joints aid in reducing the lateral deflection of the inner square tube due to the impact of fuel assemblies during the ground motion (postulated seismic motion specified in the FSAR). This type of construction leads to natural venting locations for the inter-cell space where the neutron absorber material is located.

4.1.2.1.2 Region 2 Design

The rack modules in Region 2 are fabricated from the same material as that used for Region 1 modules, i.e., ASME SA-240-304L austenitic stainless steel.

As shown in Figure 4-5 a typical Region 2 module storage cell also has an 8.65-inch nominal square cross-sectional opening . Figure 4-8 shows a horizontal cross-section of a 3 x 3 array. The rack construction varies from that for Region 1 inasmuch as the stainless steel cover plates, gap elements and top lead-ins are eliminated. Hence, the basic components of this design are as follows:

- Inner tube
- Neutron absorber material
- Side strips
- Baseplate
- Support assembly

In this construction, two channel elements form the cell of an 8.65-inch nominal square cross-sectional opening. The poison material is placed between two boxes as shown in Figure 4-8. Stainless steel side strips are inserted on both sides of the poison material to firmly locate it in the lateral direction. The bottom strip positions the poison material in the vertical direction to envelope the entire active fuel length of a fuel assembly (Figure 4-5). Two adjacent boxes and the side strip between boxes are welded together as shown in Figure 4-8, to form the honeycomb rack module.

The baseplate and support assemblies are incorporated in exactly the same manner as described for Region 1 in the preceding section.

4.1.2.2 Fuel Handling

The design of the spent fuel racks will not affect the conclusions of the fuel handling accidents presented in the FSAR (Section 15.4.3) and summarized by the NRC in the Safety Evaluation Report. That is, the radiological doses for the postulated fuel cask and fuel assembly drop accidents are well within the 10 CFR 100 criteria.

4.2 APPLICABLE CODES, STANDARDS, AND SPECIFICATIONS

The design and fabrication of the spent fuel racks and the analysis of the spent fuel pool have been performed in accordance with the applicable portions of the following NRC Regulatory Guides, Standard Review Plan Sections, and published standards:

4.2.1 NRC Documents

a. April 14, 1978 NRC OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, as amended by the NRC letter dated January 18, 1979.

b. St Lucie Plant Unit 1 Updated Final Safety Analysis Report, Docket No. 50-335.

c. NRC Regulatory Guides

1.13, Rev 2 Dec. 1981 (Draft)	Spent Fuel Storage Facility Design Basis
1.25 March 1972	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors

1.26, Rev 3 Feb. 1976	Quality Group Classifications and Standards for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants
1.29, Rev 3 Sept. 1978	Seismic Design Classification
1.31, Rev 3	Proposed Control of Ferrite Component in Stainless Steel Weld Material
1.71, Rev 0	Welder Qualification for Areas of Limited Accessibility
1.85, Rev 22	Material Code Case Acceptability ASME Section III Division I
1.92, Rev 1	Combining Modal Responses and Spatial Components in Seismic Response Analysis
1.124, Rev 1 Jan. 1978	Service Limits and Load Combinations for Class 1 Linear-Type Component Supports
3.41, Rev 1	Validation of Calculational Methods for Nuclear Criticality Safety.

d. NRC Standard Review Plan - NUREG-0800

Rev 1, July 1981	Section 3.7, Seismic Design
Rev 1, July 1981	Section 3.8.4, Other Seismic Category I Structures, Appendix D
Rev 3, July 1981	Section 9.1.2, Spent Fuel Storage
Rev 1, July 1981	Section 9.1.3, Spent Fuel Pool Cooling System
Rev 2, July 1981	NRC Branch Technical Position ASB 9-2, Residual Decay Energy for Light Water Reactors for Long Term Cooling

e. General Design Criteria for Nuclear Power Plants, Code of Federal Regulations, Title 10, Part 50, Appendix A (GDC Nos. 1, 2, 61, 62 and 63)

f. NUREG-0612 Control of Heavy loads at Nuclear Power Plants.

4.2.2 Industry Codes and Standards

ANSI N14.6-1978	American National Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials
ANSI N16.1-75	Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors
ANSI N16.9-75	Validation of Calculation Methods for Nuclear Criticality Safety
ANSI N18.2-1973	Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants
ANSI N45.2.2	Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants
ANSI N45.2.1	Cleaning of Fluid Systems and Associated Components during Construction Phase of Nuclear Power Plants
ANSI N45.2.11	1974 Quality Assurance Requirements for the Design of Nuclear Power Plants
ANSI ANS-57.2-1983	Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants
ANSI N210-76	Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations
ASME Section III (1983 Edition up to and including Summer 1984 Addenda	Nuclear Power Plant Components, Subsection NF
ACI-ASME Section III, Division 2 (1977 Edition)	Code for Concrete Reactor Vessels and Containments
ACI 318-63	Building Code Requirements for Reinforced Concrete
AISC 1980	Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, Eighth Edition
AWS D1.1	Structural Welding Code
ASNT-TC-1A June 1980	American Society for Nondestructive Testing (Recommended Practice for Personnel Qualification)

ASME II Part A & C (1983 Edition up to and including Summer 1984 Addenda)	Material Specifications Part A Ferrous, Part C Welding Rods, Electrodes and Filler Metals
ASME IX (1983 Edition up to and including Summer 1984 Addenda)	Welding & Brazing Qualifications
ASME Boiler and Pressure Vessel, Section V, (1983 Edition up to and including Summer 1984 Addenda)	Non-destructive Examination

4.3 SEISMIC AND IMPACT LOADS

The objective of the seismic analysis of the spent fuel racks is to determine the structural responses resulting from the simultaneous application of three orthogonal seismic excitations. The method of analysis employed is the time history method.

Seismic floor response spectra for the spent fuel pool floor have been developed using the methods described in Subsections 3.7.1 and 3.7.2 of the St Lucie Unit No 1 Updated FSAR. The parameters of the original lumped mass model of the Fuel Handling Building were adjusted to reflect the increased mass corresponding to the new high density spent fuel storage racks. The resulting floor response spectra are shown in Figure 4-9. These spectra were then used to generate statistically independent time history excitations, one for each of the three orthogonal directions. Since the spent fuel racks have no connection with the pool walls or with each other, the pool floor time histories are used as input to the dynamic analysis of the racks, as described in Subsection 4.5.2.2.1. Fluid coupling is also considered as described therein.

Deflection or movements of racks under earthquake loading is limited by design such that the nuclear parameters outlined in Section 3.1 are not exceeded. Impact loads have been considered as discussed in Subsection 4.6.4.

The interaction between the fuel assemblies and the rack has been considered, particularly gap effects. The resulting impact loads are of small magnitudes so there is no structural damage to the fuel assemblies.

The spent fuel pool structure has been reanalyzed for the increased dead, thermal and seismic loading resulting from the storage of additional fuel assemblies in the pool, as described in Subsection 4.5.1.

4.4 LOADS AND LOAD COMBINATIONS

4.4.1 Spent Fuel Pool

4.4.1.1 Loads

The following design loads were considered in the spent fuel pool analysis:

a) Structural Dead Load (D)

Dead load consists of the dead weight of the spent fuel racks, the pool water and the concrete structure, superstructure, walls and miscellaneous building items within the Fuel Handling Building.

b) Live Load (L)

Live loads are random temporary load conditions for maintenance which include the spent fuel cask dead weight.

c) Seismic Loads (SSE and OBE)

Seismic loads include the loads induced by Safe Shutdown Earthquake (SSE) and Operating Basis Earthquake (OBE). The hydrodynamic load during the earthquake events was also considered.

d) Normal Operating Thermal Loads (T)

The load induced by normal thermal gradients existing between the building interior and the ambient external environment was considered. The conditions are:

Summer

- Interior water temperature	150°F
- Exterior air temperature	93°F
- Soil temperature	70°F

Winter

- Interior water temperature	150°F
- Exterior air temperature	32°F
- Soil temperature	70°F

For all cases, the "as constructed" concrete temperature was assumed to be 70°F. A linear gradient through the wall and mat was assumed.

e) Accident (Loss of Fuel Pool Cooling) Thermal Load (T_A)

The thermal accident temperature for the spent fuel pool water is 217°F throughout the pool. At this temperature, the exterior air temperature at 40°F was assumed for the critical thermal gradient through the wall. 70°F soil temperature was used. The thermal gradient was assumed to be linear.

f) Fuel Cask Drop Load (M)

A 25 ton cask drop from the maximum height of 58 feet above the pool floor (Elevation 79.50') was considered. (The cask bottom must attain Elevation 77.00' for entry into the building.)

4.4.1.2 Load Combinations

In the spent fuel pool analysis, the following load combinations, from the St Lucie No. 1 Updated FSAR, Section 3.8.1.5, were considered:

a) Normal Operation

$$1.5\ (D + T) + 1.8\ L$$

b) OBE Condition

$$1.25\ (D + T + OBE + 0.2\ L)$$

c) SSE Condition

$$1.05\ (D + T + 0.2\ L) + 1.0\ SSE$$

d) Accident and Cask Drop

$$1.05\ (D + T_A + 0.2\ L)$$

$$1.05\ (D + T + 0.2\ L) + 1.0\ M$$

For the evaluation of the liner and liner anchors, the above load combinations are applicable except that load factors for all cases may be taken equal to 1.0 (in accordance with Table CC-3230-1 of ACI-ASME Section III, Division 2) in conjunction with the structural acceptance criteria of this SAR subsection 4.6.1.1.b.

Linear analyses without iterations were performed initially to determine the critical load combinations. As a result, the following loading cases were selected for the non-linear concrete cracking analysis:

i)	$1.5\ D + 1.8\ L$
ii)	$1.05\ (D + T\ \text{winter} + 0.2\ L) + 1.0\ SSE$
iii)	$1.05\ (D + T\ \text{summer} + 0.2\ L) + 1.0\ SSE$
iv)	$1.05\ (D + 0.2\ L) + 1.0\ SSE$
v)	$1.05\ (D + T_A + 0.2\ L)$
vi)	$1.05\ (D + T\ \text{winter} + 0.2\ L) + 1.0\ M$
vii)	$1.05\ (D + 0.2\ L) + 1.0\ M$

4.4.2 Spent Fuel Racks

4.4.2.1 Loads

The following loads were considered in the rack design:

Dead Load	(D) =	Dead weight-induced stresses (including fuel assembly weight).
	(D') =	Dead weight of empty rack.
Live Load	(L) =	0 for the structure, since there are no moving objects in the rack load path.
Fuel Drop Accident Load	(F_d) =	Force caused by the accidental drop of the heaviest load from the maximum possible height. (See Section 4.6.6.)
Crane Uplift Load	(P_f) =	Upward force on the racks caused by postulated stuck fuel assembly (4000 lbs).
Seismic Loads	(E) =	Operating Basis Earthquake.
	(E') =	Safe Shutdown Earthquake.
Thermal Loads	(T_o) =	Differential temperature induced loads (normal condition).
	(T_a) =	Differential temperature induced loads (abnormal design condition). For upset and emergency conditions, T_a is the differential temperature for the full core offload condition. For faulted conditions, T_a is the differential temperature for the loss of cooling condition.

The conditions T_a and T_o cause local thermal stresses to be produced. The worst situation will be obtained when an isolated storage location has a fuel assembly which is generating heat at the maximum postulated rate. The surrounding storage locations are assumed to contain no fuel. The heated water makes unobstructed contact with the inside of the storage walls, thereby producing the maximum possible temperature difference between the adjacent cells. The secondary stresses thus produced are limited to the body of the rack; that is, the support legs do not experience the secondary (thermal) stresses.

4.4.2.2 Load Combinations

Each component operating condition has been evaluated for the applicable loading combinations listed below:

a) Normal Condition

$D + L$

$D + L + T_o$

$D + L + T_o + E$

$D' + T_o$

b) Upset Condition

$D + L + T_a + E$

$D + L + T_a + P_f$

$D + T_a + F_D$

c) Emergency Condition

$D + T_a + P_f + E$

$D + T_a + F_D + E$

d) Faulted Condition

$D + L + T_a + E'$

$D + L + F_D$

$D + L + P_f$

4.5 DESIGN AND ANALYSIS PROCEDURES

4.5.1 Design and Analysis Procedures for the Spent Fuel Pool

4.5.1.1 Spent Fuel Pool Structure Finite Element Analysis

In this analysis, the EBS/NASTRAN program, developed by Ebasco and linked to the commercially available NASTRAN program, was used. Various layers of concrete and reinforcing bars were used to determine the effects of concrete cracking. The nonlinear analysis scheme based on the combination of stiffness iteration and load iteration methods, which were available in EBS/NASTRAN program, was used to automatically determine the stresses in the concrete and reinforcing bars after the concrete cracks. The finite element model used in this analysis can be summarized as follows:

a) Since the effect of the additional fuel rack load on the pool floor is limited to the mat in the pool area, the upper portion of the pool walls is not required for the re-evaluation. Therefore, the finite element model included the lower portion of walls, the pool floor (mat) and the underlying soil. The structural components included in the model are shown on Figure 4-10. The cut-off boundary of the walls is at EL. 45.25 ft.

b) The following boundary conditions were used at the model cut-off boundaries:

   i) South end of the mat - Rotational springs representing the bending resistance of the cut-off mat were provided.

ii) Top of the walls - The rotation about the axis parallel to the edge of the wall was restrained to consider the effect of the cut-off wall. This assumed boundary condition has little effect on the response of the pool mat, since the boundary is far above the mat. This was demonstrated in the linear analysis results.

iii) South end of east and west walls - Since the rigidity of the cut-off walls is very small, a free boundary condition was assumed.

A computer plot of the finite element model is presented in Figure 4-11 which shows the overall view of the model indicating the composite of the four exterior and one interior walls.

4.5.1.2 Liner and Anchorage Analysis

The liner and its anchors were evaluated for the temperature load, the strain induced load due to the deformation of the floor, and the horizontal seismic load. The program POSBUKF developed by Ebasco was used for the liner buckling analysis due to the temperature and strain induced loads. This program is capable of determining the post-buckling stress/strain if the liner plate buckles. The effect of the hydrostatic pressure was considered in this analysis. In calculating the in-plane shear due to the horizontal seismic loads transmitted from the fuel rack to the liner, the maximum assumed friction coefficient of 0.8 was used.

The liner anchors were evaluated for the unbalanced liner in-plane force due to the temperature and strain induced loads, as well as the horizontal seismic in-plane shear force.

4.5.1.3 Foundation Stability and Soil Bearing

A detailed soil bearing evaluation was performed for the increased fuel rack loading. The soil stresses were obtained at each mat corner and compared to the allowable value. Stability calculations were performed for overturning and sliding.

4.5.2 Design and Analysis Procedures for Spent Fuel Storage Racks

The purpose of this subsection is to demonstrate the structural adequacy of the spent fuel rack design under normal and accident loading conditions. The method of analysis presented herein uses a time-history integration method similar to that previously used in the Licensing Reports on High Density Fuel Racks for Fermi 2 (Docket No 50-341), Quad Cities 1 and 2 (Docket Nos 50-254 and 50-265), Rancho Seco (Docket No 50-312), Grand Gulf Unit 1 (Docket No 50-416), Oyster Creek (Docket No 50-219), V C Summer (Docket No 50-395), Diablo Canyon 1 and 2 (Docket Nos 50-275 and 50-323) and Byron Units 1 and 2 (Docket Nos 50-454 and 50-455). The results show that the high density spent fuel racks are structurally adequate to resist the postulated stress combinations associated with level A, B, C and D conditions as defined in References 1 and 2.

4.5.2.1 Analysis Outline

The spent fuel storage racks are seismic Category I equipment. Thus, they are required to remain functional during and after a Safe Shutdown Earthquake[3]. As noted previously, these racks are neither anchored to the pool floor nor are they attached to the side walls. The individual rack modules are not interconnected. Furthermore, a particular rack may be completely loaded with fuel assemblies (which corresponds to greatest rack inertia), or it may be completely empty. The coefficient of friction, μ, between the supports and pool floor is determined as follows. According to Rabinowicz[4] the results of 199 tests performed on austenitic stainless steel plates submerged in water show a mean value of μ to be 0.503 with a standard deviation of 0.125. The upper and lower bounds (based on twice the standard deviation) are thus 0.753 and 0.253, respectively. Two separate analyses are performed for the rack assemblies with values of the coefficient of friction equal to 0.2 (lower limit) and 0.8 (upper limit), respectively. Analyses performed for the geometrically limiting rack modules focus on limiting values of the coefficient of friction, and the number of fuel assemblies stored. Typical cases studied are:

- Fully loaded rack (all storage locations occupied), μ = 0.8, 0.2 (μ = coefficient of friction)
- Nearly empty rack μ = 0.8, 0.2
- Rack half full μ = 0.2, 0.8

Pool floor slab acceleration data developed for the Safe Shutdown Earthquake (SSE) are shown in Figures 4-12 through 4-14. The method of analysis employed is the time-history method. The pool slab acceleration data were developed from the building response spectra.

The objective of the seismic analysis is to determine the structural response (stresses, deformation, rigid body motion, etc) due to simulataneous application of the three independent, orthogonal excitations.

The seismic analysis is performed in three steps, namely:

1. Development of a nonlinear dynamic model consisting of inertial mass elements and gap and friction elements.
2. Generation of the equations of motion and inertial coupling and solution of the equations using the "component element time integration scheme"[6, 7] to determine nodal forces and displacements.
3. Computation of the detailed stress field in the rack (at the critical location) and in the support legs using the nodal forces calculated in the previous step. These stresses are checked against the design limits given in Section 4.6.2.2.

A brief description of the dynamic model follows.

4.5.2.2 Fuel Rack - Fuel Assembly Model

Since the racks are not anchored to the pool slab or attached to the pool walls or to each other, they can execute a wide variety of rigid body motions. For example, the rack may slide on the pool floor (so-called "sliding condition"); one or more legs may momentarily lose contact with the liner ("tipping condition"); or the rack may experience a combination of sliding and tipping conditions. The structural model should permit simulation of these kinematic events with inherent built-in conservatisms. Since these racks are equipped with girdle bars to dissipate energy due to inter-rack impact (if it occurs), it is also necessary to model the inter-rack impact phenomena in a conservative manner. Similarly, lift off of the support legs and subsequent impacts must be modelled using appropriate impact elements, and Coulomb friction between the rack and the pool liner must be simulated by appropriate piecewise linear springs. These special attributes of the rack dynamics require a strong emphasis on the modeling of the linear and nonlinear springs, dampers, and stop elements. The model outline in the remainder of this section, and the model description in the following section describe the detailed modeling technique to simulate these effects, with emphasis placed on the nonlinearity of the rack seismic response.

4.5.2.2.1 Outline of Model

a. The fuel rack structure is a folded metal plate assemblage welded to a baseplate and supported on four legs. The rack structure itself is a very rigid structure. Dynamic analysis of typical multicell racks has shown that the motion of the structure is captured almost completely by the behavior of a six degrees-of-freedom structure; therefore, the movement of the rack cross-section at any height is described in terms of the six degrees-of-freedom of the rack base.

b. The seismic motion of a fuel rack is characterized by random rattling of fuel assemblies in their individual storage locations. Assuming that all assemblies vibrate in phase obviously exaggerates the computed dynamic loading on the rack structure. This assumption, however, greatly reduces the required degrees-of-freedom needed to model the fuel assemblies which are represented by five lumped masses located at different levels of the rack. The centroid of each fuel assembly mass can be located, relative to the rack structure centroid at that level, so as to simulate a partially loaded rack.

c. The local flexibility of the rack-support interface is modeled conservatively in the analysis.

d. The rack base support may slide or lift off the pool floor.

e. The pool floor and walls have a specified time-history of seismic accelerations along the three orthogonal directions.

f. Fluid coupling between rack and assemblies, and between rack and adjacent racks, is simulated by introducing appropriate inertial coupling into the system kinetic energy, Inclusion of these effects uses the methods of References 4 and 6 for rack/assembly coupling and for rack/rack coupling (see Section 4.5.2.2.3 of this report).

g. Potential impacts between rack and assemblies are accounted for by appropriate "compression only" gap elements between masses involved.

h. Fluid damping between rack and assemblies, and between rack and adjacent rack, is conservatively neglected.

i. The supports are modeled as "compression only" elements for the vertical direction and as "rigid links" for dynamic analysis. The bottom of a support leg is attached to a frictional element as described in Section 4.5.2.2.2. The cross-section inertial properties of the support legs are computed and used in the final computations to determine support leg stresses.

j. The effect of sloshing has been shown to be negligible at the bottom of a pool and hence is neglected.

k. Inter-rack impact, if it occurs, is simulated by a series of gap elements at the top and bottom of of the rack in the two horizontal directions. The most conservative case of adjacent rack movement is assumed; each adjacent rack is assumed to move completely out of phase with the rack being analyzed.

l. The form drag opposing the motion of the fuel assemblies in the storage locations is conservatively neglected in the results reported herein.

m. The form drag opposing the motion of the fuel rack in the water is also conservatively neglected in the results reported herein.

n. The rattling of the fuel assemblies inside the storage locations causes the "gap" between the fuel assemblies and the cell wall to change from a maximum of twice the nominal gap to a theoretical zero gap. However, the fluid coupling coefficients(8) utilized are based on linear vibration theory(9). Studies in the literature show that inclusion of the nonlinear effect (viz., vibration amplitude of the same order of magnitude as the gap) drastically lowers the equipment response(10).

Figure 4-15 shows a schematic of the model. Six degrees-of-freedom are used to track the motion of the rack structure. Figures 4-16 and 4-17, respectively, show the inter-rack impact springs and fuel assembly/storage cell impact springs.

The model for simulating fuel assembly motion incorporates five lumped masses. The lower mass is assumed to be attached to the baseplate and to move with the baseplate. The four rattling masses are located at quarter height, half height, three quarter height and top of the rack. Two degrees-of-freedom are used to track the motion of each rattling mass.

The solution procedure described in the following is implemented in computer code DYNARACK, which is a validated computer code under Holtec's Q A program.

4.5.2.2.2 Model Description

The absolute degrees-of-freedom associated with each of the mass locations are shown in Figure 4-15. As shown, the discrete mass fractions are located at heights z=0, 0.25H, 0.5H, 0.75H and H respectively. Table 4-6 gives the degrees-of-freedom and the associated generalized coordinates.

$U_1(t)$ is the pool floor slab displacement seismic time-history. Thus, as tabulated in Table 4-6 and shown in Figure 4-15, there are sixteen degrees-of-freedom in the system. Not shown in Figure 4-15 are the gap elements used to model the support legs and the impacts with adjacent racks.

4.5.2.2.3 Fluid Coupling

An effect of some significance requiring careful modeling is the so-called "fluid coupling effect". If one body of mass (m) vibrates adjacent to another body (mass m), and both bodies are submerged in a frictionless fluid medium, then Newton's equations of motion for the two bodies have the form:

$$(m_1 + M_{11})\ddot{X}_1 - M_{12}\ddot{X}_2 = \text{applied forces on mass } m_1$$

$$-M_{21}\ddot{X}_1 + (m_2 + M_{22})\ddot{X}_2 = \text{applied forces on mass } m_2$$

$\ddot{X}_1$, $\ddot{X}_2$ denote absolute accelerations of mass m_1 and m_2, respectively.

M_{11}, M_{12}, M_{21} and M_{22} are fluid coupling coefficients which depend on the shape of the two bodies, their relative disposition, etc. Fritz[9] gives data for M_{ij} for various body shapes and arrangements. It is to be noted that the above equation indicates that the effect of the fluid is to add a certain amount of mass to the body (M_{11} to body 1), and an external force which is proportional to the acceleration of the adjacent body (mass m_2). Thus, the acceleration of the one body affects the force field on another. This force is a strong function of the interbody gap, reaching large values for very small gaps. This inertial coupling is called fluid coupling. It has an important effect in rack dynamics. The lateral motion of a fuel assembly inside the storage location will encounter this effect. So will the motion of a rack adjacent to another rack. These effects are included in the equations of motion. The fluid coupling is between fuel array node i and cell wall in Figure 4-17. Furthermore, the rack equations contain coupling terms which model the effect of fluid in the gaps between adjacent racks. The coupling terms modeling the effects of fluid flowing between adjacent racks are computed assuming that all adjacent racks are vibrating 180 degrees out of phase from the rack being analyzed. Therefore, only one rack is considered surrounded by a hydrodynamic mass computed as if there were a plane of symmetry located in the middle of the gap region.

Finally, fluid virtual mass is included in the vertical direction vibration equations of the rack; virtual inertia is also added to the governing equation coresponding to the rotational degree-of-freedom, q(t).

4.5.2.2.4 Damping

In reality, damping of the rack motion arises from material hysteresis (material damping), relative intercomponent motion in structures (structural damping), and fluid drag effects (fluid damping). In the analysis, a maximum of 2% structural damping is imposed on elements of the rack structure during SSE seismic simulations. This is in accordance with the St Lucie Unit 1 FSAR(13). Material and fluid damping are conservatively neglected. The dynamic model has the provision to incorporate fluid damping effects; however, no fluid damping has been used for this analysis.

4.5.2.2.5 Impact

Referring to Figure 4-18, any fuel assembly node may impact the corresponding structural mass node. To simulate this impact, four compression-only gap elements around each rattling fuel assembly node are provided (see Figure 4-17). As noted previously, fluid dampers may also be provided in parallel with the springs. The compressive loads developed in these springs provide the necessary data to evaluate the integrity of the cell wall structure and stored array during the seismic event. Figure 4-16 shows the location of the impact springs used to simulate any potential for inter-rack impacts. Section 4.5.2.4.2 gives more details on these additional impact springs.

4.5.2.3 Assembly of the Dynamic Model

The cartesian coordinate system associated with the rack has the following nomenclature:

- x = Horizontal coordinate along the short direction of rack rectangular platform
- y = Horizontal coordinate along the long direction of the rack rectangular platform
- z = Vertically upward

As described in the preceding section, the rack, along with the base, supports, and stored fuel assemblies, is modeled for the general three-dimensional (3-D) motion simulation by a fourteen degree-of- freedom model. To simulate the impact and sliding phenomena expected, 60 nonlinear gap elements and 16 nonlinear friction elements are used. Gap and friction elements, with their connectivity and purpose, are presented in Table 4-7.

If the simulation model is restricted to two dimensions (one horizontal motion plus vertical motion, for example) for the purposes of model clarification only, then a descriptive model of the simulated structure which includes gap and friction elements is shown in Figure 4-18. (Note that only the top rattling mass is shown for clarity.)

The impacts between fuel assemblies and rack show up in the gap element, having local stiffness K_I, in Figure 4-18. In Table 4-7, gap elements 5 through 8 are for the vibrating mass at the top of the rack. The support leg spring rates K_d are modeled by elements 1 through 4 in Table 4-7. Note that

the local compliance of the concrete floor is included in K_d. To simulate sliding potential, friction elements 1 through 8 in Table 4-7 are employed. Friction elements 2 and 8, and 4 and 6 (Table 4-7) are represented as K_f in Figure 4-18. The friction of the support/liner interface is modeled by a piecewise linear spring with a suitably large stiffness K_f up to the limiting lateral load, N, where N is the current compression load at the interface between support and liner. At every time step during the transient analysis, the current value of N (either zero for liftoff condition, or a compressive finite value) is computed. Finally, the support rotational friction springs K_R reflect any rotational restraint that may be offered by the foundation. This spring rate is calculated using a modified Boussinesq equation(4) and is included to simulate the resistive moment of the support to counteract rotation of the rack leg in a vertical plane. This rotation spring is also nonlinear, with a zero spring constant value assigned after a certain limiting condition of slab moment loading is reached.

The nonlinearity of these springs (friction elements 9, 11, 13 and 15 in Table 4-7) reflects the edging limitation imposed on the base of the rack support legs. In this analysis, this effect is neglected; any support leg bending, induced by liner/baseplate friction forces, is resisted by the leg acting as a beam cantilevered from the rack baseplate.

For the 3-D simulation, all support elements (listed in Table 4-7) are included in the model. Coupling between the two horizontal seismic motions is provided both by the offset of the fuel assembly group centroid which causes the rotation of the entire rack and by the possibility of liftoff of one or more support legs. The potential exists for the rack to be supported on one or more support legs or to liftoff completely during any instant of a complex 3-D seismic event. All of these potential events may be simulated during a 3-D motion and have been observed in the results.

4.5.2.4 Time Integration of the Equations of Motion

4.5.2.4.1 Time-History Analysis Using 16 DOF Rack Model

Having assembled the structural model, the dynamic equations of motion corresponding to each degree-of-freedom can be written by using Newton's second law of motion; or by using Lagrange's equation. The system of equations can be represented in matrix notation as:

$$[M]\,(\ddot{q}) = (Q) + (G)$$

where the vector (Q) is a function of nodal displacements and velocities, and (G) depends on the coupling inertia and the ground acceleration. Premultiplying the above equations by $[M]^{-1}$ renders the resulting equation uncoupled in mass.

We have: $(\ddot{q}) = [M]^{-1}\,(Q) + [M]^{-1}\,(G)$

As noted earlier, in the numerical simulations run to verify structural integrity during a seismic event, all elements of the fuel assemblies are assumed to move in phase. This will provide maximum impact force level, and induce additional conservatism in the time-history analysis.

This equation set is mass uncoupled, displacement coupled, and is ideally suited for numerical solution using a central difference scheme. The computer program "DYNARACK"* is utilized for this purpose.

Stresses in various portions of the structure are computed from known element forces at each instant of time.

Dynamic analysis of typical multicell racks has shown that the motion of the structure is captured almost completely by the behavior of a six degree-of-freedom structure; therefore, in this analysis model, the movement of the rack cross-section at any height is described in terms of the rack base degrees-of-freedom ($q_1(t)$, ...$q_6(t)$). The remaining degrees-of-freedom are associated with horizontal movements of the fuel assembly masses. In this dynamic model, five rattling masses are used to represent fuel assembly movement. Therefore, the final dynamic model consists of six degrees-of-freedom for the rack plus ten additional mass degrees-of-freedom for the five rattling masses. The remaining portion of the fuel assembly is assumed to move with the rack base. Thus, the totality of fuel mass is included in the simulation.

4.5.2.4.2 Evaluation of Potential for Inter-Rack Impact

Since the racks are closely spaced, the simulation includes impact springs to model the potential for inter-rack impact, especially for low values of the friction coefficient between the support and the pool liner. To account for this potential, five inter-rack gap elements were located at each side of the rack at the top and at the baseplate. Figure 4-16 shows the location of these gap elements. Loads in these elements, computed during the dynamic analysis, are used to assess rack integrity if inter-rack impact occurs.

4.6 STRUCTURAL EVALUATION CRITERIA

4.6.1 <u>Structural Acceptance Criteria for Spent Fuel Pool Structure</u>

4.6.1.1 Criteria

The stresses/strains resulting from the loading combinations described in Section 4.4.1 satisfy the following acceptance criteria:

a) Spent Fuel Pool Concrete Structure

The design stress limits described in Section 3.8.1.6 of St Lucie Unit No. 1 Updated FSAR were used for the evaluation of the spent fuel pool reinforced concrete structural components. The capacity of all sections was computed in accordance with ACI 318-63 Part IV-B, Ultimate Strength Design.

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*The numerical procedure underlying DYNARACK has been previously utilized in licensing of similar racks for Fermi 2 (Docket No 50-341), Quad Cities 1 and 2 (Docket Nos 50-254 and 265), Rancho Seco (Docket No 50-312), Oyster Creek (Docket No 50-219), V C Summer (Docket No 50-395), and Diablo Canyon 1 and 2 (Docket Nos 50-275 and 50-323).

Re: St. Lucie Plant
Docket No. 50-335
10CFR50.59 REPORT

St. Lucie Plant Unit 1

Report of Changes Made

Under the Provisions of

10 CFR 50.59

for the Period Ending January 22, 1988

GRMATWS.RAI

DOCUMENTS REVIEWED FOR FSAR AMENDMENT 7

NUMBER	REVISION	TITLE
118-181	0-1	AIR DRYER PLUG VALVE REPLACEMENT
040-182	0-1	AUXILIARY BUILDING JIB HOIST
092-182	6	DIESEL GENERATOR UPGRADE
100-182	0-2	REMOVAL OF STEAM GENERATOR DELTA P SIGNAL CHARACTERIZERS
046-183	0	REACTOR UPPER CAVITY HANDRAIL
078-183	4-5	NITROGEN SUPPLY SYSTEM
336-183	0	INSTRUMENT AIR SYSTEM CROSS TIE CHECK VALVE ADDITION
340-183	0	CCW HEAT EXCHANGER RETUBING
383-183	0	SECURITY CONSOLES - LED
412-183	0-1	MICROWAVE DETECTOR REPLACEMENT
174-184	0-1	MAIN STEAM SAFETY VALVE LIFT LEVER REMOVAL
187-184	0-1	EDG CFD RELAY REPLACEMENT
[illegible]-184	0	EX-CORE NEUTRON FLUX MONITORING SYSTEM UPGRADE
252-184	0-1	TRAVELING WATER SCREEN UPGRADE
015-185	0-1	DIESEL GENERATOR AIR START SYSTEM PIPING MODIFICATION
022-185	0-1	INSTRUMENT AIR UPGRADE TIE INS
024-185	0-2	EPAS INSTALLATION
039-185	0-1	DIESEL GENERATOR SUBSYSTEM FLOW DIAGRAMS
047-185	0	GE SAM RELAY PC CARD REPLACEMENT
064-185	0-1	MSR RETUBING
073-185	0	MSR PERFORATED PLATE MODIFICATION
131-185	0	REPLACEMENT OF RIS DEVICES
142-185	0	APPENDIX R CONTROL ROOM INDEPENDENCE FOR ALTERNATE SHUTDOWN
155-185	0	REACTOR CLOSURE HEAD LIFITNG RIG PIPE ASSEMBLY
169-185	0-1	TURBINE GANTRY CRANE BRAKE SYSTEM MODIFICATION

DOCUMENTS REVIEWED FOR FSAR AMENDMENT 7

NUMBER	REVISION	TITLE
174-185	0	RCP OIL LIFT SYSTEM PRESSURE SWITCH REPLACEMENT
179-185	0	DIESEL GENERATOR COOLING SYSTEM VALVE REPLACEMENT
202-185	0	CCW STRAINER BACK FLUSH DRAIN
004-186	0-1	LUBE OIL CENTRIFUGE ANNUNCIATION
007-186	0	NRV ACTUATION SOLENOIDS
023-186	0-1	MAIN FEEDWATER PUMP LUBE OIL PRESSURE SWITCH REPLACEMENT
032-186	0	EXCITER COOLER VENTS AND DRAINS TUBING MATERIAL CHANGE
037-186	0	PCB TRANSFORMER REPLACEMENT NON-SAFETY RELATED
043-186	0	MISCELLANEOUS PIPING SYSTEMS MODIFICATIONS
071-186	0	FHB HVAC PENETRATION BARRIERS
074-186	0	HEATER DRAIN PUMP DEMINERALIZED WATER SUPPLY
082-186	0	MAIN STEAM PIPING MODIFICATION
[illegible]-186	0	LOW POWER FEEDWATER CONTROL SYSTEM
086-186	0-1	MISAPPLICATION OF LIMITORQUE OPERATOR
088-186	0	ISOLATED PHASE BUS DUCT JUMPER MODIFICATION
090-186	0	CLOSE INTERCEPT VALVE CIRCUIT MODIFICATION
093-186	0	RTGB ANNUNCIATOR GROUND DETECTION
095-186	0	REPLACEMENT OF INSTRUMENT SCALES
098-186	0	HYDROGEN PURGE PENETRATIONS: REPLACE OF VLVS I-V-25-11,13,15
106-186	0-2	HIR EXCITATION SYSTEM
116-186	0-1	TURBINE CROSS UNDER PIPE REPAIR
117-186	0	EXTRACTION STEAM PIPING AND FITTING MATERIAL UPGRADE
118-186	0	REACTOR CAVITY SEAL RING
119-186	0	10CFR50.49 ENVIRONMENTAL QUALIFICATION LIST REVISION
122-186	0	PRESSURIZER MISSILE SHIELD ACCESS LADDER SAFETY CAGE

DOCUMENTS REVIEWED FOR FSAR AMENDMENT 7

NUMBER	REVISION	TITLE
126-186	0	AFAS DVM MODIFICATION
132-186	0-1	EXCESSIVE AC/DC CONTROL VOLTAGE DROP
136-186	0	REPLACEMENT OF SG LEVEL TRANSMITTERS
139-186	0	MASONRY WALL MODIFICATIONS
140-186	0	ANNUNCIATOR NUISANCE ALARMS
141-186	0-1	PRESSURIZER LEVEL INSTRUMENTATION MODIFICATION
143-186	0	PASS DISSOLVED HYDROGEN ANALYZER TIE-INS
146-186	0	MISCELLANEOUS ICW SYSTEM MODIFICATIONS
148-186	0	ICW ISOLATION VALVE REPLACEMENT
151-186	0	480 VOLT AC LOAD CENTER 1B-2 TRANSFORMER COOLING
153-186	0	ROSEMOUNT AND VALCOR EQ ENHANCEMENT
001-187	0-1	IE BULLETIN 85-03 MOV SWITCH SETTINGS
[illegible]5-187	0-1	NRC IE BULLETING 85-03 MOV POSITION INDICATION
009-187	0	OVERPRESSURE MITIGATION SYSTEM MODIFICATION
010-187	0	PROTECTIVE COATINGS REPAIR/REPL IN REACTOR CONTAINMENT BLDG
011-187	0	CONDENSER HOTWELD NITROGEN INJECTION CONNECTIONS
012-187	0	FEEDWATER HEATER AND EXTRACTION PIPE SHIELDING
017-187	0	CCW HEAT EXCHANGER SHELL SIDE DRAIN ADDITION
027-187	0-4	MISCELLANEOUS SNUBBER MODIFICATION
028-187	0	REPLACEMENT OF RWT NOZZLE FOR LINE I-3"-CS-46
032-187	0	PRESSURIZER SURGE LINE SAMPLE VALVE V1210 REPLACEMENT
034-187	0	CONDENSER OUTLET TUBE SHEET AND WATER BOX COATINGS
035-187	0-1	REPLACEMENT OF RAYCHEM SPLICES
044-187	0	MSCV DISK NUT LOCKING PLATE MODIFICATION
046-187	0	CONTAINMENT BLDG TELESCOPING JIB CRANE SEISMIC RESTRAINTS

DOCUMENTS REVIEWED FOR FSAR AMENDMENT 7

NUMBER	REVISION	TITLE
087-187	0	CODE BOUNDRY DRAWING REVISION
097-187	0	TSC BLOCK WALL 207A MODIFICATIONS
108-187	0	TURBINE GANTRY CRANE MAIN SHEAVE NEST UPGRADE
008-984	0	NON-MANUAL PARKING LOT
227-984	1	TURBINE GANTRY CRANE SEPARATION REQUIREMENTS
178-985	0-1	TIE BETWEEN CONSTRUCTION FIRE MAIN AND PLANT FIRE LOOP
199-985	0	WATER TREATMENT PLANT CAUSTIC DILUTION WATER BOOSTER PUMP
015-986	0	TELEPHONE SYSTEM UPGRADE
112-986	0	TURBINE BUILDNG GANTRY CRANE GIRDER INSPECTION AND TB ISOL
130-986	0	NEUTRALIZATION BASIN CLOSURE MONITOR WELLS
N/A	0	ST LUCIE UNIT 1 CYCLE 8 SAFETY EVALUATION REVIEW

AIR DRYER PLUG VALVE REPLACEMENT

SYSTEM DESCRIPTION

FUNCTION

The purpose of this modification is to replace the lubricated plug valves on the instrument air dryer with non-lubricated types to prevent grease intrusion into the instrument air system.

DESIGN DESCRIPTION

The grease sealed plug valves originally installed on the instrument air dryer allow grease to contaminate the system and plug up the cooler causing the blower to trip on high current. Replacement of these valves with greaseless type valves will eliminate the problem.

OPERATION

The operation of the air dryer will remain the same. The valves are hand operated and utilized when the dessicant beds are being switched for regeneration.

SAFETY ANALYSIS

The instrument air dryer is Non-Safety Related Quality Group D. It is non-seismic and has no wind, flood, or missile criteria.

Failure of the instrument air dryer will not affect any safety related systems since the instrument air system can operate without the instrument air dryer and the instrument air system is not required for safe shutdown. The instrument air dryer is located in the turbine building so in the event of a catastrophic failure, no safety related equipment will be affected.

REACTOR AUXILIARY BUILDING JIB HOIST

DESCRIPTION

The implementation of this PC/M package will provide a 5 ton capacity jib hoist in the Reactor Auxiliary Building (RAB). The hoist will be located east of the hot machine shop as detailed on the location plan, JPE-C-40-82-001, included with this package. The hoist will be used to transfer equipment in the RAB between El -0.5' and El 19.5'. The boom shall be provided with a hold down chain to limit movement when not in operation.

This design package primarily outlines Civil's requirements. Electrical's requirements are included in Appendix C.

SAFETY EVALUATION

The jib hoist will be used to transfer equipment and materials in the RAB between El 0.5' and El 19.5'. The hoist performs no nuclear safety related function therefore this PC/M is non-nuclear safety related.

The design of the jib crane attachments to the concrete wall of the RAB will be verified based on seismic loads as well as dead and live loads. This will preclude detachment during a seismic event. The requirements of NUREG 0612 will not be applicable as there will be no interactions with safety related equipment.

All structural steel and bolting material will be required to have Certificates of Compliance with the applicable material specification. This will assure material integrity.

In accordance with QI Section 3.2, no unreviewed safety questions have been introduced.

DIESEL GENERATOR UPGRADE

ABSTRACT

PC/M 92-182 was generated to perform several modifications recommended by the diesel generator vendor to upgrade our diesels with the latest design improvements for optimal reliability. The modifications included:

1) Vibration damper replacement
2) Idler gear stubshaft replacement
3) Exhaust screen inspection port addition
4) Air dryer addition
5) Lube oil modification

Items 1 through 4 were installed in 1983 per the PC/M package. Item~~s~~ ~~4~~ 5 was never installed due to difficulty in procuring parts and escalating cost.

A summary of PC/M ~~82-82~~ 92-182 is provided below.

Supplement 0 - Installed the vibration damper, idler gear stubshaft, and exhaust screen inspection port. Engineering was by FPL.

Supplement 1 - Installed the Electrical, CIvil and I&C portion of the air dryer addition and the lube oil modification. Engineering was by Ebasco.

Supplement 2 - Installed the mechanical portion of the air dryer addition. Engineering was by FPL.

Supplement 3 - Transmitted the vendor wiring diagrams for the air dryers. Engineering was by Ebasco.

Supplement 4 - Installed the mechanical portion of the lube oil modification. Engineering was by FPL.

Supplement 5 - Revised supplement 4 based on additional vendor submittals. Engineering was by FPL.

Supplements 0, 2 and 3 were fully installed. Only a portion of Supplement 1 was installed and none of supplements 4 and 5 was ever installed. Supplements 0, 1, 2, and 3 were appropriately as-built.

The scope, engineering and materials of supplement ~~5~~ 6 were all included in the original lube oil modification. Supplement 6 breaks out all of the portions required to implement the turbocharger soakback low pressure alarm and deletes the remainder of the drawings not yet as-built. When supplement 6 is installed, and as-built, the PC/M can be closed out.

This supplement does not affect the original safety analysis, does not require any technical specification changes and does not cause any changes to the operation of the diesel generator. THEREFORE, PRIOR NRC APPROVAL IS NOT REQUIRED FOR IMPLEMENTATION OF THIS EP.

SAFETY ANALYSIS

The addition of a turbocharger soakback low pressure alarm does not involve an unreviewed safety question since the alarm does not affect the operation or operability of the diesel generators, it merely alerts operators to a low pressure condition in the turbocharger soakback system while the engine is in the standby mode. As a result, it can be concluded that the probability of occcurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR has not been increased.

The possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR has not been created since the diesel generators are not considered in determining the probabilities of accidents and since the alarm was designed to the appropriate codes and it does not affect the operation of the diesel generators.

The margin of safety as defined in the basis for any Technical Specification has not been reduced since the modification does not affect the performance or operability of the engine.

REMOVAL OF STEAM GENERATOR DELTA P SIGNAL CHARACTERIZERS

SYSTEM DESCRIPTION

The system which is being affected by this modification is the Steam Generator Reactor Coolant Differential Flow Instrumentation and Control System. This system monitors the Reactor Coolant flow across each of the two Steam Generators with the purpose of protecting the core against departure from nucleate boiling DNB in the event of a coolant flow decrease.

Flow measurement signals are provided by summing the output of differential pressure transmitters across each steam generator. This measurement of differential pressure is modified by two signal characterizers in each of the four channels to provide square root of differential pressure signals which correspond to actual flow. The low flow reactor trip is actuated directly by the summed flow signal. It requires a two-out-of-four coincidence logic from the four independent channels (when the flow falls below a preselected value) to initiate a Reactor trip.

In addition, four separate indicators (one per channel) receive signals from each instrumentation loop. By means of a selector handswitch the operator can read either the summed, average flow signal of the two steam generators, as measured from that channel, or the flow signal of either steam generator by itself.

The purpose of this modification is to remove from the instrumentation and control loops the eight signal characterizers (two per channel). Because technical specifications limit plant operations to four Reactor Coolant Pumps, there is no longer a need for the characterizer function.

The characterizer modifies the delta p input signals to allow for operation under conditions of less than four reactor coolant pumps.

The characterizers will remain physically in the loops, but will be bypassed electrically so that they will serve no function. The indicators will be recalibrated and their scales will be changed to read the delta p signals directly rather than the modified flow signals. The wiring in the loops will be altered and several resistors will need to be replaced with new values to maintain the required voltage ranges.

Fuel Resources has completed an engineering evaluation that will generate the new voltage tripping setpoints for the System.

SAFETY ANALYSIS

This modification is Nuclear Safety related because the Steam Generator Reactor Coolant Delta Flow Instrumentation and Control system is Nuclear System related, and the equipment being modified is part of this system.

This PC/M proposes to remove the signal characterizers from the four redundant loops and modify the instrument and control circuits so that the loop current and voltage signals become proportional to the differences in reactor primary coolant pressures, rather than directly to the flow across the steam generators. In accordance with the Fuel Resources recommendations (Appendix C), "both steam generator delta p signals are summed and a reactor trip is initiated when the summed delta p falls below a preselected trip value." Accordingly, the transmitters in the loops will be recalibrated and the indicators rescaled for the process range of 0 to 50 psid.

The characterizers served a function under condition of less than four reactor coolant pump operation. However, because St. Lucie #1 Technical Specifications limit plant operation while at power to four reactor coolant pumps, there is no longer a need for the characterizer functions. Combustion Engineering, the original Architect Engineer for the Reactor Protection System, has recommended removal of the characterizers because the removal will reduce the calibration time of the delta p signal processing circuitry.

The change is not an unreviewed safety question because:

The probability of occurrence or the consequences of an accident or malfunction important to safety previously evaluated in the FSAR has not been increased. Fuel Resources has completed an engineering evaluation that will be used to generate the new voltage tripping setpoints for the Reactor Coolant delta pressure inputs for the Reactor Protection System based on removal of the characterizers. New values of resistors will be required to maintain the voltage ranges required by the circuitry in the absence of the signal characterizers. The new resistors are being purchased as commercial grade items to exacting standards of precision and performance.

Additionally, failure of the new resistors would have the same consequences on the Stream Generator delta p input to the Reactor Protection System as failure of the existing resistors. Hence, no possibility for an accident or malfunction of a type different from any evaluated previously in the FSAR has been created by this modification.

For the same reasons, the margin of safety, as defined in the bases for the technical specifications has not been decreased.

In conclusion, this modification does not involve an unreviewed safety question.

REACTOR BUILDING UPPER CAVITY HANDRAIL

INTRODUCTION

The reactor building refueling cavity is a stainless steel lined pool with a split level bottom at Elevations 36.0' and 21.5'. During plant shutdown, the upper cavity is occupied for various operations including reactor head removal, head stud detensioning, etc. At the present time there is no provision along the edge of the upper cavity to prevent personnel from falling into the lower cavity. This PC/M provides a handrail in this area.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The handrail provided by this PC/M has been designed to withstand those loading combinations as specified in FSAR Chapter 3. Although the handrail serves no safety related function, seismic loads have been considered in the design. In order to preclude potential missile generation during a LOCA, the handrail has been designed to withstand jet impingement loads.

There are no unreviewed safety questions associated with this PC/M and prior Commission approval is not required for implementation.

NITROGEN SUPPLY SYSTEM

SYSTEM DESCRIPTION

Functions and Design Requirements

Function

The Nitrogen System supplies low and high pressure nitrogen to various systems and vessels which require cover gas for St. Lucie Units #1 and #2.

a) Low pressure nitrogen (200 psig and below) is supplied to the following for each unit:

Spent Resin Tank
Volume Control Tank
Refueling Failed Fuel Detector
Reactor Drain Tank
Quench Tank
Pressure Reduction and Sample Cabinet
Waste Concentrator
Boric Acid Concentrators
Flash Tank
Hold Up Tanks

b) High pressure nitrogen (over 200 PSIG) is supplied solely as cover gas for the safety injection tanks.

Design Requirements

The system shall be capable of supplying both units with high and low pressure nitrogen for 30 days without refilling. The high pressure storage volume shall be of sufficient capacity to recharge all four safety injection tanks on either unit. The nitrogen compressors shall be of adequate size to recharge the high pressure storage volume within 24 hours each or 12 hours combined.

SAFETY ANALYSIS

The nitrogen supply system provides a high pressure source and normal operating source of nitrogen gas for plant use. The nitrogen system serves no safety function. The nitrogen supply system is located outdoors adjacent to the gas house. Failure of any portion of the nitrogen supply system will not inhibit the ability to mitigate the consequences of a postulated accident, achieve safe shutdown, or adequately cool spent fuel. This change is therefore not considered safety related and does not involve an unreviewed safety question.

All piping has been designed in accordance with ANSI B-31.1. The concrete foundations have been designed in accordance with all applicable codes. The power sources for the nitrogen compressors are from non-safety related MCC's 1A-3 and 1B-3. All conduits utilized are below the allowable fill. In addition, qualified cable will be used for this installation.

INSTRUMENT AIR SYSTEM CROSS TIE CHECK VALVE ADDITION

SYSTEM DESCRIPTION

Operation

This modification adds one check valve to the cross-tie line to prevent accidental bleed-down of the IA system. This valve performs its function automatically and requires no operator action. The modification also adds a differential pressure indicator to the coalescing filter. The differential pressure should be monitored on a periodic basis to determine when filter element replacement is necessary. (This should be done at 10 PSID) Operating procedures should be modified to indicate this limitation.

Function

This modification will provide a passive means of preventing bleed-down of the IA system by way of leakage to the SA system while the Unit #1/Unit #2 IA cross connect is in use or the Unit #1 SA to IA cross-tie is in use.

Design Description

This modification provides the details for installation of one check valve in the Unit #1 SA to IA cross-connect line and addition of a differential pressure indicator on the coalescing filter.

SAFETY EVALUATION

1.0 This modification is non-seismic, non-safety related and does not involve an unreviewed safety question because:

- 1.1 The IA/SA cross-tie performs no safety function and is classified as non-safety related.
- 1.2 These modifications do not interact with any safety related system or components.
- 1.3 No safety related equipment or components are compromised by any assumed failure of any existing or new equipment or components.
- 1.4 No Technical Specifications are altered or adversely affected.

2.0 Care has been taken in the design to eliminate or control aspects which could be hazardous to equipment and/or personnel.

CCW HEAT EXCHANGER RETUBING

SYSTEM DESCRIPTION

Function

The component cooling water (CCW) heat exchangers are one side mixed, one side unmixed single pass horizontal mount units with intake cooling water (ICW) on the tube side and CCW on the shell side.

The basic function of the CCW heat exchangers is to reject heat from various components associated with the reactor support and safety equipment (such as the shutdown heat exchangers, containment fan coolers and RCP seal coolers) to the ICW system.

Design Description

Due to the deteriorated condition of the existing CCW heat exchanger Aluminum-Brass tubing, these units are to be retubed during the next refueling outage. This design package includes the specification required for the procurement of the tubes and guidelines for the retubing effort.

Operation

The component cooling system (CCS) is arranged as two redundant essential supply header systems (designated A & B) each with a pump and heat exchanger and the capability to supply the minimum safety features requirements during plant shutdown or LOCA conditions.

SAFETY ANALYSIS

1a. With respect to the probability of occurrence of an accident previously evaluated in the FSAR:

The probability of occurrence of an accident previously evaluated in the FSAR has not been increased since this PC/M does not involve a design change.

b. With respect to the consequences of an accident previously evaluated in the FSAR:

The consequences of FSAR accident evaluations have not been altered since this PC/M does not involve a design change.

c. With respect to the probability of malfunction of equipment important to safety previously evaluated in the FSAR:

The probability of any equipment malfunction has not increased for the reasons outlined in 1a.

d. With respect to the consequences of malfunciton of equipment important to safety previously evaluated in the FSAR:

The consequences of any equipment malfunction has not been increased for the same reason outlined in 1b.

2a. With respect to the possibility of an accident of a different type than analyzed in the FSAR:

There is no possibility for an accident of a different type than analyzed in the FSAR since this PC/M does not involve a design change or a new design.

b. With respect to the possibility of a malfunction of a different type than analyzed in the FSAR:

There is no possibility of a malfunction of a different type than analyzed in the FSAR for the same reason as given in 2a.

3. With respect to the margin of safety as defined in the basis for any Technical Specification:

No margin of safety has been decreased or altered for the reason given in 2a.

SECURITY CONSOLE - LED GRAPHIC DISPLAY

INTRODUCTION

The NRC has determined that annunciation of the Security System power supplies is required for compliance with 10CFR Part 73 (i.e. requirements for security systems for nuclear power plants). To meet the intent of this requirement, status lights shall be installed on the security system alarm consoles to indicate the "at hand" condition of the power input to the security SUPS and therefore, to the entire security system.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The security system is a non-safety related plant system. The Central and Secondary Alarm Stations are components of this system. The modifications presented in this PC/M affect both safety and non-safety related plant equipment.

The modifications to the CAS and SAS control panels, i.e. installation and wiring of the annunciator circuitry, and the inputs to these annunciators are non-safety related. The alarm stations are located in the Turbine and Security Building, which are non-safety, non-seismic structures. The majority of required cable to these areas will be routed in non-safety related cable tray in the TGB and through the dedicated security duct banks to the Security Building. The balance of cable will be routed through appropriately dedicated raceway.

Diesel generator breaker position is monitored to provide input to the "Normal" and "Diesel" annunciator circuits. This portion of the diesel generator control circuitry is safety related. Therefore, this signal will be isolated from the non-safety security annunciation circuitry by installing safety related isolation relays in 4160V switchgears 1A3 and 1B3. These relays were purchased under RPA 432953 to be qualified to the applicable industry standards.

The balance of the control relays that are required in this modification have been purchased and will be installed as non-safety related equipment.

Control power to all relays is from the associated plant power train (safety to isolation relays, non-safety to the non-safety control relays). All cables will be routed through the appropriate raceway and the raceway will be seismically supported as required (i.e. inside the RAB).

This modification has no impact on the plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the basis that this change does not involve any unreviewed safety question, therefore prior Commission approval is not required for implementation of this PC/M.

MICROWAVE DETECTOR REPLACEMENT

INTRODUCTION

The existing Microwave 700 series Intrusion Detection System which is installed at the Unit 1 and 2 perimeter fence, will be replaced with Stellar's Model 682 E-Field. The replacement of the Microwave 700 series at Units 1 and 2 will be covered by PC/Ms 412-183 and 54-283, respectively.

PC/M 412-183 addresses Zones 25, 30, 31, and 32. Please note FPL Start-Up has previously installed the Stellar E-Field system at Zones 31 and 32 in a test-bed application. PC/M 412-183 will document these zones as a permanent installation. There is no construction impact for these zones.

As previously agreed, the change out from Microwave to E-Field in the area of the Discharge Canal has first priority. This will eliminate the need for the existing security station, which is manned 24 hours a day.

Therefore, Supplement 0 of PC/M 412-183 addresses the Unit 1 perimeter zones at the Discharge Canal (Zones 30, 31, and 32). Zone 25 will be modified via Supplement 1 to PC/M 412-183.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The security system is non-safety related. Furthermore, the modifications to the perimeter intrusion detection system presented by this PC/M will improve the overall security system operation. The change out from microwave to E-Field is required in order to eliminate the need for a security station at the Discharge Canal which is manned 24-hours a day.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the basis that this change does not involve an unreviewed safety question; therefore, prior Commission approval is not required for implementation of this PC/M.

NON-MANUAL PARKING LOT

ABSTRACT

This engineering package covers the restoration and repaving of the Non-Manual Parking Lot which is on the east side of the St. Lucie Plant. Also included in the package is the removal of the construction fire water tank in the parking lot, as well as the addition of an improved area lighting design. The parking lot is located outside of the plant security fence perimeter. The modifications included in this design package will not affect any plant safety-related system and are therefore classified as non-nuclear-safety-related. In addition, the removal of the construction fire water tank will not affect the plant fire protection system, since this work cannot be started until PCM 178-985 is implemented. PCM 178-985 ties the fire water piping downstream from the tank into the plant system. The restoration and repaving of the Non-Manual Parking Lot and the tank removal do not pose any unreviewed safety questions.

Safety Evaluation

The Non-Manual Parking Lot is located outside of the security perimeter fence and will not be in the vicinity of any plant safety-related structure or system. It does not in any way perform or affect a plant safety-related function.

The Non-Manual Parking Lot area lighting does not perform or affect any plant safety-related systems or function. It is being supplied from LP 260 which is a non-safety related lighting panel and is not loaded on the emergency diesel generator.

The removal of the construction fire water tank and piping does not affect any plant safety-related system or functions. The city water to the plant is not a safety-related system. The fire water supply from the tank is not part of the plant fire water system and does not affect that system.

The modifications to the Non-Manual Parking Lot do not change any assumptions made or conclusions drawn in the St. Lucie FSAR. The repaving of the lot does not adversely affect any site topographic features.

For the above reasons the modifications of the Non-Manual Parking Lot will not increase the probability of occurrence nor the consequences of a design basis accident or malfunction of equipment important to the safety of the plant. Additionally, there will continue to be no possibility of an accident or malfunction different than those already evaluated in the FSAR. Finally, the margin of safety as defined in the Plant Technical Specifications has not been reduced. It is therefore concluded that this modification does not pose an unreviewed safety questions pursuant to 10 CFR 50.59 and does not affect any technical specifications.

NOTE: THIS PACKAGE CONTAINS SAFEGUARD DRAWINGS.

MAIN STEAM SAFETY VALVE LIFT LEVER REMOVAL

SYSTEM DESCRIPTION

1.0 Design Description

On February 9, 1984, a PSL-2 plant trip caused actuation of the Main Steam Safety Valves (MSSV's). Following the transient, one of the MSSV's failed to reseat because the absence of the spindle nut cotter pin allowed the spindle nut to rotate down onto the forked lift lever and prevented full travel of the spindle rod. To prevent this type of event from occurring again, this design package allows the removal of the lift lever components (spindle nut, cotter pin and fork lever). A new valve cap is required to maintain ASME Code requirements concerning cap sealing.

2.0 Function

The lift lever assembly provides the capability to manually exercise the safety valve to facilitate blowing out lines. While the original component design specification required a lifting lever, the Code requirement for lift levers on Class 2 valves has been eliminated. The purchaser of the safety valves, C-E agrees that these lift levers can be removed.

The original valve design utilized the lifting lever assembly to seal the valve cap and prevent unauthorized adjustments of the valve's set pressure. Cap sealing is required by the Code. In order to maintain this requirement, new valve caps with an integral sealing device are required.

3.0 Operation

Removal of the lift levers will not impact the intended operation of the MSSV's. The MSSV's will still be capable of relieving secondary side overpressure events. However, if the need arises to manually exercise the valve, the cap will have to be removed and the lift levers reinstalled.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (1) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety defined in the basis for any technical specification is reduced.

This modification allows for the removal of the MSSV lift levers and replaces the existing valve caps with ones that will enable proper Code required sealing of the valve adjusting bolts. This modification will not in any way impact or jeopardize the intended operation of the MSSV's or their ability to provide overpressure protection for the secondary side of the steam generator.

The probability of occurrence of an accident previously evaluated in the FSAR (excessive heat removal from the steam generator caused by a stuck open safety valve) will be significantly reduced by removing the lift lever components. Also the probability of a malfunction of equipment important to safety previously evaluated in the FSAR is also reduced by this modification. No other type of accident or malfunction not previously evaluated in the FSAR is created. In addition, it does not reduce the margin of safety as defined in the bases for any technical specifications. A change to plant Tech. Specs. is not required.

The foregoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

EDG CFD RELAY REPLACEMENT

SUMMARY STATEMENT

Summary

This PC/M provides the design for replacement of the existing emergency diesel generator CFD current differential relays.

Safety Evaluation

This PC/M improves the fragility level of the D/G differential circuit by reducing the probability of relay trip due to mechanical vibration. This modification is accomplished solely by replacing the existing differential relays and cases, while implementing no internal or external wiring changes in the diesel generator control panel. This then precludes any new type of interaction with other safety related equipment. Therefore, this PC/M is nuclear safety related but does not involve an unreviewed safety question.

SAFETY ANALYSIS

This PC/M improves the vibrational fragility of the diesel generator differential relay circuit. This is accomplished by replacing the existing CFD differential relays with IJD differential relays. This makes the circuit less susceptible to spurious trips caused by control panel vibration. This modification does not adversely affect the normal operation of the diesel generator differential circuit or operation of the diesel generator.

The replacement differential relays have been seismically tested by the vendor as stated in the General Electric MIL. NO. 82-12. The seismic response spectrum (TRS) for the instrument envelopes the St. Lucie Unit 1&2 Envelope Response Spectrum for the Diesel Generator Building (RRS) for mass point #3 (elev. 23.0') which corresponds to the instrument location. A damping factor of 5% was used in the TRS in accordance with IEEE Std-344-1975. Consequently, a 4% damping factor was conservatively used in the RRS for comparison of acceleration values. The acceleration values required by the RRS are a minimum of 2.8 times less at all frequencies than those of the TRS in the non-operate mode, and 4.9 times less for the operate mode. This will sufficiently account for any acceleration amplification through the relay cabinet. In addition, the size, weight, mounting location and mounting details of the replacement relays will be the same as that of the original relays. Therefore, the original relay support and cabinet design criteria will not be adversely affected.

This modification does not affect any cable tray analysis, Appendix "R" analysis or any other safety related equipment as it only involves replacement of relays on the diesel generator control panel.

With respect to the probability of occurrence or the consequences of a design basis accident or malfunction of equipment important to safety as previously evaluated in the FSAR:

The PC/M simply involves replacing the CFD differential relays and cases, which means only a slight change of internal relay contacts without changing the relays' function or external wiring. This PC/M does not affect the operation of the emergency diesel generator as discussed in FSAR 8.0., nor does it adversely affect the operation of the D/G or any other safety related equipment. The design basis in the FSAR (Chapters 8 and 15) which addresses the operation of the D/G, is an event involving the loss-of-offsite power. The evaluation of this design basis is not affected by this PC/M.

With respect to the possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR:

This PC/M just involves replacing the existing diesel generator differential relays with another model to improve the vibrational fragility of the differential circuit and therefore, does not create the possibility of the above.

With respect to the margin of safety as defined in the basis for a Technical Specification:

As this PC/M does not adversely affect operation of the emergency diesel generator, it does not change the margin of safety as defined in the basis for the Diesel Generator Technical Specification 4.8. Additionally this PC/M does not change the margin of safety as defined in the basis for any Technical Specification as it only involves changes in the diesel generator control panel.

Therefore it can be concluded that this PC/M does not involve an unreviewed safety question.

TURBINE GANTRY CRANE PROXIMITY RESTRICTIONS

INTRODUCTION

This PCM provides restrictions on the proximity of the Units 1 and 2 turbine gantry cranes to each other in order to prevent potential overstressing of the turbine building structure.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The turbine building is a non-Category I structure and contains no safety related equipment. This PCM imposes a restriction of the proximity of the turbine building gantry crane to each other. This restriction assures that the assumptions used in the building design remain valid.

Therefore the implementation of this PCM will not increase the probability of any accident previously evaluated. Implementation of this PCM does not involve a change to the St Lucie Unit 2 Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the basis that this change does not involve any unreviewed safety question, therefore prior Commission approval is not required for the implementation of this PCM.

EXCORE NEUTRON FLUX MONITORING SYSTEM

INTRODUCTION

PC/M 228-184 provides the method to install the Ex-Core Neutron Flux Monitoring System that monitors wide and source ranges neutron flux with independent displays in the Control Room and Hot Shutdown Panel (HSP). This system meets the FP&L commitment to the NRC by providing wide range neutron flux monitoring in the hot shutdown panel which is independent of the Control Room and required by 10CFR50 Appendix R. It also satisfies the guidelines of Regulatory Guide 1.97 Rev. 3 by providing a neutron monitoring system to meet the definition and requirements for category 1 variables as indicated in the Regulatory Guide. In addition this system permits the continuity of refueling activities in the event of a failure of the existing start-up excore detectors during refueling. Technical Specifications 3/4.9.3 Refueling Operations, requires the suspension of operations involving core alterations or positive reactivity changes if at least two start-up detectors are not operational.

As a result of a malfunction of one of the two fission chambers in the Neutron Detector Assembly Channel SB, during the system commissioning, it is necessary to replace the malfunctioning detector. The replacement detector is similar to the one replaced except that it is environmentally qualified for 10 year life plus Design Basis Accident.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The Excore Neutron Monitoring System is a Nuclear Safety Related System designed to meet the requirements of 10CFR50 Appendix R by providing independent source range and wide range neutron flux monitoring in the Control Room and HSP. It also satisfies the guidelines of Regulatory Guide 1.97 Rev 3 for category 1 variables. These components are seismically and environmentally qualified to the requirements of IEEE 323-1974 and 344-1975. The inside and outside containment cable is qualified to IEEE-383-1974.

The replacement detector and its integral cable assembly, of the failed detector, is also qualified to IEEE-323-1974, IEEE-344-1975 and IEEE 383-1974, however its qualified life is 10 years normal operation plus Design Basis Accident, therefore this replacement detector is considered only a temporary replacement.

The Excore Neutron Monitoring System is a post-accident monitoring system that provides a neutron source range monitoring signal which is redundant to the existing non-post accident qualified excore neutron detector system.

The installation of the Excore Neutron Monitoring System does not increase the probability of occurrence of an accident or malfunction of equipment important to safety previously evaluated since the system performs only monitoring functions, it is seismically and environmentally qualified; and it is not interlocked with any safeguards system. It does not create the possibility for an accident or malfunction of a different type than any evaluated previously since it is an independent/redundant system designed to meet the requirements of Regulatory Guide 1.75. The instrumentation added to the existing boards (RTGB-104, PACB and HSP) has been evaluated by Ebasco and does not have any significant impact on the previous seismic qualification of the boards.

The margin of safety as defined in the technical specifications is not reduced. In fact the additional source range instrumentation added in the Control Room will permit continuity in refueling operations in the event of failure of the existing excore detectors by satisfying the requirements of Technical Specification 3.4.9.2.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the basis that this change does not involve any unreviewed safety question, therefore prior Commission approval is not required for implementation of this PC/M.

TRAVELING WATER SCREEN UPGRADE

INTRODUCTION

Recent experiences of excessive jellyfish runs have caused damage to the intake traveling water screens. This PCM replaces the existing traveling water screen frames and baskets with new, high strength, braced sections that will increase the strength of the screens to a differential pressure head capacity of 15 feet. In addition, the replacement baskets will be furnished with intermediate and lower lifting lips having serrated edges to impale debris which may slip out of the baskets on their upward travel. As a result of these changes, the screens will have an improved chance of surviving a future influx of jellyfish. The head sections, footshaft assemblies and carrying chains have also been replaced with upgraded components to further extend the useful life of the screens.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This PCM provides the details for modifications/improvements to the existing intake traveling water screens. The screens are fabricated and designed by Envirex Incorporated in accordance with Ebasco Specification FLO 8770.760, which requires that the screen be non-seismic and non-safety related. This PCM increases the strength of the screens to a differential pressure head capacity of 15 feet. The strengthening of the screens does not alter the conditions to which they may be subjected but will reduce the damage to the screens in the event of an excessive differential pressure condition. The new loads imposed on the existing embedded guide slots as a result of the screen redesign have been reviewed and are acceptable.

Each traveling screen is presently furnished with a pneumatic differential water level controller for automatic operation. Initially, the screen wash pump and rotation of the screen will automatically start when a specified differential pressure head loss is reached. When the differential pressure increases beyond this point, the control room is alerted via an annunciator which receives its input from differential pressure indicator transmitters. The modifications issued via this PCM do not alter in any way the operation of the existing control system.

Modification issued via this PCM will not impact the minimum submergence levels from the Intake Cooling Water (ICW) pump or the Circulating Water (CW) pump. In addition, this PCM will not compromise the operation or safety of these pumps.

Therefore, the implementation of this PCM will not increase the probability or consequences of any accident previously evaluated nor does it create any new types of accidents. Implementation of this PCM does not involve a change to the St Lucie Unit 1 Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the basis that this change does not involve an unreviewed safety question, therefore prior Commission approval is not required for implementation of this PCM.

DIESEL GENERATOR AIR START SYSTEM PIPING MODIFICATION

PC/M SCOPE

System Description

Each diesel generator (D/G) set at St. Lucie Unit #1 has an independent air starting system. Each system has four skid mounted air receivers which provide sufficient air charge for ten starts per diesel generator set. The air receivers are pressurized by an electrically driven compressor via an air dryer. A diesel driven air compressor is used as a backup to the electric driven compressor. The charging air flows to the number 3 and 4 air receivers in parallel and then to the number 1 and 2 air receivers through the outlet air header.

Design Description

The air receiver charging configuration will be modified by installing a branch off the air dryer outlet piping to the number 1 and 2 air receivers (1A1, 1A2, 1B1 and 1B2). The new line will enable the compressors to charge the number 1 and 2 air receivers in parallel with the number 3 and 4 air receivers. The existing diesel driven air compressor discharge check valves (V-17355A) will be relocated to be in the new charging line. New check valves will be installed at the discharge of all the air compressors. A flexible connector will be installed in the charging line to all the air receivers in order to limit the boundary of the seismic stress analysis. Drawings JPE-15-185.001, .002, .003 and .004 show the before and after D/G air start schematics.

SAFETY ANALYSIS

The proposed change isolates the two air receiver sets (and associated air start motor sets) per D/G engine by modifying the inlet air headers and closing the outlet air header cross-connect valve. The modification provides further separation and redundancy than required by the St. Lucie Unit 1 Final Safety Analysis Report (FSAR). Each emergency diesel engine starting system must be independent and physically separated from other systems serving the redundant diesel generator, such that a single failure in any one of the systems will affect only the associated diesel generator.

This modification will make each D/G engine air start motor sets independent and physically separated. Therefore, a failure associated with one of the two air receiver sets per D/G engine will not disable the air start system for the that D/G engine. This redundancy is above and beyond what is required by the FSAR, Reg. Guides or General Design Criteria for St. Lucie Unit #1.

The modification does not involve an Unreviewed Safety Question because:

1) The probability of occurrence of a design basis accident or malfunction of equipment important to safety is decreased because the air start motor sets and associated air receivers and piping are independent and separated.

2) The consequence of a malfunction of equipment important to safety is decreased because a single equipment malfunction will not disable both air start motor sets on a D/G engine.

3) The possibility for an accident or malfunction of a different type that any preciously evaluated in the FSAR is not created because the air start system operates functionally the same as before without intedependence between air motor sets.

4) The margin of safety as defined in the basis of a Technical Specification is increased, not reduced, because greater reliability to start both D/G's is provided.

This PC/M is classified as a Nuclear Safety Related change because the air receiver inlet headers up to the check valve are required to withstand all design basis events, to preclude an air receiver set from losing sufficient air pressure for ten cold starts of a Diesel Generator. The piping upstream of the air receiver inlet header check valve is not Nuclear Safety Related because the ability to charge the air receivers during or after a design basis event is not required per the FSAR.

The specific pipes which are attached to the new supports are non-safety related. As such, these supports are classified non-safety related, QA/QC required. The pipe supports have been seismically designed so that they will not interact with nearby safety related equipment or piping in the event of an earthquake.

INSTRUMENT AIR UPGRADE TIE-INS

SYSTEM DESCRIPTION

1.0 Operation

The Instrument Air (IA) tie-ins provided by this modification will have isolation valves which should remain closed at all times until the balance of the instrument air modification is implemented. These valves should be included in the valve alignment table in Operating Procedure 1-1010020.

2.0 Function

This modification functions to provide all outage related tie-ins to the IA System so that installation of new compressors and dryers can be completed during plant operations. Note that electrical tie-ins can be installed during any plant operating mode.

3.0 Design Description

This modification provides tie-ins for cooling water to the new compressors. The cooling water will be from the local Turbine Cooling Water (TCW) System supply and return headers. The modification also provides tie-ins to the IA System just upstream of the existing air receiver. Each tie-in will consist of an isolation valve, piping, pipe cap and the necessary materials to tie into the existing piping.

SAFETY EVALUATION

1.0 This modification is non-seismic, non-safety related and does not involve an unreviewed safety question because:

- 1.1 The TCW and IA Systems perform no safety function and are classified as non-safety, non-seismic Quality Group D.
- 1.2 These modifications do not interact with any safety related systems or components.
- 1.3 No safety related equipment or components are compromised by any assumed failure of any existing or new equipment or components.
- 1.4 No Technical Specifications are altered or adversely affected.

2.0 Care has been taken in the design to eliminate or control aspects which could be hazardous to equipment and/or personnel.

ELECTRIC PENETRATION ASSEMBLIES' INSTALLATION

ABSTRACT

This engineering package covers replacement of certain Electric Penetration Assemblies (EPAs) at the St Lucie Plant, Unit No 1. These modifications consist of:

1. Removal of five existing EPAs manufactured by Gulf General Atomic Company, EPA Designation Nos C8, D1, D2, D3 and D9. These EPAs are used for low voltage power and control circuits.

2. Installation of six new EPAs manufactured by Conax Buffalo Corporation. These new EPAs will be installed in the containment nozzles itemized above and spare containment nozzle C9.

3. Recircuiting of all circuits presently connected to No 4, No 8 and No 12 AWG modules associated with all the Gulf General Atomic Company EPAs installed at Unit No 1. In addition to the five removed EPAs, this modification affects nine other existing Gulf General Atomic EPAs, totalling approximately 400 circuits. The affected circuits have been provided in Attachments 4 and 5.

The EPAs are classified as Safety Class 2 (Quality Group B), Class 1E, seismic Category I components and perform a safety related function. Therefore, this PC/M is considered nuclear safety related. The implementation of this PC/M does not require a change to the plant Technical Specifications. The modifications do not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

The probability of occurrence or the consequences of an accident or malfunction is not increased since:

EPAs are being supplied by Conax Buffalo Corporation, who has supplied previously five EPAs installed in the St Lucie Plant, Unit No 1, as well as all the EPAs for Unit No 2. These EPAs have been environmentally and seismically qualified in accordance with IEEE 317-1976 for application in both St Lucie Plants.

The design, fabrication, test, inspection, installation, and qualification of the EPAs are in accordance with IEEE 317-1976 and the ASME B&PV Code, Section III, Subsection NE for Class MC Components. As discussed in the Design Analysis, all ratings associated with the new equipment meet or exceed the requirements for the application.

New wireway is manufactured by Hoffman, vendor of the existing EPA wireway system. Wireway installation has been analyzed as suitable for the application as discussed in the Design Analysis.

All new cables are qualified to the flame test requirements of IEEE 383-1974 and, except for a few CEDM circuit jumpers, are suitable for Class 1E use, as is the Raychem splice material. As discussed in the Design Analysis, the CEDM jumper cable is suitable for use under the St Lucie Plant normal conditions and is de-energized immediately upon reactor trip. Special consideration of the CEDM circuit conductors is consistent with the present St Lucie Plant, Unit No 1 FSAR (Reference Section 3.11.5.4).

The possibility for an accident or malfunction of a different type than previously analyzed is not created since:

The new EPAs are qualified to the required environment and will be able to perform their intended safety function post DBA.

The only modification to plant components is recircuiting to the new EPAs. The components being thus modified have been reviewed to assure that their removal from service for recircuiting does not violate technical specifications nor impact required plant systems.

Containment vessel structural integrity is unaffected by this modification, as discussed in the Design Analysis. After the installation of the EPAs, the overall containment boundary integrity will be verified via a 10CFR50, Appendix J, Type A leak rate test. In addition, a local Type B leak test will be performed on each EPA to verify the integrity of the electric feedthrough seals.

Configuration of the interface between the existing plant cable and the new equipment is consistent with the design bases established in the St Lucie Plant, Unit No 1 FSAR. This insures that the consequences of all analyzed accidents remain unchanged. Also, no new accidents or malfunctions are introduced by this modification.

The margin of safety as defined in the bases for any technical specification is not reduced since Special Instructions have been provided in Section 9.0 so the implementation of this modification does not violate the St Lucie Plant, Unit No 1 Technical Specifications. Therefore, the implementation of this PC/M does not require a change to the plant's Technical Specifications.

A holdpoint has been established for the rework of EPA C3 conduits and the termination locations in the RCB vertical termination boxes. This holdpoint will be removed upon access to the RCB and field verification of the existing installation. Pending the release of the hold, a revised safety evaluation will be provided.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

DIESEL GENERATOR SUBSYSTEM FLOW DIAGRAMS

Modification Description

This PC/M releases the new Diesel Generator Subsystem Flow Diagrams to the site. The following activities must be completed before the new flow diagrams can be issued as permanent plant drawings:

1.) All valves and instruments must be tagged in the field as per the new flow diagrams.

2.) Affected operating procedures must be reviewed to determine if revision is required to reflect the new tag numbers or flow diagram numbers.

Safety Analysis

1a. With respect to the probability of occurrence of an accident previously evaluated in the FSAR:

Flow diagrams are not considered in evaluating FSAR accidents.

1b. With respect to the consequences of an accident previously evaluated in the FSAR:

Flow diagrams are not considered in evaluating FSAR accidents.

1c. With respect to the probability of malfunction of equipment important to safety previously evaluated in FSAR:

Flow diagrams are not considered in determining the probabilities of safety related equipment malfunctions.

1d. With respect to the consequences of malfunction of equipment important to nuclear safety previously evaluated in the FSAR:

Flow diagrams are not considered in determining the probabilities of safety related equipment malfunctions.

2a. With respect to the possibility of an accident of a different type than analyzed in the FSAR:

Flow diagrams are not considered in evaluating FSAR accidents.

2b. With respect to the possibility of a malfunction of a different type than analyzed in the FSAR:

Flow diagrams are not considered in determining the probabilities of safety related equipment malfunctions.

3. With respect to the margin of safety as defined in the basis for any technical specification:

Flow diagrams do not impact technical specification safety margins.

Based on the above, the new flow diagrams and the tagging/retagging of diesel generator valves and instruments are determined not to involve an unreviewed safety question. There are no system modifications involved.

G.E. SAM RELAYS P.C. CARD REPLACEMENT

Introduction

The General Electric type Sam 11B utilized at St. Lucie Plant are D-C operated timing relays that employ solid-state components to provide an extremely stable time delay function. This relay finds application wherever short D. C. operated timing functions are required. Specifically this relay is used for circuit breaker failure back-up protection schemes where accurate and repeatable time settings are essential.

It has been determined that it is possible for this type relay to operate in less than the set time delay if the initiating contact experiences a very specific mode of contact bounce. The foreshortened operating time has been observed by other users and has been demonstrated in the factory under carefully controlled test conditions. Although the actual occurrence of this particular mode of initiating contact bounce appears to be rather unlikely, a minor design change internal to the relay has been recommended by the manufacturer.

It is the intent of this PC/M to incorporate the design change which replaces the existing printed circuit board in the relay to eliminate the possibility of such foreshortened timing by the SAM relay.

Safety Analysis

This modification has been reviewed with respect to Title 10 of the Code of Federal Regulations, Part 50.59, which states that a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The modification being performed under this PC/M will enhance the operation of the G. E. Sam 11B relay assuring that if the unlikely event of an initiating contact bounce occurred, the relay will time out appropriately.

The G. E. Sam 11B relay affected are utilized for circuit breaker failure back-up protection schemes and are not in any safety related circuit or performed a safety related function.

Environmental qualification is justified by the fact that these relays and thus their internal PC cards are located in a mild environment.

There is no seismic concerns affected by this modification, the relays have no seismic requirements associated with them.

Therefore, the probability of a previously reviewed accident is not increased, the possibility of an accident of a different type has not been created and the margin of safety has not been reduced. The implementation of this PC/M does not require a change to the plant technical specification. The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the basis that this change does not involve an unreviewed safety question, therefore, prior Commission approval for implementation of this PC/M is not necessary.

MSR RETUBING

ABSTRACT

This Engineering Package covers the modifications to the Moisture Separator Reheaters (MSRs) and to the Scavenging Steam Vent Condenser (SSVC) System. The major feature of this package is the installation of new tube bundles in the MSRs which will provide improved thermal performance of the plant secondary side.

A review of the changes to be implemented by this PCM was performed against the requirements of 10CFR 50.59. As a result, these MSR and SSVC modifications are classified as non-safety related, do not constitute an unreviewed safety question, and will not affect plant safety, (as addressed in Section 3, "Safety Evaluation"). However, the MSR Operating Procedure must be revised prior to plant startup.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

This modification does not involve an unreviewed safety question because:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased. The MSRs, the portion of Main Steam associated with the MSRs and the Heater Drain System piping are not used in any safety analysis for accidents or malfunction of equipment and as such are non-safety related and will have no effect on equipment vital to plant safety.

ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created. The components involved in this modification have no safety related function and no changes have been made to the operational design of the system.

iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this PCM, since the component involved in this modification are not included in the bases of any Technical Specification.

The implementation of this PCM does not require a change to the plant technical specifications.

The foregoing constitutes, per 10CFR 50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

MSR PERFORATED PLATE MODIFICATION

ABSTRACT

This Engineering Package covers the modifications to the Moisture Separator Reheaters (MSRs). This package is for the installation of perforated plates in the MSRs which will provide better steam flow distribution. This will improve thermal performance of the plant secondary side and reduce erosion/corrosion of the moisture separator vanes and supports.

A review of the changes to be implemented by this PCM was performed against the requirements of 10CFR50.59. As a result, this MSR modification is classified as non-safety related, does not constitute an unreviewed safety question,does not require a change to the Plant Technical Specification, and will not affect plant safety, (as addressed in Section 3, "Safety Evaluation"). However, the MSR Operating Procedure must be revised prior to plant startup as indicated in PCM 064-185.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

This modification does not involve an unreviewed safety question because:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased since the MSRs are not used in any safety analysis for accidents or malfunction of equipment and as such are non-safety related and will have no effect on equipment vital to plant safety.

ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created since the components involved in this modification have no safety related function and no changes has been made to the operational design of the system.

iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this PCM, since the components involved in this modification are not included in the bases of any Technical Specification.

The implementation of this PCM does not require a change to the plant technical specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provide the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

REPLACEMENT OF RIS DEVICES

INTRODUCTION

This PC/M is for the installation of fourteen (14) new transmitters by Rochester Instrument Systems model SC-1302-323 to replace existing units. The existing transmitters are reaching their qualified life expectancy. Therefore, a new replacement unit is required to satisfy the life expectancy requirement.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This modification does not involve an unreviewed safety question and the following provides the bases for this conclusion.

These new transmitters by RIS are qualified environmentally to IEEE-323-1974 and seismically to IEEE-344-1975. This PC/M replaces existing RIS transmitters with new units thus satisfying the life expectancy requirement. The seismic qualification of these devices have been previously reviewed for the particular mounting location and found acceptable. The seismic integrity of the RTGB, Post Accident Panel 1A and Radiation Monitoring Cabinet "E" are not affected since the device is a replacement for an existing, identical device at the same location.

Therefore, this modification will not increase the probability of the occurrence of any accident, whether previously evaluated of a different type than previously evaluated and will not reduce the safety of the plant.

This PC/M does not reduce the margin of safety as defined in the basis of any technical specification.

The implementation of this PC/M does not require a change to the plant technical specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

APPENDIX R CONTROL ROOM INDEPENDENCE FOR ALTERNATE SHUTDOWN

INTRODUCTION

Since the separation defined in Appendix "R" Section III.G.2 cannot be provided for essential components and circuits in the Control Room and/or cable spread room, alternative shutdown capability is provided. This ensures that in the unlikely event a fire makes the Control Room and/or cable spread room uninhabitable or renders equipment in either room inoperable, the plant can be safely taken to cold shutdown from remote locations and Hot Shutdown Control Panel (HSCP).

This PC/M installs redundant fuses, isolation switches and fuse blocks to various control circuits as identified by the "Essential Equipment List for Alternate Shutdown" transmitted by Ebasco letter P-M-SL-85-0325 dated February 28, 1985. The modifications to the control circuits will assure safe shutdown from the Hot Shutdown Panel and other local control stations should a fire disable the essential circuits in the Control Room and/or cable spreading room. The added components are located outside the Control Room and/or cable spread room.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This modification does not involve an unreviewed safety question and the following provides the bases for this conclusion.

The material, devices associated with this modification, will be Class 1E where connected to safety grade equipment. Safety grade devices shall conform to IEEE-323-1974 and IEEE-344-1975.

This modification provides the means for an alternate plant shutdown by the installation of additional switches and redundant fuses to isolate the Control Room in the unlikely event of a fire. The new cable routing is being done in accordance with St Lucie Unit No 1 ampacity and tray fill criteria. The safety related switchgears, MCC, and Diesel Generator panel have been reviewed to account for the effect of the weights of the added devices. Based on the insignificant added weight, we conclude there is no impact to the existing equipment seismic qualifications.

Therefore, this modification will not increase the probability of the occurrence of an accident, whether previously evaluated or of a different type then previously evaluated and will not reduce the safety of the plant.

This PC/M does not reduce the margin of safety as defined in the basis of any technical specification.

The implementation of this PC/M does not require a change to the plant technical specifications, nor does it require a revision of a technical specification. This modification will be performed in accordance with the requirements of Technical Specification TS-3.4.3 and TS-3.4.4.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior commission approval for the implementation of this PCM is not required.

REACTOR CLOSURE HEAD LIFTING RIG PIPE ASSEMBLY

SYSTEM DESCRIPTION

THE CLOSURE HEAD LIFTING RIG ASSEMBLY CONSISTS OF THREE MAJOR ASSEMBLIES NAMELY, (1) THE LIFTING FRAME ASSEMBLY, (2) THE LINK ASSEMBLY WHICH INCLUDES THE BOX GIRDER ASSEMBLY AND THE PIPE ASSEMBLY, AND (3) THE PLATFORM ASSEMBLY.

THE PIPE ASSEMBLY IS ATTACHED TO THE TRAMRAIL AND THE TRAMRAIL IS ATTACHED TO THE HOIST WHICH IS ATTACHED TO THE PIPE ASSEMBLY. THE ASSEMBLY ROTATES VIA THE TRAMRAIL AND ALLOWS THE TENSIONING DEVICES TO BE IN POSITION.

THE INTENT OF THIS MODIFICATION IS TO UTILIZE THE PIPE RING OF THE PIPE ASSEMBLY AS AN AIR DISTRIBUTION HEADER FOR THE PNEUMATIC TUGGERS FOR THE STUD TENSIONING DEVICES. THIS ENTAILS THE INSTALLATION OF A 2" DIAMETER AIR SUPPLY NOZZLE AND THREE 1" DIAMETER OUTLET NOZZLES FOR THE PNEUMATIC TUGGER AIR SUPPLY LINES. THIS WILL ELIMINATE THE USE OF INDIVIDUAL SUPPLY LINES TO THE TUGGERS AND WILL REDUCE PERSONNEL RADIATION EXPOSURE AND WILL ALSO PROVIDE EASE IN RIGGING AND REMOVAL OF THE TUGGER AIR SUPPLY LINES. OVERALL, THIS MODIFICATION WILL EASE UP THE ENTIRE STUD TENSIONING OPERATION.

SAFETY ANALYSIS

WITH RESPECT TO TITLE 10 OF THE CODE OF FEDERAL REGULATION, PART 50.59, A PROPOSED CHANGE SHALL BE DEEMED TO INVOLVE AN UNREVIEWED SAFETY QUESTION, (1) IF THE PROBABILITY OF OCCURENCE OR THE CONSEQUENCES OF AN ACCIDENT OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE SAFETY ANALYSIS REPORT MAY BE INCREASED, OR (II) IF A POSSIBILITY OF AN ACCIDENT OR MALFUNCTION OF A DIFFERENT TYPE THAN ANY EVALUATED PREVIOUSLY IN THE SAFETY ANALYSIS REPORT MAY BE CREATED, OR (III) IF THE MARGIN OF SAFETY AS DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATION IS REDUCED.

THE PROBABILITY OR THE CONSEQUENCES OF AN ACCIDENT OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE SAFETY ANALYSIS REPORT IS NOT INCREASED BECAUSE THE PIPE RING ASSEMBLY THAT IS BEING MODIFIED DOES NOT PERFORM A SAFETY RELATED FUNCTION AND DOES NOT AFFECT THE SAFE SHUTDOWN CAPABILITY OF THE UNIT. THIS MODIFICATION DOES NOT CREATE AN ACCIDENT OR MALFUNCTION OF A DIFFERENT TYPE THAN ANY EVALUATED PREVIOUSLY IN THE SAFETY ANALYSIS REPORT BECAUSE THE ADDED NOZZLES ARE WELDED AND MADE AS PART OF THE PIPE ASSEMBLY WHICH DOES NOT PERFORM A SAFETY RELATED FUNCTION. THE MARGIN OF SAFETY AS DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATIONS IS NOT REDUCED SINCE THIS MODIFICATION DOES NOT REQUIRE ANY REVISION TO ANY TECHNICAL SPECIFICATIONS.

THE FOREGOING CONSTITUTES, PER 10 CFR 50.59 (B), THE WRITTEN SAFETY EVALUATION WHICH PROVIDES THE BASIS THAT THIS CHANGE DOES NOT INVOLVE AN UNREVIEWED SAFETY QUESTION; THEREFORE, PRIOR COMMISSION APPROVAL IS NOT REQUIRED FOR THE IMPLEMENTATION OF THIS PC/M.

TURBINE GANTRY CRANE BRAKE SYSTEM

ABSTRACT

REA SLN-85-72 requested engineering to be provided to upgrade the Unit 1 turbine gantry crane brake system to meet the operating capabilities of the existing Unit 2 turbine gantry crane brake system. Based upon the design and hardware provided by the crane vendor (Indusco), a pneumatic hydraulic system functionally equivalent to that utilized on the Unit 2 turbine gantry crane was implemented. To support this modification, a 10 CFR 50.59 review was completed and the respective safety analysis which is now part of this document was transmitted by Ref. 3. This design package functions to endorse the brake modification implemented by the vendor. The modification is considered Non-Nuclear Safety Related and does not create an unreviewed safety question.

Revision 1 provides changes to the drawing list to reference attached instruction manuals not addressed by Revision 0. This revision does not change the original scope of work. The 10 CFR 50.59 review and safety evaluation as provided by Revision 0, therefore, remains valid and is acceptable. In addition, no changes to the Technical Specifications were required by Revision 0 or are required by Revision 1 of this design package.

SAFETY EVALUATION

The Unit 1 turbine building gantry crane is located on the turbine building and as such is not required to function during any existing analyzed accident scenario. Therefore, this modification affects only Non-Nuclear Safety Related Quality Group D equipment.

The crane design requirements of NUREG 0612 "Control of Heavy Loads at Nuclear Power Plants" are not applicable to the St. Lucie 1 Turbine Gantry Crane (FPL Letter L-81-276 from R. E. Uhrig to D. Eisenhut dated July 2, 1978).

Based on a failure mode analysis, failure of the turbine gantry crane braking system as provided by this modification will not inhibit the operation of any existing safety related equipment or components. The new braking system provides a hydraulic brake for each crane motor thus providing control of lateral movement of the crane along its rails. Failure to provide this breaking action will result in an inability to adequately control lateral movement of the crane. Additionally, all new electrical components added by this modification are powered from the crane electrical system which is powered from a nonvital source. Therefore any failure mode analyzed for the new braking system will not affect any safety related equipment or components.

Based on this information it can be demonstrated that an unreviewed safety question as defined by 10CFR50.59 does not exist since the consequences of all analyzed accidents remains unchanged. Additionally, with respect to nuclear safety, no new accidents or malfunctions are introduced as a result of using the new brake system. Finally, the margin of safety as defined in the Technical Specifications has not been reduced nor have changes to the Technical Specifications been required.

In conclusion, this modification is acceptable from the stand point of nuclear safety since it does not involve an unreviewed safety question and does not require changes to the Technical Specifications. Therefore implementation of this modification without prior NRC approval is acceptable.

RCP OIL LIFT SYSTEM PRESSURE SWITCH REPLACEMENT

INTRODUCTION

This PC/M is for the installation of twenty (20) pressure switches by Barksdale Model B1T-A48SS to replace the existing Barksdale Model 9048-4.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This modification does not involve an unreviewed safety question and the following provides the bases for this conclusion.

The new switches will be mounted in the same place and manner as the existing switches, which are mounted seismically. Both switches have the same weight 2.5 lbs. They do not perform any safety function and are non-safety related.

This PC/M does not reduce the margin of safety as defined in the bases of any technical specifications.

The implementation of this PC/M does not require a change to the plant technical specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve any unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

TIE BETWEEN CONSTRUCTION FIRE MAIN AND PLANT FIRE LOOP

Abstract

This Plant Change/Modification is for the connection of the Backfit Construction Fire Main to the St Lucie Units 1 and 2 Fire Water Loop. This connection consists of two separate tie-ins between the fire main and the fire loop.

This PCM is not classified as Safety Related since the fire main and the fire loop do not perform any safety function. Since the fire loop provides protection for safety related equipment, this PCM is classified as Quality Related.

This PCM provides additional fire protection to the plant since these tie-ins create an additional fire water supply to other portions of the Plant.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The modifications included in this PCM do not involve an unreviewed safety question because:

i The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since the fire main/fire loop tie-in is quality related and this modification will have no effect on equipment performing a safety function.

ii There is no possibility for an accident or malfunction of a different type than any previously evaluated since the fire main/fire loop tie-in has no safety function and no changes have been made to the operational design of the system.

iii This modification does not change the margin of safety as defined in the basis for any technical specification.

The implementation of this PCM does not require a change to the plant technical specification.

The foregoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

DIESEL GENERATOR COOLING SYSTEM VALVE REPLACEMENT

ABSTRACT

This engineering package covers the replacement of several valves in the Diesel Generator Cooling System and Demineralized Make Up Water System. The replacement of the Diesel Generator Cooling System Valves is classified as nuclear safety related and does not constitute an unreviewed safety question. The replacement of the valves in the Demineralized Make Up Water System is classified as non nuclear safety related.

SAFETY EVALUATION

The replacement of valves in the Diesel Generator Cooling Water System and the Demineralized Make Up Water System does not create an unreviewed safety question as defined by 10CFR 50.59.

The replacement of valves in the Demineralized Make Up Water System has no effect on nuclear safety since the D. I. Water System is not relied upon for any accident prevention or mitigation and failure of the system would not result in a release of radioactive material.

The replacement of.valves in the Diesel Generator Cooling System does not increase the probability previously analyzed accidents since the D/Gs are not utilized in determining the probabilities of accidents. Since the valve replacement does not change the operation or operability of the diesels or any other piece of equipment important to safety, the consequences of an accident previously analyzed, the probability of and consequences of a malfunction of equipment important to safety previously analyzed have not changed. Likewise, the possibility of a new accident or a new malfunction has not been created since the operation or operability of the diesels or any other piece of equipment important to safety has not changed.

In addition, the margin of safety as defined in the basis for any Technical Specification has not been changed since this modification does not change the performance, load capabilities, or operating characteristics of the diesel generators.

WATER TREATMENT PLANT REGENERATION WASTE NEUTRALIZATION TANK MODIFICATION

ABSTRACT

The subject REA requested a neutralization tank be added to the Water Treatment Plant (WTP) to meet current Department of Environmental Regulation (DER) regulations governing discharge of hazardous wastes. The neutralization tank modification (PC/M 116-985) provides the necessary details for installation of this tank and the associated piping, equipment and components necessary to allow for regeneration wastes to be automatically directed to the tank during the appropriate times in the regeneration process. During the caustic injection steps of regeneration, caustic solutions must be directed to the tank. The existing system, however, is unable to provide the necessary flows and pressures required to accommodate these regeneration steps due to the additional headloss in the new piping runs. Thus, to accommodate the new arrangement, a booster pump must be added to the caustic dilution water demineralized water supply. In addition, the caustic dilution water flow control valve and flow indicator/transmitter must be replaced to accommodate the flow requirements. This system is not required for plant safe shutdown, therefore this modification is non-nuclear safety related and its implementation does not create an unreviewed safety question.

SAFETY EVALUATION

The subject modification provides for addition of a booster pump and flow control valve in the caustic dilution water supply to the WTP. In addition, the modification provides for replacement of certain caustic dilution water flow transmitter components to accommodate the required flow rates. As defined in Section 9 of the Unit 1 FSAR, the WTP and its associated systems are classified as non-nuclear safety related and are not required to perform a safety function. Based on the failure mode analysis, as addressed in the Design Analysis, the modification has no affect on nuclear safety. Therefore, the modification is adequately classified as Non-Nuclear Safety Related Quality Group D.

Based on the above evaluation and information supplied in the design analysis, it can be demonstrated that an unreviewed safety question as defined by 10CFR 50.59 is not created. Since the modification affects only the WTP which is classified as Non-Nuclear Safety Related and cannot affect any other safety related equipment or components as addressed in the failure mode analysis, the consequences of all analyzed accidents remains unchanged. Also, with respect to nuclear safety, no new accidents or malfunctions are introduced as a result of this design change. Additionally, the margin of safety as defined in the Technical Specifications has not been reduced. Therefore, an unreviewed safety question does not exist.

Since this modification does not involve an unreviewed safety question, nor require a change to the Technical Specifications, this modification is acceptable with respect to nuclear safety thus prior NRC approval is not required for implementation of the modification.

CCW STRAINER BACKFLUSH DRAIN

Abstract

This engineering design package (EDP) modifies the CCW Strainer Backflush Drain piping. Existing cast iron and fiberglass drain piping, which is routed to the CCW sump, will be replaced with stainless steel piping which ties into the ICW discharge line. This will eliminate the flooding problem in the CCW pit area, which is causing corrosion of structural steel and piping supports mounted on or near the floor.

This EDP is classified as nuclear safety related since it modifies a safety related system. The safety evaluation has shown that this EDP does not constitute an unreviewed safety question.

This EDP has no impact on plant safety and operation.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The modification included in this engineering design package do not involve an unreviewed safety question because:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since the connection of a CCW strainer backflush drain line to the ICW discharge line will have no effect on the safety performance of the ICW or CCW systems or any of their components.

(ii) There is no possiblity for an accident or malfunction of a different type than any previously evaluted since no changes have been made to the operational design of the CCW strainer backflush system.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification.

Implementation of this engineering design package does not require a change to the plant technical specifications.

The foregoing constitutes, per 10CFR 50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

LUBE OIL CENTRIFUGE ANNUNCIATION

INTRODUCTION

Modification is required to the existing Turbine Lube Oil Centrifuge circuit. At present, only local annunciation is provided for Turbine Lube Oil abnormal conditions. A modification to the present design is required to tie-in this local signal to the annunciator at the RTGB. This will provide information to alert the operator of turbine lube oil high back pressure or low oil temperature.

SAFETY ANALYSIS

With respect to Title 10 of the Code of Federal Regulation, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This modification does not involve an unreviewed safety question and the following provides the bases for this conclusion.

The additional signal provided at the RTGB enhances the present system by providing turbine lube oil centrifuge annunciation under abnormal conditions. This information alerts the operator of turbine lube oil high back pressure or low oil temperature.

In addition, there are no safety concerns associated with the circuitry changes and new cable routing for the following reasons:

1. The turbine lube oil circuitry is non-safety related.

2. The location of the cable routing specified in the backfit sketches will not damage any safety related equipment.

Therefore, this modification will not increase the probability of the occurrence of any accident, whether previously evaluated of a different type than previously evaluated and will not reduce the safety of the plant.

This PC/M does not reduce the margin of safety as defined in the basis of any technical specification.

The implementation of this PC/M does not reqire a change to the plant technical specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question, and prior Commission approval for the implementation of this PC/M is not required.

NON RETURN VALVES ACTUATION SOLENOIDS

ABSTRACT

This engineering package covers the modifications to improve the performance of the Turbine Extraction Steam Reverse Current Valves (NRV) on overspeed turbine trip.

The modifications and details consist of the addition of a pressure switch in the turbine Overspeed Protection Control (OPC) header interlocked with six (6) NRV's actuation solenoids and the replacement of the pneumatic NRV test valves with electric test pushbuttons.

Presently the individual solenoids are actuated by high level switches installed in the corresponding feedwater heaters.

Based on the design of the Main Turbine and its Subsystems and the extraction steam lines NRV's, this Engineering Package has been classified non-safety related.

Primarily, the NRV's will improve the closing time on turbine overspeed trip. This is the main objective of this PC/M.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The Steam Turbine and Reverse Current Valves are nonsafety related. The instrumentation additions and changes to be performed have no impact on any safety related plant systems and/or operations. The modifications improve the equipment operation without changing the original design intent.

The addition of the pressure switch, the local electric test pushbutton and the deletion of the pneumatic test valve will improve the turbine protection in case of overspeed trip.

The modification to this package will not increase the probability or consequences of an accident. This system is not used in any accident mitigation scenario and therefore the systems failure will have no impact on plant safe shutdown. This modificaton is not described in the Technical Specifications and therefore, the implementation of this PCM does not require a change to the plant Technical Specifications.

The foregoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

The turbine missile criterion specified in Section 3.5 of the Unit 1 FSAR is not applicable to the components added by this PCM.

TELEPHONE SYSTEM UPGRADE

ABSTRACT

This Engineering Package covers the modifications and details required to support the installation (by AT&T) of a new AT&T System 85 PBX Telephone System. The central equipment for System 85 will be located in the Telecommunication Equipment Rooms in the Unit 1 Service Building and Unit 2 D-13 Building.

The modifications and details consist of enlargement of the telecommunication rooms to accommodate the new equipment; installation of redundant air conditioner units for each room to satisfy equipment environmental requirements; power supplies with emergency back-up; raceway between the two telecommunication rooms to install the AT&T supplied fiberoptics cable, and raceway between the D-13 Building, G-3 Building and Start-up Trailers to accommodate the AT&T supplied multipair telephone cables.

Based on the importance of the telephone system as one of the plant communication means, this Engineering Package has been classified Quality Related to enhance the system design and installation confidence.

The new "System 85" will replace the existing Dimension 600 Electronic Stored Program PBX located in the Unit 1 Service Building Telecommunications Room and the Private Automatic Telephone Exchange (PAX) located in Unit 1 Reactor Auxiliary Building (Elev 43'-0). This replacement will require modification of Section 9.5.2 "Communication Systems" of the Unit 1 and Unit 2 FUSAR, Figures 9.5-1 and 9.5-4, Table 9.5-6 of the Unit 2 FUSAR and Section 3.8 of the Unit 1 and Unit 2 Nuclear Fire Protection Program.

To energize the System 85 telephone equipment and air conditioners located in the Unit 1 Service Building upon loss of normal off-site power will require manual switching at Power Panel PP-135 located in the Security and Records Building. Resetting will also be required upon returning of normal off-site power. The System 85 telephone equipment modules and air conditioners located in the Unit 2 D-13 Building will be automatically supplied by the Non-Class 1E diesel supplying the D-13 Building upon loss of normal power.

MAIN FEEDWATER PUMP OIL PRESSURE SWITCH REPLACEMENT

ABSTRACT

The existing Main Feedwater Pump lube oil pressure switches are United Electric Series 300. The model 300 has been discontinued by the manufacturer and spare parts are very scarce and are essentially not available.

This engineering package covers the replacement, of (2) two pressure switches per pump (for a total of (4) four) with new upgraded series 400 by the same manufacturer

Both of the above models have comparable overall dimensions, weight and mounting facilities, therefore, no other modification is needed.

The function of each pressure switch will remain unchanged. All switches will have adjustable differentials.

Based on the design of the Main Feedwater Pumps and its subsystems, this Engineering Package has been classified as Quality Related. See Section 2.1.1 for additional information.

This PCM does not constitute an unreviewed safety question and has no effect on plant safety. The replacement of these pressure switches has no impact on plant operation and does not affect any safety related equipment.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The modifications included in this PCM do not involve an unreviewed safety question because:

The probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased, since the Feedwater Pump and its subsystems are non-safety related. Therefore this modification will have no effect on equipment performing any safety function.

The Feedwater Pump Lube Oil System is not used in any accident mitigation scenario, therefore there is no possibility for creating an accident or malfunction of a different type than any evaluated previously in the safety report. Its failure will have no impact on the plant safe shutdown.

This modification does not change the margin of safety as defined in the basis for any technical specification since the replacement of the Main Feedwater Pump lube oil pressure switches improves their operational quality without changing the original design intent. The Technical Specifications do not address the MFW pump/system surveillance.

The implementation of this PCM does not require a change to the plant technical specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Nuclear Regulatory Commission approval for the implementation of this PCM is not required.

EXCITER COOLER VENTS & DRAINS TUBING MATERIAL CHANGE

ABSTRACT

This Engineering Design Package covers the replacement of the Exciter Cooler Vents and Drains Tubing. The original design provided for 1/2" copper tubing which is susceptible to damage due to abrasion and vibration. This design package provides for the installation of 1/2" Stainless Steel Tubing and one additional tube vibration support to firmly locate the tubing away from pipe couplings. All of the eight tubing runs involved are located inside the Exciter Housing, each run approximately 12 feet in length. The Exciter, it's coolers, and the tubing involved in this modification are all non-nuclear safety related, non-seismic, and the implementation of this design package does not create any failure modes which could adversely affect any safety related equipment or components. The classification of this design package is therefore non-nuclear safety related. The Design Package was reviewed using the 10 CFR 50.59 criteria and it was found that the change does not involve an unreviewed safety question nor is a Tech Spec change required. Therefore prior NRC approval is not required to implement this PC/M.

SAFETY EVALUATION

The subject modification involves the replacement of 1/2" copper tubing with 1/2" stainless steel tubing containing Turbine Cooling Water from the Exciter Coolers. Both the Main Generator Exciter and the Turbine Cooling Water System are non-safety related, non-seismic systems which perform no Safety Related functions. A failure modes an effects evaluation was performed to determine if any safety related components would be affected by the components modified by this change. A failure of hardware failure of the tubing would result in possible mechanical damage to the Exciter from hardware impact or water damage due to leakage. A functional failure would result in either the inability to vent or drain the coolers, or a contained loss of Turbine Cooling Water from the Coolers. None of the above failures would in any way adversely affect any safety related equipment. Based upon the Safety classification of the affected systems, and the results of the failure mode evaluation, the implementation of this modification does not involve an unreviewed safety question as defined by 10CFR 50.59 because:

1) The consequences of analyzed accidents are not affected because no equipment required to mitigate analyzed accidents are involved in this modification.

2) The probability of an analyzed accident or the possibility of an unanalyzed accident is not increased because failure of the involved components does not affect any DBA required components.

3) The Tech Spec margin of safety is not decreased because no Tech Spec Limits or equipment are affected.

Since this modification does not involve an unreviewed safety question and does not require a change to the plant technical specifications, it may be implemented without prior NRC approval.

PCB TRANSFORMER REPLACEMENT

ABSTRACT

Due to pending Environmental Protection Agency rules on control of Polychlorinated Bephenyls (PCB) these materials will be removed from all oil filled transformers. The Main Neutral Grounding Transformer at St Lucie Unit No 1 ~~has~~ been identified as containing PCB.

This Engineering Design Package (EDP) covers the modifications and details necessary to replace the existing PCB filled Main Generator Neutral Grounding Transformer with a silicon filled transformer.

A review of the changes to be implemented by this PCM was performed against the requirements of 10CFR50.59 as indicated in Section 3.0 of this EDP. This PCM does not involve an unreviewed safety question, will not affect plant safety or operation, nor does it require a change to the Technical Specification; therefore prior Commission approval is not required for implementation of this EDP.

This EDP is non-safety related in that the Generator Neutral Grounding Transformer provides no safety-related function and as the transformer is located in the Turbine Building, it does not have any impact or interactions with any safety related equipment.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

This modification consists of replacement of the existing PCB filled Main Generator Neutral Grounding Transformer with a silicon filled transformer. In all other aspects the replacement transformer is identical in size and electrical characteristics as the existing transformer.

The Main Generator Neutral Grounding Transformer is part of the high resistance grounding system connected by 2500 amp self-cooled bus to the Main Generator Neutrals. The Neutral Grounding Transformer, located in the Turbine Building, does not interact with any safety related equipment or provide any safety related function. As a non-safety related system, the Neutral Grounding Transformer does not increase the probability of occurrence or malfunction of equipment important to safety. No accident evaluated in the FUSAR takes credit for the Neutral Grounding Transformer and this modification does not create a possibility for an accident or malfunction of a different type than previously evaluated.

The implementation of this PCM does not modify the operation of any plant system or function. Therefore, its margin of safety as defined in the bases for any Technical Specification is not reduced.

The implementation of this PC/M does not require a change to the plant technical specifications.

The foregoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve any unreviewed safety question, and prior Commission approval for the implementation of this PCM is not required.

MISCELLANEOUS PIPING SYSTEMS MODIFICATIONS

This Engineering Package is issued for the purpose of documentation (i.e.. as-building) of minor modifications made to various piping system/supports as a result of disassembly, inspection, repair, and reassembly during the 1985 fall refueling outage. The modifications were initiated via the Field Change Request/Notice form which were reviewed and approved by Engineering. The modifications are classified as nuclear safety related and do not constitute an unreviewed safety question.

NOTE: This PCM is for documentation (as-building) purposes only.

SAFETY EVALUATION

The modifications to the essential portions of the CCW, SI, AFW and ILRT systems, as described in the Project Scope, are classified as nuclear safety related because the failure of the modified component, in conjunction with a worst case single failure as analyzed by corresponding sections of the FSAR, would result in the inability of the particular system to achieve its designed safety function.

As described in the Design Analysis, the safety related modifications were performed in accordance with the requirements of ASME Section III for Class 2 and 3 components and are deemed not to involve an unreviewed safety question for the following reasons:

i) The probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR has not increased by correction of material degradation due to process or external environment since the repairs included herein did not alter the function of any affected system, create new systems or reduce the design margin of affected systems.

ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the FSAR has not been created since all repairs and replacements were performed in strict accordance with all original design bases and applicable code requirements. Since all possible accidents and malfunctions resulting from these systems have previously been analyzed, the repair and/or replacement of degraded portions of these systems is deemed not to have created any different types of failures.

iii) The margin of safety for any technical specifications has not decreased due to repair and/or replacement of parts and components, since the affected items were restored to their original design safety margin, as a minimum, in every case.

iv) All repairs, replacements, and modifications have been determined to be equivalent to or better than the existing components in terms of design, procurement, and installation. Thus the reliability of the affected systems has not decreased.

Based on the above evaluations, and pursuant to 10CFR 50.59, the forgoing constitutes the written safety evaluation which provides the basis that these changes do not involve an unreviewed safety question, consequently prior approval from NRC for the implementation of these modifications is not required.

FHB HVAC PENETRATION BARRIERS

ABSTRACT

It has been determined that steel barriers are required for the two (2) HVAC penetrations located at elevation 48 ft of the Fuel Handling Building (west exterior wall). The barriers are required in order to prevent unauthorized access into the FHB.

Both HVAC penetrations are protected by a continuous L-shaped concrete tornado missile barrier, cantilevered two (2) feet from the FHB exterior wall and extending down to approximately one (1) foot below the bottom of the penetration. For ease of construction, the access barriers will be located at the bottom of the two (2) foot opening which exists between the missile barrier and the FHB exterior wall.

The existing HVAC system has not been affected by this modification. Based on the review of the existing HVAC system, a 40% reduction of the missile barrier opening is acceptable. As a result of the addition of the access barriers, the missile barrier openings have been reduced by only 17%.

Failure of the access barriers will not adversely affect the function of any safety-related systems or components. However, since the barriers are being installed in a tornado missile barrier and the FHB exterior wall, this PCM has been classified as Quality Related. This modification does not affect the structural capability of the missile barrier or the FHB wall, nor does it pose any safety hazards.

This PCM does not constitute an unreviewed safety question and has no effect on plant safety. The addition of the access barriers has no impact on plant operation and does not affect any safety related equipment.

SAFETY EVALUATION

With respect to title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The modifications included in this PCM do not involve an unreviewed safety question because:

The probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since:

a - The failure of the access barriers for the two (2) HVAC penetrations located at elevation 48 ft of the Fuel Handling Building will not adversely affect the structural capacity of the missile barrier nor the FHB wall, for which certain quality control inspections (e.g. hole size and verification that no rebar is cut) will be performed.

b - No effect on equipment or components performing a safety function are located beneath this access barrier.

c - The HVAC ventilation system operation has not been affected by the reduction in the missile barrier opening.

There is no possibility for an accident or malfunction of a different type than any previously evaluated since this modification will have no impact on the plant safe shutdown.

This modification does not change the margin of safety as defined in the basis for any Technical Specification by the addition of these access barriers.

There is no change on the existing technical specification due to the implementation of this PCM.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the basis that this change does not involve an unreviewed safety question, therefore prior Commission approval is not required for implementation of this PCM.

HEATER DRAIN PUMP DEMINERALIZED WATER SUPPLY

ABSTRACT

This design package provides the required engineering for adding permanent piping from the demineralized water system to the Unit 1 heater drain pumps' mechanical seals. The piping will make available to the seals the necessary back up flushing water meeting the appropriate chemistry requirements. The back up water source is required during initial plant startup whenever the pumps sit idle.

Based on the failure modes analysis and 10 CFR 50.59 review, this modification does not impact any safety related equipment and is not relied upon for any accident prevention or mitigation. Thus it does not constitute an unreviewed safety question and is correctly classified as Non-Nuclear Safety Related. Implementation of this modification, therefore, does not require prior NRC approval.

Supplement 1

This package revision provides valve drawings for valves added by this PC/M and modifies the expiration date to reflect the correct format. The scope of work specified by this Engineering Package has not been affected by this revision. The safety classification and the safety evaluation as stated is correct and is not impacted.

SAFETY EVALUATION

The Unit 1 Heater Drain Pumps are located in a Non-Nuclear Safety Related system and as such are not required to function during any existing analyzed accident scenario. Therefore, modifications to these pumps affect only Non-Nuclear Safety Related, Quality Group D equipment.

Based on the failure mode analysis, failure of the demineralized water supply piping could result only in failure of the heater drain pumps. Since the piping and components are located remote from any safety related equipment or components, failure of this equipment will not inhibit operation of any safety related equipment or components.

Based on the above evaluation and information supplied in the design analysis it can be demonstrated that an unreviewed safety question as defined in 10CFR50.59 does not exist.

- The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

  Since this design change does not alter or affect equipment used to mitigate accidents, the probability of occurrence of analyzed accidents remains unchanged.

- The possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created.

  There is no new failure mode introduced by this change that has not been evaluated previously in the FSAR.

- The margin of safety as defined in the basis for any Technical Specifications has not been reduced.

  This change has no affect on any existing Technical Specifications.

MAIN STEAM PIPING MODIFICATION

ABSTRACT

During the as-building of PCM 580-79, the as-built information and field walkdowns indicated that some of the restraints on MS drain lines did not conform to the design analysis. The stress analysis performed incorporating the as-built information indicated stresses in the piping for two stress calculations exceeded the allowable stress limits for the Operational Bases Earthquake (OBE) condition while the stress limits for the Design Bases Earthquake (DBE) condition were met. In order to correct the overstress condition in the piping, two restraints are added and two existing restraints are modified through this engineering design package (EDP).

This issue was presented to Plant Personnel via FPL Power Plant Engineering Memo EPO 86-1237. This memo addresses the "functionality" aspects of the Main Steam drain piping in its present configuration.

The piping system considered in this EDP is Nuclear Safety Class 2, Quality Group B and Seismic Category I piping and hence the support/restraints for this piping are classified as Nuclear Safety Class 2 and Seismic Category I. The safety evaluation has shown that this EDP does not constitute an unreviewed safety question.

This EDP has no impact on plant safety and operation.

Safety Analyses

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This modification does not involve an unreviewed safety question and the following provides the bases for this conclusion:

(i) This modification provides two new restraints and additional restraint function to the two existing restraints on the MS drain lines. These modifications when implemented will reduce the stress levels in the piping to the acceptable stress limits established in the USAS B31.7 Code, 1969 Edition. The integral attachments (lugs) welded to the piping create additional stresses in the piping. However, the total piping stresses including those due to the welded attachment have been shown to be within allowable stress limits. Hence, the integrity of the pressure boundary of the piping has not been compromised and no new failure mechanism has been introduced. Therefore, the implementation of this PCM does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analyses Report.

(ii) Since the pipe stresses remain within the code allowable limits, this modification does not create the possiblity for an accident or malfunction of a different type than any evaluated previously in the Safety Analyses Report.

(iii) The main steam system as a whole has been considered in establishing the bases for several technical specifications. Since the MS drain line pressure boundary integrity has not been affected, the implementation of this PCM does not reduce the margin of safety as defined in the basis for any technical specification.

The implementation of this PCM does not require a change to plant technical specification.

The foregoing constitutes, per 10CFR 50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior commission approval for the implementation of this PCM is not required.

LOW POWER FEEDWATER CONTROL SYSTEM

ABSTRACT

The St Lucie Unit No 1 Feedwater Control System consists of two (2) feedwater regulators which operate in parallel. The main feedwater regulator is automatically controlled by a three-element feedwater control system and is responsive in the range of 15-100% power operation. By-pass feedwater regulators are manually operated during plant start-up in the range of 0-15% power operation.

Thermodynamic characteristics of the steam generators at low power loads are such that increased feedwater flow will tend to shrink or lower the steam generator level. At the time thermal equilibrium has been re-established the level will tend to increase due to swelling characteristics. Reactor trips therefore could result from either a high or low steam generator level.

The new Low Power Feedwater Control System (one for the LCV-9005 and one for LCV-9006), which is microprocessor based, is designed to provide stable and automatic control of the by-pass feedwater regulators at low power loads in the range of 2-15%. The system will receive and process system variables such as steam generator level, feedwater flow/temperature and neutron flux in order to provide for a smooth and steady output for automatic control of the by-pass regulators and to significantly reduce the frequency of reactor trips during unit start-up.

This new system is considered to be an extension of the present Feedwater Regulating System, does not affect other safety related equipment and as such provides no safety related functions.

A review of the changes to be implemented by this PCM was performed against the requirements of 10CFR50.59. As indicated in Section 3.0 of this PCM, this PCM does not involve an unreviewed safety question, nor does it require a revision to the technical specification; therefore, prior Commission approval is not required for implementation of this PCM.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

i) The probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased since this new Low Power Feedwater Control System (LPFCS) is an extension of the Feedwater Regulating System and as described in FUSAR Subsection 7.7.1 this systems function is not essential for the safety of the plant. The installation of the LPFCS will provide control improvements to maintain steam generator water level at set point value during unit start-up with significant reduction in the number of reactor trips due to steam generator level excursions.

ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created since:

a — This installation is in accordance with the Code of Federal Regulation 10 CFR 50.48 and no impact is incurred by this installation.

b — The new equipment mountings and added components have been seismically analyized for additional loading and it has been concluded that these additions will not alter the original stress conditions or the fundamental frequency of the RTGB Boards. Consequently, the seismic qualification of the RTG Boards will not be adversely affected.

c — Modification to the RTG Boards have been designed in accordance with NUREG 0700.

d — This installation is in accordance with the Code of Federal Regulation 10 CFR 50.49 and has been determined to have no impact on the Envrionmental Qualification criteria because the equipment is located in the Control Room which is a mild environment.

e — The LPFCS, which is an extension of the Feedwater Regulation System is neither required for safe shutdown nor for mitigating the consequences of an accident.

iii) The margin of safety as defined in the bases for any Technical Specifications is not affected by this PC/M since the components involved in this modification are not included in the bases of any Technical Specification.

The implementation of this PC/M does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10 CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Nuclear Regulatory Commission approval for the implementation of this PCM is not required.

MISAPPLICATION OF LIMITORQUE OPERATORS .

ABSTRACT

This Engineering Design Package (EDP) replaces the motors for the Main Feedwater Flow Control Station by-pass valves (MV-09-3 and MV-09-4) motor operators. The replacement of the existing motors with motors having lower RPM is required to reduce the valve stem speed, to be within the limits recommended by the valve operator manufacturer (Limitorque) for the type of operator (SMB) involved.

This EDP is classified non-safety related since the Main Feedwater Flow Control Stations, where the affected valves are installed, does not perform any safety function and are in the non-safety class portion of the Main Feedwater System.

The safety analysis has correctly concluded that no unreviewed safety concern exist and no changes to the Technical Specifications are required as a result of this modification. Therefore, previous NRC approval for the implementation of this modification is not required.

This EDP has no impact on plant safety and/or operation.

Revision 1 was for the removal of Paragraph 9.1 which required a Limitorque representative to provide technical assistance for the implementation of this EP.

Revision 2 added the Revision 1 description to the abstract and the Engineering approval signature to page III-i. Revisions 1 and 2 do not have an impact on the safety classification and/or the results of the safety evaluation of this E P.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any aevaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

This modification does not involve an unreviewed safety question because:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased. The portions of the feedwater system where this modification will be implemented are not used in any safety analysis for accidents or malfunction of equipment and as such are non-safety related and will have no effect on equipment vital to plant safety.

ii) The possiblity for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created. The components involved in this modification have no safety related function and no changes have been made to the operational design of the system.

iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this PCM, since the component involved in this modification are not included in the bases of any Technical Specification.

The implementation of this PCM does not require a change to the plant Technical Specifications.

The foregoing constitutes, per 10CFR 50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not necessary.

ISOPHASE BUS DUCT JUMPER MODIFICATION

ABSTRACT

The St Lucie Unit 1 Isolated Phase Bus supplied by Westinghouse makes use of cable jumpers to provide electrical continuity at each housing joint so that the longitudinal currents flowing in the enclosures will flow in a closed loop. These cables are continuously exposed to weather resulting in heating, overloading and further insulation degradation. This has resulted in reduced loading on the unit while repairs/cleaning take place.

This EP provides for the replacement of the existing cable jumpers with laminated shunts, welding adaptor brackets (for covers) onto the bus enclosure and installing covers over the shunts to shield them from the weather.

This engineering package is considered non-safety related in that the equipment being modified does not interface with any safety related component or function.

A review of the changes to be implemented by this was performed against the requirements of 10CFR50.59. As indicated in Section 3.0 of this Engineering package, this PC/M does not involve an unreviewed safety question, nor does it require a revision to the technical specification; therefore prior Commission approval is not required for implementation of this EDP.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve and unreviewed safety question: (1) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the analysis report may be increased, or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

The Isolated Phase Bus is described in FSAR Section 8.2. This component is part of the offsite power system and is not required to mitigate any accident. The loss of AC power has been addressed in FSAR Section 15.2.9. The results reached in that section, namely that the plant can be safely shutdown and maintained in a safe shutdown condition, is not affected. In fact, with the addition of this modification the reliability of the offsite power system will be increased.

The Isolated Phase Bus is not a safety related system. The replacement of the cable jumpers has no impact on any plant systems and operations.

The Isolated Phase Bus is not necessary to mitigate or monitor any result of an accident. Failure of this component has no impact on previously generated safety analysis reports. The margin of safety as defined in the bases for any technical specification is not impacted.

The implementation of this PCM does not require a change to the Plant Technical Specifications.

No accident previously evaluated takes credit for the Isolated Phase Bus. This modification, to improve operability and reliability of the Isolated Phase Bus, does not affect any equipment important to safety. As such, this modification will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety.

The Isolated Phase Bus, as part of the offsite power system, is not required to mitigate any accident and does not create a possibility for an accident or malfunction of a different type than previously evaluated. This modification does not impact any technical specification and as such the margin of safety as defined in the bases for any technical specification is not reduced.

The foregoing constitutes, per 10CFR50.59, the written safety evaluation which provides the bases that this change does not involve any unreviewed safety questions, and prior Commission approval for the implementation of this PCM is not required.

CLOSE INTERCEPT VALVE - CONTROL CIRCUIT MODIFICATION

ABSTRACT

This Engineering Design Package (EDP) provides for the removal of the Close Intercept Valve (CIV) anticipatory control circuit from the Westinghouse Digital Electro-hydraulic (DEH) turbine control system.

The original intent of the CIV anticipatory circuit was to provide a temporary closure of the Interceptor Valves in the event of a load mismatch between turbine steam flow and generated electrical output.

This particular circuit does not take into account the dynamic response of the turbine steam cycles, nor does the DEH model P-2000 contain the necessary programming software to perform the required calculations to automatically adjust the turbine governor valves to the new thermodynamic values.

These features, therefore, will, in most cases, maintain the Interceptor Valves closed with a resultant trip of both the turbine and the reactor.

The CIV control circuit is a downstream extension of the DEH overspeed control channel. System failure would not impact plant safety, since this system is neither required for safe shutdown nor does it perform any safety related functions. However the DEH Control System is required to be operable by the Technical Specifications. Since this modification impacts the subject control circuit, this engineering design package shall be classified as Quality Related.

A review of the changes to be implemented by this PC/M was performed against the requirements of 10CFR50.59. As indicated in Section 3.0 of this PC/M, this PC/M does not involve an unreviewed safety question, nor does it require a revision to the technical specification; therefore, prior Commission approval is not required for implementation of this EDP.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence of the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be creatd; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The probability of occurrence as the consequences of an accident or malfunction of equipment previously evaluated in the Safety Analysis Report is not increased by this PC/M. This modification to the CIV control circuit does not change or alter the turbine-generator monitoring and control system.

The possibility of an accident or malfunction of a type different than previously evaluated in the safety analysis report is not created since:

- The CIV control circuit is an independent function generated by the DEH control system software.
- The removal of the CIV anticipatory function does not alter the operation of the DEH control system.
- This modification, which will remove the partial load mismatch circuit, will reduce the number of spurious reactor trips which will occur should the Interceptor Valves fail to re-open.
- The turbine overspeed protection channels to both the Reheater Stop valves and the Intercept valves and the mechanical overspeed protection channels are not altered by implementation of this circuit modification. Therefore, the margin of safety for turbine rupture due to the probability of turbine overspeed is not reduced.

The implementation of this PC/M does not require a change to the St Lucie Unit 1 Technical Specifications.

"The foregoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required."

RTGB ANNUNCIATOR GROUND DETECTION

ABSTRACT

Primary power for the RTGB Annuniciator actuation circuits is provided by the plant 125VDC ungrounded power supply. Although the plant 125VDC ungrounded power supply is furnished with a ground fault detection system, these ground fault detector modules are ineffective in detecting a ground fault on the annunciator 125VDC actuation system. The RTGB annunciators contain a DC isolation circuit which essentially separates the plant 125VDC system from the annunciator 125VDC actuation circuit. A single ground fault on either the positive or negative terminal of an ungrounded system will remain undetected and the system will operate normally. In the event of a second ground on the opposite polarity line, a short circuit will be created causing line interruption by way of the fuses with resultant loss of the system. Corrective action following a catastropic failure is extremely difficult since a fault will now exist on both lines. It is extremely important therefore that ground fault be detected and cleared as soon as practical.

Each RTGB Annunciator (total of 6) will be furnished with an individual ground detector. These ground detectors will monitor the annunciator 125VDC actuation system for excess current leakage levels to ground. Ground fault indication will be provided via indicating lamps located on the front of each of the ground-detector modules.

In order to increase efficiency in trouble shooting and locating ground faults, a total of six ground detector modules will be installed in the RTG board 101. Each ground fault detector shall detect those ground level leakage currents which may exceed preset values.

This modification will improve both the operability and the availability of the RTGB annunciator system.

The annunciators serve no safety related function. However, since this package involves modification to the main control board, it may require work in and around safety related sub-panels, compartments, etc. Therefore, this package is considered to be Quality Related.

A review of the changes to be implemented by this PCM was performed against the requirements of 10CFR50.59. As indicated in Section 3.0 of this PCM, this PCM does not involve an unreviewed safety question, nor does it require a revision to the technical specification; therefore prior Commission approval is not required for implementation of this PCM.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence of the consequences of an accident or malfucntion of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The probability of occurrence as the consequences of an accident or malfunction of equipment previously evaluated in the Safety Analysis Report is not increased by this PCM. The implementation of ground fault detectors does not change or alter the operation of the RTGB annunicator system.

The possibility of an accident or malfunction of a type different than previously evaluated in the safety analysis report is not created since:

- The new equipment mounting has been seismically analyzed for additional loading in accordance with St. Lucie Design Criteria Manual, Section I and found not to have any impact on the seismic qualification of the boards.

- The ground fault detectors will be located in the Control Room, which is considered to be a mild environment.

- The addition of ground fault detectors to the annunciator 125 VDC system will enhance both the operability and the availability of the RTGB annunicator system.

- The use of fuses on the 125 VDC circuit provides for isolating non-class IE circuits from Class IE and associated circuits.

- The ground detector modules and their mounting devices were analyzed and it was determined that they will retain their structural integrity during and following an SSE.

The implementation of this PCM does not require a change to the Plant Technical Specifications.

"The forgoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Nuclear Regulatory Commission approval for the implementation of this PCM is not required."

REPLACEMENT OF INSTRUMENT SCALES

ABSTRACT

This Engineering Design Package covers the replacement of the scales for the following instruments; indicator TIA-1116, "Quench Tank Temperature, indicators FI-09-2A, 2B and recorders FR-09-2A, 2B "Aux. F.W. Pumps 2A & 2B discharge." The indicators are located on the RTG Board while the recorders are located on the Post Accident Panel (PAP). The present scale ranges on these indicators and recorders do not agree with the instrument ranges listed in the October 1985 RG 1.97 Rev. 3, "Parameter Summary Sheet" Type D variables. Nonconformance Reports #287 and #288 (Attachments 7.1 and 7.2) identify the discrepancies between the existing and the required scale ranges. As a result of the scale replacement the associated transmitters FT-09-2A and FT-09-2B will require recalibration in order to accommodate the extended ranges.

This PCM is classified as nuclear safety related since some of the indicators and recorders are monitoring a safety related system. However, the modification initiated by this PCM does not constitute an unreviewed safety question.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This modification does not involve an unreviewed safety question and the following provides the bases for this conclusion:

(i) This modification provides new scales for existing instruments in order to increase the viewing range of the parameters, while all the components and circuitry remain unchanged. Therefore, the implementation of this PCM has no effect on safety and does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analyses Report.

(ii) Since the components and circuitry remain unchanged, this modification does not create the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analyses Report.

(iii) The implementation of this PCM does not reduce the margin of safety as defined in the basis for any technical specification, for the reasons stated above.

The implementation of this PCM does not require a change to plant techncial specification.

The forgoing constitutes, per 10CFR50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior NRC approval for the implementation of this PCM is not required.

HYDROGEN PURGE PENETRATIONS

ABSTRACT

This Engineering Design Package (EDP) allows for the replacement of the inboard valve (valve closest to the Reactor Containment Vessel) on Containment Penetrations P-56, P-57 and P-58. The existing valves have flanged ends and the new valves are butt welded. This modification is being implemented to improve the Containment Vessel integrity with respect to post accident leakage rates by replacing flanged connections with welded connections.

This EDP is classified as nuclear safety related since it modifies a safety related system. The safety evaluation has shown that this EDP does not constitute any unreviewed safety questions, nor does it require a Technical Specification change. Therefore, prior NRC approval is not required for implementation of this EDP.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

The replacement valves for this modification are classified as Nuclear Safety Related, ASME Section III, Class 2, Quality Group B, because they are Reactor Containment Building isolation valves. The modifications included in this engineering design package do not involve an unreviewed safety question because:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since replacement of the hydrogen purge penetrations isolation valves does not affect their dual performance requirements. The performance requirement for containment isolation remains intact and is periodically verified by testing in accordance with 10CFR50, Appendix J. The performance requirement for hydrogen purge has been verified as remaining acceptable.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated. This modification does not change any existing Design Criteria, Operating Procedure or Technical Specification. This modification is a one for one replacement of existing equipment.

(iii) This modification does not affect the basis for any Technical Specification, and therefore does not reduce the margin of safety as defined in the basis for any Technical Specifications.

Implementation of this Engineering Design Package does not require a change to the plant Technical Specification.

The foregoing constitutes, per 10CFR50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question or a change to the Technical Specification; thus prior Commission approval for the implementation of this PCM is not required.

HIR EXCITATION SYSTEM

ABSTRACT

This Engineering Package covers modifications to the Turbine-Generator brushless excitation system. The brushless excitation system will be upgraded to a High Initial Response (HIR) Brushless Excitation System which will allow the generator to respond quickly to changes in system voltage.

A larger permanent magnet generator, a new stator coil in brushless exciter, a new voltage regulator and a new voltage regulator enclosure will be required to modify this system.

The Turbine-Generator does not perform a safety related function. The modifications to the Turbine Generator are classified as non safety related. However, since there will be modifications to the RTG Boards, this package is classified as Quality Related.

This PC/M does not constitute an unreviewed safety question since the modifications described above will have no impact on plant operations or safety related equipment.

Supplement 1

This supplement revised the Abstract and Project Scope pages. The original safety evaluation is not affected by this supplement.

Supplement 2

This supplement incorporates the vendor letter documenting the exciter component serial numbers; drawings for installation of voltage regulator enclosure; fabrication drawings for the HVAC duct; testing and protection requirements for the concrete insulation beneath the voltage regulator and regulator enclosure; revision to drawing list. The original safety evaluation is not affected by this supplement.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

The Turbine Generator High Initial Response (HIR) brushless excitation system is not a safety related system. A larger permanent magnet generator (PMG), stator coil in brushless exciter and voltage regulator will replace the existing equipment and have no impact on any plant system or operation. The HIR excitation system allows the generator to respond quickly to changes in system voltage.

Subsection 3.5.3.2 of the FSAR addresses External Missiles with subpart (b) addressing Turbine Missiles, specifically, missiles generated by the high pressure turbine rotor and the low pressure turbine discs. There are no changes to the high pressure turbine rotor nor the low pressure turbine discs. The modifications required to upgrade the system include a new PMG rotor, PMG stator and exciter stator which are located at the exciter end. The consequences of turbine failure and the potential for damage to critical plant structures, systems, and components from the resulting missiles has not been increased by this modification.

The modificatons to the Turbine Generator, the voltage regulator, the voltage regulator enclosure and the HVAC system in the Turbine Building are not safety related and do not affect any plant systems.

The cables for the lighting, receptacles and power feeds in the voltage reguator enclosure are routed in cable tray and conduit in the Turbine Building. They do not require seismic support and do not affect safety related equipment.

The modifications to the RTG Boards will involve the replacement of selector switches with an updated version, that is the same size and has a negligible weight differential as the existing switches, the relabling of annunciator windows and the actuation of an existing spare relay, that will have the same characteristics as the existing equipment. These modifications do not effect the safety related functions of the affected RTG Boards.

The implementation of this PC/M does not require a change to the plant technical specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

TURBINE BUILDING CRANE GURDER CONNECTIONANGLE MODIFICATIONS

ABSTRACT

Recently, cracking and excessive prying deformation were noted at some of the crane girder connections in the laydown area between the Unit 1 and 2 turbine buildings. An evaluation of the condition concluded that the failures were attributable to the inability of the connections at column line 20 to slide as originally designed.

This PC/M will provide modifications to the crane girder connections at column line 20 to restore independent thermal movement between the units. Modifications will also be implemented at the other crane girder connections in the laydown area to provide reinforcement for those connections which may have been subjected to overstress as a result of the thermal restraint of the crane girders.

This PC/M does not constitute an unreviewed safety question and has no effect on plant safety. The turbine buildings are classified as non-nuclear safety related structures and therefore the modification does not affect any safety related equipment. The connection modifications have no impact on plant operation except for restrictions on the movement of the turbine gantry cranes while the modifications are in progress.

The turbine buildings have been designed for seismic loading to preclude interaction with adjacent Seismic Category I structures during a seismic event. Consequently, this PC/M is classified as Quality Related.

SAFETY EVALUATION

Safety Analysis

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This PC/M provides modifications to the crane girder connections in the laydown area between the Unit 1 and Unit 2 turbine buildings to restore independent thermal movement between the units. It does not involve an unreviewed safety question. The following are the bases for this conclusion:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since this modification will be performed in accordance with Quality Related requirements, hence the seismic capability of the turbine buildings is not compromised and there can be no impact on any adjacent Seismic Category I structures, systems, or equipment.

(ii) There is no possibility for an accident or malfunction of a different type than any evaluated previously since the turbine buildings are non-safety related structures containing no safety-related equipment, hence this modification can have no impact on any safety-related system.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification.

The implementation of this PC/M does not require a change to plant technical specifications.

The foregoing constitutes, per 10CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

TURBINE CROSS UNDER PIPE REPAIR

ABSTRACT

This Engineering Design Package covers various repairs necessary to be performed in the HP turbine exhaust pipes. These repairs are necessitated by damage due to wet steam erosion of turning vane bracing bars, backing rings of circumferential welds, and pipe wall. The carbon steel bracing bars will be replaced with stainless steel bars which are more resistant to erosion/corrosion damage. The backing rings will be removed and any pipe wall damage will be repaired to restore the cross under pipe to acceptable thickness. This modification is classified as non-nuclear safety related and does not constitute an unreviewed safety question. Operation of the HP turbine exhaust lines has not been affected.

Based on failure mode evaluation and 10CFR50.59 reviews, it has been determined that no unreviewed safety question exists as a result of this change. Therefore, prior NRC approval for the implementation of this modification is not required.

Supplement 1

This supplement incorporated Westinghouse Electric Corp. comments regarding the use of a different type of stainless steel material for the turning vane bracing bars. This supplement does not affect the safety evaluation.

SAFETY EVALUATION

The proposed modifications to the turbine cross under pipe will restore the components to original design configuration and provide increased margin against premature erosion wear due to the service conditions. Based on the most recent inspection the carbon steel bracing bars on the turning vanes must be replaced due to extreme erosion. The remaining number of backing rings must also be removed to prevent turbulence in steam flow and subsequent pipe wall erosion. Any existing pipe wall erosion will be repaired to restore wall thickness to the nominal value.

The St. Lucie Unit 1 HP turbine cross under pipe is designated as non-nuclear safety related, Quality Group D. Accordingly, the modifications to the cross under pipe have been designated as non-nuclear safety related, Quality Group D. A failure mode analysis has demonstrated that the modifications to the cross under pipe or any of its components will have no effect on, or inhibit the operation of, any safety related systems or components. The cross under pipe is located remotely from any safety related equipment and cannot fall on, hit, or otherwise affect any such equipment.

EXTRACTION STEAM PIPING AND FITTING - MATERIAL UPGRADE

ABSTRACT

A rupture of an extraction steam line on unit 1 during cycle 7 resulted in a forced outage and a loss of approximately 39 full power hours. An examination of the failed pipe and a follow-up engineering evaluation concluded that Erosion/Corrosion was the failure mechanism. Erosion/Corrosion is an industry wide problem that is unique to wet steam piping systems. Erosion/Corrosion is an accelerated form of corrosion induced by flow due to the breakdown of a protective oxide film from the material's surface.

This PC/M provides details and instructions for plant personnel to replace eroded carbon steel piping and fittings in the Extraction Steam systems on an "as-needed" basis. The fittings to be replaced will be identified following review of ultrasonic inspection data during the 1987 refueling outage. The new materials specified, ASTM A-234 WP22 (fittings) and ASTM A-335 P22 (piping) are chromium-molybdenum alloys and will provide superior resistance to corrosion/erosion effects. Additionally, the new materials can be welded into the existing A-105 and A-106 piping and fittings. The extent of the replacement required for each situation will be based on inspection data review by Power Plant Engineering. The extent of the replacement required will be reported to construction. and details of the specific replacements will be added to the package via the FCN process.

This PC/M is classified as "Non-Nuclear Safety Related" since it affects only nonseismic, Quality Group D piping in Non-Nuclear Safety Related systems.

Based on the failure modes analysis and 10 CFR 50.59 review, this modification does not impact any safety related equipment and is not relied upon for any accident prevention or mitigation. Thus it does not constitute an unreviewed safety question. Since there are no unreviewed safety questions, and since no changes to technical specifications are involved, this PC/M may be implemented without proir NRC approval.

SAFETY EVALUATION

The Unit 1 Extraction Steam System is a Non-Nuclear Safety Related system and as such is not required to function during any existing analyzed accident scenario. Therefore, modifications to these pipes affect only Non-Nuclear Safety Related, Quality Group D equipment.

The modification is a material upgrade only. The new material has been shown, in the Design Analysis, to meet all design requirements of the previous material.

Postulated failures of the extraction steam line would have no impact on safe shutdown of the plant, or safety related systems. The extraction steam lines are not used to prevent postulated accidents, mitigate the consequences of such accidents, maintain safe shutdown conditions, or adequately store spent fuel.

The following statements demonstrate that an unreviewed safety question, as defined by 10 CFR 50.59, does not exist:

* The probablity of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

  Failure of an extraction steam line is not considered as an accident initiating event or considered in determining the probability of an accident. Also, since this design change does not alter or affect equipment used to mitigate accidents, the probability of malfunction of equipment important to safety remains unchanged.

* The possibility of an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created.

  There is no new failure mode introduced by this change that has not been evaluated previously in the FSAR. Additionally, no failure modes analyzed by the FSAR are affected by this design.

* The margin of safety as defined in the basis for any Technical Specifications has not been reduced.

  This change has no effect on any existing Technical Specifications and does not require any changes to the Technical Specifications.

Since no unreviewed saftey questions have been determined to exist, and since no revisions to the Technical Specifications are required, NRC approval is not required prior to implementation.

REACTOR CAVITY SEAL RING

ABSTRACT

This Engineering Package modifies the St. Lucie Unit 1 Reactor Cavity Seal Ring. The modifications consist of enlarging the penetrations in the seal plate for the seal air lines to ensure the air lines are not pinched during installation and adding penetrations and pipe plugs to the cavity seal ring to allow the box section toroid to be filled with water. This water provides additional shielding while the ring is in place. The water shall be removed from the toroid at the conclusion of the outage. Also, the cavity seal ring seal air lines have been changed by the vendor from a neoprene hose to a stainless steel braid hose.

The cavity seal ring based on the FSAR is non-nuclear safety related. Some quality requirements are assigned to assure that the Reactor Cavity Seal Ring will perform its intended function.

Based on a failure mode evaluation and a 10CFR 50.59 review, this modification does not involve an unreviewed safety question, nor require a change to the Technical Specifications. Therefore, prior NRC approval is not required for implementation of this engineering package.

SAFETY EVALUATION

This Engineering Package modified the St. Lucie Unit 1 Reactor Cavity Seal Ring. The modifications consist of enlarging the penetrations in the seal plate for the seal air lines to ensure the air lines are not pinched during installation and adding penetrations and pipe plugs to the cavity seal ring to allow the box section toroid to be filled with water. This water provides additional shielding while the ring is in place. The water shall be removed from the toroid at the conclusion of the outage. Also, the cavity seal ring seal air lines have been changed by the vendor from a neoprene hose to stainless steel braid hose.

Based on the above and information supplied in the design analysis it can be demonstrated that an unreviewed safety question as defined by 10CFR 50.59 does not exist.

- o The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

Since the reactor cavity seal ring is not considered by the FSAR in determining the probability of accidents, possible types of accidents, or in the evaluation of consequences of accidents, it can be concluded that the probability of occurrence of accidents previously addressed in the FSAR remains unchanged.

- o The possibilty of an accident or malfunction of a different type than any evaluated previously in the safety analysis report has not been created.

Since the sealing portion of the cavity seal ring has not changed, the possibility of an accident of a different type has not been created.

- o The margin of safety as defined in the basis for any Technical Specification has not been reduced.

Again, since the sealing portion of the cavity seal ring has not changed, the margin of safety as defined in the basis for any Technical Specification has not been reduced.

10CFR 50.59 allows changes to a facility as described in the FSAR if an unreviewed safety question does not exist and if a change to the Technical Specification is not required. As shown in the preceding sections, the change proposed by this design package does not involve an unreviewed safety question because each concern posed by 10CFR 50.59 that pertains to an unreviewed safety question can be positively answered. Also, no change to the Technical Specifications is required based on the above evaluation.

In conclusion, the change proposed in this design package is acceptable from the standpoint of nuclear safety, does not involve an unreviewed safety question, and does not require any change to Technical Specifications. Therefore, prior NRC approval is not required for implementation of the modification.

10 CFR 50.49 ENVIRONMENTAL QUALIFICATION LIST REVISION

ABSTRACT

This Engineering Package provides the vehicle for updating several areas of equipment qualification. This package includes corrections to the 10CFR50.49 list, changes in maintenance requirements, and various documentation package corrections.

This Engineering Package (EP) is considered Nuclear Safety Related because it affects equipment falling under the scope of 10CFR50.49. This package does not represent an unreviewed safety question since it deals strictly with enhancing the present documentation used to qualify equipment at St Lucie Unit No 1 and no physical plant modifications are required by the EP. The safety evaluation of this package indicates that a change to the Plant Technical Specifications is not required. Removal of equipment from the 10CFR50.49 list does not affect plant safety and operation.

Supplement 1

This EP revision adds terminal blocks to the 10CFR50.49 list and their associated Equipment Qualification Documentation Package 8770-A-451-17.0 "Amerace Terminal Blocks". The equipment and EQ Documentation Package does not affect the original safety evaluation.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

This Engineering Package provides for several changes to the present St Lucie Unit No. 1's 10CFR50.49 list. This documentation will affect the future procurement of various safety related components and assist in validating the components' ability to function before, during and after a design basis accident. Therefore, this EP is considered Nuclear Safety Related.

The documentation changes addressed in this package range from corrections of typographical errors on the 10CFR50.49 list to additions and deletions of equipment as a result of EQ documentation packages reviews. None of the changes require physical modification to any plant system. They do, however, affect the future maintenance of various equipment.

The possibility of new Design Basis Events (DBEs) not considered in the UFSAR is not created since this change does not-alter any equipment used to mitigate accidents. This modification is an enhancement of the environmental qualification documentation of various equipment and in no way affects the plant design.

Due to the fact that this EP does not affect or modify any cables essential to safe reactor shutdown or systems associated with achieving and maintaining shutdowns, this package has no impact on 10CFR50 Appendix "R" fire protection requirements. Therefore the proposed design of this package is in compliance with the applicable codes and UFSAR requirements for fire protection equipment.

Since this modification involves no physical modifications to safety related equipment and changes in the maintenance schedules will not result in failure of equipment, the degree of protection provided to Nuclear Safety Related equipment is unchanged. Removal of equipment from the 10CFR50.49 list does not affect the plant's safety. The probability of malfunction of equipment is unchanged. The probability of malfunction of equipment important to safety previously evaluated in the UFSAR remains unchanged. The consequences of malfunction of equipment important to safety previously evaluated in the UFSAR are unchanged. The possibility of malfunctions of a different type than those analyzed in the UFSAR is not created.

Based on the above, the modifications included in this Engineering Package do not involve an unreviewed safety question because of the following reasons:

(i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report will not be increased by this modification because it does not affect the availability, redundancy, capacity, or function of any equipment required to mitigate the effects of an accident.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report will not be created by this modification. Function, mounting and the ability to withstand harsh environmental conditions have not been altered and this modification does not affect any other safety related equipment.

(iii) The margin of safety as defined in the bases for any technical specification is not reduced since this modification does not change the requirements of the Technical Specifications.

The implementation of this PCM does not require a change to the Plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

PRESSURIZER MISSILE SHIELD ACCESS LADDER SAFETY CAGE

ABSTRACT

This design package consists of the fabrication and installation of a personnel safety cage for the pressurizer missile shield access ladder and modification of the ladder. The safety cage will be attached to the ladder. The modification of the ladder is required to provide safe access to the top of the pressurizer wall as well as to the missile shield.

The personnel safety cage doesn't perform or affect a safety-related function. However, this PC/M is classified Quality Related since there is a potential that, during a seismic event, the personnel safety cage could damage safety-related items that are in the vicinity. Quality Related requirements are applied to this design

This PC/M does not constitute an unreviewed safety question.

SAFETY EVALUATION

With respect to title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The pressurizer missile shield access ladder and safety cage do not perform or affect any safety-related system or function. However, this PC/M is classified as Quality Related since failure of the access ladder or safety cage during a design basis event (e.g., earthquake) could potentially affect a safety-related system or equipment, since the ladder and cage are located in the containment building which contains safety-related systems. Consequently, the ladder and safety cage have been designed for the design basis conditions specified in the FUSAR and Quality Related design requirements have been implemented, thus assuring the integrity of the installation during any design basis event.

The modifications included in this PC/M do not involve any unreviewed safety questions because:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since this modification will have no effect on equipment required to shut down the plant and monitor the plant in a safe shutdown condition.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since the ladder and cage perform no safety function and no changes have been made to any operational design. Failure of the ladder and cage could not occur since the modification has been designed for the design basis conditions.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification.

The implementation of this PC/M does not require a change to plant technical specifications.

The foregoing constitutes, per 10CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

AUXILIARY FEEDWATER ACTUATION SYSTEM DVM CIRCUIT MODIFICATION

ABSTRACT

This engineering package covers modifications to the St. Lucie Unit 2 Auxiliary Feedwater Actuation System (AFAS) that will allow the use of an external meter to monitor the various input signals, pretrip setpoint, and trip setpoint voltages. The current AFAS system has a built in Digital Voltmeter (DVM) that monitors various input voltages. However this DVM has become obsolete. The modification described in this engineering package will allow the use of a portable external meter to be used in place of the built in Digital Voltmeter.

This PC/M is classified as Nuclear Safety Related since it modifies the safety related AFAS. The modifications have been reviewed in accordance with 10CFR50.59 and have been found to neither involve an unreviewed safety question nor require a technical specification change. Therefore prior NRC approval is not required to implement this PC/M.

SAFETY EVALUATION

The Auxiliary Feedwater Actuation System (AFAS) cabinet contains four safety channels of equipment that automatically initiate auxiliary feedwater flow to either or both NSSS Steam Generators. The Auxiliary Feedwater System is actuated by the AFAS when loss of normal steam generator flow would cause a reactor trip by the RPS due to low steam generator water level.

The DVM circuit modification involves the addition of fuses, fuseholders, resistors, and a nameplate to the test circuit in order to provide an external test jack for monitoring test functions using a portable DVM instead of the built in DVM that currently exists. The test circuit is disconnected from the operating circuit in normal operating conditions. Therefore any fault occuring in the test circuit when the AFAS is in normal operation will have no affect on the capability of the AFAS to perform its safeguard function. The functional capability of the AFAS will remain the same after the modification therefore no change in the safety margin will occur. This modification will result in no new malfunctions or accidents to the safety system since all the modifications are to the test circuit which was previously analyzed to be disconnected from the safety function operating circuit during normal operation.

The AFAS test circuit in considered not to be safety related since it does not perform any safety function. However it is designed as an associated Class 1E circuit in order not to degrade the qualification of the safety function circuit of the AFAS.

10CFR50.59 allows changes to a facility if an unreviewed safety question does not exist and if a change to the Technical Specifications is not required. Based on the above evaluation and information supplied in the design analysis, it can be stated that an unreviewed safety question as defined by 10.CFR50.59 does not exist since:

(1) The change described herein does not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report, (2) The change does not create the possibility for an accident or malfunction of a different type than evaluated previously in the safety analysis report, (3) The change does not reduce the margin of safety as defined in the basis for any technical specification.

In conclusion, the change proposed in this design package is acceptable from the standpoint of nuclear safety as it does not involve an unreviewed safety question and does not change the Technical Specifications. Therefore prior NRC approval is not required to implement this procedure.

NEUTRALIZATION BASIN CLOSURE MONITORING WELLS

ABSTRACT

This engineering package covers the installation of two temporary ground water monitoring wells in the vicinity of the St. Lucie Water Treatment Plant. These wells will be used to demonstrate to the State Department of Environmental Regulation (DER) that the operation of our acid/caustic neutralization basin has not resulted in any ground water contamination.

The temporary monitor wells perform no safety related function and are located away from, and have no effect on, any safety related system. This PC/M is non-safety related, but has been classified as 'Quality Related' to ensure the wells are located as specified by the enclosed design drawings. The addition of these wells does not pose an unreviewed safety question.

SAFETY EVALUATION

The Neutralization Basin ground water monitoring wells do not perform any plant safety - related function. They will not be located in the vicinity of any safety - related equipment and therefore well drilling operations cannot adversely impact safety - related functions. A complete well failure or collapse will not impair the structural integrity of plant fill material; accordingly, safety - related structures or equipment supported by the plant fill will not be affected.

Based on the above evaluation and the information supplied in the design analyis it can be demonstrated that an unreviewed safety question as defined by 10 CFR 50.59 does not exist.

- o The probability of occurence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.
- o The possibility of an accident or malfuntion of a different type than any evaluated previously in the safety analysis report has not been created.
- o The margin of safety as defined in the basis for any Technical Specification has not been reduced.

EXCESSIVE AC/DC CONTROL VOLTAGE DROP

ABSTRACT

A study was performed by Ebasco on the St Lucie Plant Unit 1 to address the concern (per INPO SER 80-83) that excessive AC/DC control circuit voltage drops at the control power terminals of the breaker/combination starters may lead to the failure of the equipment contactor mechanism to operate. The study identified deficiencies in four (4) control circuit loops where the calculated control circuit loop resistances exceed the maximum allowable loop resistances.

However, the study found that due to conservatism in the design of the control circuit components, there will be sufficient voltage at the motor starter coil terminals to allow proper operation of the valves during under voltage and degraded grid voltage conditions. Nevertheless, FPL has decided to implement the recommendations of the study to assure the reliability of these control circuit loops.

This engineering package (EP) provides for the implementation of the recommendations of the Ebasco study. These consist of replacement of control transformers for two of the four affected control circuit loops and paralleling conductors for selected portions of all four control circuit loops. These modifications will ensure that the calculated loop resistances will be less than the maximum allowable loop resistances.

This engineering package is considered safety related in that the control circuit loops being modified are for equipment required for mitigation of an accident; Main Feedwater Pump Discharge Isolation Valves MV-09-1, MV-09-2 and Main Feedwater Isolation Valves, MV-09-7, MV-09-8.

A review of the changes to be implemented by this EP was performed against the requirements of 10CFR50.59. As indicated in Section 3.0 of this Engineering Package, this PC/M does not involve an unreviewed safety question. has no impact on plant safety or operation, nor does it require a revision to the technical specification; therefore prior Commission approval is not required for implementation of this PC/M.

Supplement 1 Addendum

This supplement incorporated seismic and environmental qualification for replacement transformer and fuse block. The holdpoint established for this installation has been removed and the original safety evaluation has not been affected by this supplement.

3.0 SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

As a result of the AC/DC Control Circuit Voltage Drop Study (FLO 62-8.5000 R1), it was determined that the calculated actual control loop cable resistance for each isolation valve exceeds the maximum allowable control loop cable resistance. Although this condition exists, the conservatism of the control circuit components allows the valves to operate during undervoltage and degraded grid voltage conditions. This modification consists of implementing the recommendation of the study which include; (1) replacing the 150VA control transformers with a 500 VA transformer for MV-09-1 & MV-09-2 control circuit loop, (2) paralleling conductors for selected portions of the control circuit loops for MV-09-1, MV-09-2, MV-09-7, and MV-09-8.

Although the calculated values indicated insufficient voltage may exist at the motor starter coils, field tests were performed to determine if an actual deficient condition existed. The results of these tests, documented in Attachment 8.3 of the voltage drop study, concluded that due to conservatism in the design of the control circuit components, there will be sufficient voltage at the motor starter coil terminals to allow proper operation of the valves during undervoltage and degraded grid voltage conditions.

The operation of these valves as described in UFSAR Amendment 4 Section 7.3.1.1.12 and 15.4.6.1 is not affected by this modification. In fact, with the implementation of this modification, the reliability of the operation of these valves will be increased. As such, this modification will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and does not create a possibility for an accident or malfunction of a different type than previously evaluated. This modification does not impact any technical specification and as such the margin of safety as defined in the bases for any technical specification is not reduced.

The foregoing constitutes, per 10CFR50.59 the written safety evaluation which provides the bases that this change does not involve any unreviewed safety questions, and prior Commision approval of this PCM is not required.

REPLACEMENT OF STEAM GENERATOR LEVEL TRANSMITTERS

ABSTRACT

This Engineering Package covers the replacement of the steam generator level transmitters. The transmitters provide input signals to the Reactor Protection System, Auxiliary Feedwater Actuation System, Steam Generator Level Indicating Controller and High Steam Generator Turbine Protection Circuitry.

The existing transmitters are part of the Reactor Protection System and classified as Nuclear Safety Related. Since this modification is a one-for-one replacement of the existing transmitters with Rosemount Model 1154DP4RAN0026 transmitter, the same classification applies. The existing wiring is not affected by this change. Plant safety and operation are not affected.

The safety evaluation of this package indicates that the transmitters' replacement does not involve an unreviewed safety question, and does not require a change in the Plant Technical Specifications.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

The modifications included in this Engineering Package do not involve an unreviewed safety question because of the following reasons:

(i) The probability of occurrence and the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report will not be increased by this modification because it does not affect the availability, redundance, capacity, or function of any equipment required to mitigate the effects of an accident.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report will not be created by this modification. Function, mounting and the ability to withstand harsh environmental conditions have not been altered and this modification does not affect any other safety related equipment.

(iii) The margin of safety as defined in the bases for any technical specification is not reduced since this modification installs transmitters with improved accuracy and acceptable response time which are seismically and environmentally qualified to withstand the normal and accident conditions which are anticipated.

The possibility of new Design Basis Events (DBEs) not considered in the FUSAR is not created since the design philosophy of the RPS has been previously discussed in the FUSAR and is not changed by the replacement of the eight (8) level transmitters (LT-9013A,B,C,D and LT-9023A,B,C,D). This modification is an enhancement of an existing system as it furnishes replacement transmitters which cover both the normal and high transient level responses of the steam generators with improved accuracy and reliability.

Due to the fact that this EP does not affect or modify any cables essential to safe reactor shutdown or systems associated with achieving and maintaining shutdowns, this package has no impact on 10CFR50 Appendix "R" fire protection requirements. Therefore the proposed design of this package is in compliance with the applicable codes and FUSAR requirements for fire protection equipment.

According to the FUSAR Sections 7.1.1 and 7.2.1, the RPS is identified as a Nuclear Safety Related System since it monitors the steam supply system and effects reliable and rapid reactor shutdown if any one of a combination of parameters deviates from a preselected operating range. Hence, this EP is considered Nuclear Safety Related. Since this modification provides for a one-for-one replacement of existing level transmitters with more reliable and accurate equipment and involves no other modifications to safety related equipment, the degree of protection provided to nuclear safety related equipment is unchanged. The probability of malfunction of equipment important to safety previously evaluated in the FUSAR remains unchanged. The consequences of malfunction of equipment important to safety previously evaluated in the FUSAR are unchanged. The possibility of malfunctions of a different type than those analyzed in the FUSAR is not created.

The implementation of Nuclear Safety Related PCM 136-186 does not require a change to the Plant Technical Specifications, nor does it create an unreviewed safety question.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

MASONRY WALL MODIFICATIONS

ABSTRACT

Certain masonry walls require a classification upgrade as a result of the installation of safety related equipment in their vicinity. 23 such walls have been identified.

This PCM, issued in response to JCO JPE-LR-87-001, Revision 0, will provide modifications to 10 of these upgraded walls to enable them to satisfy the structural acceptance criteria for safety related walls.

This PCM does not constitute an unreviewed safety question. The modification will ensure that the affected walls will have no interaction with safety related equipment and therefore has no effect on plant safety. The modification has no impact on plant operation.

The affected masonry walls and the structural modifications thereto being implemented by this PCM have been seismically designed. Consequently, this PCM is classified as Quality Related.

SAFETY EVALUATION

Safety Analysis

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This PC/M provides modifications to 10 masonry walls to improve the lateral load carrying capabilities of these walls. It does not involve an unreviewed safety question. The following are the bases for this conclusion:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since this modification will be performed in accordance with Quality Related requirements, hence the seismic capability of the affected masonry walls is not compromised. Therefore, there can be no impact on any adjacent safety related structures, systems, or equipment.

(ii) There is no possibility for an accident or malfunction of a different type than any evaluated previously since the modification will ensure that the affected walls will have no interaction with safety related equipment and hence will have no effect on plant safety.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification.

The implementation of this PC/M does not require a change to plant technical specifications.

The foregoing constitutes, per 10CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

ANNUNCIATOR NUISANCE ALARMS

ABSTRACT

This Engineering Package (EP) covers the modifications of five annunciator circuits in the Main Control Room. Existing logic, circuit configuration and components will be changed in the Reactor Turbine Generator Boards (RTGBs) so as to eliminate existing nuisance conditions caused by erroneous alarm indication of these five annunciator circuits. By implementing this EP, these circuits will be consistent with the "Dark Annunciator" concept which allows for lighted annunciators during off-normal conditions only.

This EP is classified as Nuclear Safety Related since it involves the interposing of a control relay in a safety related circuit (hydrogen analyzer) and the extension of safety related power supply cables (10482E, 10482L, and 10485H). The safety evaluation has determined that this EP does not involve an unreviewed safety question and does not require a change in the plant technical specifications. This PCM may be implemented without prior Commission approval.

This Engineering Package Revision covers modification of the six annunciator circuits associated with annunciated windows P-30, P-35, P-36, P-42, Q-40 and X-5 in the Control Room. These modifications, which include relocation of local reset switches, installation of reflashers and logic modifications, will eliminate the nuisance alarm status of the six annunciators. By implementing this PCM Supplement, these six annunciators will be brought into compliance with the "Dark Annunciator" concept of NUREG 0700 "Guidelines for Control Room Design Review".

The original Safety Evaluation has been revised. The Safety Evaluation still concludes, however, that this EP does not involve and unreviewed safety question, or a change to the technical specifications. Therefore, prior commission approval is not required for implementation of the PCM. The intent of the original Safety Evaluation is not affected by this supplement.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed changed shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased; or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The modifications included in this Engineering Package do not involve an unreviewed safety question because:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since the annunciators serve no function in the control of plant operations or safe shutdown. Electrical separation is provided between redundant safety related wiring and components and annunciator logic which is separated to protect control functions from being affected by annunciation circuit failure.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since no changes have been made to the operational design of any control circuits or associated systems.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification.

Since this EP affects equipment that is identified as Nuclear Safety Related (Hydrogen Analyzer) and requires the extention of Nuclear Safety Related power supply cables (10482E, 10482L, and 10485H), it is considered Nuclear Safety Related.

Due to the fact that the EP does not involve any cables essential to safe reactor shutdown or systems associated with achieving and maintaining safe shutdown conditions, this package has no impact on 10CFR50 Appendix "R" fire protection requirements. Therefore, the proposed design of this package is in compliance with the applicable codes and St Lucie - Unit 1 FSAR requirements for fire protection equipment.

Implementation of Nuclear Safety Related PCM 140-186 and Supplement 1 to the same PCM do not require a change to the plant technical specifications and may be implemented without prior commission approval.

The foregoing constitutes, per 10CFR 50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM and Supplement 1 to the same is not required.

PRESSURIZER LEVEL INSTRUMENTATION MODIFICATION REA-SLN-86-076

ABSTRACT

This Engineering Package (EP) modifies the Pressurizer Level Instrumentation to provide two redundant qualified channels of Control Room indication (Sigma meters). Electrical isolators will be installed so as to separate the non-safety (control) loop from the upgraded Nuclear Safety Related Pressurizer Level Instrumentation loops. Existing pressurizer pressure and pressurizer level recorders will be replaced with narrow case equivalent equipment in order to allow space for the installation of new Sigma indicators in the front panel of RTGB-103. Cable, conduit, penetrations and components which will be part of the Nuclear Safety Related loop will be re-tagged to indicate this upgrade. The two upgraded channels of pressurizer level instrumentation will meet the requirements of Regulatory Guide 1.97, Rev 3 for Variable D Category 1.

This EP is classified as Nuclear Safety Related since it provides an upgrade of an existing system to Nuclear Safety Related status. The safety evaluation has shown that the implementation of this PCM does not constitute an unreviewed safety question and prior Commission approval for its implementation is not required.

This EP does not adversely affect plant safety and operation or impact Plant Technical Specifications.

Supplement 1 to this PCM is the vehicle for the issuance of a revision to the Environmental Qualification Documentation Package and updating the 10 CFR 50.49 list to address the Safety Injection Tank sample line containment isolation valves, FCV-03-1E and FCV-03-1F. These solenoid valves were installed as part of PCM 314-77 which moved the Safety Injection Tank sample point from inside to outside the containment. These containment isolation valves contain limit switches (in the solenoid assembly) as required by Regulatory Guide 1.97, Rev 3 for Variable B Category 1. The isolation valves are located in a potentially harsh environment, thus they require Environmental Qualification as required by 10 CFR 50.49. No physical changes are needed to address Supplement 1 of this PCM since the qualified limit switches were installed as part of PCM 314-77.

Revision 1 does not alter the Nuclear Safety Related status of this EP since it only affects a Nuclear Safety Related Environmental Qualification Documentation Package. The safety evaluation has shown that the implementation of the PCM supplement does not constitute an unreviewed safety question and prior Commission approval for its implementation is not required.

This EP revision does not adversely affect plant safety and operation or impact Plant Technical Specifications.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

The proposed modification affects the Pressurizer Level Instrumentation and control loops in that it provides for the establishment of two redundant Nuclear Safety Related pressurizer level indication channels in the Control Room so as to meet the requirements of Reg Guide 1.97. In accordance with Reg Guide 1.75, physical separation is provided between the non safety (control) and the safety related segments of the loop. Supplement 1 of this PCM is the vehicle for issuance of a revision to the EQ Documentation Package for Valcor Solenoid valves to include Safety Injection Tank Sample Line Containment Isolation Valves in accordance with 10 CFR 50.49. Supplement 1 serves to add the environment qualification of the isolation valves into the document system and has no impact on plant hardware or procedures as described in the UFSAR.

The probability of occurrence of a DBE previously addressed in the UFSAR is not affected by this modification. This EP will in fact decrease the probability of pressurizer level instrumentation failure since it provides for increased reliability in the event a single failure by furnishing two redundant, qualified channels. The possibility of new DBEs not considered in the UFSAR is not created since the design philosophy has been previously discussed in the UFSAR. This modification is an enhancement to a pre-existing system and is being performed to provide increased reliability in the event of a single failure.

Pressurizer Level Instrumentation is identified as Post Accident Monitoring Instrumentation and is required to be Nuclear Safety Related per Regulatory Guide 1.97, Rev 3 as it pertains to Category 1, Type D variables. As this modification involves cables essential to reactor safe shutdown (Essential Equipment List 8770-B-049, Rev 0) and Post Accident Monitoring Instrumentation (PAMI), all applicable 10CFR50 Appendix "R" fire protection requirements have been identified and are satisfied by this package (Section 2.1.4). Therefore the proposed design of this package is in compliance with the applicable codes and UFSAR requirements for fire protection equipment.

Since this package meets the requirements of Reg Guide 1.97 as it provides for two independent, redundant Nuclear Safety Related channels of pressurizer level instrumentation, this EP is considered Nuclear Safety Related. As the evaluation of system failure mode (Section 2.1.17) indicates, this package upgrades existing redundant, independent pressurizer level instrumentation loops from IA/IB (important to safety) to SA/SB (Nuclear Safety Related) and provides adequate electrical and physical separation. Hence, this is an enhancement to the pressurizer level instrumentation and increases the degree of protection to nuclear safety related systems and equipment. The probability of malfunction of equipment important to safety, previously evaluated in the UFSAR, remains unchanged. The possibility of malfunctions of a different type than those analyzed in the UFSAR is not created.

PASS DISSOLVED HYDROGEN ANALYZER TIE-INS

Abstract

This Engineering Package (EP) covers the installation of tie-ins to the existing Unit No 1 Post Accident Sampling System (PASS) for a dissolved hydrogen analyzer.

A review of the change to be implemented by this EP was performed against the requirements of 10CFR50.59. As a result the installation of a disssolved hydrogen analyzer in the PASS is classified as non-safety related, does not constitute an unreviewed safety question, does not require a change to the plant Technical Specification and will not affect plant safety (as addressed in Section 3, "Safety Evaluation"). However, this modification is quality related, because it is required by NUREG 0737 for monitoring Reactor Coolant System (RCS) chemistry and activity resulting from a design basis accident. This change will not affect plant operations. Prior NRC approval is not required for implementation of this EP.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

This modification involves the necessary tie-ins to the existing system including tubing and valves. This EP is only for provisions to install the dissolved hydrogen analyzer in the Unit No 1 PASS.

The PASS is classified as non-safety related. This EP is classified quality related because it is required by NUREG 0737 for monitoring Reactor Coolant System chemistry and activity resulting from a design basis accident.

Based on the above, this engineering design package does not involve an unreviewed safety question because:

(i) The probability of occurrence or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased since this EP is only for provisions to install the dissolved hydrogen analyzer in the Unit No 1 PASS. The PASS is non-safety related and its normal performance requirements have not been affected by these tie-ins. There is no potential for this modification to interact with safety related system functions.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created since the components involved in this modification have no safety related function and no changes have been made to the operational design of the system.

(iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this EP, since the components involved in this modification are not included in the bases of any Technical Specification.

Implementation of this PCM does not require a change to the plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

MISCELLANEOUS ICW SYSTEM MODIFICATIONS

ABSTRACT

This engineering package enables minor modifications to be made to the Intake Cooling Water (ICW) system resulting from disassembly, inspection, repair and reassembly during the 1987 refueling outage. Those modifications that meet the criteria established by this design package shall be initiated via the Change Request/Notice form and dispositioned by engineering. Those modifications which do not meet the criteria established by this design package shall be implemented under separate design packages. Those modifications to the essential portion of the ICW System are classified as nuclear safety related, therefore the PC/M is classified as safety related. Modifications to the non-essential portion of the ICW System are classified as non-nuclear safety related unless the failure mode analysis determines an interaction with equipment important to safety. If so, quality requirements will be applied and the modification classified as Quality Related. Since the PC/M will restore the system to its original design configuration, it does not result in an unreviewed safety question.

SAFETY EVALUATION

The modifications to the essential portion of the ICW system described in the project scope are classified as nuclear safety-related because the failure of the modified component in conjunction with the worst case single failure as analyzed per FSAR Table 9.2.2 would result in the inability of the ICW system to achieve its design basis safety function. Historically, the types of modifications to the ICW System resulting from the disassembly and reassembly of the piping system for inspection and repair have been:

1. Modifications to pipe vent and drain lines (e.g., replacement of corroded material).

2. Modifications to support/restraints (e.g., documentation of weld symbols required to reassemble S/R's, excessive gap at S/R base plates, replacement of corroded material).

3. Weld repair to ICW pipe (e.g., documentation of pipe welds).

4. Pipe flange bolting material changes or bolt torque valve documentation.

As described in the design bases, these nuclear safety-related modifications shall be made in accordance with the design code requirements for Safety Class 3 pipe and pipe components and for Seismic Class I support/restraints.

In accordance with the requirements specified in the design bases. each modification to the non-nuclear safety-related portion of the ICW system shall have a failure mode evaluation performed to determine if there are any interactions with safety-related equipment or functions. Since the non-nuclear safety related portion of the ICW system is not relied upon for any accident prevention or mitigation, failures which are determined to not impact the function of the nuclear safety-related portion of the ICW system are acceptable with regard to nuclear safety. No Quality Related requirements will be applied to the design of these modifications. However, if a modification to the non-nuclear safety-related portion of the ICW system is determined by the failure mode evaluation to interact with Nuclear Safety Related equipment , Quality Related requirements will be applied to the design of these modifications.

Based on the above, it can be demonstrated that an unreviewed safety question as defined by 10CFR50.59 does not exist.

i) The probability of occurrence or the consequences of a Design Basis Accident (DBA) evaluated in the FSAR is not increased because no DBA's deal with specific ICW component failures. The modifications restore the ICW system and original design condition and ensure its safety function will be performed.

ii) The probability of occurrence or the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR is not increased because the modifications proposed by this design package are to passive components only and they will be designed/implemented in accordance with safety class/FSAR requirements. The FSAR does not evaluate passive component failures.

iii) The possibility of an accident or malfunction of a different type than any evaluated previously in the FSAR is not created because the modifications permitted by this design package do not alter the ICW system function or mode of operation. The FSAR evaluation of the ICW system envelopes the failure of the described modified components.

iv) The margin of safety as defined in the basis for a technical specification is not reduced. The modifications permitted by this design package have been reviewed and found acceptable. No changes to the design basis, function, or mode of operation of the ICW system is proposed

10CFR50.59 allows changes to a facility as described in the FSAR if an unreviewed safety question does not exist and if a change to the Technical Specifications is not required. As shown in the preceeding sections, the change proposed by this design package does not involve an unreviewed safety question because each concern posed by 10CFR50.59 that pertains to an unreviewed safety question can be positively answered since the PC/M returns the ICW system to its design condition and no Technical Specification change is required.

In conclusion, the changes proposed in this design package are acceptable from the standpoint of nuclear safety, do not involve an unreviewed safety question, do not require a change to the Technical Specifications and do not require prior NRC approval prior to implementation.

ICW ISOLATION VALVE REPLACEMENT

ABSTRACT

This engineering package is issued to provide direction for the replacement of any of the 30-inch and 36-inch normally open isolation valves in the Intake Cooling Water (ICW) system, as required. The isolation valve replacement is classified as nuclear safety related, and does not constitute an unreviewed safety question.

"Normally closed" isolation valves must be replaced with rubber lined valves and are not within the scope of this design package.

Specification MN 2.57 which is in accordance with the procurement specification used to procure the original valves. The new valves have cast stainless steel bodies (ASME SA-351 Grade CF3M) which preclude deterioration seen in the existing carbon steel/rubber lined valves. The procurement specification specified the requirement that the valves and extension must be compatible with the existing operators.

Pressure boundary components installed as part of the Intake Cooling Water system were designed for the following:

Design Pressure, psig	90
Design Temperature, F	125

Chemistry: Seawater with 2.0 maximum dissolved chlorine

Per FPL P.O C63913-67619P and Power Plant Engineering Specification MN 2.57 revision 0, the valves were bought to the requirements of ASME Section III for Class 3 components. Appropriate seismic requirements were also specified.

The new valves were designed for the following external environmental conditions:

Temperature, F	30-120
Pressure, psig	Atmospheric

100% Humidity, Salt Laden Atmosphere

A Quality Control holdpoint has been assigned in the Construction Scope section for the verification that the extension bonnet supporting details are approved by engineering prior to system turnover in order to comply with the requirements of paragraph 1.3.5.

SAFETY EVALUATION

The replacement of isolation valves as described in the Project Scope is a Nuclear Safety Related modification because it changes valves which form a part of the pressure retaining boundary of a Safety Class 3/Quality Group C system.

Based upon the requirements of 10 CFR 50.59, an unreviewed safety question does not exist because:

1. The ICW system is not considered in determining the probability of design basis accidents (i.e., LOCA, MSLB, LOOP, etc.).

2. The consequence of a malfunction of equipment important to nuclear safety is not made more serious due to the design redundancy of the ICW system. The separation criteria is maintained by the valves designed in accordance with ASME Section III requirements.

3. The possibility for an accident or malfunction of a different type than any previously evaluated in the FSAR is not created because the failure of any of these valves is a passive failure which is enveloped by the evaluations of the FSAR.

4. The margin of safety as defined in the basis of a Technical Specification remains unchanged because the redundancy of the ICW system is maintained.

   Based on the above evaluation and information contained in the Design Analysis, the modification can be implemented without prior NRC approval because an Unreviewed Safety Question does not exist and a change to the Technical Specifications is not required.

480V AC LOAD CENTER 1B-2 TRANSFORMER COOLING

ABSTRACT

This Engineering Package covers modification to the 480 Vac, 1B-2 Load Center to provide a permanent source of 240 Vac, Class 1E power for the station service transformer's cooling fans. This modification consists of installing a new control power transformer, complete with primary and secondary overcurrent protection, in the existing transition compartment of 1B-2 Load Center and the interconnection of this power source to the existing automatic/manual control scheme associated with the cooling fans.

The 1B-2 Load Center is classfied as Class 1E, seismic Category I equipment and performs a safety related function. Therefore, this Plant Change/Modification (PC/M) is classified as nuclear safety related. The implementation of this PC/M does not require a change to the plant Technical Specifications. The modification does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

This Engineering Package covers modification to the 480 Vac, 1B-2 Load Center to provide a permanent source of 240 Vac, Class 1E power for the station service transformer's cooling fans. The 1B-2 Load Center is classified as Class 1E, seismic Category I equipment and performs a safety related function. Therefore, this PC/M has been classified nuclear safety related and nuclear safety related design requirements were applied to this EP.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased since:

All modifications are being performed on the "B" load group Load Center. The only modification is to restore the installation similar to its original condition. New components installed by this modification include current limiting primary fuses and a secondary circuit breaker, which provide overcurrent protection of the new control power transformer and its secondary circuit. Should failure of the new components occur, they will be removed from their power source by these protective devices without affecting the rest of the "B" electrical system. Also, new components are environmentally and seismically qualified to the required environment and will remain functional during previously evaluated accidents. Therefore, the probability of occurrence or the consequences of previously evaluated accidents are not increased.

New components are being provided by Brown Boveri, formally ITE, the original manufacturer of the 1B-2 Load Center. In addition, the manner in which the components are mounted is per Brown Boveri/industry standard installation details. The Brown Boveri Addendum to the 1B-2 Load Center Certificate of Conformance (Reference Attachment 4) has been reviewed and it has been verified that the addition of the new components does not affect the existing equipment's environmental or seismic qualification. Thus, this modification does not compromise the operation/reliability of the existing installation and the equipment will function during accidents as previously evaluated.

The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created since:

All modifications are being performed on a auxiliary supporting feature of the "B" load group Load Center, 1B-2, with no modifications required to the independent, redundant "A" load group Load Center, 1A-2. Primary and secondary overcurrent protection have been provided for the new control power transformer.

All new components and the interconnecting wiring are environmentally and seismically qualified to the required environment and will be able to provide power to the 1B-2 Load Center station service transformer cooling fans post Design Basis Accident. Based on this and Paragraph 3.3.1, no accidents/malfunctions different than those evaluated previously are created by this modification.

The margin of safety as defined in the bases for any technical specification is not reduced since the modification ensures that the equipment will function as previously evaluated during operation/accidents. Also, as 1B-2 Load Center must be deenergized prior to performing any work, implementation of this PC/M will be coordinated with Plant Operations so the criteria established by the plant Technical Specifications, eg, Specification 3/4.8, are not violated (Reference Section 9.0). Therefore, the implementation of this PC/M does not require a change to the plant's Technical Specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

ROSEMOUNT AND VALCOR EQ ENHANCEMENTS

ABSTRACT

This Engineering Package (EP) provides for the modification of seven (7) Valcor Engineering Corporation solenoid valves and the relocation of one (1) Rosemount level transmitter. These modifications eliminate the need to perform additional equipment qualification (EQ) analysis on this equipment while bolstering the applicable qualification documentation packages. This is accomplished by raising the transmitter on the Instrument Rack 50.1 above flood level and adding NAMCO Controls conduit seal assemblies to the electrical power leads of seven (7) Valcor solenoid valves to protect against ingress of containment chemical spray into the valve controls.

This modification provides for increased protection to Nuclear Safety Related equipment and improves the margin of safety of the pressurizer level instrumentation and the hydrogen sampling system. The consequences of malfunction of equipment important to safety previously evaluated in the St Lucie - Unit 1 Updated Final Safety Analysis Report (UFSAR) are unchanged.

This PCM is classified Nuclear Safety Related since it involves equipment that serves to mitigate the consequences of a Design Bases Event (DBE).

The implementation of this PCM does not require any change to the St Lucie - Unit 1 Technical Specifications. The modifications, as provided by this package, do not involve an unreviewed safety question and prior Commission approval for the implementation of this package is not required. Plant safety and operation are not affected.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurence or the consequences of an accident or malfunction of the equipment important to safety previously evaluated in the safety analysis report may be increased, or (ii) if a possibililty for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

The proposed modification affects the location of Rosemount level transmitter LT-1110X and the electrical conduit connection to seven (7) Valcor isolation solenoid valves in the hydrogen sampling system.

The probability of occurence of an accident previously addressed in the St Lucie - Unit 1 Updated Final Safety Analysis Report (UFSAR) is not increased by this modification. This EP in fact decreases the possibility of Rosemount level transmitters' failure by reducing the probability of failure due to flooding. Probability of failure of the Valcor solenoid valves is reduced by eliminating the likelihood of the intrusion of chemical spray in the electrical conduit. The possibility of new DBEs not considered in the UFSAR is not created since the design philosophies of pressurizer level and hydrogen analyzer have been previously addressed in the UFSAR. This modification is an enhancement to existing hydrogen analyzer isolation valves as well as pressurizer level transmitter LT-1110X.

Due to the fact that this EP involves cables identified as essential to safe reactor shutdown (pressurizer level transmitter LT-1110X), Appendix "R" requirements have been considered and addressed in this package (Section 2.1.6). As such, the design of this package is in compliance with the applicable codes and UFSAR requirements for fire protection equipment.

This EP is considered Nuclear Safety Related since it involves pressurizer level instrumentation (used for safe reactor shutdown) and hydrogen sampling/analyzer (used for the mitigation of a design bases event). As the evaluation of failure mode (Section 2.2.8) indicates, the degree of protection to Nuclear Safety Related equipment (margin of safety) is increased and the consequences of malfunction of equipment important to safety previously evaluated in the UFSAR are unchanged. The possibility of malfunctions of a different type than those analyzed in the UFSAR is not created.

The implementation of Nuclear Safety Related PC/M 153-186 does not require a change to the Plant Technical Specifications, nor does it create an unreviewed safety question.

The foregoing consitutes, per 10CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

IE BULLETIN 85-03 MOV SWITCH SETTINGS

ABSTRACT

NRC IE Bulletin 85-03 requires that operating nuclear plants develop and implement a program to ensure that switch settings on selected safety-related motor-operated valves (MOV's) are correctly selected, set and maintained to accommodate the maximum differential pressures expected on these valves during all postulated events within the design basis. Item a) of the bulletin requires that the design basis for those MOV's located in AFW and HPSI systems be reviewed to determine the maximum differential pressure expected during both opening and closing strokes for all postulated events. This effort was performed for St. Lucie Units 1 and 2 by Combustion Engineering as part of the CE Owner's Group (CEOG) Tasks 528 and 531. The results of the Item a) were subsequently transmitted to the NRC via FPL letter L-86-204, dated May 15, 1986.

Item b) of Bulletin 85-03 requires that the licensee establish the correct MOV switch settings based on the previously determined maximum differential pressure. All switches, including torque switches, torque bypass switches, position limit, position indication, overloads, etc., shall be considered. This design package provides the overall switch setting guidelines for each MOV, in addition to the specific design information necessary to set both the open and close torque switches and meet the requirements of Bulletin 85-03.

Once the correct switch settings have been incorporated into the respective MOV, Item c) of IE Bulletin 85-03 requires that each MOV be stroke tested against the maximum differential pressure established in Item a) to verify operability.

Because all of the MOV's associated with Bulletin 85-03 are safety-related, this engineering package has been classified as nuclear safety-related. A review of the switch setting changes to be implemented by this PC/M was performed against the requirements of 10CFR 50.59, and it was concluded that these modifications do not constitute an unreviewed safety question and do not require a change to the plant Technical Specifications.

This supplement revises the torque switch settings for valve V-3654 to account for actual field testing. This condition had been previously justified via Safety Evaluation JPE-M-87-038, Rev. 1. The Engineering Package safety classification and safety evaluation are unaffected.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, the modification described in this engineering package does not constitute an unreviewed safety question because:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the FSAR is not increased. This engineering package only provides the necessary design information required to set MOV switch settings utilizing MOVATS signature analysis techniques. The recommended switch settings are considered enhancements to the existing settings to further ensure valve operability. Also, FSAR design bases were reviewed to determine the maximum loading conditions on each MOV to ensure the switch settings were properly selected. Furthermore, Item c) of Bulletin 85-03 requires that each MOV be stroke tested under maximum differential pressure conditions to ensure valve operability.

ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created. No hardware modifications are performed as part of this PC/M. The proposed MOV switch settings alter accident mitigating equipment to further enhance operability. However, malfunctions of these MOV's do not in themselves initiate an accident. Therefore, no new accidents have been created.

Additionally, the specified modifications do not introduce any new failure modes for the equipment. Therefore, no different malfunctions of the equipment than those previously analyzed are introduced.

iii) The margin of safety as defined in the basis for any Technical Specification has not been reduced. This modification does not impact the Technical Specification requirements for the associated equipment. Valve stroke times are not impacted. Therefore, the margin of safety controlled by the Technical Specifications is preserved.

In conclusion, the change proposed in this engineering package is acceptable from the standpoint of nuclear safety does not involve an unreviewed safety question and prior NRC approval for implementation is not required.

NRC IE BULLETIN 85-03 - MOV POSITION INDICATION

ABSTRACT

This Engineering Package covers modifications to the safety related Motor Operated Valves (MOV's) in the Auxiliary Feedwater (AFW) and the High Pressure Safety Injection (HPSI) systems.

This Engineering Package will provide the engineering and design details required to implement the close to open torque bypass switch and closed position indication wiring modifications.

The MOV's in the AFW and HPSI systems are required for plant safe shutdown and classified as Class 1E, are seismically qualified and perform a safety related function. Therefore, this PC/M is considered Nuclear Safety Related.

This PC/M does not constitute an unreviewed safety question since the modifications described above will not have an adverse impact on plant operations or safety related equipment.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

This Engineering Package provides the engineering and design details required to install additional rotors and/or internal wiring changes to MOV's in the AFW and HPSI systems. PC/M 001-187 increases the closed to open torque bypass switch settings which impact the closed position indicating light. Increasing the number of rotors from two to four will allow the limit switch for the closed position indicating light to be located on a rotor other than that used for the torque bypass switch. Motor-operated valves that have four rotors will only require internal wiring changes. The addition of the new rotors does not affect the existing equipment qualifications.

The implementation of this Engineering Package increases the availability of the MOV's during safe shutdown conditions and improves the MOV position indication provided to the control room operators.

The MOV's that are being modified perform safety related functions within the AFW and HPSI systems and are designed for operation under conditions that could be imposed by a Design Basis Accident (DBA). This EP has been classified as Nuclear Safety Related.

Based on the preceeding, the following conclusions can be made:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased, since the modifications to the MOV's enhances the operability of the equipment. The addition of rotors and/or internal wiring changes to the valves will prevent the possibility of inaccurate remote closed position indication resulting from the increased bypass limit switch settings.

(ii) As a result of this modification, there is no possibility for accident or malfunction of a different type than any previously evaluated. This modification alters accident mitigating equipment to enhance their operation. There was no introduction of any new failure mode for the equipment.

(iii) This modification does not reduce the margin of safety as defined in the bases for any Technical Specification. The safety function that is controlled by the various applicable Technical Specifications is maintained by this change. The proposed design ensures that the MOV's will function as assumed during an accident. Thus the margin of safety provided by the Technical Specifications is preserved.

The implementation of this PC/M does not require a change to the plant Technical Specifications.

The foregoing constitutes, per 10CFR50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

OVERPRESSURE MITIGATION SYSTEM MODIFICATIONS

ABSTRACT

This Engineering Package (EP) details the modifications required on the Overpressure Mitigation System (OMS) to provide changes to the Power Operated Relief Valves (PORVs) setpoints. The new setpoints are derived from Combustion Engineering's report on Pressure/Temperature (P/T) limits and Low Temperature Overpressure Protection (LTOP) for 10 Effective Full Power Years (EFPY). The re-analysis of the OMS setpoints ensures that the Reactor Coolant Systems Pressure will be maintained below the applicable P/T limits during the operating period ending at 10 EFPY. The new P/T limits are identified to provide adequate protection against rapid propogation of a flaw in the reactor vessel with consideration given to the optimization of heatup and cooldown rates.

The OMS setpoint modifications are classified Quality Related because any failure to the OMS setpoint bistable alarms under normal operating conditions or anticipated transients can result in exceeding a safety limit specified in section 3/4.4.13 and 3/4.4.14 of the Technical Specifications. Furthermore, the OMS setpoint modifications affect the RTG Boards located in the control room. This EP change will not impair the efficient operation of the OMS, nor does it create an unreviewed safety question, therefore, prior Commission approval for its implementation is not required.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report may be increased, or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report may be created, or (iii) if the margin of safety as defined in the bases for any technical specification is reduced.

The OMS protects the RCS from being pressurized beyond the curves defined by the minimum pressurization temperature curves of the Technical Specifications, while the RCS is at low temperatures. The OMS achieves its purpose of protecting the RCS from overpressurization at low, non-ductile temperatures by continuously comparing actual pressurizer pressure to two (2) pressure setpoints and corresponding temperature setpoints and by actuating the PORVs when actual pressure approaches these setpoints. The pressure and temperature comparisons and PORV actuation are both accomplished in two independent redundant OMS trains, one for each PORV.

The P/T setpoint changes outlined per this EP package are classified Quality Related because any failure to the OMS setpoint alarms under normal operating conditions or anticipated transients can result in exceeding a safety limit specified in Section 3/4.4.13 and 3/4.4.14 of the Technical Specifications. In addition, these changes also affect the RTG Boards located in the Control Room.

The following provides the justification that an unreviewed safety question does not exist.

i. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased because the OMS modification serves only to change the setpoints for the PORVs whenever an overpressurization event occurs in low temperature modes when the RCS may be water-solid. The new setpoints for overpressure protection in low temperature modes maintain the design philosophy of the OMS system.

ii The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis is not created because the new setpoints allow better control over OMS events and prevent violation of the 10CFR Appendix G pressure/temperature limits during the operating period ending at 10 EFPY. The CE report provides assurance that the system is able to perform its function assuming any single active component failure in addition to the failure that initiated the pressure transient.

iii The margin of safety as defined in the basis for any technical specification is not reduced since this OMS modification revises the setpoints for the PORVs to actuate whenever an overpressurization event occurs in low temperature modes of operation when the RCS may be water-solid. The proposed system prevents violation of the Appendix G pressure/temperature limits during an operating period ending at 10 EFPY. The implementation of the proposed LTOP system does not result in the reduction in a margin of safety; instead, it provides additional setpoints and thereby increases the margin of safety. This EP change will not impair the efficient operation of the OMS.

The implementation of this PCM does not required a change to the Technical Specifications. Although new P/T limits for reactor operation are being developed, those Technical Specification changes apply to operation beyond 7.4 EFPY, and are not directly related to the setpoint changes accomplished by this EP.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this EP is not required on that basis.

PROTECTIVE COATINGS REPAIR AND/OR REPLACEMENT IN REACTOR CONTAINMENT BUILDING

ABSTRACT

This engineering package covers the maintenance of Service Level I protective coatings on concrete and steel surfaces inside the Reactor Containment Building. This project is classified as quality related and does not constitute an unreviewed safety question.

SAFETY EVALUATION

The function of the coatings used inside the RCB is to provide protection against corrosion and contamination. The only safety related aspect is for the coatings to remain intact throughout a design basis accident to insure that no engineered safety features are interfered with. The coatings to be used are DBA tested and their composition, functional and testing requirements are addressed in the Unit 1 FSAR. For these reasons the probability of occurrence or consequences of a design basis accident or malfunction of equipment important to the safety of the plant has not been increased. In addition, there will continue to be no possibility of an accident or malfunction different than those previously evaluated in the Unit 1 FSAR. Finally, the margin of safety as defined in the plant technical specifications has not been reduced. It is therefore concluded that the repair and/or replacement of protective coatings on surfaces inside the Reactor Containment Building as outlined in this package does not pose an unreviewed safety question pursuant to 10 CFR 50.59.

CONDENSER HOTWELL NITROGEN INJECTION CONNECTIONS

ABSTRACT

This Engineering Package is to provide connections to the condensers to inject nitrogen into the condenser hotwells. Test have shown that injecting 1 cfm of nitrogen into a condenser shell reduces the dissolved oxygen level in the hotwell condensate by approximately 2 ppb. It is theorized that because of low air in-leakage into the condensers (condensers are designed based upon the fact there will be air in-leakage), the flow of the non-condensibles in the air removal section of the tube bundle is not established. Therefore, oxygen is entrained as the condensate drips through the air pockets which form as a result of the stagnant conditions. The injection of an inert gas such as nitrogen enables the air removal section of the condenser to establish the flow required to remove non-condensibles without introducing oxygen into the system.

A review of the changes to be implemented by this Engineering Package was performed against the requirements of 10CFR 50.59. As a result, these condenser modifications are classified as non-safety related, do not constitute an unreviewed safety question and will not affect plant safety or operation (as addressed in Section 3 "Safety Evaluation").

Safety Evaluation

With respect to Title 10 of the Code of Federal Regualtion, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

This Engineering Package is to provide connections to the condensers to inject nitrogen into the condenser hotwells. Tests have shown that injecting 1 cfm of nitrogen into a condenser shell reduces the dissolved oxygen level in the hotwell condensate by approximately 2 ppb. It is theorized that because of low air in-leakage into the condensers (condensers are designed based upon the fact there will be air in-leakage), the flow of the non-condensibles in the air removal section of the tube bundle is not established. Therefore, oxygen is entrained as the condensate drips through the air pockets which form as a result of the stagnant conditions. The injection of an inert gas such as nitrogen enables the air removal section of the condenser to establish the flow required to remove non-condensibles without introducing oxygen into the system.

Based on the above description, the modification included in this Engineering Package (EP) is considered to be non-safety related. This EP does not involve on unreviewed safety question, and the following are bases for this justification:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased. The condensers are not used in any safety analysis for accidents or malfunction of equipment and as such are non-safety related and will have no effect on equipment vital to plant safety.

ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created. The components involved in this modification have no safety related function and no changes have been made to the operational design of the system.

iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this PCM, since the components involved in this modification are not included in the bases of any Technical Specification.

The implementation of this PCM does not require a change to the plant Technical Specifications.

The foregoing constitutes, per 10CFR 50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

ONE AND TWO FEEDWATER HEATER AND EXTRACTION PIPE SHIELDING

ABSTRACT

This Engineering Package provides generic details for the repair or replacement of damaged shielding for the Extraction Stress Pipe and Expansion Joints and the Feedwater Heaters inside the Condenser.

The Engineering Package is classified as non-safety related since it is a modification to a non-safety related system. The safety evaluation has shown that this EP does not constitute any unreviewed safety question.

This system is non-safety related and will have no effect on equipment vital to plant safety.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Speciification is reduced.

This PCM involves the repair and replacement of the shielding for the extraction piping and feedwater heaters located in the condenser. It has been classified as non-safety related and does not involve an unreviewed safety question because:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased. The St Lucie Unit No 1 FSAR, Section 10.4 reads as follows: Except for a portion of the feedwater system piping, the features, components and system described in this section serve no safety function since they are not required for safe shutdown or to mitigate the effects of a LOCA and their failure will not result in the release of significant uncontrolled radioactivity. The St Lucie Unit No 1 FSAR, Section 10.4.1 describes the condenser where the extraction piping and low pressure Feedwater heaters 1 and 2 are located. This system is non-safety related and will have no effect on equipment vital to plant safety.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis is not created. The components involved in this modification do not perform any safety related function. No changes have been made to the operational design of the condenser or the extraction piping.

(iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this PCM, since the components involved in this modification are not directly included in the bases of any Technical Specification.

The implementation of this PCM does not require a change to the plant Technical Specification.

The foregoing constitutes, per 10CFR50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

CCW HEAT EXCHANGERS - SHELL SIDE DRAIN ADDITION

ABSTRACT

THIS ENGINEERING PACKAGE PROVIDES DETAILS FOR THE ADDITION OF TWO FLANGED SIX INCH SHELL SIDE DRAIN CONNECTIONS TO THE COMPONENT COOLING WATER HEAT EXCHANGERS. THE DRAINS ARE DESIRED TO IMPROVE FLUSHING EFFECTIVENESS AND TO REDUCE CRITICAL PATH DRAIN TIME BETWEEN FLUSHES FOLLOWING RETUBING UNDER PC/M 340-183 DURING THE 1987 UNIT 1 REFUELING OUTAGE.

THIS MODIFICATION IS CLASSIFIED AS NUCLEAR SAFETY RELATED SINCE IT AFFECTS THE PRESSURE BOUNDARY OF THE COMPONENT COOLING WATER HEAT EXCHANGERS. THE COMPONENT COOLING WATER HEAT EXCHANGERS FORM A PORTION OF THE COMPONENT COOLING WATER AND INTAKE COOLING WATER SYSTEMS WHICH ARE CLASSIFIED AS QUALITY GROUP C, SEISMIC SYSTEMS.

THIS PC/M HAS BEEN REVIEWED TO THE CRITERIA SET FORTH IN 10 CFR 50.59 AND HAS BEEN DETERMINED NOT TO INVOLVE AN UNREVIEWED SAFETY QUESTION, NOR DOES IT REQUIRE CHANGES TO THE TECHNICAL SPECIFICATIONS. PRIOR COMMISSION APPROVAL IS NOT REQUIRED FOR IMPLEMENTATION OF THIS MODIFICATION.

SAFETY EVALUATION

THIS MODIFICATION CONSISTS ONLY OF THE ADDITION OF TWO SIX INCH FLANGED PIPE STUB DRAINS TO THE UNDERSIDE OF EACH COMPONENT COOLING WATER HEAT EXCHANGER SHELL TO REDUCE DRAINAGE TIME AND TO IMPROVE FLUSHING EFFECTIVENESS. THE NEW DRAINS WILL PERFORM NO "ACTIVE" SAFETY RELATED FUNCTION, ONLY THE PASSIVE FUNCTION OF RETAINING THE PRESSURE BOUNDARY INTEGRITY OF THE COMPONENT COOLING WATER SYSTEM. THIS MODIFICATION IS NUCLEAR SAFETY RELATED SINCE IT AFFECTS THE PRESSURE BOUNDARY OF THE CCW HEAT EXCHANGERS WHICH ARE QUALITY GROUP C, SEISMIC COMPONENTS (REFERENCE UNIT 1 FSAR TABLE 3.2-2).

BASED ON THE FOLLOWING ARGUEMENTS, IT IS DEMONSTRATED THAT NO UNREVIEWED SAFETY QUESTIONS EXIST AND THAT THE MODIFICATION MAY BE IMPLEMENTED WITHOUT PRIOR COMMISSION APPROVAL:

-THE PROBABILITY OF OCCURENCE OR THE CONSEQUENCES OF AN ACCIDENT OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE SAFETY ANALYSIS REPORT HAS NOT BEEN INCREASED.

THIS MODIFICATION DOES NOT AFFECT THE PROBABILITY OF OCCURENCE OF AN ACCIDENT PREVIOUSLY EVALUATED SINCE THE CCW HEAT EXCHANGERS ARE NOT CONSIDERED IN DETERMINING THE PROBABILITIES OF ACCIDENTS.

THE CONSEQUENCES OF POSTULATED ACCIDENTS HAVE NOT BEEN MADE MORE SEVERE SINCE THIS MODIFICATION DOES NOT AFFECT THE FUNCTIONAL PERFORMANCE OF THE COMPONENT COOLING WATER HEAT EXCHANGERS AND DOES NOT AFFECT ANY OTHER EQUIPMENT WHICH WOULD MITIGATE THE CONSEQUENCES OF POSTULATED ACCIDENTS.

-THE POSSIBILITY OF AN ACCIDENT OR MALFUNCTION OF A DIFFERENT TYPE THAN PREVIOUSLY EVALUATED IN THE FINAL SAFETY ANALYSIS REPORT HAS NOT BEEN CREATED.

THE NEW FLANGED DRAIN CONNECTIONS ARE TYPICAL IN TERMS OF DESIGN CODES TO THE OTHER FLANGED PIPING IN THE COMPONENT COOLING WATER SYSTEM THEREFORE NO NEW ACCIDENTS OR MALFUNCTIONS ARE INTRODUCED.

-THE MARGIN OF SAFETY AS DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATION HAS NOT BEEN REDUCED.

THE MODIFICATION DESCRIBED HEREIN IN NO WAY AFFECTS ANY TECHNICAL SPECIFICATION.

10CFR50.59 ALLOWS MODIFICATIONS TO NUCLEAR FACILITIES AS DESCRIBED IN THE FSAR WITHOUT PRIOR COMMISSION APPROVAL IF AN UNREVIEWED SAFETY QUESTION IS DEMONSTRATED NOT TO EXIST AND IF TECHNICAL SPECIFICATIONS ARE UNAFFECTED. THE PRECEDING ARGUEMENTS DEMONSTRATE THAT NO UNREVIEWED SAFETY QUESTIONS EXIST AND THAT REVISION TO THE TECHNICAL SPECIFICATIONS IS NOT REQUIRED, THEREFORE THIS MODIFICATION MAY BE IMPLEMENTED WITHOUT PRIOR COMMISSION APPROVAL.

MISCELLANEOUS SNUBBER MODIFICATION

ABSTRACT

This EP provides engineering and design for miscellaneous modifications to snubbers as a result of the inservice inspection findings. The modification generally includes a replacement of the existing snubber and/or its components with an upgraded snubber or its components of a different or same manufacturer.

This EP has been classified as Safety Related because the modification affects safety related piping system. The modifications have been reviewed under the criteria of 10CFR 50.59 and no unreviewed safety questions have been demonstrated to exist.

This EP has no adverse impact on the plant safety and operation.

The implementation of this EP will not require a change to the Plant Technical Specification as snubbers have not been removed or added to the Table 3.7-2b of the Plant Technical Specification 3.7.10.

Supplement 1 provides engineering and designs for modification to additional snubbers as a result of the Inservice Inspection findings.

This supplement has no adverse impact on the plant safety and operation.

The implementation of this supplement will require a change to the plant Technical Specification Table 3.7-2a and 2b. However, per the foot note of these tablesto include the safety related snubber for restraint Mk No RC-247-H3 in the Table 3.7-2b. However, per the foot note of these tables, a snubber may be added or deleted from safety related systems without prior License Amendment to these tables provided a revision to these tables are included with the next License Amendment request.

Supplement 2 provides engineering and designs for modification to additional snubbers as a result of the Inservice Inspection findings.

This supplement has no adverse impact on the plant safety and operation.

The implementation of this supplement will require a change to plant Technical Specification 3.7.10 to include the safety related snubber for restraint Mk No RC-247-H3 in Table 3.7-2b. However, per the foot note of these tables, a snubber may be added or deleted from safety related systems without prior License Amendment to these tables provided a revision to these tables are included with the next License Amendment request.

Supplement 3 provides engineering and design for modification to two restraints which were found to have been damaged during the inservice inspection.

The safety evaluation has shown that this modification does not constitute an unreviewed safety question; however, as indicated implementation of supplement 1 and 2 of this PCM will require a change to Plant Technical Specification 3.7.10 which must be included in the next License Amendment request. Therefore, prior NRC approval is not required for implementation.

This supplement has no adverse impact on the plant safety and operation.

The implementation of this supplement will not require a change to the Plant Technical Specification.

Supplement 4 provides engineering and design for modifications to additional snubbers as a result of Inservice Inspections findings. The original pipe stress analysis & pipe support design has not been modified for this PC/M.

The implementation of this supplement does not change the previous conclusion that Technical Specification Section 3.7.10 and tables 3.7-2a or 3.7-2b require modification. This modification requires a change to Plant Technical Specification 3.7.10, which must be included in the next License Amendment request. The safety evaluation has shown that the modifications included in Revision 4 of this EP does not constitute an unreviewed safety question. Based on these points, it is concluded that prior NRC approval is not required for implementation.

This supplement has no adverse impact on the plant safety and operation.

The implementation of this PC/M will require a change to Plant Technical Specification.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any Technical Specification is reduced.

This EP is for modification of existing restraints to either replace snubbers of one manufacturer with that of the same or larger size and load rating from a different manufacturer or to replace existing restraint components with a different design. This is required to resolve restraint anomalies found during the Inservice Inspection of the restraints. This modification affects safety related piping systems. Accordingly, this modification is classified as safety related. This EP does not constitute an unreviewed safety question and the following are the basis for this justification:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased since the restraint systems for the piping will remain functionally identical to existing configuration. In addition, since the restraint configuration is not changed all previous analysis conclusions are still valid.

(ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created because, no changes have been made to the operational design of either the snubbers or the restraints and the system remains functionally identical to the requirements specified in the existing stress anlaysis of record.

(iii) The margin of safety as defined in the bases for any Technical Specification is not affected by this modification because the replacement components utilized perform the same restraining function as those they replace.

The implementation of this PCM will require a change to the Plant Technical Specification as snubbers have been removed from Table 3.7-2b and added to Table 3.7-2a of the Plant Technical Specification 3.7.10. The safety related snubber for restraint Mark No RC-247-H3 is required to be added to Table 3.7-2b. As per the footnote of these tables, snubbers may be added to or removed from safety related systems without prior License Amendment to these tables provided a revision to Tables 3.7-2a and 3.7-2b are included with the next License Amendment request.

REPLACEMENT OF RWT NOZZLE FOR LINE I-3"-CS-46

ABSTRACT

This Engineering Package (EP) provides the design for the replacement of the Refueling Water Tank (RWT) nozzle for line I-3"-CS-46. The existing nozzle has extensive pitting and corrosion, therefore it was considered advisible to replace it.

This EP is classified as nuclear safety related since it repairs a safety related piece of equipment. The safety evaluation has shown that this EP does not constitute any unreviewed safety quetsions, nor does it require a Technical Specification change. Therefore, prior NRC approval is not required for implementtion of this PCM.

This EP has no adverse impact on nuclear plant safety and operation.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

This modification replaces a 3 inch nozzle in the RWT with an identical nozzle except for the welding detail which is equivalent but not identical. This welding detail minimizes radiation exposure by reducing welding requirements inside the RWT.

The modification included in this Engineering Package is considered to be safety related and does not involve an unreviewed safety question because:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since the replacement of the nozzle will not impact the quantity of water supplies to the charging pumps.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since no changes have been made to the operational design of the RWT and the new nozzle is equivalent in design.

(iii) This modification does not change the margin of safety as defined in the bases for any technical specification.

Implementation of this PCM does not require a change to the plant technical specifications.

The foregoing constitutes, per 10CFR50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

PRESSURIZER SURGE LINE SAMPLE VALVE (V1210) REPLACEMENT

ABSTRACT

Existing pressurizer surge line sample valve V1210 is leaking, has a damaged valve stem and cannot be repaired. The valve will be replaced with a new valve originally procured for Unit 2 for similar service.

The Reactor Coolant System and Sampling System in which this valve is located in safety related. Accordingly, this Engineering Package is classified as nuclear safety related. The safety evaluation has shown that this EP does not a constitute an unreviewed safety question and prior NRC approval is not required for implementation.

This EP neither reduces the margin of safety, as defined in the bases for any Technical Specification, nor has any impact on the plant safety and operation.

SAFETY EVALUATION

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specification is reduced.

This safety related modification does not involve an unreviewed safety question because:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased. The replacement valve has the same qualification for the location and service on the valve which it replaces. Total failure of the replacement valve will result in the same consequences or the total failure of the existing valve. Total failure of this valve will cause a small loss of coolant which is limited by a orifice to be less than the capacity of a single charging pump. This condition is within the existing FSAR Analyses.

ii) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis is not created. This modification does not change the system function or design. This modification is the replacement of a valve with a similar but not identical valve. The failure mode of this valve will be identical with the existing valve and, as stated above, is within the FSAR anlayses

iii) The margin of safety is defined in the bases for any Technical Specification is not affected by this modification since the valve involved does not form the bases for any Technical Specification.

The implementation of this PCM does not require a change to the plant Technical Specification.

The foregoing constitutes, per 10CFR 50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for implementation of this PCM is not required.

CONDENSER OUTLET TUBE SHEET AND WATERBOX COATINGS

ABSTRACT

This engineering package address the addition of an epoxy coating to the to the condenser outlet tube sheets and waterboxes. This modification will enhance the corrosion resistance of the tube sheets and waterboxes and allow reduction of the cathodic protection system potentials and current densities.

The condensers and the plant circulating water system are classified as non-nuclear safety related and therefore, the modification addressed in this engineering does not constitute an unreviewed safety question. Furthermore, the addition of a protective coating to the condenser outlet tube sheets and waterboxes does not require a change to the plant Technical Specifications.

SAFETY EVALUATION

As noted in FSAR Sections 9.2.3 and 10.4.5, the condensers and circulating water system perform no nuclear safety related function. A failure mode evaluation of the proposed condenser outlet tube sheet and waterbox coatings has determined there is no potential for interaction with equipment or functions important to nuclear safety. Accordingly, the modification addressed by this engineering package is classified as non nuclear safety related.

Based on the above evaluation and information supplied in the design analysis, it has been demonstrated that an unreviewed safety question as defined by 10 CFR 50.59 does not exist.

- The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report has not been increased.

  Since there is no potential for interaction between the modification addressed by this engineering package and equipment of functions important to safety, previous safety analysis report evaluations related to safety remain unaffected.

- The possibility of an accident or malfunction different than those previously evaluated in the safety analysis report has not been created.

  No new accidents or malfunctions associated with the failure of the condenser outlet tube sheet and waterbox coatings have been created.

- The margin of safety as defined in the basis for any Technical Specification has not been reduced.

  Since there is no potential for interaction between the modification addressed by this engineering package and equipment or functions important to safety, the margin of safety as defined in any Technical Specification remains unaffected.

conclusion, the modification proposed in this engineering package is acceptable from standpoint of nuclear safety, does not involve an unreviewed safety question and does not require a change to any Technical Specifications. Accordingly, NRC approval prior to implementation is not required.

REPLACEMENT OF RAYCHEM SPLICES AND CONAX CONDUIT SEALS

ABSTRACT

This Engineering Package covers modifications to the boxes and/or conduit seals and splices associated with the safety related instruments listed in the Environmental Qualification List for 10CFR50.49 and located in the Reactor Containment Building.

This Engineering Package will provide the engineering and design details required to implement the replacement of the boxes that are used for splicing the Conax conduit seal pigtail cable to the plant cable and replacement of Raychem splices at the boxes and the Electrical Penetration Assemblies.

The majority of instruments listed in Attachment 7.4 and their associated splices and conduit seals are classified as Class 1E, are seismically qualified and perform a safety related function. All the instruments listed in Attachment 7.4 are required for plant safe shutdown. Therefore, this EP is considered Nuclear Safety Related.

This EP does not constitute an unreviewed safety question and the modifications described were reviewed in accordance with 10CFR50.59 and determined to have no adverse impact on plant operations or safety related equipment.

The implementation of this PC/M does not require a change to the plant Technical Specification.

This change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

Supplement 1 incorporates additional safety related instruments, their associated Equipment Qualification Documentation Package and the removal of the holdpoints for Equipment Qualification Documentation Packages for the Raychem splices and the Conax conduit seals. The additional equipment, EQ Documentation Package and removal of holdpoints, does not affect the original safety evaluation, except for the removal of the holdpoints.

Supplement 2 revises attachment 7.4 and incorporates detail drawings for the Raychem splices. This additional information completes the modifications covered in this EP and does not affect the conclusions of the original safety evaluation.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety defined in the bases for any Technical Specification is reduced.

This Engineering Package provides the splice inspection criteria and the engineering and design details to implement, as required, the replacement of splice boxes and splices.

The implementation of this Engineering Package increases the availability of the equipment during safe shutdown conditions by improving the reliability of the splices at the equipment and penetration. This modification shall be implemented prior to entering Mode 4.

The equipment, listed in Attachment 7.4.1 and 7.4.3, whose associated boxes and/or conduit seals and splices need replacement, perform safety related functions within the various plant systems and are designed for operation under conditions that could be imposed by a Design Basis Accident (DBA). The power and control terminations for the equipment listed in Attachment 7.4.2, though non-safety in nature, could have an adverse affect on the safety related terminations due to the high energy levels associated with these non-safety circuits. Therefore, these non-safety circuits (with the exception of low energy annunciator circuits and circuits identified via FPL letter JPE-PSL-87-0787 dated 3/20/87), which could be energized following a LOCA event, have been provided with splices or connectors (as per Chapter 3, Section 3.11.5.4 of the FSAR), which effectively eliminated the potential for adverse interaction between safety and non-safety related terminations by eliminating the high energy circuits from the harsh LOCA environment.

This EP has been classified as Nuclear Safety Related.

Based on the preceeding, the following conclusions can be made:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased, since the modifications to the boxes and/or conduit seals and splices associated with the equipment listed in Attachment 7.4 enhances the operability of the equipment in a harsh environment post DBA.

(ii) As a result of this modification, there is no possibility for accident or malfunction of a different type than any previously evaluated. This modification alters associated components of accident mitigating equipment to enhance their operation. Existing accuracies with respect to control or monitoring functions of the instrumentation loops, under modification, will not be affected by this change. There is no introduction of any new failure mode for the equipment.

(iii) This modification does not reduce the margin of safety as defined in the bases for any Technical Specification. The safety function that is controlled by the various applicable Technical Specifications is maintained by this change. The proposed design ensures that the equipment will function as assumed during an accident. Thus the margin of safety provided by the Technical Specifications is preserved.

The implementation of this PC/M does not require a change to the plant Technical Specification.

The foregoing constitutes, per 10CFR50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

MSCV DISK NUT LOCKING PLATE MODIFICATION

ABSTRACT

This Engineering Package (EP) is to replace the locking washer and nut on each Main Steam Check Valve (MSCV) disk stud with a new locking plate and nut. The existing locking washer design has been adequate in service but is showing some wear indicating that replacement would be required prior to the end of the valve design life. The new locking device will provide an enhanced method of locking the nut and will preclude the possibility of the nut rotating and allowing the disk to become loose.

The valves considered in this EP are in the Main Steam System. This system is designated as nuclear safety related and seismically qualified on the ASME Section XI Code Boundary drawings, and therefore this modification is classified as safety related. The safety evaluation has shown that this EP does not constitute an unreviewed safety question and prior NRC approval is not required for implementation.

The implementation of this EP will have no impact on plant safety or operation.

Safety Evaluation

With respect to Title 10 of the Code of Federal Regulation, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question; (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the bases for any Technical Specificatin is reduced.
3/24/87

The modifications included in this Engineering Package are for the replacement of the locking device on the MSCV disk/tail link connection. The two (2) valves affected are in the Main Steam System.

Based on the above description, the modificatin included in this Engineering Package (EP) is considered to be safety related. This EP does not involve an unreviewed safety question, and the following are bases for this justification:

i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased since the modification will eliminate the possibility of the valves' disk becoming loose from the mount. Accordingly, the modification ensures reliable operation of the valves and consequently of the system in which they are installed.

ii) As a result of this modification, there is no possibility for an accident or malfunction of a different type than any previously evaluated because the modification is simply to replace a component of the valve with a component with a higher level of conservatism in the valve. No changes are made to the operational design of the system in which the modification is made.

iii) This modification does not reduce the margin of safety as defined in the basis for any Technical Specification because it neither changes the design parameter of the locking device nor does it change the system design flow or functional requiremets.

The implementation of this PCM does not require a change to the plant Technical Specifications.

The foregoing constitutes, per 10CFR 50.59 (b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PCM is not required.

REACTOR CONTAINMENT BUILDING TELESCOPING JIB CRANE SEISMIC RESTRAINTS

ABSTRACT

This engineering package is being issued in response to the commitment made in the disposition to plant NCR 1-099. This package addresses the modification of the seismic restraints on the Reactor Containment Building (RCB) Telescoping Jib Crane and does not constitute a new design. The specified slip hooks would not engage the existing restraint pad eyes. The seismic restraints are designed to hold the crane in place during a design basis earthquake.

This engineering package will document the design change and will provide guidelines for establishing a maintenance procedure to assure reinstallation of the seismic restraints at the end of each outage. The modification of the seismic restraints was completed under the disposition to NCR 1-099.

The jib crane seismic restraints do not perform or affect any safety-related function. However, this PC/M is classified Quality Related since there is a potential that during a seismic event the telescoping jib crane could interact with safety-related items that are in the vicinity. Quality Related requirements are applied to this modification.

The implementation of this PC/M does not require a change to plant technical specifications. This modification does not affect plant operations or safety.

This PC/M does not constitute an unreviewed safety question and therefore does not require prior NRC approval.

SAFETY EVALUATION

Safety Analysis

In response to NCR 1-099, this engineering package addresses the modification of the RCB Telescoping Jib Crane seismic restraints.

With respect to title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The seismic restraints do not perform or affect any safety-related system or function. However, this PC/M is classified as Quality Related since failure of the seismic restraints during a design basis event (e.g. earthquake) could potentially affect safety-related systems or equipment since the jib crane is located in the RCB.

Consequently, the revised restraints have been analyzed for the design basis conditions specified in the FUSAR and Quality Related design requirements have been implemented, thus assuring the integrity of the installation.

CODE BOUNDARY DRAWING REVISION

Code boundary drawing 8770-G-090 is revised/expanded to 30 sheets to include all nuclear safety related systems. The new drawings will facilitate testing and examination under the "inservice inspection - ten year plan." No unreviewed safety questions exist as defined by 10 CFR 50.59, and no Technical Specifications are impacted by this modification. Therefore, prior commission approval is not required.

NUCLEAR SAFETY EVALUATION CHECKLIST

The written evaluation of the proposed design change to demonstrate that the change does not alter the plants design basis and is bounded by the design analyses is attached to the Design Equivalent Engineering Package. The answers below are supported by this evaluation.

TYPE OF CHANGE

Yes	No	
____	X	A change to the plant as described in the FSAR?
____	X	A change to procedures as described in the FSAR?
____	X	A test or experiment not described in the FSAR?
____	X	A change to the plant technical specifications?

EFFECT OF CHANGE

Yes	No	
____	X	Will the probability of an accident previously evaluated in the FSAR be increased?
____	X	Will the consequences of an accident previously evaluated in the FSAR be increased?
____	X	May the possibility of an accident which is different than any already evaluated in the FSAR be created?
____	X	Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
____	X	Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
____	X	May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
____	X	Will the margin of safety as defined in the bases to any technical specification be reduced?

TECHNICAL SUPPORT CENTER BLOCK WALL NO 207A MODIFICATION

ABSTRACT

This engineering package addresses the installation of a status board in the Technical Support Center (TSC), located at Elev. 62.00 in the Reactor Auxiliary Building (RAB). The board will be attached to a masonry block wall and will be used during emergency drills and plant operation.

The board and wall do not perform or affect any safety related function. However, this PC/M is classified Quality Related since there is a potential for the wall to interact with safety related items. Quality Related requirements are applied to this modification.

The implementation of this PC/M does not require a change to plant technical specifications. This modification does not affect plant operations or safety.

This PC/M does not constitute an unreviewed safety question and therefore does not require prior NRC approval.

SAFETY EVALUATION

With respect to title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously calculated in the safety analysis report may be increased; or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the safety report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The status board and masonry block wall 207A do not perform or affect any safety-related system or function. However, this PC/M is classified as quality related since failure of the wall during a design basis event (e.g. , earthquake) could potentially affect safety-related systems or equipment.

Consequently, the wall has been analyzed for the design basis conditions specified in the FUSAR and Quality Related design requirements have been implemented, thus assuring the integrity of the installation.

The modifications included in this PC/M do not involve any unreviewed safety questions because:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since this modification will have no effect on equipment required to shut down the plant and monitor the plant in a safe shutdown condition.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since the status board and masonry block wall perform no safety function and no changes have been made to any operational design. Failure of the wall could not occur since the modification has been analyzed for the design basis conditions.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification. The implementation of this PC/M does not require a change to plant technical specifications.

The foregoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question and prior Commission approval for the implementation of this PC/M is not required.

TURBINE GANTRY CRANE MAIN SHEAVE NEST UPGRADE

ABSTRACT

This engineering package is being issued in response to REA SLN-87-(Later). This package will provide the engineering documentation required for modifications to the turbine gantry crane main sheave nest. The modifications are required because of bearing failures on at least two sheaves.

The turbine gantry crane is classified as Non-Nuclear Safety Related. However, this PC/M is classified Quality Related to provide Q.C. inspection of critical load bearing welds and assure realignment of the sheave nest shaft.

A safety evaluation and failure mode evaluation has determined that the modifications addressed in this package do not constitute an unreviewed safety question as defined in 10 CFR 50.59. Furthermore, the implementation of this PC/M does not require a change to plant technical specifications and does not affect plant operations or safety. Based on the above, implementation of this PC/M does not require prior NRC approval.

SAFETY EVALUATION

This package addresses the turbine gantry crane main sheave nest support plate modifications and sheave nest shaft replacement required because of bearing failures on at least two sheaves.

With respect to title 10 of the Code of Federal Regulations, Part 50.59, a proposed change shall be deemed to involve an unreviewed safety question: (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or (ii) if the possibility for an accident or malfunction of a different type than any evaluated previously in the safety report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

The turbine gantry crane does not perform or affect any safety related system or function. However, this PC/M is classified as Quality Related to ensure Q.C. inspection of the installation.

The modifications included in this PC/M do not involve any unreviewed safety questions because:

(i) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated is not increased since this modification will have no effect on equipment required to shut down the plant and monitor the plant in a safe shutdown condition.

(ii) There is no possibility for an accident or malfunction of a different type than any previously evaluated since the turbine gantry crane does not perform any safety function and no changes have been made to any operational design.

(iii) This modification does not change the margin of safety as defined in the basis for any technical specification because the turbine gantry crane is not addressed by any technical specification. The implementation of this PC/M does not require a change to plant technical specifications.

The foregoing constitutes, per 10 CFR 50.59(b), the written safety evaluation which provides the bases that this change does not involve an unreviewed safety question. Furthermore, the implementation of this PC/M does not require a change to plant technical specifications. Based on the above, prior Commission approval for the implementation of this PC/M is not required.

INTRODUCTION

The St. Lucie Unit 1 Cycle 8 Safety Analysis was performed to support operation with 133 assemblies of Batch H and J fuel, along with 84 fresh Batch K natural uranium axial blanket assemblies. Thirty six of the Batch K assemblies contain no burnable absorber rods, 24 assemblies contain eight 4 w/o Gd_2O_3 rods, and the remaining Batch K assemblies contain 4 B_4C-Al_2O_3 rods in addition to eight gadolinia bearing rods. The bottom 3.04 inches of the fuel rod is composed of a long Zircaloy end cap in order to reduce the possibility of debris related fuel rod failures. A Technical Specification amendment for the long end cap change was issued and is found in Reference 5. In addition to this change, the reload fuel is characterized by an improved spacer spring design and, in the top and bottom spacers, by the addition of backup dimples in the spacer peripheral rod cells.

The Cycle 8 energy requirement is 10390 EFPH, based on an end-of-cycle 7 burnup of 9300 EFPH.

SAFETY EVALUATION

The Cycle 8 SAR covers the fuel management, fuel design and evaluation of the physics characteristics, shutdown margin calculations, power distributions and peaking factors throughout the cycle. Core Physics has performed 1) a detailed review of the vendor's methods and neutronic calculations 2) an independent verification of the vendor results by comparing the SAR reported physics parameters to those calculated with the core models generated by the Core Physics Group documented in Reference 6 and, 3) a comparison of calculated values to the plant Technical Specifications in regards to limiting power peaking factors, moderator temperature coefficients, shutdown margin and linear heat rates.

Based on the results of the aforementioned evaluation, it can be concluded that:

a) The vendor's physics data was calculated with approved methods and documented satisfactorily.

b) FPL's independent core physics models are in reasonable agreement with the vendor's results, and

c) The cycle 8 reload design meets the Technical Specification Limits with regards to Fr, Fxy, MTC, minimum required shutdown margin and maximum linear heat rate.

Based on projected burnups, 5 assemblies (4 Batch H-1 and 1 Batch H-4) will exceed the currently analyzed mechanical and radiological design limits of 44,500 MWD/MTU prior to the end-of-cycle 8. The most limiting of these assemblies will reach a burnup of 44,500 MWD/MTU at 9450 EFPH cycle exposure. An analysis to extend the mechanical and radiological burnup limit beyond the current value will be performed prior to that time. As a result, operation of cycle 8 with these assemblies in the core does not constitute a safety concern up to 9450 EFPH and provided the aforementioned analysis is completed prior to that cycle exposure and shows acceptable results, operation beyond 9450 EFPH will not constitute an unreviewed safety question.

As noted in the introduction, the reload fuel design incorporates minor modifications to the fuel design used in Cycle 7. The end cap in the reload fuel is lengthened from 0.4 inches to 3.04 inches while the active fuel length is reduced by the same amount, therefore maintaining the overall length of the rod. This change has been previously evaluated and was approved by the NRC (Reference 5). The reload fuel also incorporates an improved spacer spring design and spacer backup dimples have been added to the top and bottom spacer peripheral cells. The effect of these changes has been evaluated and determined to have no adverse impact relative to the fuel design basis. The Cycle 8 safety evaluation demonstrated that these changes are bounded by previous analyses.

The St. Lucie Unit 1 Cycle 8 Safety Analysis Report (SAR) (Reference 1), presents the evaluation of the reload characteristics with respect to the safety analysis presented for Cycle 6 (Reference 2)which serves as the reference cycle. The basis of the safety analysis for Cycle 8 is the same as that used for Cycle 6 with the exception of the Local Power Density (LPD) versus Axial Shape Index (ASI) Limiting Condition for Operation (LCO). The peak power density during Cycle 8 could increase over that assumed in the reference analyses so that it is necessary to limit the maximum power level when relying on the ex-core detectors for determination of the peak linear heat rate. A proposed Technical Specification change will reduce the fraction of maximum allowable power from 0.88 to 0.85 when the in-core detectors are inoperable. The proposed Technical Specification change has been submitted to the NRC per L-86-510 dated December 18, 1986. NRC approval is expected by February 28, 1987. Until approval of this Technical Specification change, this Safety Evaluation is only applicable for reactor operation up to 40% rated power when this Technical Specification is applicable. Upon approval of this Technical Specification change, power operation above 40% rated power up to 100% rated power for the St. Lucie Unit 1 Cycle 8 reload core is acceptable and will not constitute an unreviewed safety question.

The St. Lucie Unit 1 Cycle 8 SAR presents the evaluation and review of the Chapter 15 events for the reload core. FPL has reviewed the SAR and has determined that Cycle 8 is bounded by the results of the analysis presented for Cycle 6 with the exception of the Reactor Coolant Pump Rotor Seizure and the CEA Ejection Accidents which were re-evaluated and re-analyzed respectively and are presented in the SAR.

The Reactor Coolant Pump (RCP) Rotor Seizure was evaluated to assess the effects of minor fuel design changes and increased axial power peaking on the percent of fuel to experience DNB. Pressure characteristics are not affected by these changes and the results are bounded by the Cycle 6 analysis. The results of the evaluation show that the percent of fuel predicted to experience DNB is well below the 10% limiting criteria. Off-site dose rates are a small fraction of 10CFR100 dose guidelines and are bounded by the results of analysis presented in Reference 4.

The CEA Ejection event was re-analyzed using the generically approved methodology (Reference 3) and results show significant margin to the limiting criteria.

Since all the events have been reviewed and proved acceptable it can be stated that for Cycle 8:

i. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased.

   The Cycle 8 reload does not change the overall configuration of the plant. The minor changes in fuel design do not adversely affect the mechanical integrity nor significantly change the coolant flow characteristics through the core. The mode of operation of the plant remains unchanged. Therefore, the probability of occurrence of an accident or malfunction of equipment important to safety is not impacted. The safety analysis performed for the Cycle 8 reload core demonstrates that the consequences of an accident or malfunction have not been increased beyond those evaluated in the previous analyses.

ii. A possibility for an accident or malfunction of a different type than any previously analyzed in the safety analysis is not created.

The Cycle 8 reload does not change the overall configuration of the plant. The minor changes in fuel design do not adversely affect the mechanical integrity nor significantly change the coolant flow characteristics through the core. The mode of operation of the plant remains unchanged. Therefore, a new accident or equipment malfunction has not been created.

iii. The margin of safety as defined in the basis for every Technical Specification is not reduced.

The re-evaluation of the RCP Rotor seizure and the re-analysis of the CEA Ejection accidents have shown that the results are well within the design basis. All other events have been determined to be bounded by previous analyses. Therefore, there is no reduction in the margin of safety relative to the Technical Specification basis for operation of Cycle 8 up to 40% rated power. With the proposed change in the LPD LCO implemented, there is no reduction in the margin of safety relative to the Technical Specification basis for operation of Cycle 8 up to 100% rated power.

Conclusion

As per Federal Regulation 10 CFR50.59 (b), the above Safety Evaluation provides the basis to conclude that the Cycle 8 reload configuration does not involve any changes which introduce an unreviewed safety question. Therefore, implementation of this change is permissible without prior NRC approval.