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March 6, 1984
RBG-17201
File Code No. G9.5
G9.8.6.2

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

River Bend Station - Unit 1
Docket No. 50-458

Enclosed for your review are Gulf States Utilities Company responses to the open items identified in the Draft Safety Evaluation Report by the Auxiliary Systems Branch and responses to the request for additional information identified in part by Staff letters dated August 5, 1981 and December 31, 1981. This letter supplements docketed correspondence from Mr. Booker to Mr. Denton dated December 1, 1983, December 30, 1983 and February 2, 1984. Attachment 1 summarizes the open items and indicates changes to be made in the River Bend Station FSAR. Attachment 2 provides a brief discussion of each open item, the response and reference material for each item. Where indicated, these responses will be provided in a future amendment to the FSAR.

Sincerely,

William J. Reed Jr.
for J. E. Booker
Manager-Engineering,
Nuclear Fuels and Licensing
River Bend Nuclear Group

JEB/~~WJR~~/ERG/JEP
Enclosures

B001
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Attachment 1

ITEM NUMBER	DSER SECTION	SUBJECT	FSAR REVISIONS
6a.	6.7.3 Pg. 6-73	MSIV Leak Rate	Enclosure 1
8.	9.1.2 Pg. 9-10,11	Spent Fuel Racks Criticality Analysis	Enclosure 2 Q410.6
9a.	9.1.3 Pg. 9-16	Spent Fuel Pool Heat Load Assumptions	Enclosure 3
10c.	9.1.4 Pg. 9-27	Light Load Maximum Kinetic Energy	Enclosure 4 Q410.36
11	9.1.5 Pg. 9-30, 33	Heavy Loads NUREG-0612	RBG-17,168 March 1, 1984

Attachment 2

RESPONSES TO DSER OPEN ITEMS

6a. Specify the maximum allowable leakage rate across the main steam isolation valves. (DSER Page 6-73.)

RESPONSE

As indicated in Enclosure 1, the MSIV allowable leakage rate will be included in the Technical Specifications. The proposed Technical Specification will provide a limit on the allowable leakage through all valves served by a division of the Main Steam-Positive Leakage Control System (MS-PLCS). This proposed limit is 309 SCFH total for each of the inboard and outboard divisions. The contribution of this leakage combined with postulated allowable leakage through the Penetration Valve Leakage Control System does not exceed fifty percent (50%) of the containment design pressure in thirty days.

As indicated in NUREG-0800, Standard Review Plan 15.6.5, Appendix D, Section III, "No release of activity from the MSIVLCS is assumed up to the time of system actuation." The basis for this assumption can be found in Part B of Regulatory Guide 1.96.

8. The applicant has not provided a criticality analysis to confirm the criticality limits to be attained in the spent fuel storage facility. (DSER Pages 9-10 &11).

RESPONSE

A summary of the criticality analysis was provided with our letter of February 2, 1984. This summary is reflected in the FSAR revisions provided as Enclosure 2. Any additional required information can be provided within two weeks of its identification to GSU.

9a. The applicant should provide the number of fuel bundles and frequency assumed for "normal" refueling so that the heat load calculations for the spent fuel pool may be verified.

RESPONSE

Assumptions and results of the design heat load calculations for the fuel pool cooling system are provided in revised Section 9.1.3.1.1, Tables 9.1-5 and 9.1-8, and Figure 9.1-8. (See Enclosure 3.) These revisions reflect the heat inputs of an eighteen month refueling cycle which provides a more conservative design basis than any shorter refueling cycle.

Attachment 2 (cont.)

10c. The applicant should provide an analysis to show that the maximum kinetic energy resulting from a fall of any object which weighs less than a fuel bundle and its handling tool, when over spent fuel in either containment or the fuel building storage facilities will not exceed that obtained in the fall of a fuel bundle and its handling tool (fuel handling accident). (DSER Pages 9-24, 26 & 27.)

RESPONSE

Enclosure 4 provides the results of this analysis for the fuel building spent fuel storage building. Similar results are expected for the containment fuel storage pool and will be submitted by March 30, 1984.

11. The Staff will require that the applicant implement the guidelines contained in NUREG-0612 before any operating license can be issued.

RESPONSE

GSU committed in a letter from J.E. Booker to D.G. Eisenhut dated August 9, 1983 to provide an evaluation of River Bend Station to the guidelines contained in NUREG-0612 in two phases. The first phase of this evaluation was recently provided in our letter from J.E. Booker to J.R. Denton dated March 2, 1984. The second phase of this evaluation will be submitted by October 1, 1984.

period following the postulated LOCA and prior to MS-PLCS initiation, leakage is not a significant contributor to the 2-hr site boundary dose, since conservative allowances for transport delay effects indicate that actual transport times are well in excess of the 20-min sealing system start time.

6.7.3.6 Failure Mode and Effects Analysis

The consequences of component malfunctions are shown in Table 6.7-1. System level qualitative Type A analysis is provided in Appendix 15A.

6.7.3.7 Influence on Other Safety Features

The use of MS-PLCS as a positive leakage barrier results in in-leakage and gradual pressure buildup within the containment. As previously stated, the total allowable MSIV in-leakage rate does not have radiological consequences. ~~neither does it allow containment pressurization to exceed design values.~~ System design includes the use of instruments to continuously monitor in-leakage rate and trip the system to isolate whenever recorded leak rate is in excess of the allowable level.

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Instrumentation necessary for control and status indication are adequately provided to assist the operator to initiate appropriate action, if necessary.

6.7.4 Instrumentation Requirements

The instrumentation necessary for control and status indication of the MS-PLCS are designed and qualified in accordance with applicable IEEE Standards to function under Seismic Category I and LOCA environmental loading conditions. The control circuits are designed to satisfy the mechanical and electrical separation criteria (see Section 7.3.1.1.3 for a control and instrumentation description).

6.7.5 Inspection and Testing

Preoperational tests for the MS-PLCS are discussed in Chapter 14. During plant operation, valves, piping, instrumentation, electrical circuits, and other components outside the steam tunnel can be inspected visually at any time. Complete system functional testing is performed only during extended reactor shutdown or plant refueling. This precludes inadvertent steam discharge.

<u>INSERT</u> for Page 6.7-8

The total allowable air inleakage rate from the MSIV's and from valves served by the penetration valve leakage control system (PVLCS, see Section 9.3.6) is limited such that containment pressurization does not exceed 50 percent of the design value in a 30-day period due to these sources.

Since the MS-PLCS establishes a containment leakage barrier, direct MSIV out-leakage to the environment is eliminated. Therefore, the only concern regarding MSIV leak rate is possible containment pressurization when the MS-PLCS is in operation. The specific MSIV allowable leak rate ~~will be~~ is provided in ~~a future amendment.~~ the Technical Specifications and is established such that the total air inleakage from the MSIV's and from valves served by the penetration valve leakage control system (PVLCS- see Section 9.3.6) does not contribute greater than fifty percent of the containment design pressure in a thirty day period.

If the SGTS filter trains are not treating the annulus atmosphere or the exhaust air of the shielded compartments in the auxiliary building, the containment and drywell purge can be manually diverted through both SGTS filter trains. By utilizing both SGTS filter trains, a maximum of 25,000 cfm of containment/drywell purge air can be processed by the filter trains (Section 9.4.6).

The SGTS is designed to maintain a negative pressure of at least 0.50 in W.G. in the annulus during post-LOCA operation. With the annulus at a negative pressure, any potential leakage is directed inward (away from the shield building). Therefore, if a primary containment DBA occurs, airborne radioactivity which exfiltrates the steel primary
7 containment is collected and passed through a filter train of the SGTS before being released.

Potential paths exist for bypass of the annulus mixing system and the SGTS. Potential leakage paths and measures taken to mitigate their consequences are discussed in Sections 6.7 and 9.3.6, and Table 9.3-3 summarizes the maximum leakage rates for the PVLCS process line valves. Leakage rates from the MSIVs ~~will be established in a future amendment~~ are provided in the Technical Specifications.

Since the annulus and the auxiliary building are exhausted via the SGTS filter train following a LOCA, the only leakage considered to bypass the SGTS is that which leaks to areas other than those previously mentioned. This would be from systems listed in Table 9.3-3. The PVLCS is designed to minimize this potential leakage in compliance with the guidelines of Regulatory Guide 1.96 as described in Section 9.3.6.

The primary containment and penetration valve leakage are monitored during the periodic tests of the containment and during the tests to measure local leakage, as discussed in Section 9.3.6.

The integral welded design of the guard pipes precludes leakage from the drywell into the containment portion of the main steam tunnel (see Fig. 3.8-4 for the sleeved penetration design). The electrical penetrations in the primary containment can leak only into the annulus and this leakage is treated by the SGTS.

The maximum inleakage rate across the shield building barrier when the annulus is at a pressure of -3 in W.G. is 2,000 cfm. The building is tested under wind loading conditions characteristic of the plant site prior to

TABLE 6.2-39

POSTULATED POST-LOCA LEAKAGE PATHS TO OUTSIDE ATMOSPHERE[1,2]

Name of Line	Bypass Leakage	Location of Interface with Outside Atmosphere	
Phase I			
Floor drain discharge	[7]	Radwaste building	11
Equipment drain discharge	[7]	Radwaste building or turbine building	
Containment and drywell purge supply	(Later)	Outside or turbine building	11
			11
Phase IIa			
Main steam	~~11.5 SCFH per Valve~~[4]	Turbine building	
Phase IIb			
Feedwater	[5,6]	Turbine building	
Turbine plant misc. drains	[5,6]	Turbine building	
Service air supply	[5,6]	Turbine building	
Condensate makeup water	[5,6]	Outside	
Fire protection header	[5,6]	Outside	
Instrument air supply	[5,6]	Turbine building	
Reactor water cleanup system backwash	[5,6]	Radwaste building	
Ventilation chilled water supply	[5,6]	Radwaste building or turbine building	
Ventilation chilled water return	([5],[6])	Radwaste building or turbine building	11

Guard Pipes Terminating in the Main Steam Tunnel

Main steam[3,4]
Feedwater[1]
Turbine plant misc. drain[1]
RCIC and RHR steam supply line[1]

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[1] See Section 6.2.6 for definition of outside atmosphere.
[2] Radiation monitoring is discussed in Section 11.5.
[3] No leakage barriers other than those listed on Table 6.2-40 are assumed to exist in these lines.
[4] These valves and/or systems supplied with a sealing system (MS-PLCS); refer to Section 6.7 for additional description.
[5] The combined leakage rates for these lines plus the leakage rates for any components subject to Type B tests whose leakage may bypass either the SGTS or the fuel building charcoal filtration system does not exceed 5 percent of L_a. The bypass leakage for each of these lines is not specified, only the total leakage of all of these lines.
[6] These valves and/or systems supplied with a sealing system (PVLCS); refer to Section 9.3.6 for additional description. 11
[7] See Section 6.2.3.2.6.

QUESTION 410.6 (9.1.2.3.1.2)

Provide a schedule for submittal of the criticality analysis for the fuel building spent fuel storage or provide the analysis.

RESPONSE

The criticality analysis for the fuel building spent fuel storage racks ~~will be provided by December 1983,~~ is provided in Section 9.1.2.3.1.2. 7

3. Fuel stored in control rod guide tube racks (Fig. 9.1-5)

4. Pool water temperature increases to 212°F

5. Two bundles placed side by side while separated from the storage rack area by 12 in of water (Fig. 9.1-6)

6. Three-bundle linear array separated from the storage rack area by 12 in of water (Fig. 9.1-6)

7. Three-bundle tee array separated from the storage rack area by 12 in of water (Fig. 9.1-6)

8. Normal storage array of ruptured fuel

9. Abnormal condition of pool being drained and ruptured fuel containers being flooded

10. Moving fuel bundle between work rack and storage area

11. Moving fuel bundle in aisle between storage racks

12. Grapple drop displacing two fuel bundles (Fig. 9.1-6)

13. Four-bundle square array separated from the storage rack area by 12 in of water (Fig. 9.1-6).

Concerning safety implications related to sharing, no limitation is placed on the size of the spent fuel storage array from a criticality standpoint, since all calculations are performed on an infinite basis.

9.1.2.3.1.2 Fuel Building Fuel Storage

The fuel building fuel pool storage design incorporates poison-type, high-density spent fuel storage racks. ~~Details of the criticality analysis will be provided in a future amendment.~~ INSERT

9.1.2.3.2 Structural Design

7 9.1.2.3.2.1 Containment Fuel Storage (Refer to Figs. 9.1-2 and 9.1-2a)

1. The spent fuel pool in the containment building contains 20 sets of racks which may contain up to

INSERT for Page 9.1-10

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poison between assemblies. Structural design of the racks is discussed in Section 9.1.2.3.2.2.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (K_{eff}) of the fuel assembly array is less than 0.95 as recommended in ANSI N210-1976[4] and in an NRC staff position letter.[1]

In meeting this design basis, some of the conditions assumed are: General Electric 8 x 8 BWR/6 fuel with an enrichment of 3.80 w/o U-235 are stored, the pool water has a density of 1.0 gm/cm^3, the storage arrray is infinite in lateral and axial extent which is more reactive than the actual finite array, mechanical and method biases and uncertainties are included, the minimum poison loading is used, and no credit is taken for any burnable poison in the fuel assemblies.

The design method which determines the criticality safety of fuel assemblies in the spent fuel storage rack uses the AMPX system [5] of codes for cross-section generation and the KENO IV Code [6] for reactivity determination. A set of 27 critical experiments [7,8,9] has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and variability which are then included in reactivity analysis of the rack.

The result of the above considerations is that the nuclear design of the River Bend Station racks meet the requirements of its specification as well as U.S. Nuclear Regulatory Commission guidelines [1] and criteria.[10]

11. The fuel storage racks are designed to handle irradiated fuel assemblies. The expected radiation levels are well below the design levels.

12. The containment spent fuel storage racks have the capability of also storing 9 control rod guide tubes and 9 defective fuel containers. These special castings prevent fuel from exceeding k_{eff} of 0.95 in the event that they are positioned in these positions.

9.1.2.3.2.2 Fuel Building Fuel Storage ~~(Fig. 9.1-3)~~

1. The spent fuel pool contains three ~~four~~ sizes of racks: 12x13, ~~11x13,~~ 13x13, and a modified 12x13 containing nine defective fuel cells. A maximum of
11 3,172 fuel assemblies may be stored along with the nine defective fuel cells. See Fig. 9.1-3.

2. Each rack utilizes individually fabricated cells. All cells are assembled to each other and welded to a grid base to form an integral structure.

3. All structural components of racks are made from type 304 stainless steel. The optimum storage density is provided by the incorporation of neutron-absorbing material between adjacent cells. The material choice is based on a consideration of the susceptibility of various metal combinations to electrochemical reactions.

11 4. All racks are designed to allow for remote underwater installation and/or removal. Lifting attachments are
11 provided in the lower structure of each rack to facilitate moving and handling.

5. The center-to-center spacing for the fuel assemblies within a rack is 6.28 in and 8.5 in between cell centers
11 of adjacent racks. Fuel assembly placement between adjacent storage cells or between racks is not possible.

6. The storage rack structure is designed to withstand the impact resulting from a falling object possessing 3,800 ft-lb of kinetic energy, which represents the maximum credible fuel drop accident. The structural

References - 9.1

1. NRC letter from Mr. Brian K. Grimes, Assistant Director for Engineering and Projects, Division of Operating Reactors addressed to all Power Reactor Licensees, dated April 14, 1978.

2. Martin, C. L., Lattice Physics Methods, General Electric Company, NEDO-20913, June 1975.

3. American Welding Society Publication C1.1-1966, Recommended Practices for Resistance Welding.

4. American Nuclear Society, American National Standard, Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations, ANS-57.2, ANSI N210-1976, April 12, 1976.

5. Greene, N. M., et al, AMPX: A Modular Code System for Generating Coupled Multigroup Neutron-Gamma Libraries from ENDF/B, ORNL/TM-3706, March, 1976.

6. Petrie, L. M. and Cross, N. F., KENO IV - An Improved Monte Carlo Criticality Program, ORNL-4938, November, 1975.

7. Bierman, S. R., et al, Critical Separation Between Subcritical Clusters of 2.35 wt % ^{235}U Enriched UO_2 Rods in Water with Fixed Neutron Poisons Battelle Pacific Northwest Laboratories PNL-2614, March, 1978.

9. Thomas, J. T., Critical Three-Dimensional Arrays of U (93.2) - Metal Cylinders, Nuclear Science and Engineering, Volume 52, pages 350-359 (1973).

10. Title 10, Code of Federal Regulations, Part 50, Appendix B, GDC 62, Prevention of Criticality in Fuel Storage and Handling.

severe radiation environments. These coupons are designed to simulate the material and support conditions of the poison material in the racks. Periodically, a coupon is removed and evaluated. B_{10} concentration and mechanical properties will be evaluated and compared against acceptable ranges established by the criticality calculation and poison material qualification reports.

9.1.3 Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system consists of two separate subsystems: the fuel pool cooling subsystem and the fuel pool purification subsystem.

The fuel pool cooling subsystem provides heat removal for spent fuel and maintains the spent fuel covered with water during all storage conditions. The purification subsystem maintains required water purity under normal conditions and is not required under accident conditions.

9.1.3.1 Design Bases

9.1.3.1.1 Fuel Pool Cooling Subsystem

The fuel pool cooling subsystem is designed to remove decay heat produced by stored spent fuel assemblies during all anticipated plant operation, refueling, and accident conditions. The design criteria for the fuel pool cooling subsystem are as follows:

1. The fuel pool cooling subsystem and the connecting piping for the backup source of fuel pool makeup are classified as Safety Class 3 and Seismic Category I and are designed in accordance with Regulatory Guides 1.13, 1.26, and 1.29 and General Design Criteria 2, 4, 5, 44, 45, 46, 61, and 63.

2. The spent fuel storage capacity is 3,172 fuel
11 assemblies, approximately 5.08 cores, of which
4.08 cores are designated for routine spent fuel
storage in the fuel building. This provides fuel
pool cooling capacity for an offload of a full
reactor core in addition to normal storage. Under
normal operating conditions, spent fuel assemblies
11 in the pool do not exceed 4.08 cores.

3. The fuel pool cooling subsystem is designed to
maintain the temperature of the water in the fuel
building fuel storage pool at or below ~~129°F~~ during
3 normal operation, with one cooler and one cooling
pump in service.
139 F

The heat load for normal operation was calculated based on the following: 3

a. Storage of 408 percent of an equilibrium core is in the pool. 11

This storage is comprised of 200 fuel assemblies removed from the first core after the first 18 months of operation, and 248 assemblies removed each 18-month refueling 11
INSERT cycle thereafter. Residual decay energy release rates are calculated in accordance with Branch Technical Position ASB 9-2, Revision 1.

b. A batch of equilibrium core from the most recent refueling outage is assumed to have been in the pool 150 hr after reactor shutdown, with the batches from previous refueling outages in the pool.

4. The fuel pool cooling subsystem is designed to remove the decay heat from the combined fuel storage pools' capacity at a rate sufficient to maintain the temperature of the water at or below 156°F, when 508 percent of an equilibrium core is 11
stored in the pools (Fig. 9.1-8). The calculation of the water temperature for this abnormal load with the storage of 508 percent of an equilibrium 11
core was based on the following:

a. A full core removal event is assumed to be required at the time when batches from each of the previous refueling outages, totaling 4.08 11
cores, are in the pool.

b. The last refueling outage required 30 days to complete. At the end of the 30-day period, the reactor was started up and brought to full power, and was forced to be shut down immediately and the full core was removed,
14 which required ~~10~~ days. Two hundred spent fuel assemblies from the full core were stored in the containment fuel storage pool.

5. The fuel pool cooling subsystem is designed to maintain the temperature of the containment fuel storage pool water at or below 127°F, when 30 percent of an equilibrium core is stored in the fuel storage area of the containment pool during refueling operations.

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(See Table 9.1-8). The heat load imposed on the spent fuel storage pool by assuming an 18-month refueling cycle envelopes (i.e., is higher than) the 12-month refueling cycle case.

service water is also supplied to the heat exchangers for cooling (Section 9.2.7).

Normal pool makeup water from the condensate storage tank is sized for normal evaporation and equipment leakage losses, as well as leakage rates associated with potential damage to the fuel storage pool (Section 9.1.4.2.2.1).

The fuel pool cooling subsystem is designed with complete redundancy during both normal, abnormal, and accident plant conditions. Either one of the two cooling loops can maintain fuel building fuel storage pool temperature at or
139 below ~~129°~~F, with the design decay heat load (criteria discussed in Section 9.1.3.1) at the time of plant startup after the refueling is completed. Fig. 9.1-8 is a graphical representation of an analysis of fuel building fuel pool temperature versus heat load. If an abnormal operating condition requires full core removal (criteria discussed in Section 9.1.3.1), the fuel building fuel storage pool temperature may rise up to 156°F. This temperature is within the design limits of the pool concrete structure.

During a refueling outage, spent fuel may be stored in the containment fuel storage racks. A fuel pool cooling pump and heat exchanger are then aligned to cool the containment fuel pool. The other fuel pool cooling pump and heat exchanger are aligned to cool the fuel building storage pool. In the event of failure of either train during this 3
situation, the operating train is used to cool the fuel building pool and the RHR train in standby is aligned and initiated as necessary to cool the containment fuel pool. Figures 5.4-12 (RHR) and 9.1-23a and b (fuel pool cooling) show this capability.

The spent fuel cask pool is normally isolated from the fuel storage pool with a watertight gate, which closes the opening through which spent fuel passes as it is being transported to the cask area. The bottom of this opening is above the top of the spent fuel storage racks so that if water in the spent fuel cask pool is lost while the gate is open, the fuel storage racks are not uncovered.

The consequence of fuel pool cooling subsystem component failures are presented in the Failure Modes and Effects Analysis (FMEA) report submitted under separate cover.

A radiological evaluation of the fuel pool purification subsystem is presented in Chapter 12.

TABLE 9.1-5

DESIGN PARAMETERS OF FUEL POOL COOLING
AND PURIFICATION SUBSYSTEMS
(18 MONTH REFUELING CYCLE)

<u>Cooling Subsystem</u>		
Fuel Storage Pool (Fuel Building)		
Normal Temp (~~4.25~~ 4.08 cores)		~~129°F~~ 139 F
Containment Storage Pool Temp (0.30 core)		127°F
Combined Pools Maximum Temp (~~5.25~~ 5.08 cores)		156°F
Cooling Water Temperature		
RPCCW		105°F (maximum)
SSW		95°F (maximum)
Pump		
Capacity		2 @ 100%
Type		Horizontal Centrifugal
Design Flow		2,500 gpm
Design Total Head		87.6 ft H_2O
Cooler	<u>Shell Side</u>	<u>Tube Side</u>
Fluid	RPCCW	Fuel Pool Water
Flow	1,000,000 lb/hr	1,250,000 lb/hr
Design Pressure	150 psig	150 psig
Heat Exchanger Normal	~~11.87~~ 16.17 $\times 10^6$ Btu/hr	
Temp, in	105°F	~~119°F~~ 139 F
Temp, out	~~116.5°F~~ 121.2 F	~~129°F~~ 124.6 F
Heat Exchanger Abnormal	24.73 $\times 10^6$ Btu/hr	
Temp, in	105°F	135.5°F
Temp, out	130°F	156°F

KEY: RPCCW - Reactor Plant Component Cooling Water
SSW - Standby Service Water

TABLE 9.1-8

FUEL DISCHARGE - HEAT LOAD FOR NORMAL OPERATION (18 MONTH REFUELING CYCLE)

Discharge No.	No. of Fuel Assemblies	Decay Time (Yrs)
1	200	14.5
2	248	13.5
3	248	12
4	248	10.5
5	248	9
6	248	7.5
7	248	6
8	248	4.5
9	248	3
10	248	1.5
11	248	150 hours

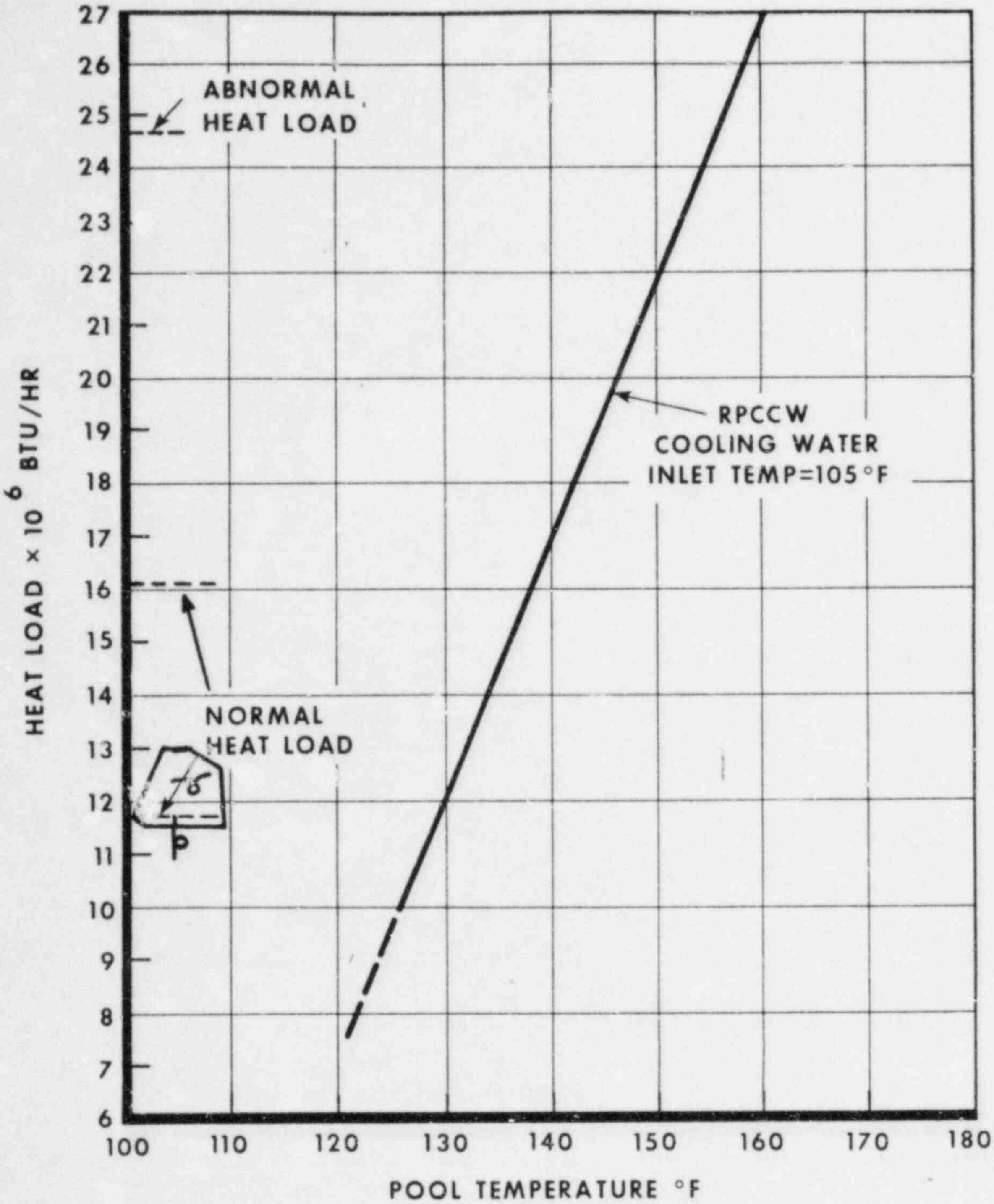

FIGURE 9.1-8

FUEL POOL TEMPERATURE VS HEAT LOAD

RIVER BEND STATION
FINAL SAFETY ANALYSIS REPORT

QUESTION 410.36 (9.1.4)

Discuss how you intend to control the handling of objects over the spent fuels in both the containment and spent fuel buildings so that the maximum kinetic energy of any such object, if dropped from the height at which it is normally handled above a storage rack, will not exceed the kinetic energy of one fuel assembly and its associated handling tool. Include objects of less weight than a spent fuel assembly in your consideration.

RESPONSE

The response to this request is provided in revised Section 9.1.4.3 and new Table 9.1-6. 3

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cable" signal from the lifting cables indicates that the fuel assembly is seated.

In addition to the main hoist on the trolley, there is an auxiliary hoist on the trolley, and another hoist on its own monorail. These three hoists are precluded from operating simultaneously because control power is available to only one of them at a time. The two auxiliary hoists have load cells with interlocks which prevent the hoists from moving anything as heavy as a fuel bundle.

The two auxiliary hoists have electrical interlocks which prevent the lifting of their loads higher than 8 ft under water. Adjustable mechanical jam-stops on the cables back up these interlocks.

Administrative procedures and training are provided to control the handling of objects over the spent fuels in both the containment and spent fuel buildings so that the maximum kinetic energy of the object, if dropped from the height at which it is normally handled above a storage rack, does not exceed the kinetic energy assumed in any previously accepted analysis.

The only items that are transported across the spent fuel pool or the containment fuel storage pool are the fuel channels, the fuel assemblies, or the individual subassemblies or components of these fuel assemblies. These items are handled by the same grapple that handles the entire fuel assembly, and the retraction elevation is limited to ensure that the minimum safe water level is maintained over the fuel assembly during transport. 11

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~~The lighter components of the fuel assembly and the channels, which are much lighter than the fuel assemblies, cannot be raised any higher than a fuel assembly. Thus, the kinetic energy associated with an accidental drop is much less than that of a dropped fuel assembly.~~

In summary, the fuel handling system complies with General Design Criteria 2, 3, 4, 5, 61, 62, and 63, applicable portions of 10CFR50, and Regulatory Guide 1.13. In addition, procedures and training, inspection and maintenance programs will be developed in accordance with Section 5.1 and NUREG-0612. 3

A system-level, qualitative-type failure mode and effects analysis relative to this system is discussed in Section 15A.6.5.

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Components of the fuel assembly and the channels and other objects which are lighter than the fuel assemblies, either cannot be raised any higher than a fuel assembly or have been shown that the kinetic energy associated with an accidental drop is less than that of a dropped fuel assembly as analyzed in Section 15.7.4. Examples of analyzed objects and results of the analyses are shown in Table 9.1-6.

TABLE 9.1-6

Light Loads Over the Spent Fuel Pool
Fuel Building

Item	Distance (ft) Above Pool Surface	Total Above Rack	Dry Weight (lbs) (2)	Kinetic Energy at Impact (top of rack) (ft-lbs)
1. Channel bolt wrench	4	31	15	465
2. Channel handling tool	11.5(1)	38.5(1)	25	963
3. Channel gauging fixture		17.8(1)	210	3738
4. General purpose grapple	3.6(1)	30.6(1)	45	1377
5. Actuating pole	4	31	61	1891
6. Drop Light	4	31	25	775
7. Local area underwater Light	4	31	35	1085
8. Viewing aid		27	13	351
9. Control rod		12(1)	218	2616
10. Handrail, removeable	4	31	122	3782
11. Underwater TV camera	4	31	50	1550
12. Defect. Fuel Contain w/fuel assy		5.4	573	3094
13. Fuel channel		24(1)	99	2376
14. Dummy Fuel assembly		6.1	621	3803 (3)
15. Fuel bundle		6.1	538	3296 (3)
16. Fuel bundle & channel		6.1	621	3803 (3)

Notes:

(1) Distance from CG to top of fuel storage rack and pool surface.
(2) Kinetic energy calculated without credit taken for buoyancy and drag for conservation, except for defective fuel container & fuel bundles.
(3) See FSAR Section 15.7.4 for fuel assembly fuel handling accident Kinetic energy value differs because of rounding-off of factors.