The Light company
Houston Lighting & Power South Texas Project Electric Generating Station P. O. Box 289 Wadsworth, Texas 77483

May 30, 1995
ST-HL-AE-5015
File No.: G20.01, G21.01
10CFR50.90

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

South Texas Project
Units 1 and 2
Docket Nos. STN 50-498, STN 50-499
Proposed License Amendment Concerning an
Increase in Spent Fuel Pool Heat Loads

Pursuant to 10CFR50.90, Houston Lighting & Power Company proposes to amend its Operating Licenses NPF-76 and NPF-80 by incorporating the attached changes to the Updated Final Safety Analysis Report for the South Texas Project Units 1 and 2. The purpose of this license change is to revise the Spent Fuel Pool heat load licensing basis to provide greater flexibility for normal refueling practices.

Current South Texas Project licensing basis calculations for heatload to the Spent Fuel Pool are based on the assumption that the entire core is offloaded to the Spent Fuel Pool during refueling, and all but typically 65 assemblies are returned to the reactor. This is based on a twelve-month refueling cycle. Under the present 18-month refueling cycle, refueling results in typically 88 fuel assemblies being left in the Spent Fuel Pool. As a result, precautions have been taken to ensure that the heat balance limits are met so that the heat load remains within the licensing basis. To ensure South Texas Project remained within the design basis, fuel offload to the Spent Fuel Pool was not permitted until the decay heat was less than that assumed in the licensing basis.

The licensing basis calculations have been reperformed using the methodology given in Section 9.1.3 of the Standard Review Plan, assuming more fuel assemblies being offloaded as a basis for the heat load. In this case the calculated maximum temperatures may exceed the limits identified in the Safety Evaluation Report.

Pursuant to 10CFR50.59, the calculated increase in the Spent Fuel Pool temperature decreases the margin of safety and therefore requires review as an unreviewed safety question by the Nuclear Regulatory Commission. However, the attached safety evaluation shows that the increase does not constitute a significant hazard.

Houston Lighting & Power has reviewed the attached proposed amendment pursuant to 10CFR50.92 and determined that it does not involve a significant hazards consideration. In addition, Houston Lighting & Power has determined that the proposed amendment satisfies the criteria of 10CFR51.22(c)(9) for categorical exclusion from the requirement for an environmental assessment.

The South Texas Project Nuclear Safety Review Board and the Plant Operations Review Committee have reviewed and approved the proposed changes.

Houston Lighting & Power requests that the effective date of this amendment be 30 days after the date of Nuclear Regulatory Commission approval. Although this request is neither exigent nor an emergency, issuance of this amendment by the Nuclear Regulatory Commission by August 31, 1995, is requested.

In accordance with 10CFR50.91(b), Houston Lighting & Power is providing the State of Texas with a copy of this proposed amendment.

If you should have any questions concerning this matter, please contact Mr. K. J. Taplett at (512) 972-8416 or me at (512) 972-8664.

John Groth

J. F. Groth
Vice President,
Nuclear Generation

PLW/lf

Attachments:
1) Summary and Description of the Proposed Changes
2) No Significant Hazards Consideration Determination
3) Marked-Up Updated Final Safety Analysis Report Pages

c:

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Houston Lighting & Power	)	Docket Nos. 50-498
Company, et al.,	)	50-499
	)	
South Texas Project	)	
Units 1 and 2	)	

AFFIDAVIT

J. F. Groth, being duly sworn, hereby deposes and says that he is Vice President, Nuclear Generation of Houston Lighting & Power Company; that he is duly authorized to sign and file with the Nuclear Regulatory Commission the attached proposed amendment to the South Texas Project Units 1 and 2 concerning an increase in spent fuel pool heat loads; is familiar with the content thereof; and that the matters set forth therein are true and correct to the best of his knowledge and belief.

John Groth

J. F. Groth
Vice President,
Nuclear Generation

STATE OF TEXAS)
)
)

Subscribed and sworn to before me, a Notary Public in and for The State of Texas this 30th day of May, 1995.

Lois J. Mills

Notary Public in and for
The State of Texas

MISC-95\95-018.002

ATTACHMENT 1

SUMMARY AND DESCRIPTION OF THE PROPOSED CHANGES

1.0 SUMMARY

Currently, the South Texas Project spent fuel pool heatup calculations assume the entire core is discharged to the Spent Fuel Pool, and all but typically one-third of the core (65 assemblies) is reinserted in the reactor during each refueling outage. However, this is based on use of a twelve-month refueling cycle. Houston Lighting & Power has performed an analysis based on an eighteen-month refueling cycle under which typically eighty-eight assemblies are left in the Spent Fuel Pool during each refueling outage. Increasing the assumed number of fuel assemblies being discharged results in an increased heat load to the Spent Fuel Pool. Houston Lighting & Power proposes to revise the Updated Final Safety Analysis Report Section 9.1 and Tables 9.1-1 and 9.1-5 to reflect the recalculated Spent Fuel Pool heatup temperatures.

Using methods more severe than prescribed in Section 9.1.3 of the Standard Review Plan, Houston Lighting & Power has determined that the calculated maximum temperature in the Spent Fuel Pool, with the 18-month cycle Vantage 5H fuel, may exceed the limits specified in the Safety Evaluation Report for the South Texas Project. Pursuant to 10CFR50.59, this increase in temperature is a decrease in the margin of safety, and therefore requires review as an unreviewed safety question. However, the following safety evaluation shows that the increase does not constitute a significant hazard.

2.0 BACKGROUND

The current refueling practice at the South Texas Project is as follows:

1. Remove the entire core (193 fuel assemblies) and place it in the Spent Fuel Pool. This is a full-core offload and is defined as the Abnormal Maximum condition in Section 9.1.3 of the Standard Review Plan.

2. Transfer a fraction of the 193 fuel assemblies from the Spent Fuel Pool back to the reactor for use in the next cycle. The fuel assemblies not transferred remain in the Spent Fuel Pool. This is the Normal Maximum condition as defined in Section 9.1.3 of the Standard Review Plan.

Currently, the South Texas Project licensing basis assumes one-third of the core (65 assemblies) is discharged into the Spent Fuel Pool for each normal refueling. This assumption was based on 12-month refueling cycles. In April 1991 and December 1991, respectively, South Texas Project Units 1 and 2 began the first 18-month fuel cycles. For an 18-month fuel cycle, typically 88 fuel assemblies can be discharged to the Spent Fuel Pool for a given fuel cycle. To ensure the South Texas Project remains within the licensing basis for heat load to the Spent Fuel Pool, fuel offload to the Spent Fuel Pool is not permitted until the decay heat is less than that assumed in the licensing basis.

Houston Lighting & Power has revised the Spent Fuel Pool heatup analysis to incorporate the routine refueling practice of full-core offload and 18-month cycles. In the revised Spent Fuel Pool heatup analysis, Houston Lighting & Power has also conservatively accounted for the higher peaking factor and enthalpy rise factor due to the Vantage 5H Fuel Upgrade program. The revision changes the licensing basis of the plant.

3.0 PROPOSED CHANGES

Houston Lighting & Power proposes to modify the Updated Final Safety Analysis Report Section 9.1 and Tables 9.1-1. The proposed change would revise the Spent Fuel Pool heatup temperatures in the Updated Final Safety Analysis Report Table 9.1-1. Particularly, the peak Spent Fuel Pool temperatures for the following cases would exceed the licensing limits given in Safety Evaluation Report, Supplement 6, Appendix BB:

- **Normal Maximum Case:**

Current	145.7°F	(65 assemblies discharged 140 hours after shutdown, one cooling train in operation)
Proposed	155.0°F	(88 assemblies discharged 150 hours after shutdown, one cooling train in operation)

- **Rapid Refueling Case:**

Current	129.2°F	(65 assemblies discharged 80 hours after shutdown, two cooling trains)
Proposed	135.0°F	(88 assemblies discharged 100 hours after shutdown, two cooling trains)

- **Peak Spent Fuel Pool Temperature:**

Rapid Refueling Case:

Current	150.7°F	(65 assemblies discharged 80 hours after shutdown, one cooling train)
Proposed	155.0°F	(193 assemblies discharged 100 hours after shutdown, two cooling trains)
	202.0°F	(193 assemblies discharged 100 hours after shutdown, one cooling train)

Abnormal Maximum Case:

Current	155.4°F	(193 assemblies discharged 120 hours after shutdown, two cooling trains)
Proposed	205.0°F	(193 assemblies discharged 150 hours after shutdown, one cooling train)
	156.0°F	(193 assemblies discharged 150 hours after shutdown, two cooling trains)

Temperatures for the other cases presented in the revised Updated Final Safety Analysis Report Table 9.1-1 would remain below the Standard Review Plan criteria of 140°F for the Normal Maximum cases, and no boiling condition for the Abnormal Maximum cases.

The following Updated Final Safety Analysis Report changes clarify the refueling practices and design of the Spent Fuel Pool and are included in this submittal for completeness:

(a) Updated Final Safety Analysis Report Section 9.1.2.3, Safety Evaluation for Spent Fuel Storage, will be revised to reflect that a complete loss of Spent Fuel Pool water inventory is not a credible event.

(b) Updated Final Safety Analysis Report Section 9.1.3.1.2, describing dewatering protection for the Spent Fuel Pool Cooling and Cleanup System, will be revised to describe that draindown is not possible for credible design basis breaks.

(c) A new Updated Final Safety Analysis Report Section 9.1.3.2.2, "Spent Fuel Pool Cooling During Refueling Operation," will be added to describe the current Spent Fuel Pool cooling method during refueling operation.

(d) Updated Final Safety Analysis Report Section 9.1.3.3.1, describing availability and reliability of the Spent Fuel Pool Cooling and Cleanup System, will be revised to discuss 18-month refueling cycles.

(e) Updated Final Safety Analysis Report Table 9.1-1, "Spent Fuel Pool Cooling and Cleanup System Design Parameters," will be replaced to provide the revised Spent Fuel Pool temperatures and heat loads for various scenarios and the 18 month fuel cycles. The title of the table will also be changed to read "Spent Fuel Pool Heatup Analysis Results."

(f) Updated Final Safety Analysis Report Table 9.1-5, "Spent Fuel Pool Cooling and Cleanup System Failure Modes and Effects Analysis," will be revised to include effects of failure that may result in a complete loss of Spent Fuel Pool cooling.

(g) Editorial changes are included in various other sections.

Updated Final Safety Analysis Report markups of the proposed changes are provided as Attachment 3.

4.0 SAFETY EVALUATION

This safety evaluation addresses the following topics:

1. Spent Fuel Pool time to boil
2. Spent Fuel Pool boil-off rate
3. Normal Maximum case temperature
4. Rapid Refueling case temperature
5. Abnormal Maximum case temperature
6. Peak clad temperature
7. Peak Boraflex (poison) temperature
8. Stainless steel fuel box temperature
9. Spent Fuel Pool liner temperature
10. Spent Fuel Pool concrete temperature
11. Boiling dose consequences
12. Spent Fuel Pool calculated heatup rate comparison with plant data

The safety evaluation includes the effects of the following:

(a) Full-core offload for routine refueling operation,
(b) Fuel reload cycles of 18 months,
(c) Higher peaking factor (Fq) to incorporate the Vantage 5H fuel upgrade effects,
(d) Higher enthalpy rise factor (FΔh) to incorporate the Vantage 5H fuel upgrade effects.

4.1 SPENT FUEL POOL TIME TO BOIL

The impact of the proposed changes on the Spent Fuel Pool boiling conditions have been evaluated. In the event of a postulated complete loss of Spent Fuel Pool cooling capability and Abnormal Maximum case conditions, the time-to-boil is calculated to be 2.8 hours, compared to the 2.86 hours under the current licensing basis. The analysis conservatively assumes that both cooling trains simultaneously fail just when the Spent Fuel Pool reaches its maximum temperature of 156°F. The 0.06 hour difference in time-to-boil is well within the error bounds of the calculation, and is partially due to the added conservatism of 2% higher heat load for power measurement uncertainty. The South Texas Project Emergency Operating Procedures conservatively require 2.5 hours to restore cooling to the Spent Fuel Pool following a loss of cooling incident.

4.2 SPENT FUEL POOL BOIL-OFF RATE

Houston Lighting & Power has determined that the maximum revised Spent Fuel Pool boil-off rate would be 131 gpm for the Abnormal Maximum case, while the current boil-off rate given in the Safety Evaluation Report is 135 gpm. Therefore, there is no adverse impact on the Spent Fuel Pool boil-off rate as a result of the proposed changes. The current means of makeup, described in Updated Final Safety Analysis Report Section 9.1.3.3.2, would not be affected.

4.3 NORMAL MAXIMUM CASE TEMPERATURE

Houston Lighting & Power has evaluated the effects of the proposed changes on the Spent Fuel Pool bulk water temperature for the Normal Maximum Case. The peak bulk water temperature for this case (150 hours after shutdown) would be 155°F. In the current safety analysis, the maximum pool temperature is 145.7°F, and is given in Supplement 6 to the Safety Evaluation Report, Appendix BB, Section 5.1. In the current safety analysis, the Spent Fuel Pool temperature remains above the Standard Review Plan limit of 140°F for 11.5 days. For the proposed changes, the maximum bulk temperature would be greater than the 140°F Standard Review Plan criteria for approximately 710 hours (30 days).

The 30-day period in which the Spent Fuel Pool bulk water temperature would remain above 140°F would not have an adverse affect on the structural integrity of the Spent Fuel Pool (see Sections 4.8 and 4.9.) The design temperatures for all Spent Fuel Pool components (e.g., demineralizers, heat exchangers, etc.) are greater than the proposed Spent Fuel Pool temperatures presented in the revised Updated Final Safety Analysis Report Table 9.1-1. The peak temperature calculated for the Normal Maximum Case would be below the 225°F used in the original structural steel analysis.

4.4 RAPID REFUELING CASE TEMPERATURE

Houston Lighting & Power has evaluated the effects of the proposed changes on the Spent Fuel Pool bulk water temperature for the Rapid Refueling Case (full-core offload 100 hours after shutdown). The peak bulk water temperature for this case, with two Spent Fuel Pool cooling trains in operation, would be 155°F. The current safety analysis assumes one-third of the core is offloaded to the Spent Fuel Pool 80 hours after shutdown, resulting in a maximum pool temperature of 129.2°F.

Since a full-core offload is the normal refueling practice at the South Texas Project, Houston Lighting & Power has also performed an analysis for this case considering a single active failure (e.g., loss of one Spent Fuel Pool cooling train). The peak bulk temperature was calculated to be 202°F. This temperature is lower than the "no boiling" criterion for the full-core offload case specified in Standard Review Plan 9.1.3. This case was not previously considered in the Safety Analysis Report but is included here for completeness. This case is bounded by the Abnormal Maximum case (see Section 4.5).

4.5 ABNORMAL MAXIMUM CASE TEMPERATURE

For the Abnormal Maximum Case with full-core offload, Standard Review Plan Section 9.1.3 requires that the temperature of the pool remain below the boiling point. Also, a single active failure (e.g. loss of one Spent Fuel Pool cooling train) need not be considered for the Abnormal Maximum case. In the current safety analysis, the maximum Spent Fuel Pool temperature for this case is 155.4°F. The peak Spent Fuel Pool bulk water

temperature for the proposed change would be 156°F. This is much lower than the "no boiling" criterion specified for this case in Standard Review Plan 9.1.3. Therefore, the revised Spent Fuel Pool water temperatures would be acceptable, given the conservative assumptions and calculational models used to determine the heat loads and temperatures.

Since a full-core offload is the normal refueling practice at the South Texas Project, Houston Lighting & Power has also performed an analysis for this case considering a single active failure involving loss of one Spent Fuel Pool cooling train. The peak bulk temperature was calculated to be 205°F. This temperature is also lower than the "no boiling" criterion specified in Standard Review Plan 9.1.3. This case was not previously considered in the Safety Analysis Report but is included here for completeness.

In the event of a postulated complete loss of Spent Fuel Pool cooling capability, the time-to-boil was calculated to be 2.8 hours. The time-to-boil analysis has been performed for the Abnormal Maximum case with complete loss of cooling. The analysis conservatively assumes that both cooling trains fail just when the Spent Fuel Pool reaches its maximum temperature of 156°F. The South Texas Project design basis time-to-boil, given in Supplement 6 to the Safety Evaluation Report, Appendix BB, is 2.86 hours.

4.6 PEAK CLAD TEMPERATURE

Houston Lighting & Power's analysis shows that the maximum fuel cladding temperature would be 223°F for the South Texas Project Rapid Refueling case (full core offload, 100 hours decay), which is higher than the current value of 202°F. To conservatively account for the Vantage 5H Fuel effects, the calculation for the peak clad temperature uses a higher peaking factor (Fq) of 2.7 versus 1.2 used previously and an enthalpy rise factor (FΔh) of 1.62 versus 1.0 used previously. At the top of the fuel racks (about 23 ft below the Technical Specification minimum water level), the saturation temperature is approximately 238°F. This gives a subcooling margin of over 15°F for the peak clad temperature. Therefore, no boiling would occur at the location of the peak clad temperature and the fuel cladding integrity would be maintained. Note that for the Abnormal Maximum case, the peak clad temperature was calculated to be 216°F.

4.7 PEAK BORAFLEX (POISON) TEMPERATURE

Houston Lighting & Power's analysis shows that the maximum Boraflex poison temperature due to gamma heating will not exceed the previously calculated temperatures. Input from consultants currently investigating this phenomenon indicates that the short duration of the maximum temperature will not substantially affect the integrity of the Boraflex. Therefore, the change does not have an adverse effect on the integrity of the Boraflex.

4.8 STAINLESS STEEL FUEL BOX TEMPERATURE

Thermal analysis for the stainless steel rack walls shows the temperature drop through the stainless steel is less than 0.5°F. The thermal stresses in the rack walls resulting from this temperature difference would be negligible. Therefore, the stainless steel structural integrity will be maintained.

4.9 SPENT FUEL POOL LINER TEMPERATURE

The Spent Fuel Pool liner plate and gates are designed to be exposed to water containing boric acid solution at a pool temperature of 212°F. A Spent Fuel Pool temperature of 225°F was used for evaluating material properties for determining the allowable stresses. The calculated Spent Fuel Pool temperatures are less than 212°F; therefore, the design conditions have been met.

4.10 SPENT FUEL POOL CONCRETE TEMPERATURE

The concrete structure is designed for a pool temperature of 212°F. A maximum pool water temperature of 156°F has been calculated for the Abnormal Maximum case of full-core offload (the current licensing basis temperature for this case is 155.4°F). Therefore, the existing concrete design conditions have been met.

4.11 BOILING DOSE CONSEQUENCES

The Spent Fuel Pool boiling dose consequences following a complete loss of Spent Fuel Pool cooling for the full-core offload and 18-month reload cycles are described in the Updated Final Safety Analysis Report Section 9.1.3.3.4. This calculation assumes a full core discharge at 120 hours after shutdown and instantaneous boiling in the Spent Fuel Pool. The Spent Fuel Pool heatup analysis for the bounding case was performed for a full core discharge at 100 hours after shutdown. The heatup calculation shows the Spent Fuel Pool reaches 202°F in 120 hours. Therefore, the boiling dose assumption of Spent Fuel Pool boiling in 120 hours is conservative and bounding. The results of the evaluation show that the dose consequences of iodine release due to Spent Fuel Pool boiling are significantly below the allowable dose requirements of 10 CFR 100. Therefore, there will be no significant increase in hazards to the health and safety of the public.

4.12 SPENT FUEL POOL CALCULATED HEATUP RATE COMPARISON WITH PLANT DATA

Spent Fuel Pool heatup rate data was obtained during loss of Spent Fuel Pool cooling while motor-operated valves were being tested. The measured heatup rate was 0.3°F/hr. Based on the same Spent Fuel Pool conditions, the calculated heatup rate was determined using the Standard Review Plan Branch Technical Position ASB 9-2 assumptions and decay heat formulations. The heatup rate was calculated to be 0.7°F/hr. Since the calculated heatup rate over-estimates the measured heatup rate, the results presented in Updated Final Safety Analysis Report Table 9.1-1 are conservative.

5.0 CONCLUSIONS

The safety evaluation includes the effects of full-core offload, 18-month reload cycles, higher peaking factors (Fq), and higher enthalpy rise factors (FΔh) which incorporate the effects of Vantage V5H fuel upgrade. The evaluation shows that the small reduction in margin of safety does not significantly increase the hazards and is not a safety concern because the following conditions are acceptable:

1. The time-to-boil due to loss of all Spent Fuel Pool cooling is 2.8 hours, which is higher than the 2.5 hours used in the South Texas Project's Emergency Operating Procedures.

2. In the event of Spent Fuel Pool boiling, the reactor makeup water pump can provide sufficient makeup water to meet the boil-off rate of 131 gpm. The assured pool makeup source is the Seismic Category I Low Head Safety Injection system. The Low Head Safety Injection pump has the capability to pump 1900 gpm, much in excess of the 131 gpm Spent Fuel Pool boil-off rate. The Low Head Safety Injection system mode of recovery is described in Supplement 6 to the Safety Evaluation Report, Appendix BB.

3. During a full-core offload refueling operation, various means of cooling, described in the Updated Final Safety Analysis Report Section 9.1.3.3.2, are available which ensure that the safety analysis described in Updated Final Safety Analysis Report Section 9.1.3 will remain valid.

4. For the design basis case of Rapid Refueling full-core offload, the fuel clad integrity is not compromised.

5. For the design basis case, the integrity of the Spent Fuel Pool Boraflex is not adversely impacted.

6. Thermal stresses in the Spent Fuel Pool stainless steel rack walls are negligible and the stainless steel structural integrity is maintained.

7. The Spent Fuel Pool liner plate and gates are designed to be exposed to water containing boric acid solution at a pool temperature of 212°F. The calculated Spent Fuel Pool temperatures are less than 212°F; therefore, the design conditions have been met.

8. The Spent Fuel Pool concrete structure is designed to be exposed to water at a pool temperature of 212°F. The calculated Spent Fuel Pool temperatures are less than 212°F; therefore, the design conditions have been met.

9. The calculated Spent Fuel Pool boiling doses are conservative and well below the limits of 10 CFR 100.

10. The Spent Fuel Pool calculation overestimates the heatup rate as compared to plant measured data. Therefore, the calculated results presented in Updated Final Safety Analysis Report Table 9.1-1 are conservative and bounding.

6.0 REFERENCES

1. NUREG-0800, US NRC Standard Review Plan, Section 9.1.3 "Spent Fuel Pool Cooling and Cleanup System," Revision 1, July 1981.

2. NUREG-0781, Safety Evaluation Report Related to the Operation of South Texas Project, Units 1 and 2, including Supplements 3, 6, and 7, Section 9.1 and Appendix BB "Safety Evaluation by the Office of the Nuclear Reactor Regulation Related to the Increase in the Spent Fuel Capacity Through the Use of High Density Storage Racks."

3. NUREG-1346, Technical Specifications, South Texas Project, Unit Nos.1 and 2, Docket Nos. 50-498 and 50-499, Appendix "A" to License Nos. NPF-76 and NPF-80, Section 3/4.9.3 "Decay Time," March 1989.

4. South Texas Project Updated Final Safety Analysis Report, Sections 9.1.2, 9.1.3, & 9.2.2, Revision 3.

5. EPRI TR-103300, "Guidelines for Boraflex Use in Spent-Fuel Storage Racks," December 1993.

6. ST-HL-AE-2417, "Expansion of the Spent Fuel Pool Storage Capacity Using High Density Spent Fuel Racks;" letter from G. E. Vaughn to U. S. Nuclear Regulatory Commission, dated March 8, 1988.

ATTACHMENT 2

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Houston Lighting & Power has determined that the proposed changes do not involve a significant hazards consideration as defined in 10 CFR 50.92:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

   (a) The Spent Fuel Pool conditions are not indicative of accident initiators.
   (b) Design and operability requirements of equipment important to safety are not affected.
   (c) If only one Spent Fuel Pool cooling train is available, boiling would not occur and the Spent Fuel Pool components would remain within their design bases.
   (d) The complete loss of Spent Fuel Pool cooling event has previously been analyzed and described in Supplement 6 to the Safety Evaluation Report, Appendix 3B. The dose consequences for this event have been evaluated and the safety evaluation is described in Updated Final Safety Analysis Report Section 9.1.3.3.4. The results of the evaluation show that the Spent Fuel Pool components would remain within their design bases. Also, the dose consequences of iodine release as a result of Spent Fuel Pool boiling are significantly below the allowable dose limits of 10 CFR 100.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because:

   (a) The operability of safety-related equipment is not impacted.
   (b) The probability of safety-related equipment malfunctioning is not increased.
   (c) The scope of the change does not establish a potential new accident precursor.

(d) The Spent Fuel Pool design considers design basis heat loads for the modified refueling procedure which includes a full-core offload.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION - (Continued)

(e) For the design basis case, the integrity of the Spent Fuel Pool Boraflex is not adversely impacted.

3. The proposed changes do not involve a significant reduction in a margin of safety because:

(a) No fuel damage would occur as a result of the proposed change.
(b) Technical Specification operability and surveillance requirements are not reduced.
(c) The Spent Fuel Pool boiling doses would be significantly below the allowable dose limits of 10 CFR 100.
(d) The modified refueling procedure (full-core offload) continues to have acceptable margins of safety.
(e) For the design basis case, the integrity of the Spent Fuel Pool Boraflex is not adversely impacted.

Based on the safety evaluation presented above for the proposed changes, Houston Lighting & Power has determined that the health and safety of the public will not be jeopardized. Therefore, the proposed changes do not involve a significant hazards consideration.

IMPLEMENTATION PLAN

Houston Lighting & Power requests an implementation time of 30 days from the effective date of the approved license amendment to facilitate distribution and to make appropriate changes to plant documents.

ATTACHMENT 3

Marked-up Updated Final Safety Analysis Report Pages

SPENT FUEL POOL HEATUP FOR FULL-CORE OFFLOAD AND 18-MONTH CYCLES

The following Updated Final Safety Analysis Report pages are provided in support of this amendment. Proposed revisions are indicated as appropriate.

TC 9-1 (Table of Contents)
TC 9-12 (List of Tables)

9.1-1*
9.1-2*
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*Pages with no changes shown are provided to support review of proposed License Amendment.

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CHAPTER 9

AUXILIARY SYSTEMS

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CHAPTER 9

9.1 FUEL STORAGE AND HANDLING

Facilities for the receipt and storage of new fuel and the storage and transfer of spent fuel are housed in the Fuel Handling Building (FHB). A separate and independent FHB is provided for each unit of the South Texas Project Electric Generating Station (STPEGS). Each FHB is designed as a controlled-leakage seismic Category I structure. The design of the FHB Heating, Ventilating and Air-Conditioning (HVAC) System is discussed in Section 9.4.2. The structural design considerations are described in Section 3.8.4.

9.1.1 New Fuel Storage

9.1.1.1 Design Bases. The new fuel storage pit is a reinforced concrete pit and an integral part of each seismic Category I FHB. This pit provides temporary dry storage for approximately one-third of a core (66 fuel assemblies) of new fuel. The fuel is stored in racks (Figure 9.1.1-1) composed of individual vertical cells fastened together to form three 2 x 11 modules which may be bolted to anchors in the floor and walls of the new fuel storage pit. The new fuel racks are classified as seismic Category I components, as defined by Regulatory Guide (RG) 1.29, and American Nuclear Society (ANS) safety class (SC) 3 (Section 3.2).

The new fuel racks are designed with a center-to-center spacing of 21 inches. This spacing provides a minimum of 12 in. between adjacent fuel assemblies. This separation is sufficient to maintain a subcritical array assuming optimum moderation. Space between storage positions is blocked to prevent insertion of fuel. All rack surfaces that come into contact with the fuel assemblies are made of annealed authentic stainless steel, and the support structure is painted carbon steel.

The racks are designed to withstand normal operating loads, as well as to remain functional with the occurrence of a Safe Shutdown Earthquake (SSE). The new fuel racks are designed to withstand a maximum uplift force of 5,000 pounds and to meet the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Appendix XVII.

The new fuel storage pit access hatch is a three-section cover. This cover will minimize the introduction of dust and debris into the pit. The cover is designed to withstand the impact force of a new fuel assembly dropped from the maximum elevation allowed by the 2-ton hoist of the FHB overhead crane.

In addition, space is provided for the storage of fuel during refueling inside the Reactor Containment Building (RCB). See Section 9.1.2.1 for a description of the racks.

9.1.1.2 Facilities Description. The FHB abuts the south side of the RCB and is adjacent to the west side of the Mechanical-Electrical Auxiliaries Building (MEAB) of each unit. The locations of the two FHBs are shown in the station plot plan on Figure 1.2-3. For general arrangement of the new fuel storage facilities, refer to Figures 1.2-39 through 1.2-48.

New fuel assemblies are received in the receiving area of each FHB and temporarily stored in the shipping containers in the new fuel handling area.

In the new fuel handling area, each new fuel assembly is removed from its shipping container and inspected visually to confirm the assembly has not been damaged during shipment. The new fuel assemblies are transported from the inspection area to the new fuel storage pit or to the new fuel elevator by the 15/2-ton, dual-service FHB crane. The 2-ton hoist of this crane is designed to handle new fuel assemblies. New fuel handling is discussed in detail in Section 9.1.4. Use of the 2-ton hoist of the 15/2-ton crane or of the fuel-handling machine to handle new fuel ensures that the design uplift of the racks will not be exceeded.

The new fuel storage pit is situated in the approximate center of each FHB. The floor of the new fuel storage pit is at El. 50 ft-3 inches. The new fuel storage pit access hatch is provided with a three-section protective cover at El. 68 ft. The fuel assemblies are loaded into the new fuel storage racks through the top and stored vertically.

9.1.1.3 Safety Evaluation. Units 1 and 2 of the STPEGS are each provided with separate and independent fuel handling facilities.

Flood protection of each FHB is discussed in Section 3.4.1. Flooding of the new fuel storage pit from fluid sources inside either FHB is not considered credible since all fluid systems components are located well below the elevation of the new fuel storage pit access hatch. A floor drain is provided in the new fuel storage pit to minimize collection of water.

The applicable design codes and the ability of the FHB to withstand various external loads and forces are discussed in Section 3.8.4. Details of the seismic design and testing are presented in Section 3.7. Missile protection of the FHBs is discussed in Section 3.5. Failure of nonseismic systems or structures will not decrease the degree of subcriticality provided in the new fuel storage pit.

In accordance with American National Standards Institute (ANSI) N18.2, the design of the normally dry new fuel storage racks is such that the effective multiplication factor will not exceed 0.98 with fuel of the highest anticipated enrichment in place, assuming optimum moderation (under dry or fogged conditions). For the unborated flooded condition, assuming new fuel of the highest anticipated enrichment in place, the effective multiplication factor does not exceed 0.95. Credit may be taken for the inherent neutron-absorbing effect of the materials of construction.

The new fuel assemblies are stored dry, the 21-in. spacing ensuring a safe geometric array. Under these conditions, a criticality accident during refueling and storage is not considered credible. Consideration of criticality safety analysis is discussed in Section 4.3.

Design of the facility in accordance with RG 1.13 ensures adequate safety under both normal and postulated accident conditions. The new fuel storage racks also meet the requirements of General Design Criterion (GDC) 62.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases. The spent fuel pool (SFP) is a stainless steel-lined reinforced concrete pool and is an integral part of each FHB. All spent

fuel racks are classified as seismic Category I, as defined by RG 1.29, and as ANS SC 3. 3

The spent fuel storage facility provides storage capacity for 1,969 high density poison spent fuel racks in a honeycomb array in each unit. Two storage regions are provided in the SFP. There are 288 storage cells in 3 Region 1 type racks and 1,681 storage cells in Region 2 type racks. Figure 9.1.2-2 shows the pool layout for both Units 1 and 2. The six Region 1 rack modules are located in the northwest corner of the spent fuel pool.

The Region 1 racks have 10.95-in. nominal center-to-center spacing with removable poison assemblies between the cells. This region is conservatively designed to accommodate unirradiated fuel at enrichments to 4.0 weight 3 percent. Region 1 storage cells are each bounded on four sides by a water box except on the periphery of the pool. The neutron poison material (Boraflex^R) is located in these water boxes. This is accomplished by capturing two sheets of the poison material on two outside opposite faces of a thin-walled rectangular box. The poison sheets are captured under thin stainless steel sheets which are intermittently welded all around to the thin-walled rectangular box. This configuration allows any irradiation gas formation in the poison to escape. A locking device engages the structure under the lead-in guide to hold the assembly in place. Special tools are provided for unlocking, removing, reinstalling, and locking this poison assembly. The axial location of the poison with respect to the active fuel region is provided and maintained by this welded assembly structure. Figure 9.1.2-3 shows a typical Region 1 spent fuel rack.

The reactivity characteristics of fuel assemblies which are to be placed in the spent fuel storage racks are determined and the assemblies are categorized by reactivity. Alternately, if necessary, all assemblies may be treated as if each assembly is of the highest reactivity class until the actual assembly reactivity classification is determined. Section 5.6 of the Technical Specifications provides the definitions of the reactivity classifications and the allowed storage patterns. Fuel assemblies are loaded into the racks in a geometrically safe configuration to ensure rack subcriticality.

3

Fuel assembly reactivity requirements for close packed storage and checkerboard storage are specified in the Technical Specifications. The boron concentration of the water in the spent fuel pool is maintained at or above the minimum value needed to ensure that the rack K_{eff} is less than or equal to 0.95 in the event of misplaced assemblies in the close packed storage areas or in checkerboard storage areas. Consideration of criticality safety is discussed in Section 4.3.

The Region 2 racks have a 9.15-in. nominal center-to-center spacing with fixed poison material surrounding each cell. A sheet of neutron poison material is 3 captured between the side walls of all adjacent boxes. To provide space for the poison sheet between boxes, a double row of matching flat round raised areas are coined into the side walls of all boxes. The raised dimension of these locally formed areas on each box wall is half the thickness of the poison sheet. The boxes are fusion welded together at all these local areas. The poison sheets are scalloped along their edges to clear these areas. Figure 9.1.2-4 shows a typical Region 2 spent fuel rack.

The axial location of the poison with respect to the active fuel region is provided and maintained by the structure of each box. At the outside periphery of each rack, a sheet of poison material is captured under thin stainless sheets which are intermittently welded all around to the box.

All rack surfaces that come into contact with fuel assemblies are made of annealed austenitic stainless steel. These materials are resistant to corrosion during normal and emergency water quality conditions. The racks are designed to withstand normal operating loads as well as to remain functional with the occurrence of an SSE. The racks are designed with adequate energy absorption capabilities to withstand the impact of a dropped spent fuel assembly from the maximum lift height of the spent fuel pit bridge hoist. The racks are designed to withstand a maximum uplift force equal to the uplift force of the bridge hoist. The 14-in. and 16-in. racks also meet the requirements of ASME Code, Section III, Appendix XVII. The high-density spent fuel racks meet the criteria of Appendix D to Standard Review Plan (SRP) 3.8.4.

Shielding for the SFP is adequate to protect plant personnel from exposure to radiation in excess of published guideline values as stated in Section 12.1. A depth of approximately 10 ft of water over the top of the spent fuel assemblies will limit direct radiation to 2.5 mR/hr (surface dose rate).

The FHB Ventilation Exhaust System is designed to limit the offsite dose in the event of a significant release of radioactivity from the fuel, as discussed in Sections 12.3.3, 15.7.4, and 9.4.2.

The FHB is designed to prevent missiles from contacting the fuel. A more detailed discussion on missile protection is given in Section 3.5.

In addition, space is provided for storage of fuel during refueling inside the RCB for 64 fuel assemblies in four 4 x 4 modules having 16-in. center-to-center spacing (Figure 9.1.2-1A). These modules are firmly bolted in the floor.

9.1.2.2 <u>Facilities Description</u>. The FHB abuts the south side of the RCB and is adjacent to the west side of the MEAB of each unit. The locations of the two FHBs are shown in the station plot plan on Figure 1.2-3. For general arrangement of the spent fuel storage facilities, refer to Figures 1.2-39 through 1.2-48.

The spent fuel storage facilities are designed for the underwater storage of spent fuel assemblies and control rods after their removal from the reactor vessel. The spent fuel is transferred to the FHB and handled and stored in the spent fuel pool underwater. The fuel is stored to permit some decay, then transferred offsite. For a detailed discussion of spent fuel handling, see Section 9.1.4.

The SFP is located in the northwest quadrant of each FHB. The floor of the pool is at El. 21 ft-11 in., with normal water level at El. 66 ft-6 inches. The top of a fuel assembly in a storage rack does not extend above the top of the storage rack which is El. 39 ft-10 in. maximum. The fuel assemblies are loaded into the spent fuel racks through the top and are stored vertically.

9.1.2.3 Safety Evaluation. Units 1 and 2 of the STP are each provided with separate and independent fuel handling facilities. Flood protection of each FHB is discussed in Section 3.4.1. A detailed discussion of missile protection is provided in Section 3.5.

The applicable design codes and the various external loads and forces considered in the design of the FHB are discussed in Section 3.8.4. Details of the seismic design and testing are presented in Section 3.7.

Design of this storage facility in accordance with GDC 62 and RG 1.13 ensures a safe condition under normal and postulated accident conditions. The K_{eff} of the spent fuel storage racks is maintained less than or equal to 0.95, even if unborated water is used to fill the spent fuel storage pool, by both the designs of the fuel assemblies and the storage rack and the use of administrative procedures to control the placement of burned and fresh fuel.

Under accident conditions, the K_{eff} is maintained well below 0.95 assuming 700 ppm borated water. The boron concentration of the water in the spent fuel pool is maintained at or above the minimum value needed to ensure that the rack K_{eff} is less than or equal to 0.95 in the event of misplaced assemblies in the close packed storage areas or in checkerboard storage areas. Consideration of criticality safety is discussed in Section 4.3.

The SFP is designed to maintain leaktight integrity. To ensure such integrity, the pool is lined with stainless steel plate, and plate welds are backed with channels to detect and locate leakage. Leakage entering these channels is directed to the Liquid Waste Processing System (LWPS) via the FHB sump. Should a leak be detected, either by a low-level alarm (setpoint: 6 in. below normal water level) or by the fuel pool liner channel leak detection method, the operator would initiate makeup to the spent fuel pool. Makeup capability is provided by permanently installed connections to: (1) the Demineralized Water System (DWS), (2) the Reactor Makeup Water System (RMWS), and (3) the refueling water storage tank (RWST) in the Emergency Core Cooling System (ECCS).

~~A complete loss of SFP cooling is not considered a credible event since the components involved are designed to SC 3 seismic Category I requirements and could be powered from redundant Engineered Safety Features (ESF) power supplies. Further, the systems providing cooling are redundant. Therefore, no single failure would result in a complete loss of fuel pool cooling. For a more detailed discussion of SFP cooling, refer to Section 9.1.3~~
[Replace with Insert #1]

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The Spent Fuel Pool Cooling and Cleanup System (SFPCCS) is designed to remove the decay heat generated by spent fuel assemblies stored in the SFP and/or the in-Containment storage area. A second function of the system is to maintain visual clarity and purity of the spent fuel cooling water and the refueling water.

9.1.3.1 Design Bases. The SFPCCS design heat loads are given in Table 9.1-1. System capabilities to withstand natural phenomena and piping rupture are addressed in Chapter 3. The spent fuel pool cooling portions of the SFPCCS are designed to seismic Category I requirements, and are located in the FHB, a seismic Category I building. The spent fuel pool water purification

INSERT #1 - UFSAR Section 9.1.2.3 (page 9.1-5)

A complete loss of SFP water inventory is not considered a credible event since the components involved are designed to Safety Class 3, Seismic Category I requirements. System piping is arranged so that a loss of piping integrity, for all credible accidents, does not result in draining of the SFP below a minimum depth of 23 feet above the top of the fuel. Details of the system design are provided in Section 9.1.3.2. Loss of SFP heat removal is described in Section 9.1.3.3.

portions of the SFPCCS are not required for safety functions and are not designed to seismic Category I requirements.

9.1.3.1.1 Spent Fuel Cooling: The SFPCCS is designed to remove the amount of decay heat produced by the number of spent fuel assemblies that are stored following refueling. The system design incorporates two trains of equipment. Each train is capable of removing 100 percent of the normal maximum design heat load and 50 percent of the abnormal maximum design heat load. The system can maintain the spent fuel cooling water temperature at or below the maximum allowable temperatures specified by Table 9.1-1. [Add Insert #2] This temperature is based on the heat exchangers (HXs) being supplied with component cooling water (CCW) at the design flow and temperature. The flow through the spent fuel storage areas provides sufficient mixing to maintain uniform water conditions.

~~If it is necessary to remove a complete core from the reactor~~For a full core offload, the system can maintain the spent fuel cooling water below the maximum allowable temperature specified by Table 9.1-1. Makeup water requirements will be provided by either reactor makeup water, demineralized water, or refueling water. The makeup flowpath from the reactor makeup water storage tank (RMWST) is seismic Category I. The flowpaths from the demineralized water storage tank (DWST) and from the RWST are nonseismic Category I.

9.1.3.1.2 Dewatering Protection: A depth of approximately 10 ft of water over the top of the stored spent fuel assemblies will limit direct radiation to 2.5 mR/hr. System piping is arranged so that failure of any pipeline cannot drain the spent fuel pool or the in-Containment temporary storage area below a depth of approximately 23 ft of water over the top of the stored spent fuel assemblies. [Add Insert #3] Additionally, means are provided to detect component or system leakage. Refer to Section 9.3.3 for the detailed description of leak detection via the floor drains. In addition, the water level instrumentation provides a means of leakage detection.

9.1.3.1.3 Water Purification: The system's demineralizers and filters are designed to provide adequate purification to permit unrestricted access for plant personnel to spent fuel storage areas and to maintain optical clarity of the spent fuel cooling water and the refueling water. The optical clarity of the spent fuel pool surface is maintained by use of the system's skimmer pump, skimmer/strainer assemblies, and skimmer filter. The optical clarity of the refueling cavity water is maintained by the reactor cavity filtration system. The Nuclear Steam Supply System (NSSS) vendor-recommended specifications and guidelines for the spent fuel pool water purity are provided in Table 9.1-4 and the monitoring frequency is provided in Table 9.3-3.

9.1.3.2 System Description. The SFPCCS, shown on Figures 9.1.3-1 and 9.1.3-2 (piping and instrumentation diagrams [P&IDs]), consists of two cooling trains with a common header, two purification trains, a surface skimmer loop and a reactor cavity filtration system.

The SFPCCS removes decay heat produced by spent fuel after it is removed from the reactor. Spent fuel is removed from the reactor core during the refueling sequence and placed in the SFP, where it is stored until it is shipped offsite for reprocessing or permanent storage. If, for some reason, it is desirable

INSERT #2 - UFSAR Section 9.1.3.1.1 (page 9.1-6)

Table 9.1-1 incorporates various SFP loading scenarios for 18-month fuel reload cycles.

INSERT #3 - UFSAR Section 9.1.3.1.2 (page 9.1-6)

The return line contains an anti-siphon hole just below the low water level to prevent gravity drainage due to an open drain valve or due to all design basis credible breaks (non-seismic one-inch piping).

or necessary to delay the transfer of the spent fuel to the SFP, the in-Containment storage area can be used for temporary storage of up to one-third of a core. The system normally handles the heat load from one core region freshly discharged from the reactor. Heat is transferred from the SFPCCS through the HXs to the Component Cooling Water System (CCWS).

When the SFPCCS is in operation, water drawn from the SFP (and/or from the in-Containment storage area) by the SFP pumps is pumped through the tube side of the HXs, and then is returned to the spent fuel pool (and/or the in-Containment storage area). Each suction connection, which is provided with a strainer, is located at an elevation 4 ft below the normal water level (approximately 23 ft above the top of the fuel assemblies). The return line contains an antisiphon hole near the surface of the water to prevent gravity drainage.

To maintain spent fuel cooling water purity, a bypass circuit composed of a demineralizer and a filter is connected to each cooling train. The demineralizers are charged with either a mixed resin (cation and anion resin) or cation resin only, dependant on the type of contamination indicated by the required chemical analyses. While the heat removal operation is in process, a portion of the spent fuel cooling water is diverted upstream of each HX and passed through the purification circuit, returning downstream of the HXs. The demineralizers remove ionic corrosion impurities and fission products. Filters are provided to remove any additional particulates and to prevent any resin fines from entering the system from the demineralizer discharge. Transfer canal water may be circulated through the same purification circuits by removing the gate between the canal and the spent fuel pool. These purification loops are sufficient for removing fission products and other contaminants which may be introduced into the spent fuel cooling water.

One purification loop may be isolated from the heat removal portion of the SFPCCS. By so doing, the isolated equipment may be used in conjunction with either the reactor coolant drain tank pumps or the refueling water purification pump to clean and purify the refueling water while spent fuel cooling and spent fuel cooling water cleanup operations proceed. Connections are provided such that the refueling water may be pumped from either the RWST or the refueling cavity through the demineralizer and filter, and discharged to either the refueling cavity or the RWST. Samples are periodically taken to determine the need for purification of the water as well as the purification efficiency.

To further assist in maintaining spent fuel cooling water clarity, the spent fuel pool is cleaned by a skimmer loop. Water is removed from the surfaces via two skimmer/strainer assemblies located in the SFP. Water is pumped through a filter by a skimmer pump and returned to the pool surface at a single location remote from the skimmer/strainer assemblies. Piping for future addition of a third skimmer/strainer for cleaning the surface of the fuel transfer canal water is also provided.

The SFP is initially filled with water having the same boron concentration as that in the RWST. Borated water may be supplied from the RWST via the SFPCCS return header, or by running a temporary line from the boric acid blending tee, located in the Chemical and Volume Control System (CVCS), directly into the pool. Demineralized water can also be added for makeup purposes (i.e., to replace evaporative losses) through a connection in the SFPCCS return header.

The water in the spent fuel pool may be separated from the water in the transfer canal by a gate. The gate is installed so that the transfer canal may be drained to allow maintenance of the fuel transfer equipment. The water in the transfer canal is first pumped, via a portable pump, into the spent fuel pool and then is transferred to the recycle holdup tanks in the Boron Recycle System (BRS) by a SFP pump. When maintenance on the fuel transfer equipment is completed, the water is returned directly to the SFPCCS by the recycle evaporator feed pumps (BRS). A portable pump is again used to return water to the transfer canal.

When spent fuel assemblies are stored in the in-Containment storage area, either of the cooling trains may be utilized to remove the decay heat. The in-Containment storage area is sized to temporarily store one-third of a core.

The in-Containment storage area is directly connected to the refueling cavity and is filled with refueling water whenever the refueling cavity is filled. Thus, during refueling, the in-Containment storage area is always ready for use. During refueling outages, the clarity of the water in the reactor cavity is maintained by the Reactor Cavity Filtration System. Water is removed from the reactor cavity pool through a submerged strainer located one foot above the reactor cavity floor, pumped through four cartridge-type filters, and returned to the reactor cavity pool.

9.1.3.2.1 Component Description: The design codes and classifications of the components are given in Section 3.2. Equipment design parameters are given in Table 9.1-2.

Spent Fuel Pool Pumps

The pumps are horizontal, centrifugal units, with all wetted surfaces being stainless steel. The pumps draw water from the spent fuel pool (and/or the in-Containment storage area) and deliver it to the HXs for cooling and to the purification trains for cleanup.

Spent Fuel Pool Skimmer Pump

This horizontal, centrifugal pump takes suction from the SFP via adjustable surface skimmer/strainer assemblies and from the fuel transfer canal and circulates the water through a filter and returns it to the SFP and the fuel transfer canal. All wetted surfaces of the pump are austenitic stainless steel.

Refueling Water Purification Pump

This centrifugal pump is used to circulate water from the RWST through a SFP demineralizer and filter.

Spent Fuel Pool Heat Exchangers

The HXs are the shell and U-tube type. Spent fuel cooling water circulates through the tubes while CCW circulates through the shell. Each HX is sized for 50 percent of the heat load. ~~The design heat load of the SFP HXs is based on the decay heat generated by one-third of a core placed in the SFP shortly after reactor shutdown during a refueling operation with one-third of a core from the previous refueling already in the pool.~~ [Replace with Insert #4.]

INSERT #4 - UFSAR Section 9.1.3.2.1 (page 9.1-8)

The heat exchangers are designed to accommodate the Abnormal Maximum heat loads and temperatures given in Table 9.1-1.

Spent Fuel Pool Demineralizers

The two flushable demineralizers are designed to provide adequate spent fuel cooling water purity for unrestricted access of plant personnel to the spent fuel storage areas.

Spent Fuel Pool Filters

A filter is located in each purification train, downstream of the demineralizer, to collect possible particulates and resin fines passed by the demineralizer. The filter assembly utilizes a disposable cartridge filter and is readily accessible for filter change.

Spent Fuel Pool Skimmer Filter

The SFP skimmer filter is used to remove particles swept from the spent fuel pool surface which are not removed by the skimmer/strainer assembly. The filter assembly utilizes a disposable cartridge filter and is readily accessible for filter change.

Spent Fuel Pool Strainers

A strainer is located in each SFP pump suction line from the SFP to prevent introduction of relatively large particles that might clog the spent fuel demineralizers or damage the SFP pumps.

Spent Fuel Pool Skimmer/Strainer Assemblies

Two assemblies are provided. These assemblies make it possible to take suction from the pool surface and remove debris from the skimmer process flow.

Fuel Transfer Canal Skimmer/Strainer Assembly (Future Expansion)

Piping is provided for future addition of one assembly which would take suction from the transfer canal surface. Debris would be removed via the skimmer filter.

In-Containment Storage Area Strainer

A strainer is located in the SFP pump suction line from the in-Containment storage area to prevent introduction of relatively large particles that might clog the SFP demineralizers or damage the SFP pumps.

Reactor Cavity Filtration System

The Reactor Cavity Filtration System is a skid-mounted package system including a horizontal, centrifugal pump with an electric motor driver, four filter housings with cartridge-type filters, suction screen, and the necessary valves, instrumentation, and piping.

[Add Insert #5.]

INSERT #5 - Add New UFSAR Section 9.1.3.2.2 After Section 9.1.3.2.1 (page 9.1-9)

9.1.3.2.2 Spent Fuel Pool Cooling During Refueling Operation

The normal method for refueling at STP is to perform a full-core offload. Under the guidance provided in Standard Review Plan Section 9.1.3, this is the abnormal maximum condition. According to SRP Section 9.1.3, during a full-core offload, the abnormal maximum criteria applies and a single active failure need not be considered. For this condition, the SRP requires the temperature of the SFP water to be kept below boiling and level maintained with the available normal systems in operation. Table 9.1-1 shows that this criteria is met for the STP Abnormal Maximum case.

Table 9.1-1 gives the SFP temperature limits for various fuel load and SFP cooling configurations. Cycle specific calculations will be performed, if necessary, to ensure that these SFP temperature limits are met. Cycle specific calculations are not required if the typical STP fuel load conditions, specified in Table 9.1-1, represent the intended refueling operation.

During full-core offload conditions at STP, two SFP cooling trains are administratively required to be available. At least one SFP cooling train will be available at all times backed by an on-site power source. The second cooling train will at least be functional, backed by either an on-site power source or a power source available from the switchyard.

9.1.3.3 Safety Evaluation.

9.1.3.3.1 Availability and Reliability: The SFPCCS has no emergency function during an accident except to provide adequate cooling to the SFP. Since it is not necessary for automatic initiation post-accident, the SFP pumps are manually placed on the emergency power bus after completion of automatic load sequencing. In the event of failure of a SFP pump or loss of cooling to a SFP HX, the second cooling train would provide continued cooling of the stored spent fuel with the spent fuel cooling water at a higher equilibrium temperature (Table 9.1-1). [Add Insert #6.] A failure modes and effects analysis for the SFPCCS is given in Table 9.1-5.

9.1.3.3.2 Spent Fuel Storage Area Dewatering: The most serious failure of this system would be complete loss of water in one of the storage areas. To protect against that possibility, the SFP pump suction connections enter near the normal water level so that the storage areas cannot be siphoned. ~~The cooling water return line to each storage area contains an antisiphon hole to prevent the possibility of siphoning.~~[Add Insert #7.] These design features assure that neither the SFP nor the in-Containment storage area can be drained more than 4 ft below the normal water level (normal water level is approximately 27 ft above the top of the stored spent fuel).

If a seismic event results in the failure of a non-seismic pipe in the reactor makeup water tank compartment, both reactor makeup pumps may be lost because of flooding in this compartment. As a result, makeup for the SFP from the RMWS would be lost. However, the following means may still be available to provide makeup to the SFP.

1. Water from the RWST (seismic Category I) through the nonseismic Category I refueling water purification pump 1A and SFP demineralizer 1A to the SFP cooling return line.

2. Demineralized water through a 2-in. line connecting the demineralized water system to the SFP cooling return line.

3. Fire water from either one or both of the hose reel and hose cabinet located near the SFP.

In the unlikely event that all of these backup sources were not available, a seismically qualified makeup water source would be provided by connecting temporary hoses to the vent and drain valves located on the low head safety injection (LHSI) pump discharge piping. The LHSI pumps are located in the FHB at the lowest level. The hoses will be routed through building stairways and equipment hatches to the SFP. ~~Assuming a complete loss of SFP cooling and a heat load as described in 9.1.3.3.4 below, the heatup time to boiling is approximately 8 hours based upon a water loss equivalent to 4 ft of pool level.~~[Replace with Insert #8.]This allows sufficient time to route the hoses through the building.

9.1.3.3.3 Water Quality: Whenever a fuel assembly with defective cladding is removed from the reactor core, a small quantity of fission products may enter the spent fuel cooling water. The purification loops provide a means of removing fission products and other contaminants from the water. By maintaining radioactivity concentrations in the spent fuel cooling water at 5×10^{-3} μCi/cc (β and γ) or less, the dose at the water surface is 2.5 mR/hr or less.

INSERT #6 - UFSAR Section 9.1.3.3.1 (page 9.1-10)

Table 9.1-1 shows various SFP loading scenarios for 18-month fuel reload cycles. Table 9.1-1 incorporates the routine refueling practice of full-core offload, 18-month cycles.

INSERT #7 - UFSAR Section 9.1.3.3.2 (page 9.1-10)

The cooling water return line to each storage area contains an anti-siphon hole to prevent the possibility of siphoning for all design basis credible breaks (non-seismic one-inch piping).

INSERT #8 - UFSAR Section 9.1.3.3.2 (page 9.1-10)

For a complete loss of SFP cooling, the maximum heat load occurs for the STP Abnormal Maximum case listed in Table 9.1-1. This analysis assumes a SFP water level 4 feet below the normal water level, and that both SFP cooling trains simultaneously fail just when the SFP reaches its maximum temperature. This worst case scenario time-to-boil is calculated to be 2.8 hours.

9.1.3.3.4 Spent Fuel Pool Boiling Dose Analysis: In the event of a fire or moderate energy line crack in the FHB that disables both trains of SFP cooling, the SFP temperature would begin to rise and, assuming no corrective action, would eventually boil. The following analysis examines the dose consequences of a loss of the SFPCCS and the use of the seismic Category 1 makeup water source (RMWS).

It is assumed that a loss of the SFPCCS occurs after a refueling where a full core has been removed and placed into the SFP 120 hours after shutdown. The heat loads supplied to the pool are comprised of the following sources: 1) the full core removed prior to the event; 2) 92 assemblies which have decayed 36 days after shutdown; and 3) spent fuel from the previous 20 refueling offloads. The last full core offload fills the SFP to the maximum capacity of 1969 assemblies. For the purpose of this calculation, the pool is conservatively assumed to boil instantaneously after the loss of the SFPCCS. This 3
loss of SFPCCS is assumed to occur at 120 hours after shutdown. Throughout the event, the leakage rate for iodine is assumed to be the normal full power rate (1.3×10^{-8} sec^{-1}). The iodine available for release is based upon the gap activity containing 10 percent of the rod inventory and the leakage occurs from the defective 1 percent of the rods. The activity of the refueling water prior to initiation of the event is assumed to be negligible.

Using these assumptions and those found in Table 9.1-6, the thyroid dose consequences of releasing the iodine as a result of SFP boiling are well below the dose requirements of 10CFR, Part 100.

9.1.3.4 Instrumentation Application. The instrumentation provided for the SFPCCS is discussed below. Alarms and indications are provided as noted.

9.1.3.4.1 Temperature: Instrumentation is provided to measure the temperature of the water in the SFP and in the in-Containment storage area and to give local indication as well as annunciation at the main control board when normal temperatures are exceeded.

Instrumentation is also provided to give local indication of the temperature of the spent fuel cooling water as it leaves each HX.

9.1.3.4.2 Pressure: Instrumentation is provided to measure and give local indication of the pressures in the suction line of the SFP skimmer pump and in the suction and discharge lines of the refueling water purification pump and of each SFP pump. Instrumentation is also provided at locations upstream and downstream of each SFP filter, each SFP skimmer filter, and each of the SFP demineralizer so that the pressure differential across these filters can be determined.

9.1.3.4.3 Flow: Instrumentation is provided to measure and give local indication of the flow in the outlet line of each SFP filter. Instrumentation is also provided to measure discharge flow from the refueling water purification pump and to provide a low-flow alarm on the main control board in addition to providing local flow indication.

9.1.3.4.4 Level: Instrumentation is provided to give an alarm in the control room when the water level in the SFP or in the in-Containment storage area reaches either the high or low-level setpoints (6 in. above or below normal water level).

9.1.3.5 Tests and Inspections. Active components of the SFPCCS are in either continuous or intermittent use during normal system operation. The SFPCCS is included in the inservice inspection requirements described in Section 6.6 and the inservice testing requirements described in Section 3.9.6.

9.1.4 Fuel Handling System

9.1.4.1 Design Bases. The FHS consists of equipment and structures utilized in the transporting and handling of the fuel from the time it reaches the station until it leaves the station.

The following design bases apply to the FHS:

1. Fuel-handling devices have provisions to avoid dropping or jamming of fuel assemblies during transfer operation.

2. Fuel lifting and handling equipment will not fail in such a manner as to damage seismic Category I equipment in the event of an SSE.

3. The Fuel Transfer System (FTS), where it penetrates the Containment, has provisions to preserve the integrity of the Containment pressure boundary, including a means to test for leak tightness.

4. Each machine used to lift spent fuel has a limited maximum lift height so that the minimum required depth of water shielding is maintained.

5. The cask handling components and the FHB layout limit vertical lift above the floor of the cask to less than 30 ft above the floor during any moving sequence.

6. The Spent Fuel Cask Handling System (SFCHS) utilizes a wet handling technique.

7. The pool gates are designed to maintain their integrity in the event of an SSE.

9.1.4.2 System Description. The equipment in the FHS is comprised of lifting equipment, handling equipment, an FTS, and the SFCHS. The structures associated with the FHS are the refueling cavity, refueling canal, and in-Containment fuel storage area inside the RCB; and the fuel transfer canal, SFP, new fuel storage pit and inspection area, and cask loading pool and decontamination platform in the FHB. The equipment is located in seismic Category I buildings.

9.1.4.2.1 New Fuel Handling: The new fuel arrives onsite either by rail or truck. The rail track and truck receiving areas are located on the ground floor of the FHB. When new fuel is delivered to the receiving area within the FHB, the shipping containers are unloaded from the transport vehicle and examined for shipment damage. The shipping containers are then lifted to the operating floor by the FHB overhead crane. The shipping containers can be placed directly in the new fuel inspection laydown area on the operating floor or lowered through the equipment hatch in the operating floor to the new fuel handling area below by means of the FHB overhead crane. The shipping containers may be unloaded in either area. The shipping containers are placed horizontally on the floor. The containers are then stacked

using the overhead crane servicing this area. One by one, these shipping containers are unstacked, their covers removed, and the pivotal, fuel support structure within the shipping container is elevated from the horizontal to the vertical position. In the new fuel handling area, this is accomplished using the new fuel handling area overhead crane. In the new fuel inspection laydown area on the operating floor, this is accomplished using the FHB overhead crane. The various clamping devices securing the fuel assembly to the support structure are then removed. The fuel assembly is lifted from the shipping container support structure. Inspection activities may now be conducted in either areas. Alternatively, inspection activities may be conducted at a later time following transfer of the fuel to the new fuel storage pit.

Following inspection, unacceptable new fuel assemblies are set aside for dispositioning. Acceptable new fuel assemblies are engaged by the new fuel handling tool, which is in turn attached to the hook of the FHB overhead crane. If the fuel was unloaded in the new fuel handling area, the new fuel assembly must be secured beneath the equipment hatch, released from the new fuel handling area overhead crane, and then engaged by the FHB overhead crane and lifted through the equipment hatch to the operating floor. Fuel assemblies are either inserted into the new fuel storage racks in the new fuel storage pit, placed in the new fuel elevator which is located in the fuel transfer canal, or placed directly into the SFP (Cycle 1 fuel only). Those assemblies placed in the new fuel elevator are lowered to the bottom of the fuel transfer canal. They are then engaged by the spent fuel handling tool, which is in turn suspended from the fuel handling machine. The fuel handling machine either transfers the assembly to the spent fuel storage racks (before initial refueling) or to the FTS upender for transfer to the RCB for refueling operations. The upender pivots the assembly to the horizontal position and the FTS fuel container carries it through the fuel transfer tube to an upender inside the Containment.

9.1.4.2.2 Refueling Procedure: The refueling operation follows a detailed procedure to ensure a safe, efficient refueling operation. Prior to initiating refueling, shutdown conditions are as specified in the Technical Specifications. Criticality protection for the refueling operation, including a requirement for checks of boron concentration, is specified in the Technical Specifications.

Protection against uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical condition is described in Section 15.4.1 and includes source-range, intermediate-range, and power-range high neutron flux trips. The transient is assumed to be terminated by the power-range high neutron flux (low setting) reactor trip. Protection against uncontrolled boron dilution is described in Section 15.4.6.

The following significant points are assured by the refueling procedure:

1. The refueling water and the reactor coolant contain approximately 2,800 ppm boron. This concentration is sufficient to keep the core approximately 5 percent $\Delta k/k$ subcritical during the refueling operations with all control rods removed and the core refueled to provide sufficient excess reactivity for operation to the next refueling outage.

2. The water level in the refueling cavity is high enough to keep the radiation levels within acceptable limits when the fuel assemblies are being removed from the core.

The refueling operation is divided into four major phases: (1) preparation, (2) reactor disassembly, (3) fuel-handling, and (4) reactor assembly. A general description of a typical refueling operation through the four phases is given below.

This description applies to rapid (unrodded) refueling which will normally be used during a refueling shutdown to maximize plant availability. The description also points out the different steps that would be included in a nonrapid (rodded) refueling operation, which is typically used for extended shutdowns involving nonroutine maintenance.

9.1.4.2.2.1 Phase I - Preparation - The reactor is shut down (rods in) and then simultaneously borated and cooled down to refueling shutdown condition. Following initiation of normal purge and a radiation survey of the Containment Building, refueling operations may proceed. The fuel transfer equipment and refueling machines are checked for proper operation. Each new fuel assembly is brought from dry storage in the FHB as described in Section 9.1.4.2.1. After transfer through the fuel transfer tube, the FTS fuel assembly container is pivoted to the vertical position by the in-Containment upender. The refueling machine transfers the new fuel into the in-Containment storage racks (shown in Figure 9.1.2.1.a). Refer to Figures 1.2-14 through 1.2-18 for general arrangement of the in-Containment fuel storage area.

When the Reactor Coolant System (RCS) has been cooled to 150°F, RCS draining is started. The RCC assemblies (control rods) are withdrawn to their full-out position, and each control rod's holdout device is activated to ensure that the rod is held in its withdrawn position inside its upper internals guide tube and reactor head pressure housing. As soon as RCS draining has lowered reactor coolant level to the reactor vessel nozzle centerline, degassing operations using the RCS Vacuum Degassing System (Setion 11.3) may be used, if necessary, to remove radioactive gases prior to head removal.

For nonrapid refueling, the preparation for refueling is similar except that the control rods are not withdrawn from the core and the RCS temperature is reduced to 140°F before draining.

9.1.4.2.2.2 Phase II - Reactor Disassembly - The seismic tie rods attached to the missile shield are disconnected and stored. The insulation is removed from the vessel head flange area, and the Roto-Lok studs are detensioned and removed from the vessel flange. A stud hole plug is installed in each hole after the stud is removed to prevent entry of water. In addition, all flux mapping detectors and thimbles are retracted through the bottom of the reactor vessel. The refueling cavity is prepared for flooding by removing all tools, closing the refueling canal drain holes and by installing underwater lights. The reactor cavity is then flooded to 12 in. below the top of the head flange. The upper head package (i.e., head, missile, cable bridge, upper internals, control rods, and rod drives) is lifted by the polar crane until the closure head guide pins are clear. Water from the RWST is pumped into the RCS by the LHSI pumps, causing the water to overflow into the refueling cavity. The vessel head is lifted in conjunction with the water level in the refueling

cavity. When the refueling cavity is full, the upper package is moved to storage at the end of the refueling cavity opposite the refueling canal. If a radiation survey indicates the need, additional water shielding may be provided by pulling a vacuum through the reactor vessel head vent connection.

For nonrapid refueling, reactor disassembly includes additional steps starting with disconnecting the cables that run across the cable bridge. This allows the bridge to be lifted by the polar crane separately from the reactor head. The control rod drive shafts are disconnected leaving the rods in the core. The upper internals assembly is then disconnected from the reactor vessel head. After the cable bridge is lifted clear of the reactor head, the head is lifted about 24 in. to permit a visual inspection of the RCC assembly drive shafts. This ensures that they are free in the control rod drive mechanism (CRDM) housings and were not raised with the reactor head. The refueling cavity is then flooded to a level just below the closure head. Simultaneous lifting of the reactor head and filling of the refueling cavity proceeds in the same manner as described for rapid refueling.

The vessel head is placed on a dry storage pedestal in a roped-off area of the operating floor at the north end of the containment. RCC drive shafts are unlatched from their respective RCC assemblies. Finally, the upper internals are removed from the vessel by using the reactor internals lifting device suspended from the polar crane. The internals package is wet-stored on a stand in the north end of the refueling cavity.

9.1.4.2.2.3 Phase III - Fuel Handling - Fuel assemblies are removed from and inserted in to the reactor core by the refueling machine. The spent fuel assemblies are removed from the core in a sequence which is planned before each refueling. The positions of partially spent fuel assemblies are shuffled, and new fuel assemblies are added to the core.

The general fuel handling sequence is:

1. The refueling machine is positioned over a spent fuel assembly in the most depleted region of the core.

2. The spent fuel assembly is lifted by the refueling machine to a predetermined height sufficient to clear the reactor vessel and still leave sufficient water covering the spent fuel assembly to eliminate any radiation hazard to the operating personnel.

3. The fuel transfer car is moved into the refueling canal from the fuel transfer canal.

4. The FTS fuel assembly container is pivoted to the vertical position by the upender.

5. The refueling machine is moved from over the core to line up the spent fuel assembly with the fuel assembly container.

6. The refueling machine loads the spent fuel assembly into the fuel assembly container of the FTS transfer car.

7. The container is pivoted to the horizontal position by the upender.

9. The fuel assembly container is pivoted to the vertical position by the upender. The spent fuel assembly is unloaded by the spent fuel handling tool, which is suspended from the fuel handling machine hoist.

10. The spent fuel assembly is placed in the spent fuel storage rack after being transferred through the gate between the fuel transfer canal and the SFP.

11. Partially spent fuel assemblies are moved either to the SFP in the FHB or to new positions in the reactor core, and new fuel assemblies are moved from the in-Containment storage racks to the core to replace the spent fuel assemblies that were removed during the preceding fuel handling steps.

12. This procedure is continued until refueling is completed.

During nonrapid refueling, some of the spent fuel assemblies that are removed from the reactor core will contain a rod cluster control (RCC) (control rod) element. Such assemblies are placed in the RCC change fixture by the refueling machine. The RCC change fixture is located adjacent to the in-Containment storage racks. Here the RCC element is removed from the spent fuel assembly and deposited in a partially spent or new fuel assembly previously placed in the RCC change fixture.

Another step generally performed during either rapid or nonrapid refueling is the removal of an irradiated specimen from the reactor core for examination. Also, remote boroscope and television camera inspections of the core and reactor vessel are performed using equipment suspended from the refueling machine.

9.1.4.2.2.4 Phase IV - Reactor Assembly - Reactor assembly, following refueling, is essentially achieved by reversing the operations given in "Phase II - Reactor Disassembly".

9.1.4.2.3 Spent Fuel Shipment: Spent fuel is shipped offsite either by rail or truck in heavily shielded spent fuel shipping casks licensed for use by the Department of Transportation (DOT). General arrangements of the cask handling area are provided in Section 1.2.

Upon receipt in the FHB, the spent fuel cask shipping vehicle (rail or truck) is braked and blocked in position for removal of the spent fuel cask. The cask is then inspected for shipment damage and to ascertain the degree of additional cleaning that will be required to remove road dirt or radioactive contamination from the cask surface. The gate between the cask loading pool and the decontamination area is removed using the 15/2-ton FHB overhead crane. The upper and lower spent fuel cask impact structures are removed, and spent fuel cask appurtenances are disconnected. The spent fuel cask yoke is removed from its storage area and attached to the spent fuel cask trunnions using the 150-ton, overhead cask-handling crane. As the spent fuel cask is upended, the yoke is maintained in a vertical position by lateral movement of the overhead spent fuel cask handling crane trolley. (The reverse procedure is performed when the cask is loaded onto the shipping vehicle.) The spent fuel cask is then lifted clear of the shipping vehicle, moved over to the access bay, and lowered to the decontamination area, where any additional cleaning of road dirt or surface contamination is performed.

Using the 150-ton, overhead cask-handling crane, which is still connected to the cask, the spent fuel cask is lifted, moved horizontally, and lowered into the cask loading pool. The lifting yoke is removed and placed in its storage location. The spent fuel cask head is unbolted and moved, using the FHB overhead crane, to a temporary storage location. The gate between the cask loading pool and the decontamination area is replaced. The cask loading pool and channel are filled with borated water to the same level as the SFP (nominally El. 66 ft-6 in.). The two gates between the cask loading pool and the SFP are removed, by the 15/2-ton (15-ton main hook and a 2-ton auxiliary hook) FHB overhead crane, and fuel transfer is initiated. Loading of the spent fuel assemblies is accomplished in the following manner: the spent fuel assembly in the SFP is engaged by the long spent fuel handling tool, which is in turn attached to the fuel-handling machine; the fuel assembly is removed from the storage rack and transferred through the gate area into the cask loading pool and then lowered into the spent fuel cask. The height to which the spent fuel assembly can be lifted is restricted by the length of the spent fuel handling tool and the fuel-handling machine design in order to provide a minimum of 10 ft of shielding water above the spent fuel assembly. This spent fuel handling procedure is repeated until the spent fuel cask is full. The gates sealing the cask loading pool from the SFP are then replaced using the 15/2-ton FHB overhead crane.

The spent fuel cask head is lifted from its storage area using the 15/2-ton overhead FHB crane and placed on the spent fuel cask. The cask head bolts that secure the cask head to the cask are installed and tightened "finger tight" using long-handled tooling operated from the fuel-handling machine.

The cask loading pool level is lowered to at least 6 in. below the bottom of the gate between the cask loading pool and the decontamination area. This gate is removed using the 15/2-ton overhead crane. Using the 150-ton cask-handling crane, the cask-lifting yoke is attached to the cask-lifting trunnions and the cask is transferred from the cask loading pool to the cask decontamination area. A radiological survey is made of the cask to determine the extent of cask decontamination necessary. The head bolts are torqued to the proper value, and various preshipment testing requirements characteristic of the spent fuel cask being used are completed. These tests may include a cask leak test and monitoring of cask coolant activity. The contaminated outside surface areas are manually decontaminated using high-pressure sprayers or by using scrub brushes with detergent, rinses, and wipes. Once the cask is decontaminated and has passed the preshipment testing requirements, it is lifted back through the access bay and returned to its horizontal position on the shipping vehicle. The cask holddown mechanisms are secured, the cask top and bottom impact limiters are replaced, and all appropriate appurtenances are reconnected. Shipping papers are completed and the cask is removed from the FHB and released for shipment.

9.1.4.2.4 <u>Component Description</u>:

9.1.4.2.4.1 <u>Refueling Machine</u> - The refueling machine (Figure 9.1.4-1) is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling cavity. The bridge spans the refueling cavity and runs on rails set into the edge of the refueling cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly in the core. A long tube, with a pneumatic gripper on the end, is lowered down out of the mast to grip a fuel assembly. The gripper tube is long enough so that

the upper end is still contained in the mast when the gripper end contacts the fuel. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. The fuel is transported while inside the mast tube to its new position.

The refueling machine gripper tube also includes a secondary gripper mechanism (located above the primary fuel assembly gripper) to remove and insert a thimble plug into a new fuel assembly.

The refueling machine drive uses a direct current control device that gives stepless, variable speeds from zero to full speed.

All controls for the refueling machine are mounted in a console on the trolley. The bridge and trolley are positioned by a servo system in relation to an X-Y coordinate grid pattern referenced to the reactor core. Bridge and trolley position is indicated by an electric position-repeat-back system. Readout dials are read directly by the operator at the console. The drives for the bridge, trolley, and winch are variable speed and include a separate inching control on the bridge and trolley. The maximum speed is 60 ft/min for the bridge and 20 ft/min for the trolley and hoist. An auxiliary monorail hoist on the refueling machine uses a two-step magnetic controller to give hoisting speeds of approximately 7 ft/min and 20 ft/min for use in handling accessory equipment.

Electrical interlocks and limit switches on the bridge and trolley drives prevent damage to the fuel assemblies. The winch is also provided with redundant limit switches plus a mechanical stop to prevent a fuel assembly from being raised above a safe shielding depth should the limit switch fail. In an emergency, the bridge, trolley, and winch can be operated manually using a handwheel on the motor shaft.

The refueling machine is provided with a television system which permits viewing of all fuel assembly positions and fuel movements to within 6 in. of the top of the core. This system includes a telescoping boom, a monitor, and a videotape recorder.

The refueling machine is designed to permit failed fuel detection by means of a sip test. A fuel assembly will be withdrawn inside the mast and held stationary above the core for a period of time. Doors at the bottom of the mast are shut to enclose the fuel assembly completely. The water surrounding the fuel assembly will then be circulated through detection equipment located on the crane trolley. Any activity increase will be evaluated to determine the severity of cladding failure.

9.1.4.2.4.2 Fuel Handling Machine - The fuel handling machine consists of an electric monorail hoist carried on a wheel-mounted bridge (Figure 9.1.4-2), which spans the SFP, fuel transfer canal, and cask loading pool. The fuel handling machine is used exclusively for handling fuel assemblies and core components by means of handling tools suspended from the hoist. The hoist travel and tool lengths are designed to limit the maximum lift of a fuel assembly or core component to a safe shielding depth.

The fuel handling machine has a two-step magnetic controller for the bridge and hoist. The bridge speeds are 10 ft/min and 30 ft/min, and the hoist speeds are 7 ft/min and 20 ft/min. A hydraulic coupling is used in the bridge

drive to limit starting acceleration. The hoist trolley is manually positioned along the monorail by a chainfall.

9.1.4.2.4.3 New Fuel Elevator - The new fuel elevator (Figure 9.1.4-3) consists of a box-shaped elevator assembly with its top end open and sized to house one fuel assembly.

The new fuel elevator is normally used to lower a new fuel assembly into the fuel transfer canal, where the fuel handling machine can transport it to the FTS equipment for transfer into the Containment. The new fuel elevator can also be used to raise a new or spent fuel assembly provided administrative controls and procedures are utilized.

9.1.4.2.4.4 Fuel Transfer System - The FTS (Figure 9.1.4-4) includes an underwater, electric-motor-driven transfer car that runs on tracks extending from the refueling canal in the RCB through the fuel transfer tube and into the fuel transfer canal in the FHB, and a hydraulically actuated lifting arm (upender) at each end of the transfer tube. In the refueling canal the fuel container mounted on the transfer car receives a fuel assembly in the vertical position from the refueling machine. The upender then lowers the fuel assembly to a horizontal position for passage through the transfer tube. After passing through the tube, the fuel container is raised to a vertical position by the other upender for removal of the fuel assembly. The fuel handling machine lifts the fuel assembly out of the fuel container and moves it to the desired storage position in the spent fuel storage racks.

The transfer car is driven by a pusher arm connected to two continuous roller chains. The roller chains are driven by an electric motor mounted near the operating floor of the FHB next to the fuel transfer canal. They are connected to the chain-drive sprockets by a vertical drive shaft. The fuel container is center-pivoted, and the pivot structure serves to attach the container to the transfer car so they move together as a single unit. There is a mechanical stop at each end of the rail on which the car travels. Also, there is another mechanical stop for the vertical limit of travel when the fuel container is upended. Each of the upender lifting arms has its own hydraulic power unit and control console located on the operating floor of its respective side of the Containment wall.

During reactor operation, the transfer car is stored in the fuel transfer canal. A blind flange is bolted on the refueling canal end of the transfer tube to seal the Reactor Containment. The end of the tube in the FHB is closed by a gate valve.

9.1.4.2.4.5 Rod Cluster Control Change Fixture - The RCC change fixture is located in the in-Containment fuel storage pit. The change fixture is used for periodic RCC element inspections and for transfer of the RCC elements from one fuel assembly to another, or transfer of secondary source assemblies from one fuel assembly to another (Figure 9.1.4-5). During rapid refueling operations, the RCC change fixture is not required for transfer of RCC elements since the RCC elements are removed from the reactor core along with the upper head package. The major subassemblies which constitute the change fixture are the frame and track structure, the carriage, the guide tube, the gripper, and the drive mechanism. The carriage is a moveable container supported by the frame and track structure. The tracks provide a guide for the four flanged carriage wheels and allow horizontal movement of the carriage during the changing operation. The positioning stops on both the carriage and frame locate each of the three carriage compartments directly below the guide

tube. Two of these compartments are designed to hold individual fuel assemblies while the third is made to support a single core component (e.g., RCCA or secondary source assembly). Situated above the carriage and mounted on the in-Containment fuel storage pit wall is the guide tube. The guide tube provides proper orientation of the gripper and core component as they are being raised and lowered. The gripper is a pneumatically actuated mechanism which engages the core component. It has two flexure fingers which can be inserted into the top of the core component when air pressure is applied to the gripper piston. Normally, the fingers are locked in a radially extended position. Mounted on the operating deck is the drive mechanism assembly, which is composed of the manual carriage drive mechanism, the revolving stop operating handle, the pneumatic selector valve for actuating the gripper piston, and the electric hoist for elevating the gripper.

The pneumatic gripper and winch lift the core component out of a spent fuel assembly and up into the guide tube. Then by repositioning the carriage, a new fuel assembly is brought under the guide tube and the gripper lowers and releases the core component into the new fuel assembly. The refueling machine loads and unloads the fuel assemblies into and from the carriage.

9.1.4.2.4.6 Spent Fuel Assembly Handling Tool - The spent fuel assembly handling tool (Figure 9.1.4-6) is used to handle new and spent fuel assemblies in the spent fuel pit. It is a manually actuated tool, suspended from the fuel handling machine. The gripping portion of the tool consists of a mechanical finger which engages the fuel assembly top nozzle when actuated. The operating handle to actuate the fingers is located at the top of the tool. When the fingers are latched, a pin is inserted into the operating handle, to prevent the fingers from being accidentally unlatched during fuel handling operations.

9.1.4.2.4.7 New Fuel Assembly Handling Tool - The new fuel assembly handling tool (Figure 9.1.4-7) is used to lift and transfer fuel assemblies from the new fuel shipping containers to the new fuel storage racks, to transfer fuel assemblies from the new fuel storage racks to the new fuel elevator, or to directly place new fuel in the SFP (Cycle 1 only). It is a manually actuated tool, suspended from the FHB overhead crane, which uses four cam-actuated latching fingers to grip the underside of the fuel assembly top nozzle. The operating handles which actuate the fingers are located on the side of the tool. When the fingers are latched, a safety screw is turned in to prevent the accidental unlatching of the fingers.

9.1.4.2.4.8 Reactor Vessel Head and Upper Internals Lifting Device - The reactor vessel head and upper internals lifting device consists of a welded and bolted structural steel frame with suitable rigging to enable the crane operator to lift the head or the head and upper internals for storage during refueling operations. The lifting device normally remains attached to the reactor vessel head during plant operation. The missile shield and the control rod drive mechanism cooling shroud are attached to the head lifting device.

9.1.4.2.4.9 Reactor Internals Lifting Device - For rapid refueling, the upper internals are normally lifted in one lift as part of the upper head package. For nonrapid refueling, the head may be separated from the upper internals and lifted separately. In this case, the reactor internals lifting rig (Figure 9.1.4-8) is used to remove the upper internals.

9.1.4.2.4.10 Reactor Vessel Stud Tensioner - The stud tensioners (Figure 9.1.4-9) are employed to secure or release the head closure joint at every refueling. The stud tensioner is a hydraulically operated device that uses oil as the working fluid. The device permits preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. Stud tensioners minimize the time required for the tensioning or unloading operation. Three tensioners are provided and are applied simultaneously to three studs located 120 degrees apart. A single hydraulic pumping unit operates the tensioners, which are hydraulically connected in series. The studs are tensioned to their operational load in two steps to prevent high stresses on the reactor vessel flange region and unequal loading of the studs. Relief valves on each tensioner prevent overtensioning of the studs due to excessive pressure.

9.1.4.2.4.11 Spent Fuel Cask - The design of cask storage and cask handling facilities is based on a design cask weighing up to 150 tons and measuring approximately 21 ft long by 10 ft in diameter. The shipping vehicle (rail car or truck) will take the cask to and from the FHB and will be equipped with a cask-cooling system and a storage area for the cask yoke. The FHB arrangement and cask-handling equipment are designed to preclude the occurrence of any accident to a loaded spent fuel shipping cask beyond the regulatory-specified design accident conditions for the cask. Overland offsite transportation of the cask will conform to transportation rules and regulations, 49CFR173.

9.1.4.2.4.12 Fuel Handling Building Overhead Crane - This 15/2-ton overhead crane runs over the entire FHB area. Travel of this crane is shown on Figure 9.1.4-13. The design services of this crane include the following:

1. Transfer of new fuel assembly shipping containers from the shipping vehicle (truck or rail car) to the new fuel handling area.

2. Transfer of new fuel assemblies from new fuel shipping containers to new fuel elevator, new fuel storage pit, or SFP (cycle 1 only).

3. Transfer of spent fuel shipping cask head from the cask to temporary storage on the FHB operating floor and then back onto the shipping cask when the fuel loading is complete.

4. Replacement of safety injection and Containment spray pumps and SFP cooling HXs.

5. Removal and replacement of pool gates for fuel transfer operations.

9.1.4.2.4.13 New Fuel Handling Area Overhead Crane - This 5-ton overhead crane is used exclusively to handle new fuel assemblies and their shipping containers in the new fuel handling area.

9.1.4.2.5 Industrial Codes and Standards: The following industrial codes and standards are used in the design of fuel-handling equipment.

1. Cranes: Crane Manufacturers Association of America (CMAA) specification No. 70, Class A-1.

2. Structural: ASME Code, Section III, Appendix XVII.

3. Electrical: Applicable standards and requirements of the National Electric Code, National Fire Protection Association No. 70, and National Electrical Manufacturers Association standards MGI and ICS are used in the design, installation, and manufacturing of all electrical equipment.

4. Materials: Materials conformed to the specifications of the American Society for Testing Materials standard.

5. Safety: The design meets the applicable requirements of Section 1910.179 of Subpart N of the Occupational Safety and Health Act Code.

6. Others: American Institute of Steel Construction; American National Standards Institute; American Society of Testing Materials; Institute of Electrical & Electronic Engineers; National Electric Manufacturers Association; Occupational Safety & Health Administration; American Welding Society; Expansion Joint Manufacturers Association; ASME B&PV Code Sections VIII and XI; American Concrete Institute; Hydraulic Institute Standards.

9.1.4.3 Safety Evaluation. Design of the FHS in accordance with RG 1.13 and with GDCs 2, 5, 61, and 62 ensures a safe condition under normal and postulated accident conditions.

9.1.4.3.1 Safe Handling:

9.1.4.3.1.1 Design Criteria for the Refueling Machine and the Fuel-Handling Machine - The primary design requirement of the machine is reliability. A conservative design approach is used for all load-bearing parts. Where practicable, components are used that have a proven record of reliable service. Throughout the design, consideration is given to the fact that the machine will spend long idle periods stored in an atmosphere of 80°F and high humidity. In general, the crane structure is considered in the Class A1, Standby Service, as defined by CMAA Specification No. 70.

All components critical to the operation of the machine and parts which could fall into the reactor are positively restrained from loosening.

9.1.4.3.1.2 Refueling Machine - The refueling machine design includes the following provisions to ensure safe handling of fuel assemblies:

1. Electrical Interlocks

   The electrical interlocks which ensure safe operation of the crane are designed to meet single-failure criteria.

2. Bridge, Trolley, and Hoist Drive Mutual Interlocks

   Bridge, trolley, and winch drives are mutually interlocked, using redundant interlocks to prevent simultaneous operation of any two drives, and therefore can withstand a single failure.

3. Bridge Trolley Drive Gripper Tube Up

   Bridge and trolley drive operation is prevented except when the "gripper tube up" position switches are actuated. The interlock is redundant and can withstand a single failure.

4. Gripper Interlock

   An interlock is supplied which prevents the opening of a solenoid valve in the air line to the gripper except when there is less than 600 pounds of suspended weight indicated on a load cell. As backup protection for this interlock, the mechanical weight-actuated lock in the gripper prevents opening of the gripper under load even if air pressure is applied to the operating cylinder. This interlock is redundant and can withstand a single failure.

5. Excessive Suspended Weight

   An excessive suspended weight load cell limit switch and a backup deflection-actuated limit switch prohibit raising the guide tube if the load suspended from it significantly exceeds the weight of the gripper, a fuel assembly, and an RCCA. The interlock is intended to prevent inadvertent damage to a fuel assembly or adjacent components if the assembly becomes stuck during its removal.

6. Hoist-Gripper Position Interlock

   An interlock in the hoist drive circuit in the up direction permits the hoist to be operated only when either the open or closed indicating switch on the gripper is actuated. The hoist-gripper position interlock consists of two separate circuits that work in parallel such that one circuit must be closed for the hoist to operate. If one or both interlocking circuits fail in the closed position, an audible and visual alarm on the console is actuated. The interlock, therefore, is not redundant but can withstand a single failure since both an interlocking circuit and the monitoring circuit must fail to cause a hazardous condition.

7. Bridge and Trolley Hold-Down Devices

   Both the refueling machine bridge and trolley are horizontally restrained on the rails by two pairs of guide rollers, one pair at each wheel location on one truck only. The rollers are attached to the bridge truck and contact the vertical faces on either side of the rail to prevent horizontal movement. Vertical restraint is accomplished by antirotation bars located at each of the four wheels for both the bridge and trolley. The antirotation bars are bolted to the trucks and extended under the rail flange for the bridge restraint; the trolley restraints extend beneath the top flange of the bridge girder which supports the trolley rail. Both horizontal and vertical restraints are adequately designed to withstand the forces and overturning moments resulting from an SSE.

8. Design Load

   The design load for structural components is their deadweight plus 5,700 pounds (three times the weight of a fuel assembly and RCCA).

9. Main Hoist Braking System

   The main hoists are equipped with two independent braking systems. A solenoid-release, spring-set electric holding brake is mounted on the motor shaft. The brake operates normally to release upon application of current to the motor and to set when current is interrupted. The second brake is a mechanically actuated load control brake internal to the hoist gear box; it sets if the load starts to overhaul the hoist. It is necessary to apply torque from the motor to raise or lower the load. In raising the motor cams, the brake opens; in lowering, the motor slips the brake, allowing the load to lower. The brake actuates upon loss of torque from the motor for any reason and is not dependent upon any electrical circuits. On the main hoist, the motor brake is rated at 350 percent operating load and the mechanical brake at 300 percent.

   The Main Hoist Braking System is supplied with redundant paths of load support such that failure of any one component will not result in free fall of the fuel assembly. Two wire ropes are anchored to the winch drum and carried over independent sheaves to a load equalizing mechanism on the top of the gripper tube. In addition, supports for the sheaves and equalizing mechanism are backed up by passive restraints to pick up the load in the event of failure of this primary support. Each cable system is designed to support 13,750 pounds or 27,500 pounds acting together.

10. Hoist Down Limit

   A geared limit switch on the main hoist prevents lowering the gripper tube significantly below the position that would normally engage fuel assembly in the reactor core.

The working load of fuel assembly, RCCA, and gripper is approximately 2,850 pounds.

The gripper itself has four fingers gripping the fuel, any two of which will support the fuel assembly weight.

The gripper mechanism contains a spring-actuated mechanical lock which prevents the gripper from opening unless the gripper is under a compressive load.

The refueling machine gripper and hoist system are routinely load tested prior to refueling operations in accordance with the surveillance requirements of the Technical Specifications.

9.1.4.3.1.3 Fuel-Handling Machine - The fuel-handling machine includes the following safety features:

1. The bridge and hoist controls are interlocked to prevent simultaneous operation of both the bridge drive and the hoist. The interlocks are redundant and can withstand a single failure.

2. A redundant overload protection device is included on the hoist to limit the uplift force which could be applied to the spent fuel storage racks. The protection device limits the hoist load to 125 percent (5,000 pounds) of the rated 2-ton hoist capacity. This device can withstand a single failure. A load-monitoring device is provided between the hoist and spent fuel handling tool. By monitoring drag load when raising or lowering assemblies, it can be determined immediately if the assembly is hanging or experiencing unusual loads.

3. The design load on the hoist is the weight of one fuel assembly and RCCA (1,900 pounds), one failed fuel container (1,700 pounds), and the tool (400 pounds), which gives it a total weight of approximately 4,000 pounds.

4. Restraining bars are provided on each truck to prevent the bridge from overturning.

9.1.4.3.1.4 Fuel Transfer System - The following safety features are provided for in the FTS:

1. Transfer Car Permissive Switch

   The transfer car controls are located in the spent fuel pit area; therefore, conditions in the Containment are not visible to the operator. The transfer car permissive switch allows a second operator in the Containment to exercise some control over car movement if visible conditions warrant it.

   Transfer car operation is possible only when both upenders are in the down position as indicated by the limit switches. The permissive switch is a backup for the transfer car upender interlock. Assuming the fuel container is in the upright position in the Containment and the upender interlock circuit fails in its permissive condition, the operator in the spent fuel pit area still cannot operate the car because of the transfer car permissive switch interlock. The interlock, therefore, can withstand a single failure.

2. Lifting Arm Transfer Car Position

   Two redundant interlocks allow lifting arm operation only when the transfer car is at the respective end of its travel and therefore can withstand a single failure.

   Of the two redundant interlocks that allow lifting arm operation only when the transfer car is at the end of its travel, one interlock is a position limit switch in the control circuit. The backup interlock is a mechanical latch device on the lifting arm that is opened by the car moving into position.

3. Transfer Car Valve Open

Two redundant interlocks on the transfer tube valve permit operation of the transfer car only when the transfer tube valve position switch indicates the valve is fully open and, therefore, can withstand a single failure.

4. Transfer Car Upender

The transfer car upender interlock is primarily designed to protect the equipment from overload and possible damage if an attempt is made to move the car when the fuel container is in the vertical position. This interlock is redundant and can withstand a single failure. The basic interlock is a position limit switch in the control circuit. The backup interlock is a mechanical latch device that is opened by the weight of the fuel container when in the horizontal position.

5. Upender Refueling Machine

The refueling canal upender is interlocked with the refueling machine. Whenever the transfer car is located in the refueling canal, the upender cannot be operated unless the refueling machine mast is in the fully retracted position or the machine is over the core.

9.1.4.3.1.5 Fuel Handling Tools and Equipment - All fuel handling tools and equipment which are used over the open reactor vessel are designed to prevent inadvertent decoupling from crane hooks; i.e., lifting rigs are pinned to the crane hook and safety latches are provided on hooks supporting tools. Tools required for handling internal reactor components are designed with fail-safe features that prevent disengagement of the component in the event of operating mechanism malfunction. These safety features apply to the following tools:

1. Control Rod Drive Shaft Unlatching Tool

The air cylinders actuating the gripper mechanism are equipped with backup springs which close the gripper in the event of loss of air to the cylinder. Air valves are equipped with safety locking rings to prevent inadvertent actuation.

2. Spent Fuel Handling Tool

When the fingers are latched, a pin is inserted into the operating handle to prevent inadvertent actuation. The tool weighs approximately 400 pounds and is preoperationally tested at 125 percent of the weight of one fuel assembly and RCCA (1,900 pounds).

3. New Fuel Assembly Handling Tool

When the fingers are latched, a safety screw is inserted to prevent inadvertent actuation. The tool weighs approximately 100 pounds and is preoperationally tested at 125 percent of the weight of one fuel assembly and RCCA (1,900 pounds).

9.1.4.3.1.6 <u>Overhead Cranes</u> - Overhead cranes used in refueling and fuel handling operations include the polar crane (417/15-ton, Unit 1 and 500/15 ton, Unit 2), the 150-ton cask-handling crane, the 15/2-ton FHB crane, and the 5-ton new fuel handling area crane. These cranes are classified as non-nuclear safety (NNS) Class since they neither provide nor support any safety system function. However, during and after a seismic event, the cranes and their supports are designed to retain structural integrity and prevent collapse and damage to safety-related equipment and structures. Operability need not be retained.

A report under separate cover has been submitted to the Nuclear Regulatory Commission (NRC) concerning control of heavy loads. This report contains details of crane/load combinations and the safeguards that prevent damage to spent fuel, the reactor core, equipment required for safe shutdown, and decay heat removal.

1. Polar Crane

   The polar crane is used for general handling operations in the containment during refueling. These operations include:

   a. Removal of the upper package

   b. Removal of pumps, pump motors, and heat exchangers

   c. Handling of pool gate

   d. Handling of inservice inspection (ISI) rig

   e. Movement of hatch covers.

   A head drop analysis is discussed in Letter NS-CE-1101 (June 11, 1976) and received NRC approval by letter on November 30, 1976. This crane is provided with seismic restraints to prevent derailment in the event of an SSE.

2. Cask-Handling Overhead Crane

   This 150-ton crane is provided for handling the spent fuel shipping cask. Crane design and building arrangement preclude travel of this crane over the SFP; consequently, the shipping cask cannot be lifted or dropped over the spent fuel racks. This crane is designed to maintain its structural integrity and hold its load under the dynamic loading conditions of the SSE. Building arrangement and lifting rig design prevents this crane from lifting the cask higher than 30 ft above the floor. The spent fuel cask drop accident is discussed in Section 15.7.5.

3. Fuel-Handling Building Overhead Crane

The 15/2-ton capacity crane is to be used for general handling operations in the FHB. These operations include:

a. Movement of new fuel assemblies

b. Removal of pumps and heat exchangers

c. Handling of the pool gates

d. Movement of the cask head

e. Movement of hatch covers

This crane is designed to maintain its structural integrity under the dynamic loading of the SSE. The crane will retain its load under such dynamic loadings.

This crane main hoist is also provided with a redundant reeving system. With this redundancy, the crane can withstand a single failure without dropping its load and therefore meets the intent of RG 1.104. A more detailed description of compliance of the 15/2-ton FHB crane with RG 1.104 is given in Table 9.1-3.

4. New Fuel Handling Area Overhead Crane

The 5-ton new fuel handling area overhead crane is used for movement of new fuel assemblies within the new fuel handling area. Dropping of new fuel assemblies due to SSE-induced dynamic loading of the crane will not result in an offsite radiological hazard. The crane travels over no safety-related equipment.

9.1.4.3.2 Seismic Considerations: The safety classifications for all fuel handling and storage equipment are listed in Table 3.2.B-2. SC 1, 2, and 3 equipment is designed to withstand the effects of an SSE without loss of capability to perform its safety function. Further, the combined normal and SSE stresses are limited to the allowable stresses as defined by ASME Code, Section III, Appendix XVII-2110. SC 1 and 2 equipment is designed to withstand the forces of an Operating Basis Earthquake (OBE), with the combined normal and OBE stresses being limited to the allowable stresses, as defined by ASME Code, Section III, Appendix XVII. For SC 3 equipment, consideration is given to the OBE only insofar as failure of the SC 3 equipment might adversely affect SC 1 or 2 equipment.

For NNS equipment, design for the SSE is considered if failure might adversely affect a SC 1, 2, or 3 component. Design for OBE is considered if failure of the NNS component might adversely affect an SC 1 or 2 component.

9.1.4.3.3 Containment Pressure Boundary Integrity: The fuel transfer tube, which connects the refueling canal (inside the RCB) and the SFP (outside the Containment), is closed on the refueling canal side by a blind flange at all times except during refueling operations. Further discussion on the fuel transfer tube can be found in Section 3.8.2.1.3.3.

9.1.4.3.4 Radiation Shielding: During all phases of spent fuel transfer, the gamma dose rate at the surface of the water is 2.5 mR/hr or less. This is accomplished by maintaining approximately 10 ft of water above the top of the fuel assembly during all handling operations.

The two machines used to lift spent fuel assemblies are the refueling machine and the fuel handling machine. The refueling machine contains positive stops that prevent the top of a fuel assembly from being raised to within approximately 10 ft of the normal water level in the refueling cavity. The hoist on the fuel handling machine moves spent fuel assemblies with a long-handled tool. Hoist travel and tool length likewise limit the maximum lift of a fuel assembly to within approximately 10 ft of the normal water level in the SFP.

9.1.4.4 Tests and Inspections. As part of normal plant operations, the fuel handling equipment is inspected for operability before each refueling operation. During the operational testing of this equipment, procedures are followed that will verify the correct performance of the FHS interlocks.

9.1.4.5 Instrumentation Requirements. A description of the instrumentation and controls is provided in Section 9.1.4.3 for the refueling machine, the fuel handling machine, the FTS, and the SFCHS.

~~TABLE 9.1-1~~

~~SPENT FUEL POOL COOLING AND CLEANUP SYSTEM~~
~~DESIGN PARAMETERS~~

~~MODE~~	~~FUEL LOAD (PER SRP 9.1.3)~~	~~STPEGS FUEL LOAD (ACTUAL)~~	~~MAX ALLOWABLE POOL TEMPERATURE (PER SRP 9.1.3)~~	~~MAX STPEGS POOL TEMPERATURE (1 COOLING TRAIN)~~	~~MAX STPEGS POOL TEMPERATURE (2 COOLING TRAINS)~~	~~HEAT LOAD (10^6 BTU/HR)~~
~~Normal~~	~~1/3 core - 150 hrs~~ ~~1/3 core - 1 yr~~ ~~1/3 core - 400 days~~	~~N/A~~	~~N/A~~	~~131.2°F~~	~~118.7°F~~	~~16.6~~
~~Normal Maximum~~	~~N/A~~	~~1/3 core - 140 hrs~~ ~~1/3 core - 1 yr~~ ~~1/3 core - 2-26 yrs~~(1)	~~140°F~~	~~145.7°F~~	~~126.0°F~~	~~25.5~~
	~~N/A~~	~~1/3 core - 80 hrs~~ ~~1/3 core - 1 yr~~ ~~1/3 core - 2-26 yrs~~(1)	~~N/A~~	~~150.7°F~~	~~129.2°F~~	~~29.3~~
~~Abnormal Maximum~~	~~1 core - 150 hrs~~ ~~1/3 core - 36 days~~ ~~1/3 core - 400 days~~	~~1 core - 120 hrs~~ ~~1/3 core - 36 days~~ ~~1/3 core - 1 yr~~ ~~1/3 core - 2-26 yrs~~(5)	~~No Boiling~~(2,3)	~~N/A~~	~~155.4°F~~(4)	~~63.2~~

~~1. Full core discharge capability is maintained, i.e., 1776 fuel assemblies.~~

~~2. In the event of a fire or moderate energy line crack in the Fuel Handling Building that disables both trains of spent fuel pool cooling, the spent fuel pool may eventually boil. Makeup can be provided via the reactor makeup pumps. In addition, makeup water can also be supplied to the spent fuel pool using local hose stations in the FHB. See Section 3.3 of the Fire Hazards Analysis Report (FHAR).~~

~~3. If both reactor makeup water pumps are lost as a result of flooding in the Mechanical Auxiliary Building (MAB), a seismic Category I makeup source would be available by connecting temporary hoses to the vent and drain valves located on the low head safety injection pump discharge piping so that refueling water could be delivered to the spent fuel pool.~~

~~4. Temperature based on STPEGS fuel load. SRP fuel load value would be lower.~~

~~5. All fuel storage locations filled with spent fuel, i.e., 1969 fuel assemblies.~~

[Replace with new Table 9.1-1]

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TABLE 9.1-1
SPENT FUEL POOL HEATUP ANALYSIS RESULTS FOR 18-MONTH RELOAD CYCLES

SPENT FUEL POOL MODE	FUEL LOAD (SRP 9.1.3 Criteria)	TYPICAL STP FUEL LOAD	MAXIMUM ALLOWABLE POOL TEMP. (SRP 9.1.3 Criteria)	MAXIMUM POOL TEMPERATURE (1 COOLING TRAIN)	MAXIMUM POOL TEMPERATURE (2 COOLING TRAINS)	MAXIMUM HEAT LOAD (10^6 Btu/hr)[2]
SRP NORMAL MAXIMUM	1 LOAD @ 150 HRS 1 LOAD @ 1 YR 1 LOAD @ 400 DAYS	N/A	140°F	142°F	124°F	23.6
STP NORMAL MAXIMUM	N/A	88 ASSEMBLIES @ 150 HRS 88 ASSEMBLIES @ 1 YR ACTUAL OR 88 ASSEMBLIES @ 2-28 YRS[3]	140°F	155°F	131°F	31.4
STP RAPID REFUELING	N/A	1 FULL-CORE @ 100 HRS 88 ASSEMBLIES @ 1 YR ACTUAL OR 88 ASSEMBLIES @ 2-28 YRS[4]	N/A	202°F	155°F	62.5
SRP ABNORMAL MAXIMUM	1 FULL-CORE @ 150 HRS 1 LOAD @ 36 DAYS 1 LOAD @ 400 DAYS	N/A	NO BOILING [5,6]	189°F	148°F	53.6
STP ABNORMAL MAXIMUM	N/A	1 FULL-CORE @ 150 HRS 88 ASSEMBLIES @ 36 DAYS 88 ASSEMBLIES @ 1 YR ACTUAL OR 88 ASSEMBLIES @ 2-28 YRS[4]	NO BOILING [5,6]	205°F	156°F	63.1

NOTES:

1. SRP = STANDARD REVIEW PLAN (NUREG-0800), SFP = SPENT FUEL POOL, FHB = FUEL HANDLING BUILDING, MAB = MECHANICAL AUXILIARY BUILDING, LHSI = LOW HEAD SAFETY INJECTION
2. MAXIMUM HEAT LOAD BASED ON FUEL EXPOSURE OF 40000 EFFECTIVE FULL POWER HOURS AND SRP ASB 9.2 DECAY HEAT FORMULATIONS.
3. MAINTAINS FULL CORE DISCHARGE CAPABILITY, i.e., 1776 FUEL ASSEMBLIES IN SFP AND 193 LOCATIONS AVAILABLE.
4. ALL FUEL STORAGE LOCATIONS ARE FILLED WITH SPENT FUEL, i.e., 1969 FUEL ASSEMBLIES ARE IN THE SFP.
5. IN THE EVENT OF FIRE OR MODERATE ENERGY LINE BREAK IN THE FHB THAT DISABLES BOTH TRAINS OF SFP COOLING, THE SFP MAY EVENTUALLY BOIL. MAKEUP WATER CAN BE PROVIDED VIA THE REACTOR MAKEUP PUMPS, IN ADDITION, SFP MAKEUP WATER CAN ALSO BE SUPPLIED USING LOCAL HOSE STATIONS IN THE FHB. DETAILS PROVIDED IN SECTION 3.3 OF THE FIRE HAZARDS ANALYSIS REPORT.
6. IF BOTH REACTOR MAKEUP WATER PUMPS ARE UNAVAILABLE BECAUSE OF FLOODING IN THE MAB, THE SEISMIC CATEGORY- LHSI PUMP WOULD BE AVAILABLE FOR REFILLING THE SFP BY CONNECTING TEMPORARY HOSES TO THE VENT AND DRAIN VALVES LOCATED ON LHSI PUMP DISCHARGE PIPING SO THAT REFUELING WATER COULD BE DELIVERED TO THE SFP.

TABLE 9.1-2

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM
COMPONENT DESIGN PARAMETERS

Spent Fuel Pool Cooling Pump	
Number	2
Design pressure, psig	150
Design temperature, °F	200
Design flow, gal/min	2,500
Design head, ft	200
Material	Stainless steel
Spent Fuel Pool Skimmer Pump	
Number	1
Design pressure, psig	150
Design temperature, °F	200
Design flow, gal/min	100
Design head, ft	50
Material	Stainless steel
Refueling Water Purification Pump	
Number	1
Design Pressure, psig	150
Design temperature, °F	200
Design flow, gal/min	200
Design head, ft	200
Material	Stainless steel

TABLE 9.1-2 (Continued)

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM
COMPONENT DESIGN PARAMETERS

Spent Fuel Pool Heat Exchanger		
Number		2
Design heat transfer, Btu/hr		9.1×10^6
	Shell	Tube
Design pressure, psig	150	150
Design temperature, °F	200	200
Design flow, lb/hr	1.5×10^6	1.4×10^6
Inlet temperature, °F	105	120
Outlet temperature, °F	111.1	113.6
Fluid circulated	Component cooling water	Spent fuel cooling water
Material	Carbon steel	Stainless steel
Spent Fuel Pool Demineralizer		
Number		2
Design pressure, psig		300
Design temperature, °F		250
Design flow, gal/min		250
Resin volume, ft^3		75
Material		Austenitic stainless steel

TABLE 9.1-2 (Continued)

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM
COMPONENT DESIGN PARAMETERS

Spent Fuel Pool Filter	
Number	2
Design pressure, psig	300
Design temperature, °F	250
Design flow, gal/min	250
Filtration requirement	98% retention of particles above 5 microns
Material, vessel	Austenitic stainless steel
Spent Fuel Pool Skimmer Filter	
Number	1
Design pressure, psig	300
Design temperature, °F	250
Design flow, gal/min	250
Filtration requirement	98% retention of particles above 5 microns
Material, vessel	Austenitic stainless steel
Spent Fuel Pool Strainer	
Number	2
Design pressure, psig	Not applicable
Design temperature, °F	200
Design flow, gal/min	5,000
Material	Austenitic stainless steel

TABLE 9.1-2 (Continued)

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM
COMPONENT DESIGN PARMETERS

Spent Fuel Pool Skimmer/Strainer Assembly	
Number	2
Design pressure, psig	50
Design temperature, °F	200
Design flow, gal/min	50
Material	Austenitic stainless steel
In-Containment Storage Area Strainer	
Number	1
Design pressure, psig	Not applicable
Design temperature, °F	200
Design flow, gal/min	2,500
Material	Austenitic stainless steel
Reactor Cavity Filtration System	
Number	1
Design Pressure, psig	150
Design temperature, °F	200
Design flow, gal/min	250
Design head of pump, ft	48
Filtration requirement of filters	10 micron
Material	Austenitic stainless steel

TABLE 9.1-3

FHB 15/2-TON CRANE - COMPLIANCE WITH REGULATORY GUIDE 1.104

Regulatory Position	STPEGS Complies with Rev. 0	STPEGS Meets the Intent of Rev. 0 and Complies with a Proposed Revision Dated January 1978	STPEGS Takes Exception to Rev. 0	Not Applicable by Either Rev. 0 or the Proposed Revision Dated January 1978
C.1.a				X
1.b(1)	X			
(2)				X
(3)				X
(4)				X
1.c	X			
1.d	X			
1.e	X			
1.f		X		
C.2.a	X			
2.b	X			
2.c	X			
2.d	X			
C.3.a	X			
3.b	X			
3.c	X			
3.d	X			
3.e	X			
3.f	X			
3.g		X		
3.h		X		
3.i		X		
3.j		X		
3.k	X			
3.l		X		
3.m		X		
3.n	X			
3.o			Note 1	
3.p		X		
3.q		X		
3.r	X			
3.s			Note 2	
3.t				X
3.u	X			

1. Controlled plugging measures shall be provided so that if the operator reverses a drive while it is in motion, the torque during reverse shall be automatically controlled to a predetermined torque limit during deceleration.
2. The crane is designed to CMAA standards (i.e, a 5:1 minimum factor of safety for each component) for 15-ton lifts.

TABLE 9.1-3 (Continued)

FHB 15/2-TON CRANE - COMPLIANCE WITH REGULATORY GUIDE 1.104

Regulatory Position	STPEGS Complies with Rev. 0	STPEGS Meets the Intent of Rev. 0 and Complies with a Proposed Revision Dated January 1978	STPEGS Takes Exception to Rev. 0	Not Applicable by Either Rev. 0 or the Proposed Revision Dated January 1978
C.4.a	X			
4.b		X		
4.c	X			
4.d				X
C.5.a	X			
5.b	X			

TABLE 9.1-4

NSSS VENDOR RECOMMENDED SPECIFICATIONS AND GUIDELINES FOR SPENT FUEL POOL WATER PURITY

Specification Parameters	
Boric Acid, ppm B	≥2800
Chloride, ppb	≤150
Fluoride, ppb	≤150
Guideline Parameters	
pH @ 77°F	4.0 - 4.7
Aluminum, ppb	≤500
Calcium + Magnesium, ppb	≤500
Magnesium	≤250

TABLE 9.1-5

SPENT FUEL POOL COOLING AND CLEAN-UP SYSTEM
FAILURE MODES AND EFFECTS ANALYSIS*

Description of Component	Safety Function	Plant Operating Mode**	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
SFP Cooling Pumps (Typical)	Circulate the water to the spent fuel pool	1 - 6	One pump fails to provide adequate flow	Status monitoring Temperature indication of SFP water	None - A redundant cooling train is available which will provide adequate cooling of the SFP	
Class 1E AC Power (Typical)	Provide 1E power to the pumps	1 - 6	Loss one train of power to its associated pump	Bus under-voltage alarms ESF status monitoring of pumps Pump status lights	None - A redundant power train exists to power the redundant pump	
Channell III DC Power (Train B)	Provide DC control power	1 - 6	Loss of DC power	ESF monitoring on UPS failure, DC trouble alarm ESF monitoring for pump (not running, no control power)	None - Redundant trains provide system capability	
Channel IV DC Power (Train C)	Provide DC control power	1 - 6	Loss of DC power	ESF monitoring on UPS failure DC trouble alarm ESF monitoring for pump (not running, no control power)	None - Redundant trains provide system capability	

[Add Insert #9]

* Single failure only applies to normal modes of SFPCCS operation.
** Plant Modes
1. Power Operation 3. Hot Standby 5. Cold Shutdown
2. Startup 4. Hot Shutdown 6. Refueling

UFSAR - INSERT #9

Description of Component	Safety Function	Plant Operating Mode**	Failure Mode(s)	Method of Failure Detection	Failure Effect on System Safety Function Capability	General Remarks
MOVs 0447 & 0032 (Normally Open)	Circulate CCW water to SFP Cooling Heat Exchangers A & B. **Note:** The FMEA of MOVs 0447 & 0032 is applicable to the SFPCCS. This does not address the ESF function of the valves.	1 - 6 (Full Power to Refueling with full-core offload)	MOV fails closed. (Fails to Reopen following an ESF Signal.)	Loss of CCW Flow Alarm at Main Control Board (non-Class 1E). High SFP Temperature Alarm at Main Control Board (non-Class 1E).	NONE. The valve can be opened either **manually** or **locally** by an operator.	Frequency of occurrence is minimal since the valves are normally open and close only on an ESF signal. After receiving an ESF signal, Emergency Operating Procedure instructs the operator to open the MOVs within an allotted time frame. Failure can be prevented by checking to ascertain the valves reopen after closing due to an ESF signal. If the valve fails to reopen manually after receiving an ESF signal, the valve can be reopened locally. The consequences of loss of SFP cooling are mitigated by administratively providing makeup water through: (1) Reactor makeup water system, (2) Water from RWST (3) Demineralized water, or (4) Fire water.

Table 9.1-6

SPENT FUEL POOL BOILING

Parameters		
Core Thermal power, MWt	3,800	
Decay assumed for the last full core offload	120 hours	3
Hours of reactor exposure per assembly	40,000 hours	
Decay heat generator methodology	SRP 9.1.3	
Volume of water in spent fuel pool, ft^3 (water level 3 ft below normal water level with 10% of the pool volume assumed to be stainless steel)	47,105	3
Dispersion Factors (χ/Q)	Table 15.B-1	

Dose Consequences		
Exclusion Zone Boundary (0-2 hours) Thyroid, rems	1.3×10^{-3}	3
Low Population Zone (0-30 days) Thyroid, rems	7.1	

9.2.2 Component Cooling Water System

9.2.2.1 Design Bases. The CCWS meets the requirements of 10CFR50, Appendix A, General Design Criteria (GDC) 1, 2, 3, 4, 5, 44, 45, 46, 54, 56 and 57. STPEGS Units 1 and 2 have separate but identical CCWSs.

The CCWS is designed to:

1. Provide cooling water to various nuclear plant components during all modes of plant operation. This includes plant equipment required for safe shutdown and ESF equipment required after a postulated DBA.

2. Provide an intermediate fluid barrier between potentially radioactive systems and the ECWS to reduce the possibility of leakage of radioactive contamination to the outside environment.

3. Perform its cooling function following a DBA with offsite or standby power sources, automatically and without operator action, assuming a single active or passive failure.

4. Provide cooling water at 60°F to 105°F temperature during normal operation. The maximum temperature during DBA is 120.5°F (refer to Table 9.2.5-5 for temperature for the individual scenarios).

5. Conform to seismic Category I requirements and safety classifications as indicated on Figures 9.2.2-1 through 9.2.2-5 and in Table 3.2.A-1.

6. Permit periodic inspection of important components and periodic and functional testing to assure the integrity and operability of the system. See Sections 3.9.6 and 6.6.

In addition, the CCWS is protected from the effects of tornado loadings, missiles, flooding, pipe whip, and jet forces from pipe breaks. See Sections 3.3.2, 3.4.1, 3.5, and 3.6.

9.2.2.2 System Description.

9.2.2.2.1 Description: The CCWS consists of three separate redundant trains, each with a pump, HX, associated piping, and valves, that service two RCFCs, Residual Heat Removal (RHR) HX, and RHR pump, as shown on Figures 9.2.2-1 through 9.2.2-3. The three trains are connected to a common header which services other equipment as shown in Figures 9.2.2-4 and 9.2.2-5. In addition, a compartmentalized surge tank is used to accommodate the water thermal expansion and contraction, and a chemical addition tank is used to balance the water chemistry (Table 9.2.2-2) of the system.

A CCW HX bypass line is provided to maintain 60°F minimum CCWS temperature. This line is only used when the ECW temperature is very low.

For heat removal following a DBA, all three CCWS trains will operate if available, but two trains are capable of performing the heat removal function. Except for the seal water HX, reactor coolant pump (RCP) lube oil coolers and thermal barrier, RCP motor air coolers, RHR pump seal coolers, centrifugal charging pump (CCP) supplementary coolers, CCP lube oil coolers, and positive displacement pump supplementary cooler, the remaining equipment is isolated by valves which close on an SI signal. Flow to the RCP lube oil coolers, thermal barrier, and motor air coolers is automatically isolated upon reaching the Containment pressure HI-3 setpoint. An SI signal opens the pneumatic valve (closed during normal operation) to provide cooling water to each RHR HX. Also, an SI signal shifts the cooling water supply to the RCFCs from the chilled water system to the CCWS by closing the chilled water and opening the CCWS motor-operated supply and return valves.

Cooling water to the spent fuel pool (SFP) HXs is manually restored within ~~eight hours~~ the time frame permitted by plant Emergency Operating Procedures following the SI-induced isolation.

Upon a LOOP signal, the RCB chilled water supply and return valves are closed automatically. The CCWS motor-operated valves (MOVs-0057, 0069, 0148, 0136, 0210, 0197) to and from the RCFC remain closed. The CCWS motor-operated valves (MOVs-0235, 0236, 0393, 0297) to the letdown and excess letdown heat exchangers remain open. Operator action from the main control room is required to restore flow to the RCFCs by closing MOVs-0235, 0236, 0393, 0297 and opening MOVs-0057, 0069, 0148, 0136, 0210, 0197 within 30 minutes after LOOP.

The following components are cooled by the CCWS:

1. ESF Loads

   a. RHR HXs

   b. RCFCs

2. Non-ESF Loads

   a. RCPs - lube oil coolers and thermal barrier

   b. RCP motor air coolers

   c. RHR pumps - seal coolers

   d. Centrifugal charging pumps (CCPs) - lube oil coolers

   e. Excess letdown HX

   f. Reactor coolant drain tank (RCDT) HX

   g. Seal water HX

   h. Boron Recycle System (BRS) recycle evaporator package

   i. Boron Thermal Regeneration System (BTRS) chiller unit

   j. Post-accident sampling system (PASS) sample coolers, primary sampling coolers, boric acid sample coolers, and radiation monitor sample coolers

   k. Liquid Waste Processing System (LWPS) evaporator package

   l. SFP HXs

   m. Letdown HX

   n. CCP supplementary coolers

   o. Positive displacement pump supplementary cooler.

The maximum heat loads and required flow rates for each component during the different modes of operation are listed in Table 9.2.2-4.

Each CCW HX is designed to meet the normal operational heat loads. The design maximum CCW outlet temperature from the CCW HX is 105°F. During normal operation, one CCW pump, one compartment of the CCW surge tank, one ECWS pump, and one CCW HX are in use to provide a source of cooling water. The system has sufficient capacity to meet the required heat removal rates for the remaining operating conditions such as startup, shutdown, and recirculation.

The CCWS is required to remove residual and sensible heat from the Reactor Coolant System (RCS) through the RHR HXs from approximately 4 hours after normal shutdown (when the RHR HXs are placed in service) until the reactor coolant temperature reaches 150°F. Operation of the system is then continued in order to maintain cold shutdown. The CCW flow from each RHR HX is indicated on the main control board and displayed through the QDPS. The position of each CCW block valve in the outlet line of each RHR HX is monitored by open/close indicating lights on the main control board and by the ESF Status Monitoring System. Since the RHR HX does not need to function until after recirculation switchover is completed, following a design basis LOCA, the operator would have sufficient time to open the CCW block valve by using the manual switch in the control room, if the SI signal failed to actuate the valve. If this attempt to open the block valve fails, the corresponding SI train should be shut down and the core cooling requirement will be provided from the two remaining SI trains. The CCW trains should also be shut down because of flow considerations (minimum pump flow). The normal cooldown heat load is removed by all three CCW pumps and HXs. If one of the three trains is inoperative, the cooldown function is still effective, although the cooldown period is extended beyond the three train cooldown time of 12 hours. However, this does not affect the safe shutdown and cooldown of the reactor.

Pumps, safety-related valves, and instrumentation required to operate each CCW train are provided with power from their respective standby power sources if normal and offsite power fail. Two of the three cooling trains are available within 60 seconds for a safe shutdown and cooldown of the reactor in case of a LOOP and a simultaneous failure of one standby power source.

Each CCW pump is connected to a separate ESF bus which is powered from either offsite or standby power sources. The pump control logic provides for automatic starting of the standby pump(s) in the event of low system pressure or low ECW pressure during normal operation.

The CCW surge tank is partitioned into three equal volumes by internal baffles. Each compartment is connected to the inlet piping of one of the CCW pumps. The surge tank is sized to accommodate the water thermal expansion and contraction and the inleakage or outleakage in the system until the leaking component is isolated. The internal baffles are designed to provide separation between redundant CCW trains, so excess leakage from a pipe break in one train does not affect the operability of the other trains. The surge tank is located at the highest point in the system to ensure the required NPSH for proper operation of the CCW pumps and to provide a path for collecting entrapped gases. A radiation monitor activates an alarm if the CCW radioactivity increases beyond a preset level (see Section 11.5) due to inleakage from systems being cooled. Makeup to the CCWS is added automatically from the Demineralized Water System (DWS). Low surge tank level

initiates makeup; high surge tank level terminates makeup. As a backup by manual valve alignment, makeup water can be obtained from the Reactor Makeup Water System (RMWS) which is a safety-related and seismic Category I makeup source. A high-high surge tank level alarms in the control room. An alarm is also indicated in the control room if the level in the surge tank continues to decrease due to inadequate makeup. Further decrease in the surge tank level closes the valves on the lines to some non-ESF equipment (BTRS chiller unit, LWPS evaporator package, sample coolers, radiation monitor sample coolers, letdown HX, excess letdown HX, BRS evaporator package, and RCDT HX).

If the level in the surge tank continues to decrease, each CCW train is automatically isolated from the other trains by closing the motor-operated supply and return valves located upstream and downstream of the common supply and the return headers, respectively. Simultaneously, the pneumatic cross-connect valves (FV-4656 and FV-4657) in the CCP and positive displacement pump coolers supply and return headers (CCW train A and CCW train B from CCW train C) and motor-operated valves MOV-0768 and MOV-0772 are closed to prevent loss of more than one CCW train at a time. As soon as the CCW common header is isolated, the low pressure switch located in the common header will automatically start the standby CCW pump(s).

CCW supply to the RCPs is terminated as a result of the isolation of the CCW common header. However, the seal water to the RCPs from the CCPs is maintained since CCW supply to the CCP lube oil coolers is not isolated. Should the water level fall below the top of the surge tank dividing baffles, an alarm is indicated in the control room. Low-low surge tank compartment level initiates another alarm in the control room. The operator may trip the pump to respond to these alarms.

Self-actuated, spring-loaded relief valves are provided for lines and components which could be pressurized beyond their design pressure due to improper operation or failure of the physical barrier separating the CCW from other high-pressure systems.

The system, except for NNS portions, is designed to seismic Category I requirements and is located inside seismic Category I structures. A failure in the NNS portion would not affect the operation of the remainder of the system since low (makeup inadequate) surge tank level isolates the NNS section. The system equipment and piping are classified as SC 2 and 3 and NNS, as shown on Figures 9.2.2-1 through 9.2.2-5. The codes and standards to which the system is designed are listed in Table 3.2.A-1.

Table 9.2.2-2 specifies the water chemistry of the system. The system is initially filled with water from the demineralized water storage tank (DWST) through the surge tank. The corrosion inhibitor is added during the system filling via the surge tank manhole. During normal operation the corrosion inhibitor concentration is maintained via the chemical addition tank utilizing the CCW pump discharge pressure for injection to the surge makeup header.

9.2.2.2.2 <u>Components</u>:

1. Heat Exchangers

The CCW HXs are of the shell and straight tube type. The ECW circulates through the tubes while CCW circulates through the shell. Titanium tubes are

used to provide maximum corrosion resistance to the brackish ECW. The shell is carbon steel. The shell-side (CCW) pressure is maintained higher than the tube-side pressure to prevent leakage of ECW into the system. Design parameters are listed in Table 9.2.2-1.

2. Pumps

The CCW pumps are motor-driven horizontal, double-suction, dual-volute centrifugal pumps with antifriction bearings. Design parameters are listed in Table 9.2.2-1.

3. Surge Tank

The CCW surge tank is constructed of carbon steel. The tank is partitioned to form three separate volumes in the lower portion of the tank. The upper portion of the tank is open to each compartment. Design parameters are listed in Table 9.2.2-1.

4. Chemical Addition Tank

The chemical addition tank is a vertical cylindrical tank constructed of carbon steel. Its design parameters are given in Table 9.2.2-1.

5. Valves

The valves used in the CCWS are constructed of carbon steel. Self-actuated, spring-loaded relief valves are provided for lines and components which might be pressurized beyond their design pressure by improper operation or malfunction. Table 9.2.2-5 lists those relief valves in the CCWS for which Code Case N-242-1 is permitted. This Code Case provides alternate rules which may be used for the acceptance of metallic materials which were not manufactured or supplied in complete conformance with the rules of NCA-3800 (or NA-3700) and which are used in the construction of items for which the applicable code is Winter 1973 Addendum or later. The valves are listed by tag number.

6. Piping

All CCW piping is carbon steel with welded joint end connections, except at components which might be removed for maintenance. In this case, flanged connections are used.

9.2.2.3 Safety Evaluation.

9.2.2.3.1 Availability and Reliability: The normal operation of each CCW pump and HX train is rotated to monitor operational capability and balance operating time.

The operability of safety-related valves and equipment, as required during various modes of operation of the system, is tested by simulated signals corresponding to each mode on a routine basis.

The CCWS performs vital cooling functions during and after a postulated DBA, and therefore is designed to meet the single-failure criterion. Sufficient cooling capacity is provided to fulfill all system requirements under accident conditions, assuming a single failure in the CCWS.

To meet the ESF cooling requirements, the CCW pumps are automatically sequenced on standby power in the event of a LOOP. The electric power supplies to the pumps, valves, instrumentation, and control cabling for each cooling train are physically separated. Power is supplied to each pump from an independent ESF bus.

The CCW pumps, HXs, SC 2 and 3 valves, and piping are designed to seismic Category I requirements and are located in seismic Category I structures. They are protected against internally generated missiles, pipe whip, jet forces, and flooding due to pipe rupture. The CCWS equipment is located in areas shielded from radiation and is accessible for maintenance or inspection during power operation.

A failure mode and effect analysis (FEMA) for the CCWS is given in Table 9.2.2-3.

The CCWS design incorporates automatic isolation of cooling water to the RCPs by the Containment isolation phase B signal. Loss of cooling water to the RCPs due to a spurious isolation signal or the operator closing a single isolation valve is avoided by parallel Containment isolation valves, as shown on Figure 9.2.2-5. The control logic for the valves is designed so that a spurious isolation signal in any one actuation train does not isolate flow; however, isolation is assumed on a penetration basis even assuming the single failure of one of the three actuation trains.

9.2.2.3.2 Leakage Minimization, Detection, and Isolation: To minimize the possibility of leakage, welded construction is used throughout the CCWS where practicable.

Makeup to the system is automatic. The surge tank level is monitored by the main plant computer and the QDPS. Opening of the valve providing tank makeup is alarmed by the computer to give an indication of system leakage. If the normal source of makeup (the DWS) fails or is inadequate, the surge tank level instrumentation provides alarms and actuations as discussed in Section 9.2.2.2.1.

Should a large inleakage into the CCWS develop, the level in the surge tank will rise, and the high-high-level condition will be annunciated. If a leaking fluid is radioactive, then a high radiation level will be alarmed. Refer to Section 11.5 for further information on the process radiation monitor. If the level in the surge tank continues to increase, the CCW will discharge to the CCW sump through the open vent. The vent is designed to accommodate the maximum inleakage due to the rupture of one RCP thermal barrier, which is estimated at 275 gal/min. The increased pressure due to a rupture of the RCP thermal barrier will close two active self-actuated pressure regulated valves located at the CCW outlet from the RCP thermal barrier and isolate the RCS inleakage. In addition, a high flow or high temperature in the return line from the thermal barrier, indicating failure of the RCP thermal barrier pressure-retaining boundary, will close the CCW motor-operated valve downstream of the two pressure regulated valves; an alarm in the control room alerts the operator as well. The portion isolated is designed for RCS pressure and temperature.

The operator, by checking flow and temperature readings against normal values, can locate the affected portion of the system and isolate this portion by

closing the appropriate remotely operated or manual valves. Very small leaks will be detected by periodic inspection of the system piping and valves.

The relief valves on the CCW lines to the various HXs are sized to relieve the volumetric expansion occurring if the shell side of the HX is isolated and high-temperature process fluid flows through the HX tubes.

9.2.2.4 Tests and Inspections. During preoperational testing, the following will be checked:

1. Calibration of all instrumentation

2. Actuation of automatic controls at their proper setpoints

3. Actuation of alarms at the setpoints

4. Operation of power-operated valves

5. Operation of CCW pumps and checking of flow and discharge pressure

6. Checking and adjusting required flow to each component serviced by CCWS

7. System water chemistry

All the above functions are checked periodically during the life of the plant.

Inservice inspections of ASME B&PV Code, Section III, Class 2 and 3 components are performed in accordance with ASME B&PV Code, Section XI.

9.2.2.5 Instrumentation Application. The Engineered Safety Features Actuation System (ESFAS) for the CCWS is discussed in Section 7.3.1. Controls for remote manual operation of each CCW pump and selection of standby CCW pumps are provided in the control room. Remote manual control of pneumatically operated valves under normal operating conditions is provided in the control room. Pneumatically operated valves are not required for control purposes and fail in the safe position on LOOP. Remote manual control of all motor-operated and solenoid-operated valves necessary for post-LOCA cooling, for surge tank makeup, for maintenance of the CCWS trains, and for Containment isolation is provided in the control room. The valves that are power locked out will require that the motor control center (MCC) breaker be closed to enable remote operability. CCW pump operating status and valve position indicating lights are provided in the control room. For valves that are power locked out the indicating lights are not illuminated and control room indication is given via bypass/inoperable for valves not in the power locked out position. Provisions are made for local indication and control in the switchgear room for each CCW pump and selected valves in the CCWS.

The temperature in each CCW main loop is monitored by the QDPS. The temperature at each CCW pump discharge is also monitored by the plant computer. The CCW return temperatures from all equipment are monitored through the plant computer except for the sample coolers, boron recycle evaporator, positive displacement pump supplementary cooler, CCP lube oil coolers, LWPS evaporator package, and CCP supplementary coolers. Temperatures for this equipment, with the exception of the sample coolers, are monitored locally.

Flows through the CCW main loops, RHR HXs, and RCFCs are displayed in the control room on panel indicators and logged by the QDPS while the remaining equipment, with the exception of the letdown HX and sample coolers, has local flow indications or test connections. The temperature difference in conjunction with the flow data from individual components is used to monitor the particular component's performance. Each CCW main loop outlet pressure is indicated in the control room through indicators and the QDPS. The radiation level in the CCWS is available in the control room through the Radiation Monitoring System (RMS) display.

Local pressure gauges are provided on the suction and discharge lines of each CCW pump. Local level gauges are provided for each compartment of the surge tank. Surge tank level indication is displayed in the control room through indicators and the QDPS. High and low levels in the surge tank are alarmed in the control room. Measurements taken downstream of each CCW HX, which indicate system malfunction, are annunciated in the control room. Local flow indication is provided on each RCP lube oil cooler and motor air cooler outlet. Low flow condition at each RCP lube oil cooler and motor air cooler outlet is annunciated in the control room.

9.2.3 Makeup Demineralized Water System

9.2.3.1 Design Bases. The Makeup Demineralized Water System (MDWS) uses ozonated and filtered service water as influent and removes the ionic impurities of the raw water to provide high-purity demineralized water suitable for use in the primary and secondary cycles of the plant. Sodium hypochlorite is used as a backup in the event the ozone generator fails.

The purity of the water produced is suitable for preoperational tests, wet layup, startup, and normal operation of the plant primary and secondary systems. The system is designed for a specific nominal capacity, and transient demands for abnormal operating conditions are allowed for by the buffer effect of the storage volumes of the Demineralized Water Distribution System. For a more complete description of the Demineralized Water Distribution System and supply interfaces with other plant systems, refer to Sections 9.2.6 and 9.2.7 and Figures 9.2.6-1, 9.2.6-2, and 9.2.7-1. The capability of the MDWS to store, handle, and dispense any chemicals used in the demineralization and regeneration process is described in Section 3.6 of the Environmental Report.

Piping, ion exchangers, and associated equipment of the MDWS are constructed of corrosion-resistant material or lined carbon steel to prevent contamination of the water by corrosion products. The equipment is designed in accordance with ASME B&PV Code, Section VIII, Division 1.

The MDWS is designed to produce demineralized water to meet NSSS vendor recommended guidelines as follows:

Specific Conductivity[1]	Less than 0.1 micro mhos/cm at 25°C
Oxygen	Less than 100 ppb
Suspended Solids[2]	Less than 50 ppb
Total Silica, SiO_2	Less than 50 ppb