5.3 AUXILIARY EQUIPMENT

5.3.1 Fuel Storage

A. The fuel storage facilities are designed and shall be maintained with a K-effective equivalent to less than or equal to 0.95 including all calculational uncertainties.

B. 1. Loads greater than the weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility, except as noted in 5.3.1.B.2.

2. The shield plug and the associated lifting hardware may be moved over irradiated fuel assemblies that are in a dry shielded canister within the transfer cask in the cask drop protection system.

C. The spent fuel shipping cask shall not be lifted more than six inches above the top plate of the cask drop protection system. Vertical limit switches shall be operable to assure the six inch vertical limit is met when the cask is above the top plate of the cask drop protection system.

D. The temperature of the water in the spent fuel storage pool, measured at or near the surface, shall not exceed 125°F.

E. The maximum amount of spent fuel assemblies stored in the spent fuel storage pool shall be 2645.

BASIS

The specification of a K-effective less than or equal to 0.95 in fuel storage facilities assures an ample margin from criticality. This limit applies to unirradiated fuel in both the dry storage vault and the spent fuel racks as well as irradiated fuel in the spent fuel racks. Criticality analyses were performed on the poison racks to ensure that a K-effective of 0.95 would not be exceeded. The analyses took credit for burnable poisons in the fuel and included manufacturing tolerances and uncertainties as described in Section 9.1 of the FSAR. Calculational uncertainties described in 5.3.1.A are explicitly defined in FSAR Section 9.1.2.3.9. Any fuel stored in the fuel storage facilities shall be bounded by the analyses in these reference documents.

The effects of a dropped fuel bundle onto stored fuel in the spent fuel storage facility has been analyzed. This analysis shows that the fuel bundle drop would not cause doses resulting from ruptured fuel pins that exceed 10 CFR 100 limits (1,2,3) and that dropped waste cans will not damage the pool liner.

Administrative controls over crane movements, which include mechanical rail stops, serve to prevent travel of the crane outside the analyzed load path over the cask drop protection system. A safety factor greater than 10 with respect to ultimate strength, and redundant shield plug lift cables provide adequate margin for the shield plug lift. These features, combined with operator training and required inspections, contribute to the determination that dropping the shield plug onto a loaded dry shielded canister in the spent fuel pool is not a credible event.

The elevation limitation of the spent fuel shipping cask to no more than 6 inches above the top plate of the cask drop protection system prevents loss of the pool integrity resulting from postulated drop accidents. An analysis of the effects of a 100-ton cask drop from 6 inches has been done (4) which showed that the pool structure is capable of sustaining the loads imposed during such a drop. Limit switches on the crane restrict the elevation of the cask to less than or equal to 6 inches when it is above the top plate.

Detailed structural analysis of the spent fuel pool was performed using loads resulting from the dead weight of the structural elements, the building loads, hydrostatic loads from the pool water, the weight of fuel and racks stored in the pool, seismic loads, loads due to thermal gradients in the pool floor and the walls, and dynamic load from the cask drop accident. Thermal gradients result in two loading conditions; normal operating and the accident conditions with the loss of spent fuel pool cooling. For the normal condition, the containment air temperature was assumed to vary between 65°F and 110°F while the pool water temperature varied between 85°F and 125°F. The most severe loading from the normal operating thermal gradient results with containment air temperatures at 65°F and the water temperature at 125°F. Air temperature measurements made during all phases of plant operation in the shutdown heat exchanger room, which is directly beneath part of the spent fuel pool floor slab, show that 65°F is the appropriate minimum air temperature. The spent fuel pool water temperature will alarm control room before the water temperature reaches 120°F.

Results of the structural analysis show that the pool structure is structurally adequate for the loadings associated with the normal operation and the condition resulting from the postulated cask drop accident (5) (6). The floor framing was also found to be capable of withstanding the steady state thermal gradient conditions with the pool water temperature at 150°F without exceeding ACI Code requirements. The walls are also capable of operation at a steady state condition with the pool water temperature at 140°F (5).

Since the cooled fuel pool water returns at the bottom of the pool and the heated water is removed from the surface, the average of the surface temperature and the fuel pool cooling return water is an appropriate estimate of the average bulk temperature; alternately the pool surface temperature could be conservatively used.

References

1. Amendment No. 78 to FDSAR (Section 7)
2. Supplement No. 1 to Amendment No. 78 to the FDSAR (Question 12)
3. Supplement No. 1 to Amendment 78 of the FDSAR (Question 40)
4. Supplement No. 1 to Amendment 68 of the FDSAR
5. Revision No. 1 to Addendum 2 to Supplement No. 1 to Amendment No. 78 of FDSAR (Questions 5 and 10)
6. FDSAR Amendment No. 79
7. Deleted

I. TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) No. 244

GPU Nuclear requests the following replacement pages be inserted into existing Technical Specifications:

Replace existing pages 5.3-1 and 5.3-2 with the attached revised replacement pages 5.3-1 and 5.3-2.

II. REASON FOR CHANGE

The current specification 5.3.1.B requires that "Loads greater than weight of one fuel assembly shall not be moved over stored irradiated fuel in the spent fuel storage facility". This restriction is based upon the structural strength of the fuel racks in which the spent fuel is stored and the damage that would occur if the load were dropped. The process of transferring spent fuel assemblies to the Oyster Creek Independent Spent Fuel Storage Installation (ISFSI) includes placing a dry shielded canister (DSC) and a transfer cask in the cask drop protection system (CDPS). That movement does not handle a heavy load over irradiated fuel. The DSC is then loaded with spent fuel assemblies. Before the DSC and the transfer cask in which it is contained can be removed from the spent fuel pool, the DSC shield plug must be lowered into the CDPS and placed atop the DSC. The current specification prohibits this movement since the shield plug and the lifting yoke weigh more than one fuel assembly and the DSC contains irradiated fuel.

III. SAFETY EVALUATION JUSTIFYING CHANGE

GPU Nuclear has evaluated the process of transferring spent fuel assemblies from the spent fuel pool to the ISFSI. That evaluation considers the safe load paths, the design features of the reactor building crane and the requirements of NUREG 0612.

The CDPS has been designed to mitigate a cask drop into the spent fuel pool. The transfer path for the cask centerline is on a controlled path width of six inches in the north-south and east-west directions. Visual aids are used to control the motion of the cask centerline to the prescribed transfer path. Mechanical rail stops are installed to prevent travel of the crane outside the analyzed load path over the cask drop protection system. Stops are installed for limiting bridge movements in the north-south direction and for limiting trolley movements in the west direction. The movement of the shield plug would be in accordance with these same constraints. The weight of the load, however, would be considerably different. The shield plug weighs approximately 8,000 pounds and the lifting yoke weighs about 6,200 pounds.

A series of modifications have been made to enhance the crane's performance and reliability by improving the instrumentation and controls. These modifications include:

- Various crane monitoring systems have been installed. These include drum over-speed detection, mechanical drive train continuity detection, wire rope spooling monitor, fault display and reset panel and hoist speed indication.

- Phase loss/phase reversal protection has been installed. Phase loss results in substantial loss of drive motor torque and possible load drop.

- A power circuit upper limit switch to directly interrupt power to the hoist motor was installed. This reduces the possibility of two-blocking as a result of failure of existing control circuit limit switches.

- A load cell weight display was installed in the cab to provide an indication for load hang-ups and over-capacity lifts.

- The magnetic drive controllers were replaced. The new variable frequency drive (VFD) controllers provide smooth and precise speed control along with torque limitation, reducing the possibility of a load snatch.

- New controls were installed in the cab that provide spring control to normal function. These controls considered human factors in their design.

The reactor building (RB) crane has a main hoist capacity of 100 tons. The actual safety factors of the main crane for its 100 ton rated load are: cables 6.5:1; main hoist gearing 5.2:1; and main hoist brakes 120% capacity. As a result, when moving the shield plug and the lifting yoke with a combined weight of approximately 7 tons, a safety factor greater than 14 will be provided, based on the RB crane 100 ton rated capacity. For the lifting yoke, a safety factor greater than 26 will be provided, based on the lifting yoke 105 ton rated capacity. The least conservative safety factor is that for the wire rope assemblies. That safety factor is 11:1, based on the ultimate load of 22,800 lbs. Furthermore, the wire rope assemblies are redundant and each of the four has sufficient capacity to support the total weight of the shield plug.

In addition, GPU Nuclear has developed an error free plan for the movement of spent fuel assemblies to the ISFSI. That plan includes a dedicated management team and a dedicated crew who will be trained and on shift. Detailed operating instructions/procedures will be developed and mock-up training and a dry run will be conducted. A special crane inspection will be performed prior to each dry fuel storage campaign. The main hoist coupling, shafts, and hook will be examined by NDE prior to each campaign. Plant procedures for the reactor building crane satisfy the inspection, testing and maintenance criteria of ANSI B30.2.

The design features and modifications to the reactor building crane increase its reliability and enhance its performance. The safety factors of the reactor building crane relative to this load exceed 10 to 1. Personnel training, and crane inspections, testing, and maintenance will be in accordance with ANSI B30.2. Therefore, dropping the DSC shield plug onto a loaded DSC in the spent fuel pool is not considered a credible event.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION

GPU Nuclear has determined that this TSCR poses no significant hazard as defined by 10 CFR 50.92.

1. State the basis for the determination that the proposed activity will or will not increase the probability of occurrence or consequences of an accident.

   The design features and capacity of the reactor building crane provide a significant safety factor. In addition, personnel training and other administrative controls further reduce risk. Thus, the dropping of the DSC shield plug onto a loaded DSC and causing damage to the spent fuel assemblies is not a credible event. Therefore, it does not increase the probability of or consequences of an accident.

2. State the basis for the determination that the activity does or does not create the possibility of an accident or malfunction of a different type than any previously identified in the SAR.

   This activity will not create the possibility of a new or different type of accident than previously evaluated in the SAR because the proposed heavy load handling exception does not create a new credible accident scenario. Dropping the shield plug on a loaded DSC and damaging spent fuel assemblies is not considered a credible event.

3. State the basis for the determination that the margin of safety is not reduced.

   This activity will not involve a significant reduction in the margin of safety because the proposed heavy load handling evolution does not create a credible accident scenario.

V. IMPLEMENTATION

GPUN requests that the amendment authorizing this change be effective upon issuance.