APPENDIX

DAVIS-BESSE NUCLEAR POWER STATION

UPDATED SAFETY ANALYSIS REPORT EXCERPTS

9.1 FUEL STORAGE AND HANDLING

9.1.1 New Fuel Storage

9.1.1.1 Design Bases

The new fuel storage facility is capable of storing a maximum of 80 new fuel assemblies. However, due to optimum moderation ("mist") criticality considerations, rows "C" and "F" must be blocked so that new fuel assemblies cannot be stored in those rows. Therefore, the new fuel storage area can effectively store only 60 new fuel assemblies. The new fuel assemblies are stored vertically in parallel rows with a nominal 21 inch center-to-center fuel assembly spacing in both directions. New fuel assemblies containing up to 5.00 weight percent uranium-235 (wt% U-235) may be stored in the new fuel storage
facility. Although the new fuel storage facility is normally dry, this spacing, 19
combined with the blocking of rows "C" and "F" to new fuel assemblies, is
sufficient to maintain a k_{eff} of less than 0.95 when flooded with unborated 20
water, and to maintain a k_{eff} of less than 0.98 if the storage area is immersed 19
in a hydrogenous "mist" producing optimum moderation. The design loads to be withstood are seismic, tornado, and thermal loads and are discussed in Subsection 9.1.1.3.

The criticality analyses discussed above assumed that fuel assemblies were uniformly loaded with 5.0 wt% U-235 fuel rods. For zone-loaded fuel assemblies (fuel assemblies containing fuel rods with multiple U-235 enrichments), specific analyses are required to demonstrate that the analysis with a uniform 5.0 wt% U-235 enrichment loading remains bounding.

9.1.1.2 Description

The new fuel storage area is located inside the fuel-handling area in the auxiliary building. The location of the new fuel storage is shown in Figures 1.2-5 and 1.2-6.

The storage racks are of individual cells assembled in four 20-rack assembly units which are braced together to form a rack frame as shown in Figure 9.1-2. The rack assemblies are constructed entirely of type 304 stainless steel.

9.1.1.3 Safety Evaluation

The new fuel storage area and racks are Seismic Class I structures which are designed to withstand seismic loadings of 0.40 g in horizontal and 0.11 g in vertical directions acting simultaneously at the floor. Details of seismic, tornado, and thermal loads are discussed in Sections 3.7 and 3.8. Figure 9.1-3 shows the mass model for the new fuel racks.

The spacing of the racks is such as to preclude the possibility of criticality even if flooded with unborated water. 20

The fuel assemblies are removed from the racks without exerting any uplift force on the rack. The fuel assemblies make free contact at the bottom and on the sides. All the projections and corners on the racks are carefully tapered to eliminate any possibility of fuel assembly sticking in the rack. Removal of the new fuel assembly from the new fuel storage racks is done manually. The operator can stop the crane at any time by a pendant control during handling of the new fuel assemblies. The racks can withstand an uplift force of 2700 pounds.

There is no equipment located adjacent to the racks whose failure can damage the racks or the fuel assemblies. The only equipment in close proximity to the new fuel racks is the spent fuel handling bridge which is restrained to Seismic Class I rails by Seismic Class I restraints to prevent its jumping the tracks in the event of earthquake. The winch mechanism is also anchored to the floor by Seismic Class I anchors.

A 4-inch-diameter open floor drain at the bottom of the pit is provided to prevent any possibility of accumulation of water.

The following codes and standards have been followed, as applicable, in the design of the new fuel storage facility:

AISC	American Institute of Steel Construction, Inc. - Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings.
AWS	American Welding Society - Welding in Building Construction.
ASTM	American Society for Testing and Materials specification and standards A 36 Structural Steel A 240 Stainless Steel Plates A 276 Stainless Steel Bars and Shapes.

9.1.2 Spent Fuel Storage

9.1.2.1 Design Bases

The spent fuel pool storage facility is designed to store the irradiated fuel 20
assemblies under water for decay prior to shipment offsite for reprocessing.
The storage pool is sized to store 735 irradiated fuel assemblies which
includes storage for 15 failed fuel containers. The spent fuel storage cells
are installed in parallel rows with a center-to-center spacing of 12-31/32
inches in one direction, and 13-3/16 inches in the other, orthogonal direction.
The spacing and the "flux trap" design, whereby the fuel assemblies are
inserted into neutron absorbing stainless steel cans surrounded by water, is 19
sufficient to maintain a k_{eff} of less than 0.95 for fuel assemblies with
initial enrichments not exceeding 3.56 weight percent uranium-235 (wt% U-235),
assuming the storage racks are flooded with unborated water. However, 20
credit has been taken in the criticality analyses for the effects of fuel 19
depletion (burnup) upon spent fuel rack criticality. Using appropriate
requirements for fuel assembly location (checkerboarding), as defined in the
Technical Specifications, fuel assemblies having an initial enrichment of up to
5.00 wt% U-235 may be stored in the spent fuel storage facility while still
maintaining a k_{eff} of less than 0.95 with the storage racks flooded with
unborated water. Shielding and seismic classification are discussed in 20
Subsection 9.1.2.2.

The criticality analyses to support the checkerboarding scheme discussed above 19
assumed that fuel assemblies contained only fuel rods with a uniform U-235 enrichment. For zone-loaded fuel assemblies (fuel assemblies containing fuel rods with multiple U-235 enrichments), specific analyses are required to demonstrate that the analysis with uniform enrichments within the fuel assemblies remains bounding and to determine which enrichment should be used in determining the categorization of the fuel assembly for storage in the spent fuel storage racks, in accordance with the Technical Specifications.

The design of the spent fuel storage area closely follows the intent of Safety Guide 13.

There is also a Dry Fuel Storage Facility (DFSF) onsite. After a minimum 20
required storage period in the spent fuel pool, the spent fuel assemblies may be transferred to the onsite dry fuel storage facility. The design of the DFSF is described in the Davis-Besse Site Certified Safety Analysis Report (CSAR).

9.1.2.2 Description

After removal from the reactor, the spent fuel is stored under water within the spent fuel storage pool. The storage pool is a reinforced concrete pool lined with 1/4-inch-thick stainless steel. It is located inside the fuel handling area in the auxiliary building as shown in Figures 1.2-5 and 1.2-6. The auxiliary building, as well as the storage pool, is a Seismic Class I structure which is designed to withstand seismic, tornado, and thermal loads as discussed in Sections 3.7 and 3.8. The spent fuel storage racks are also Seismic Class I structures which are designed to withstand seismic loadings. The mass model is shown in Figure 9.1-3a. The fuel handling area is also protected against tornado-generated missiles and other potential missiles.

Struts have been installed across the fuel transfer tube pit to prevent overstressing of the 3-foot wall between the spent fuel pool and the fuel transfer tube pit during a seismic event. The struts do not interfere with fuel handling operations or necessary maintenance.

Adequate shielding is provided for station personnel by the 5-1/2-foot-thick concrete walls and borated water in the pool. The radiation zones around the spent fuel pool are shown in Figures 12.1-2 and 12.1-3.

The spent fuel racks (not including the failed fuel container locations) are arranged in a 16 X 45 array constructed of six 7 X 8 modules and six 8 X 8 modules. The arrangement is shown in Figure 9.1-4. The location of the storage pool within the station complex is shown in Figures 1.2-5 and 1.2-6.

A separate space is provided for loading the dry fuel storage canister or a 20
spent fuel shipping cask. The spent fuel cask pit is independent of and separated from the spent fuel pool by a 3-foot-thick concrete wall. The only communication between the spent fuel pool and the cask pit is through the
36-inch-wide slot opening provided for the transfer of the spent fuel 20
assemblies from storage to the shipping cask or dry fuel storage canister. This opening is provided with a watertight bulkhead which can isolate the spent fuel pool when needed. Following sufficient decay, the spent fuel assemblies can be removed from storage and loaded into the spent fuel shipping
cask or dry fuel storage canister under water for removal from the site. 20
Casks up to 140 tons in weight can be handled by the spent fuel cask crane.

A cask-wash-and-decontamination area is also provided adjacent to the cask pit. In this area, outside surfaces of the cask can be decontaminated before
movement to the onsite Dry Fuel Storage Facility (DFSF) or shipment from the 20
site. The DFSF is described in the Davis-Besse Site Certified Safety Analysis Report (CSAR).

9.1.2.3 Safety Evaluation

The spent fuel pool storage facility is designed for noncriticality by use of 20
adequate spacing, and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans. The spent fuel storage racks are designed to prevent accidental insertion of a fuel assembly in other than the prescribed locations, thereby ensuring a safe geometric array.

All spent fuel assembly transfer operations are normally conducted under a 19
minimum of 9-1/2 feet of borated water above the top of the active fuel to ensure adequate biological shielding. All piping penetrations into the spent fuel pool penetrate at least 9 feet above the top of the fuel assemblies to avoid any possibility of completely draining the pool in case of a pipe rupture. Isolation valves are provided on the pipes penetrating the pool. These valves are located as close to the concrete wall as practicable to minimize the possibility of pipe failure between the isolation valves and the pool.

The spent fuel pool water is cooled by the spent fuel pool cooling system as discussed in Subsection 9.1.3.

The spent fuel cask crane is electrically interlocked to prevent the crane from traveling over the spent fuel pool while any load is hanging on the main

hook. This interlock can be bypassed only with a key. Even upon bypassing this interlock, the main hook stays inoperative; only the auxiliary hook can be used.

The cask pit is separated and isolable from the pool to preclude the possibility of draining the spent fuel pool in case of damage to the cask pit by an accidental drop of a cask in the pit. The base of the cask pit is solid concrete extending down to the foundation. Thus, a cask drop is not postulated to cause any significant damage to the structure.

Additionally, the cask wash pit and train bay floor have been evaluated for 20
a dry fuel storage canister transfer cask drop. Since these two areas are located west of the cask pit, there is no possibility that a cask drop in these areas could impact the spent fuel pool. There are no safety related equipment/components located in either of these rooms. Nor are there any rooms located beneath either of these two areas. Therefore, any postulated cask drop accident in the cask wash pit or the Auxiliary Building train bay will not adversely impact the Spent Fuel Pool or any safety related equipment/components. The evaluation of the Dry Fuel Storage Facility (DFSF) components is described in the Davis-Besse Site Certified Safety Analysis Report (CSAR).

The storage racks are designed to eliminate any possibility of fuel assembly sticking in the racks. All projections and corners are properly tapered and rounded off. The spent fuel assemblies are placed into, and removed from, the racks by the spent fuel handling bridge crane. Since the fuel assemblies make free contact with the storage cells, there would be no uplift force exerted on the racks. The spent fuel handling bridge crane is provided with an overload interlock on the hoist which shuts off the power to the hoist any time the load on the hoist exceeds 2700 pounds. The racks are designed to withstand this uplift force.

The following design codes and regulatory guides have been used in the design and analysis of the spent fuel storage rack:

AISC Manual of Steel Construction, 7th edition, 1970.

USNRC Regulator Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants", October 1973.

USNRC Regulatory Guide 1.92, "Combination of Modes and Spatial Components in Seismic Response Analysis, Rev. 1, February 1976.

USNRC Standard Review Plan, Section 3.8.4

USNRC Standard Review Plan, Section 9.1.2

NUREG-75/087, Section 9.2.5, Branch Technical Position APCSB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling", November 1975.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

9.1.3.1 Design Bases

The spent fuel pool cooling system is designed to maintain the borated spent fuel pool water quality and clarity and to remove the decay heat from the stored fuel in the spent fuel pool. It is designed to maintain the spent fuel pool water at approximately 125°F, with a heat load based on removing the decay heat generated from 1/3 of the core fuel assemblies plus the decay heat generated by the previous 11 batches from prior refuelings. The fuel 20
assemblies are assumed to have undergone infinite irradiation and to have been

9.1.4 . Fuel Handling System

9.1.4.1 Receiving New Fuel

The new fuel assemblies are received either by truck or by rail in specially designed shipping containers. The container design is licensed by the fuel 19
supplier to ensure it meets all design requirements.

Upon receipt, the shipping containers are stored in an area where access 19
is limited to authorized personnel only. The shipping containers are opened one at a time and the assemblie(s) are given an inspection for any damage incurred during shipment and are then stored. Storage is in the new fuel storage racks until they are moved to the new fuel elevator for placement in the Spent Fuel Pool or Transfer Mechanism.

9.1.4.2 Refueling

9.1.4.2.1 Preparation

Refueling of the reactor core takes place approximately every 24 months. At 20
this time, as dictated by the fuel management program, spent and partially-spent fuel assemblies are replaced with new fuel assemblies. In addition, some partially-spent fuel assemblies in the spent fuel pool may be re-inserted into the reactor core to enhance fuel utilization or replace partially-spent fuel assemblies which were determined to have defective fuel rods. The actual reactor core configuration shall be identified by the cycle Reload Report incorporated as USAR Appendix 4B.

Prior to the start of the refueling cycle, the following steps are taken to ensure proper and safe refueling operation.

a. All refueling equipment and tools are given routine and scheduled maintenance and inspection to ensure proper functioning and availability.

b. All refueling personnel are thoroughly trained in the use and maintenance of all handling equipment and tools.

c. All refueling personnel are thoroughly briefed on all refueling operations.

9.1.4.2.2 Handling Equipment

The major components of the fuel handling system are shown in Figures 9.1-7 and 9.1-8.

The following Standards and Codes are used in the design, fabrication, installation, and testing of cranes, rails, supporting structures, bridge trolley, hoists, cables, lifting hooks etc.

Crane Manufacturers Association of America (CMAA): 19

Specification 70

American National Standards Institute Inc. (ANSI):

B30.2.0, 1967

Safety Code for Overhead and Gantry Cranes

American Welding Society (AWS):

AWS - D14.1,70

"Specification for Welding Industrial and Mill Cranes"

American Society for Testing of Material (ASTM):

A36 Steel plates (girder, girt, end truck)

A273 Forged steel (hook)

A235, class E Forged steel (hook)

A307, 325, 409, Bolting

A242 Steel plates (girder, end tie)

A514 Steel plates (end tie)

American Iron & Steel Institute (AISI):

4147, 4150 - Alloy steel

1035, 1045, 1070 Forged steel (hook and sheave)

Federal Specifications:

RR-W-410 for wire rope

Steel Structures Painting Council (SSPC):

SSPC-SP-6-63 "Commercial Blast Cleaning"

SSPC-VIS-1 "The Pictorial Surface Preparation Standards for Painting Steel Structures"

AISC 171 lbs (Bethlehem) Crane rail (bridge)

175 lbs (USS & Bethlehem) crane rails (bridge and trolley)

The following is the list of equipment and tools required during the refueling and servicing operations:

a. Containment vessel polar crane

b. Spent fuel handling bridge and hoist mechanism

c. Auxiliary fuel handling bridge and hoist mechanism

d. Fuel transfer mechanisms, including carriage drive system, fuel basket rotation system, and controls

e. Main fuel handling bridge and hoist mechanism

f. Fuel grapple

g. Spent fuel cask crane

h. Portable underwater (submarine) lights

i. Rod assembly handling tool

j. Long-handled hook and guide tool

k. Long-handled wrench tool

l. Phones and intercoms

m. New fuel handling tool

n. New fuel elevator

o. Stud tensioners

p. Stud tensioner slings

q. Stud and nut handling equipment

    1. Stud handling tools and stud handling adapters

    2. Slings

    3. Elongation gauges

    4. Stud impact wrench and adapter

    5. Stud nut wrench

6. Stud nut handling fixture

r. Portable auxiliary neutron detectors and supports

s. Stud hole chase tool

t. Portable radiation monitors

u. Ladders

v. In-core instrument tube plug tool

w. Spare gaskets - drive housing and instrumentation flanges

x. Binoculars and cameras

y. A sufficient stock of protective clothing and badges for the maximum number of personnel required.

z. Blotting paper, cleaning equipment, rags, plastic bags, tape, log books, etc.

aa. Complete complement of tools

The above list is not meant to be all inclusive but is indicative of the type of tools and equipment required for refueling and servicing operations.

The containment vessel polar crane and fuel handling cask crane are provided with Seismic Class I bridge and trolley structures, crane hook brakes, and rigging. The main fuel handling bridge, auxiliary fuel handling bridge, and spent fuel handling bridge are all provided with Seismic Class I railing and restraints to prevent the bridge and associated structure from tipping over in the event of a seismic occurrence. None of the other tools or equipment are designed to Seismic Class I specifications, since they are not related to nuclear plant safety. All miscellaneous tools and equipment are stored in locations in which no damage could occur to essential equipment during a seismic event.

Two horizontal transfer tubes are provided to convey fuel between the containment vessel and the fuel storage pool. Each tube contains tracks for the fuel transfer carriage, a gate valve on the spent fuel pool side, and a flanged closure on the containment vessel side. Each fuel transfer tube penetrates into the refueling canal, inside the containment vessel, where space is provided for the rotation of the fuel transfer carriage basket, containing a fuel assembly. The other end terminates in the spent fuel storage pool, where space is provided for rotation of the fuel transfer carriage basket.

The refueling canal is a passageway in the containment vessel extending from the reactor vessel to the fuel transfer tube. This reinforced concrete enclosure, lined with stainless steel, forms a canal above the reactor vessel,

which is filled with borated water for refueling. The refueling canal is also used for storage of the reactor vessel upper plenum and core barrel assemblies.

Fuel assemblies are handled, in the refueling canal, by a pneumatically actuated grapple attached to a grapple tube. A hoist mechanism raises and lowers the grapple assembly inside of a fixed outer mast. The entire mast assembly can be manually rotated through an angle of 270 degrees. This assembly is mounted on a trolley which can move laterally on a set of rails on the main bridge. 20

Rod assemblies are transferred, in the refueling canal, by an electric hoist driven rod grapple assembly mounted inside of an inner mast. Control, orifice and burnable poison rod assemblies are handled by a grapple attached to the rod grapple assembly. The inner mast is raised or lowered by a hoist similar to the fuel assemblies and mounted on the main and auxiliary fuel handling bridge trolleys adjacent to the fuel mast. 20

The main and auxiliary fuel handling bridges span the refueling canal. The main fuel handling bridge is used to shuffle fuel and rod assemblies between the reactor core and fuel transfer mechanisms or refueling canal racks. It can also be used to shuffle fuel and rod assemblies within the reactor core. During refueling operations, when the main fuel handling bridge is away from the reactor core, the auxiliary fuel handling bridge, which is identical to the main fuel handling bridge, may be used to shuffle fuel and rod assemblies within the reactor core, as specified by the fuel management program. 20

Fuel assemblies inserted or removed from the reactor core are transported between the fuel transfer tube pit and refueling canal via the fuel transfer tubes by means of a fuel transfer mechanism. There are two fuel transfer mechanisms, each consisting of a fuel transfer carriage, fuel basket, and two tilt mechanisms. The fuel transfer carriage is cable/motor driven on tracks extending from the fuel transfer tube pit to the refueling canal. The fuel basket on the carriage is rotated to a horizontal or vertical position in either the fuel transfer tube pit or refueling canal by a hydraulically operated tilt mechanism. The horizontal position is for carriage transport through the fuel transfer tube, and the vertical position is for fuel assembly insertion or withdrawal.

The spent fuel handling bridge spans the spent fuel pool and fuel transfer tube pit or cask pit, which enables the bridge to transfer fuel assemblies between the spent fuel pool storage rack positions, fuel transfer mechanisms, new fuel elevator, and the cask pit. The spent fuel handling bridge is equipped with two systems for moving a fuel assembly. The first is the mast system which is similar to the refueling canal bridge mast described above. The other system employs an auxiliary hoist mounted to the south side of the bridge's overhead frame.

The new fuel elevator is in the fuel transfer tube pit and is provided to lower new fuel to an underwater position for handling by the spent fuel handling bridge.

The spent fuel cask crane auxiliary hook is used to transport new fuel from the new fuel storage racks to the new fuel elevator, in its elevated position, by use of a new fuel handling tool.

9.1.4.2.3 Loading and Removing Fuel

Following the reactor shutdown, containment vessel entry, and missile shield removal, the refueling procedure is begun by removing the reactor closure head and control rod drives and their service structure. Head removal and

replacement time is minimized by the use of stud tensioners. The stud tensioner is a hydraulically-operated device that permits preloading and unloading of the reactor closure studs at cold shutdown conditions. The studs are tensioned to their operational load in two steps in a predetermined sequence. Stud elongation is verified after tensioning. 20

Following removal of the studs from the reactor vessel tapped holes, the studs and nuts are supported in the closure head bolt holes with specially designed spacers. The studs and nuts are removed from the reactor closure head for inspection and cleaning using special stud and nut handling fixtures. A stud storage rack is provided. 20

The reactor closure head assembly is handled by a handling fixture supported from the containment vessel polar crane. It is lifted out of the canal onto a head storage stand located on the operating floor. The stand is designed to protect the gasket surface of the closure head. The lift is guided by two closure head alignment studs installed in two of the stud holes. These studs also provide proper alignment of the reactor closure head with the reactor vessel and internals when the closure head is replaced after refueling.

The annular space between the reactor vessel flange and the bottom of the refueling canal is sealed off before the canal is filled, by a seal clamped to the canal shield plate flange and the reactor vessel flange. If maintenance activities are desired to be performed in the OTSG's (e.g. tube inspection) concurrent with refueling then nozzle dams are placed in the cold leg outlets of the OTSG. The refueling canal is then filled with borated water. 14

The plenum assembly is removed from the reactor by the polar crane, using an internals handling adapter, and is stored under water on one of two stands in the refueling canal. 20

The refueling operations of reactor core offload and reload, or in-core shuffle are conducted by main and auxiliary fuel handling bridges, two fuel transfer mechanisms, and a spent fuel handling bridge. Reactor core offload and reload are performed when plant maintenance activities require all fuel assemblies to be removed from the reactor core. Fuel assembly movement is performed in accordance with a plant procedure that directs fuel handling equipment operations described in section 9.1.4.2.2. 20

If it is suspected that there are partially-spent fuel assemblies with defective fuel rods in the reactor core, then the fuel assemblies may be tested. The testing for defective fuel rods is performed in the refueling canal, spent fuel pool, or fuel transfer tube pit as directed by the fuel management program. The testing is performed with a leakage/failure detection system and associated vendor procedures which are site approved. When a fuel assembly is identified with defective fuel rods, the repair process of either fuel reconstitution or recaging may be implemented during or after the refueling outage. In reconstitution, the defective fuel rods are replaced. In recaging, all of the sound fuel rods are transferred to a new fuel assembly cage and the defective fuel rods are replaced. Both repair processes are limited to a maximum number of replacement rods per fuel assembly by Technical Specifications. The repair processes are controlled by

fuel vendor procedures that are site approved and implemented under work control administrative site procedures. The site approval of vendor procedures is based on a safety evaluation of the repair process prior to procedure approval. Fuel assemblies with defective fuel rods may be reloaded without repair. 20

New fuel assemblies may be prestaged in the spent fuel pool for temporary storage. Otherwise, new fuel assemblies are placed in the fuel transfer mechanism as described in section 9.1.4.2.2 concurrently with other fuel handling refueling operations.

The discharged fuel assemblies are stored in the spent fuel pool to decay prior to being placed in on-site dry fuel storage or off-site shipment. Some of the discharged fuel assemblies may be reused in later reactor core configurations to enhance fuel utilization or to replace partially-spent fuel assemblies determined to have defective fuel rods.

Once refueling is completed, one of the two cables to each fuel transfer carriage is disconnected and the carriage is parked, permitting the gate valve in the fuel transfer tube pit and the blind flange in the refueling canal, on each fuel transfer tube, to be closed and installed, respectively. The gate valve closure enables the refueling canal water to be drained and pumped to the borated water storage tank allowing access to the refueling canal floor to install the blind flange and reactor closure head assembly.

9.1.4.3 Shipping Spent Fuel

The spent fuel assemblies will be stored in the spent fuel pool prior to their being placed in onsite dry fuel storage or shipment offsite. 20

The spent fuel shipping cask can be received at the site either by truck or rail.

Upon arrival, the cask, on the railroad car (or truck), is inspected for any evidence of physical damage. Any loose road dirt and grime is washed off, then the cask and transport vehicle are moved inside the fuel handling building. The cask is then unloaded from the railroad car (or truck) using the main hook on the spent fuel cask crane and placed in the cask wash area as shown in Figure 1.2-5. The cask is washed, scrubbed, and steam cleaned to remove all road dirt and grime. After thorough cleaning, the lid on the cask is unbolted, removed and stored. The cask is lifted from the wash area and lowered into the cask pit. Figure 9.1-9 shows the limits of the main hook travel on the fuel handling area cask crane. If the cask pit is empty to start with it is filled with the borated water from the borated water storage tank to elevation 601 feet 5 inches. The bulkhead between the spent fuel pool and the cask pit is removed to establish communication between the two. The spent fuel is now picked up from the storage racks by the spent fuel bridge crane and placed into the cask. Depending on the size of the cask, as many as 10 spent fuel assemblies may be shipped in one cask. When the cask is fully loaded, still in the cask pit, the lid is placed on top of the cask to provide shielding when the cask is lifted out of the water. When the cask is partially out of water, two or three bolts are loosely installed to keep the lid in place. The cask is now lifted out of the pit and placed in the cask wash area. The cask is connected to a cooling system for the removal of decay

heat from the fuel assemblies. After all of the head bolts are installed and properly torqued, the cask is washed and decontaminated, and the surface radiation level is checked. When it is below the Department of Transportation limits specified in 49CFR171-178 it is ready for shipment. The cask is then placed on the railroad car (or truck) and connected to its cooling system and shipped offsite to the reprocessing plant.

Transfer of spent fuel to the onsite dry fuel storage facility is described in the Davis-Besse Site Certified Safety Analysis Report (CSAR). The radiological consequences due to a postulated transfer cask drop have been evaluated in Section 15.4.7.5. 20

9.1.4.4 Safety Provisions

Safety provisions are designed into the fuel handling system to prevent the development of hazardous conditions in the event of component malfunctions, accidental damage, or operational and administrative failures during refueling or transfer operations. A mechanical lock prevents disengagement of the fuel assembly grapple latches as long as a fuel assembly weight is carried by the grapple mechanism. Bridge and trolley controls are interlocked to prevent movement until the fuel assembly has been completely withdrawn into the
protective mast tube. This auxiliary hoist is equipped with a load brake and 20
manual operation features to ensure the safe handling of fuel during a loss of
power, etc., to the hoist motor. The auxiliary hoist uses a manually actuated 19
fuel grapple to handle fuel. The grapple design has a mechanical interlock on the grapple fingers to ensure that a fuel assembly cannot be inadvertently disengaged while suspended. Administrative controls will be used to ensure that spent fuel handling bridge drive motor is not used while fuel is being raised or lowered with the auxiliary hoist.

The new and spent fuel assembly storage facilities are designed for noncriticality by use of adequate spacing, and, in the case of the spent fuel racks, by use of a stainless steel "flux trap" design. The new and spent fuel storage racks are designed to prevent insertion of a fuel assembly in other than the prescribed locations. This design, combined with the placement of fuel assemblies in a checkerboard pattern as described in Subsection 9.1.2.1,
ensures a safe geometric array. A safe condition is ensured even if new fuel 12
assemblies are stored adjacent to each other in the spent fuel racks in unborated water. Under these conditions, a criticality accident during refueling or storage is not credible.

All spent fuel assembly transfer operations are conducted under water. The
water level in the refueling canal normally provides a minimum 9-1/2 ft of 19
water over the top of the active fuel in the spent fuel assemblies during movement from the core into storage. The depth of the water over the fuel assemblies, as well as the thickness of the concrete walls of the refueling canal, is sufficient to limit the maximum continuous radiation levels in the working area to values consistent with the radiation zoning described in Chapter 12.

The spent fuel storage pool water is cooled by the spent fuel cooling system as described in Subsection 9.1.3. A power failure during the refueling cycle will create no immediate hazardous condition due to the large water volume in both the refueling canal and spent fuel storage pool.

During the refueling period the water level in both the refueling canal and the spent fuel storage pool is the same, and the fuel transfer tube valve is continuously open. This eliminates the necessity for an interlock between the fuel transfer carriage and fuel transfer tube valve operations except to verify full open valve position.

The simplified movement of a transfer carriage through the horizontal fuel transfer tube minimizes the danger of jamming or derailing. All operating mechanisms of the system are located in the fuel handling area for ease of maintenance and accessibility for inspection before the start of refueling operations.

During reactor operation, a bolted closure plate and gasket on the containment
vessel flange of the fuel transfer tube and the fuel transfer tube valve on 19
the fuel handling area end of the tube provide containment vessel isolation as
described in Subsection 6.2.4. Both the spent fuel storage pool and the
refueling canal are completely lined with stainless steel for sealing and ease
of decontamination. The fuel transfer tubes are appropriately attached to
these liners to maintain leak integrity. The spent fuel storage pool cannot
be accidentally drained.

The fuel transfer mechanism is designed to permit initiation of the carriage fuel basket rotation from the building in which the carriage fuel basket is being loaded or unloaded.

All electrical gear is located above water for greater reliability and ease of
maintenance. The hydraulic system that actuates the rotating fuel basket uses
demineralized water for operation to minimize contamination. 14

The refueling canal and storage pool water will have a boron concentration 20
high enough to ensure $\leq 0.95\ k_{eff}$. The requirement of 0.95 k_{eff} may be met
using a combination of boron concentration and control rods. However, the
boron concentration shall provide a 1% $\Delta k/k$ subcritical condition if all
control rods were removed. This ensures that the Startup Accident initial
conditions in Section 15.2.1.1 are satisfied. The cycle specific boron
concentration and control rod configuration may be obtained from the cycle 18
specific Refueling Specification. Although not required for safe storage of
spent fuel assemblies, the spent fuel storage pool water will also be borated
so that the refueling canal water will not be diluted during fuel transfer
operations.

Each fuel handling bridge mast travel is designed to limit the maximum lift of
a fuel assembly to a safe shielding depth with the water at normal refueling 14
level. The length of the connecting rod, between the spent fuel handling 19
bridge auxiliary hoist and the fuel grapple, was designed to provide a safe
shielding depth with the pool water at normal refueling level.

Relief valves are provided on each stud tensioner to prevent overtensioning of the studs due to excessive pressure.

Suspected leaking fuel assemblies can be removed from the core, verified for 14
leakage, and placed in the spent fuel pool for repair, if required. During 20
the fuel assembly repair process, the site approved fuel vendor procedures
and site procedures shall ensure the following: the depth of shielding water
over the fuel rods during extraction/insertion limit the radiation levels in
the work area to values consistent with the radiation zoning levels described
in chapter 12 and accident levels described in chapter 15, the thermal
characteristics within the fuel assembly at the repair station shall not
result in a boiling condition, the fuel assembly repair process shall not
compromise the fuel assembly's structural integrity or reliability during
operation, and the fuel repair process shall have been safety evaluated by a
10CFR50.59a process prior to implementation.

Structurally failed fuel is placed in a failed fuel container prior to its 14
transfer to the spent fuel storage rack. Offsite shipment, following a
suitable decay will require that fuel be transferred to a shipping container
compatible with the shipping cask design to comply with 10CFR71. Evaluation 20
of the onsite Dry Fuel Storage Facility (DFSF) components is described in the
Davis-Besse Site Certified Safety Analysis Report (CSAR).

15.4.7 Fuel-Handling Accident

15.4.7.1 Identification of Causes

Spent fuel assemblies are handled entirely under water. Before refueling, the reactor coolant and the refueling canal water above the reactor are increased in boron concentration so that, with all control rods removed, the k_{eff} of a core is no greater than 0.99. In the spent fuel storage pool, the fuel assemblies are stored under water in storage racks having an "ever-safe" geometric array. Under these conditions, a criticality accident during refueling is not considered credible. Mechanical damage to the fuel assemblies during transfer operations is possible but improbable. The mechanical damage type of accident is considered the maximum potential source of activity release during refueling operations.

Radiological consequences due to a postulated dry fuel storage cask drop accident 20
have been evaluated in Section 15.4.7.2.5.3. Additional evaluations of potential accidents related to onsite dry fuel storage facility operations are discussed in the Davis-Besse Site Certified Safety Analysis Report (CSAR).

15.4.7.2 Accident Analysis - Accident Outside Containment

15.4.7.2.1 Safety Evaluation Criterion

The safety evaluation criterion for this accident is that resultant doses shall not exceed 10CFR100 guideline values.

In support of 18 month cycle operation, re-analyses of the Fuel Handling Accident (FHA) outside containment were performed (reference 30) based on three 450 EFPD cycles. Dose calculations were performed using more conservative source terms. Results of the evaluations showed that the 14
offsite radiological dose for this accident was below the acceptance criterion value in the Standard Review Plan (NUREG 0800). This evaluation criterion was approved by the NRC via approval of the Cycle 6 Reload Report.

15.4.7.2.2 Methods of Analysis

The assumptions and guidelines of Safety Guide 25 were used in this analysis. For convenience, the major assumptions made for this analysis are shown in Table 15.4.7-1. The reactor is assumed to have been shut down for 72 hours, which is the minimum time for Reactor Coolant System cooldown, reactor closure head removal, and removal of the first fuel assembly. It is further assumed that the entire outer row of fuel rods in the assembly, 56 of 208, suffers mechanical damage to the cladding. Since the fuel pellets are cold, only the gap activity is released. The fuel rod gap activity is calculated using the escape rate coefficients and calculational methods discussed in Section 11.1.

See 15.4.7.2.1 referenced re-analyses for a description of the methods of 14
analysis performed to support 18 month cycles.

15.4.7.2.3 Results of Analysis

The gases released from the fuel assembly pass upward through the spent fuel storage pool water prior to reaching the fuel-handling-area atmosphere. Although there is experimental evidence that a portion of the noble gases will remain in the water, no retention of noble gases is assumed. In experiments whereby air/steam mixtures were bubbled through a water pond, Diffey, et al.

(reference 3) demonstrated decontamination factors of about 1000 for Iodine. Similar results for iodine were demonstrated by Barthoux, et al. (reference 4) and predicted by Eggleton (reference 5). Based conservatively on these references, 99 percent of the iodine released from the fuel assembly is assumed to remain in the water. The fuel-handling area is ventilated and discharged through Emergency Ventilation Systems 95 percent efficient HEPA 16
filters and charcoal adsorbers to the station vent.

See 15.4.7.2.1 referenced re-analyses for results of calculations performed 14
to support 18 month operation.

See 15.4.7.2.5 for analyses to support fuel enrichments to 5.0 wt% U-235 and 19
fuel burnups to 60,000 megawatt days per metric ton (MWD/MTU).

15.4.7.2.4 Environmental Consequences

The activity is assumed to be released over two hours from the unit vent. Atmospheric dilution (for site and LPZ boundary) is calculated using the two-hour atmospheric dispersion coefficient developed in Section 2.3. Table 15.4.7-2 gives the total integrated dose at the exclusion distance for the whole body and the thyroid gland. These results are less than 10CFR100 guideline values.

See 15.4.7.2.1 referenced re-analyses for environmental consequences 14
re-analyzed to support 18 month operation.

See 15.4.7.2.5 for analyses for environmental consequences involving fuel 19
enrichments up to 5.0 wt% U-235 and fuel burnups up to 60,000 megawatt days per metric ton (MWD/MTU).

15.4.7.2.5 Additional Analyses

The following analyses supersede all previously documented USAR fuel handling accident analyses for an outside containment fuel handling accident.

15.4.7.2.5.1 Effects Of Extended Burnup And Increased Fuel Enrichments

In order to evaluate the effects of extended fuel burnup and increased fuel enrichment on the consequences of the fuel handling accident outside containment, the source term and offsite dose calculations were reperformed, assuming that fuel assembly-average burnups could be as great as 60,000 MWD/MTU and that initial fuel enrichments could be as great as 5.0 weight percent uranium-235 (wt% U-235).

All assumptions were made in accordance with Nuclear Regulatory Commission Regulatory Guide 1.25 ("Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," dated March 25, 1972).

15.4.7.2.5.2 Results

References 38 and 48 contain a detailed description of these fuel handling 20
accident analyses. Source terms were determined to be extremely weak 19
functions of fuel enrichment, with lower enrichments in general producing larger offsite doses than higher enrichments. Table 15.4.7-6 contains the source term results for fuel assemblies having initial enrichments of 3.0 and 5.0 wt% U-235. The values in Table 15.4.7-6 are fuel assembly-average fission product activities, and do not represent atmospheric release activities, as is the case in Table 15.4.7-3.

The offsite doses for these fuel handling accident outside containment analyses are shown in Table 15.4.7-2a. In summary, these results meet the acceptance criteria provided in NUREG-0800 and, therefore, are well within the dose guidelines set forth in 10CFR100.

15.4.7.2.5.3 Dry Fuel Storage Cask Drop

As a part of the evaluation of the onsite dry fuel storage facility, additional evaluations related to a postulated cask drop accident have been performed.

USAR Section 9.1.4.3, implies that the maximum number of fuel assemblies in a shipping cask is ten. Since the dry fuel storage canister transfer cask contain 24 assemblies, the offsite radiological evaluations were performed as a part of the onsite dry fuel storage facility evaluation using the guidance contained in Standard Review Plan (SRP) 15.7.5; Spent Fuel Cask Drop Accidents. SRP 15.7.5 states that evaluation of radiological consequences need not be performed if the cask drop distances are less than 30 feet and appropriate impact limiting devices are employed during cask movement. Since impact limiting devices are not employed during cask movement, the radiological consequences are evaluated below.

The following assumptions were considered in performing the analysis:

a. The top shield plug will be secured to prevent displacement of assemblies out of the cask in the postulated cask drop accident.

b. The minimum decay time for Spent Fuel Assemblies that will be stored in the dry fuel storage canister is five years. During this decay period all the gaseous activity except for Kr-85 and I-129 will decay to negligible levels.

c. The gap activity for these isotopes is assumed to be 30 percent of the total activity of these isotopes in the assembly. 20

d. All the gap activity is released. No credit is considered for plateaus or any other passive removal mechanisms.

e. Although the station emergency ventilation system will be activated if the radiation levels exceed the radiation monitor setpoints, no credit is taken for this system for removal of iodine.

f. The activity is released over a two hour period. The X/Q value applicable for this time period is 1.9×10^{-4} sec/m^3.

The Resultant Site Boundary Dose are:

WHOLE BODY: 0.008 REM

THYROID: 0.07 REM

These dose are less than the offsite consequences given for a Fuel Handling Accident Outside of Containment, and they are significantly less than the SRP 15.7.5 acceptance criteria values (i.e., 75 REM to the Thyroid and 6 REM to the Whole Body).

TABLE 15.4.7-1[1] |14

Fuel-Handling Accident Parameters Outside Containment

Fuel burnup, full power days	1017
Power level for the assembly during operation, MWt	27.9
Filter efficiency for iodine removal, %	95
Atmospheric dispersion at exclusion distance, s/m^3, assuming ground release	1.9×10^{-4}

The gap activity of the highest power fuel assembly is given in Table 15A-3, and the release by isotope is given in Table 15.4.7-3.

Accident duration, hr	2

Bases:

a. 56 fuel pins fail, releasing gap activity. |20

b. 99 percent iodine remains in water.

c. 72-hour decay

d. Hottest assembly

e. The control room ventilation system is isolated upon receipt of a high-radiation signal from the station vent. The isolation requires a maximum of ten seconds from attainment of the setpoint to closure of damper, during which time the intake rate of outside air is 10,960 cfm.

f. Control room inleakage of outside air is 1 cfm.

g. The release point (station vent) is 160 feet horizontal distance and 180 feet vertical distance.

(1) See 15.4.7.2.5 referenced re-analyses for the analysis assumptions. |19

TABLE 15.4.7-2(1) 14

Resultant Doses From Fuel-Handling Accident Outside Containment

	Exclusion Area Boundary 0 to 2 hours	LPZ Boundary 0 to 30 days
Thyroid dose (rem)	0.106	5.58×10^{-3}
Whole body dose (rem)	0.106	5.59×10^{-3}

	Control Room 0 to 2 hours
Thyroid dose (rem)	0.116
β-skin dose (rem)	0.024
Total body gamma dose (rem)	5.53×10^{-3}

(1) See 15.4.7.2.5 referenced re-analyses for resultant doses. 19

TABLE 15.4.7-2a

Resultant Doses From Fuel Handling Accident Outside Containment - Extended Fuel Burnup (60,000 MWD/MTU) 19

	Exclusion Area Boundary 0 to 2 hours	LPZ Boundary 0 to 30 days
Thyroid Dose (rem)	0.85	4.4×10^{-2}
Whole Body Dose (rem)	0.15	8.0×10^{-3}

TABLE 15.4.7-3 (1) |14

Activity Released to the Atmosphere Due to the Postulated Fuel-Handling Accident Outside Containment (Ci)

I-131	2.18×10^{1}
I-132	1.47×10^{-9}
I-133	5.52×10^{-1}
I-134	3.66×10^{-26}
I-135	1.09×10^{-3}
Xe-131m	1.40×10^{2}
Xe-133m	7.91×10^{1}
Xe-133	1.19×10^{4}
Xe-135m	0
Xe-135	1.52×10^{2}
Xe-137	0
Xe-138	1.82×10^{-73}
Kr-83m	5.00×10^{-11}
Kr-85m	1.27×10^{-3}
Kr-85	1.23×10^{3}
Kr-87	4.60×10^{-16}
Kr-88	3.45×10^{-6}
Kr-89	0

(1) See 15.4.7.2.5 referenced re-analyses for activity released to atmosphere. |19

15.4.7.3 Accident Analysis - Accident Inside Containment

15.4.7.3.1 Safety Evaluation Criterion

The safety evaluation criterion for this accident is that resultant doses shall not exceed 10CFR100 guideline values.

In support of 18 month cycle operation, re-analyses of the Fuel Handling Accident (FHA) inside containment were performed (reference 30) based on three 450 EFPD cycles. Dose calculations were performed using more conservative source terms. Results of the evaluations showed that offsite radiological dose for this accident was below the acceptance criterion value in the current NRC Standard Review Plan (NUREG 0800). This evaluation criterion was approved by the NRC via approval of the Cycle 6 Reload Report. 14

15.4.7.3.2 Analysis

The following assumptions were used in the analysis:

a. The accident occurs at 72 hours following reactor shutdown.

b. Noble gas and iodine gap activities are based on Regulatory Guide 1.25.

c. One entire assembly is considered damaged.

d. A time-averaged radial peaking factor on an assembly basis of 1.4 is utilized.

e. The gap activity in the damaged fuel assembly is assumed to be released to the pool. All the noble gas activities that are released to the pool are assumed to escape from the pool; one percent of the iodine activities that are released to the pool are assumed to escape from the pool. 19

f. Containment isolation is not assumed.

g. An instantaneous release (very high escape rate) from the containment is assumed to ensure that all the activity coming out of the pool is released to the environment in a short time (see Table 15.4.7-5).

h. Atmospheric dispersion factor (χ/Q) at site boundary is $1.9\text{x}10^{-4}$ sec/m^3 and at LPZ it is $9.9\text{x}10^{-6}$ sec/m^3.

i. No filtration is assumed.

See 15.4.7.3.1 referenced re-analyses for a description of 18 month cycle re-analysis and assumptions used. 14

See 15.4.7.3.4 for analyses supporting fuel enrichments up to 5.0 wt% U-235 and fuel burnups up to 60,000 megawatt days per metric ton (MWD/MTU). 19

15.4.7.3.3 Environmental Consequences

The activity is assumed to be released over 2 hours. Atmospheric dilution for site boundary and LPZ is calculated using the 2-hour atmospheric dispersion coefficient developed in Section 2.3. Table 15.4.7-4 gives the calculated doses. These results are well within the 10CFR100 guideline values.

See 15.4.7.3.1 referenced re-analyses for the 18 month cycle re-analysis environmental consequences. 14

See 15.4.7.3.4 for analyses for environmental consequences involving fuel enrichments up to 5.0 wt% U-235 and fuel burnups up to 60,000 megawatt days per metric ton (MWD/MTU). 19

15.4.7.3.4 Additional Analyses

The following analyses supersede all previously documented USAR fuel handling accident analyses for an inside containment fuel handling accident.

15.4.7.3.4.1 Effects Of Extended Burnup And Increased Fuel Enrichments

In order to evaluate the effects of extended fuel burnup and increased fuel enrichment on the consequences of the fuel handling accident inside containment, the source term and offsite dose calculations were repreformed, assuming that fuel assembly-average burnups could be as great as 60,000 MWD/MTU and that initial fuel enrichments could be as great as 5.0 weight percent uranium-235 (wt% U-235).

All assumptions were made in accordance with Nuclear Regulatory Commission Regulatory Guide 1.25 ("Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Safety Guide 25)," dated March 25, 1972).

15.4.7.3.4.2 Results

Reference 38 contains a detailed description of these fuel handling accident analyses. Source terms were determined to be extremely weak functions of fuel enrichment, with lower enrichments in general producing larger offsite doses than higher enrichments. Table 15.4.7-6 contains the source term results for fuel assemblies having initial enrichments of 3.0 and 5.0 wt% U-235. The values in Table 15.4.7-6 are fuel assembly-average fission product activities, and do not represent atmospheric release activities, as is the case in Table 15.4.7-5.

The offsite doses for these fuel handling accident inside containment analyses are shown in Table 15.4.7-4a. In summary, these results meet the acceptance criteria provided in NUREG-0800 and, therefore, are well within the dose guidelines set forth in 10CFR100.

15.4.7.3.4.3 Control Room Dose Analysis

Additional analysis was performed in order to support Technical Specification Amendment 202 (Reference 43). This amendment allows both doors of the containment personnel air lock to be open during core alterations or movement of irradiated fuel within containment. This analysis (Reference 44) calculated the control room dose for a fuel handling accident inside containment. The following assumptions were used in the analysis:

1. The accident occurs 72 hours after reactor shutdown.

2. The noble gas and iodine fuel activities are based on the 3 wt% enrichment U-235 for an extended burnup of 60,000 MWD/MTU and are consistent with Table 15.4.7-6.

3. Noble gas and iodine release fractions are consistent with RG 1.25: 10% of fuel assembly Xe, Kr and I except for Kr-85 which is 30%.

4. All fuel pins (208) of one assembly are assumed to release their activity instantaneously to the pool.

5. All the noble gases and 1% of the iodine are released from the pool to the containment. The activity is assumed to be released from containment to the atmosphere over 2 hours. No credit is taken for containment isolation. The 2 hour atmospheric dispersion coefficient for the control room is 5.85×10^{-4} sec/m^3. No filtration is assumed prior to atmospheric release.

6. The control room normal HVAC air intake, which is more than 160 feet from the release point, is isolated prior to the release from containment reaching it. The HVAC air intake is automatically isolated upon receipt of a high radiation signal from the station vent.

7. The volume of the control room is 55,400 ft^3. The in-leakage into the control room is 54.4 CFM. This equates to .06 air changes per hour.

8. No credit is taken for the control room emergency ventilation system to remove iodine activity leaked into the control room.

9. Normal control room HVAC is established 2 hours following initiation of the accident.

Table 15.4.7-4a gives the results of the control room dose calculation. The doses are well within the control room dose acceptance criteria as given in GDC 19 of 10 CFR 50 Appendix A.

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TABLE 15.4.7-4(1) 14

Resultant Doses From Fuel-Handling Accident Inside Containment

	Exclusion Area Boundary	LPZ Boundary
Thyroid dose (rem)	44.7	2.33
Whole body dose (rem)	0.17	8.86×10^{-3}

(1) See 15.4.7.3.4 referenced re-analyses for resultant doses from the FHA inside containment. 19

TABLE 15.4.7-4a

Resultant Doses From Fuel Handling Accident Inside

Containment - Extended Fuel Burnup (60,000 MWD/MTU)

	Exclusion Area Boundary 0 to 2 hours	LPZ Boundary 0 to 30 days
Thyroid Dose (rem)	62.6	3.26
Whole Body Dose (rem)	0.55	3.0×10^{-2}

	Control Room(1) 0 to 2 hours
Thyroid Dose (rem)	21.7
Whole Body Dose (rem)	0.10

20

(1) See 15.4.7.3.4.3 referenced reanalysis from the FHA inside containment.

TABLE 15.4.7-5[(1)] 14

Activity Released to the Atmosphere Due to the Postulated Fuel-Handling Accident Inside Containment (Ci)

I-131	4.26×10^{2}
I-132	3.11×10^{-7}
I-133	1.18×10^{2}
I-134	1.47×10^{-22}
I-135	6.86×10^{-1}
Xe-131m	3.50×10^{2}
Xe-133m	1.28×10^{3}
Xe-133	8.55×10^{4}
Xe-135m	0
Xe-135	5.07×10^{2}
Xe-137	0
Xe-138	5.67×10^{-70}
Kr-83m	2.43×10^{-8}
Kr-85m	2.93×10^{-1}
Kr-85	2.48×10^{3}
Kr-87	3.70×10^{-13}
Kr-88	1.23×10^{-3}
Kr-89	0

(1) See 15.4.7.3.4 referenced re-analyses for the activity released from the FHA inside containment. 19 14

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TABLE 15.4.7-6

Fuel Assembly Fission Product Activities (Curies)

For Extended Fuel Burnup (60,000 MWD/MTU) (1)

Isotope	Enrichment = 3 wt% U-235	Enrichment = 5 wt% U-235
I-131	5.86E+05	5.81E+05
I-132	5.58E+05	5.56E+05
I-133	1.29E+05	1.30E+05
I-134	1.05E-18	1.07E-18
I-135	7.02E+02	7.03E+02
Xe-131m	8.32E+03	8.18E+03
Xe-133m	2.56E+04	2.55E+04
Xe-133	1.12E+06	1.12E+06
Xe-135m	1.13E+02	1.13E+02
Xe-135	1.35E+04	1.35E+04
Xe-138	0	0
Kr-83m	2.49E-04	2.62E-04
Kr-85m	1.71E+00	1.85E+00
Kr-85	8.18E+03	9.53E+03
Kr-87	1.87E-12	2.06E-12
Kr-88	6.59E-03	7.29E-03

(1) Corrected to include Regulatory Guide 1.25 Radial Peaking Factor and Radioactive Decay for 72 hours.