# Chapter 5

# 5.0 OPERATIONS

## 5.1 GENERAL DESCRIPTION

The methods and sequences described below provide an overview of the operational controls ~~which~~ *that* personnel performing spent fuel loading and storage activities will implement to assure that operations utilize the passive safety features of the Trojan ISFSI design described in Chapter 4. Fuel loading and ~~PWR Basket~~ *MPC* sealing operations (including non-destructive examination and pressure testing) will be performed within the Fuel Building in order to utilize the existing systems and equipment for heavy lifts, radiation monitoring and controls, decontamination and any necessary auxiliary support (i.e., electrical, crane, service air, etc.). Fuel handling and cask loading operations in the Fuel Building will be performed in accordance with Portland General Electric Company's 10 CFR 50 license for the Trojan Plant. Certain restrictions related to ~~PWR Basket~~ *MPC* loading are also contained in the ISFSI Technical Specifications. Storage at the ISFSI will be subject to the requirements of the ISFSI license issued in accordance with 10 CFR 72. Once the loaded Concrete Cask is ~~placed on air pads in the Fuel Building Bay and~~ moved to the ISFSI concrete slab area, operational activities are essentially limited to monitoring proper decay heat removal.

### 5.1.1 OPERATION DESCRIPTION

The following sections describe the spent fuel handling, ~~PWR Basket~~ *MPC* sealing, and Concrete Cask loading activities relevant to the operation of the Trojan ISFSI. As previously described in Chapter 3, the Trojan ISFSI will contain intact and ~~failed~~ *damaged* spent nuclear fuel assemblies and fuel debris. The ~~PWR Basket~~ *MPC* is vertically loaded with fuel assemblies and/or special canisters designed to hold ~~failed~~ *damaged* fuel or fuel debris. Section 5.1.1.1 describes the operational controls for loading the individual canisters. Section 5.1.1.2 describes operational controls for loading *and sealing* the individual ~~PWR Basket~~ *MPC*.

Specific procedures will define and control classification criteria, loading sequence, and individual ~~PWR Basket~~ *MPC*/Concrete Cask inventory. Fuel and debris ~~will be~~ *have been* visually inspected ~~prior to loading~~ to verify that each assembly/item conforms to the established classification criteria. Fuel *has been*~~will be~~ examined to verify that pellets are structurally contained within the cladding *of intact fuel assemblies*. Fuel will *also* be visually inspected during the loading process to ensure conditions have not changed since the previous inspection, which would cause the need for special handling of the component. In addition, item identification and/or serial numbers will be verified and recorded. Fuel loading operations will utilize videotape to record fuel assembly serial numbers and to provide an independent record of loaded inventory. Additional procedures will control placement and use of impact limiters, allowable travel path inside the Fuel Building, and limit lifting heights to assure compliance with bounding analysis.

### 5.1.1.1 *Damaged* ~~Failed~~ Fuel and Fuel Debris Process Can Capsule Loading

Special containers are used to segregate ~~failed~~ *damaged* fuel and fuel debris within the confines of the ~~PWR Basket~~ *MPC*. The individualized containers provide containment properties by constraining the material to fixed storage locations which maintains the assumptions in the criticality analysis and heat transfer modeling.

~~Failed~~ *Damaged* fuel is contained in special cans designed to fit in one of four oversized peripheral storage ~~sleeves~~ *cells* of the ~~PWR Basket~~ *MPC*. ~~Failed~~ *Damaged* or suspect fuel that cannot structurally contain pellets within the cladding will be placed in a Failed Fuel Can. Because the cans are open to the internal ~~PWR Basket~~ *MPC* atmosphere, they are ~~vacuum~~ dried and backfilled at the same time as other ~~PWR Basket~~ *MPC* contents.

~~Fuel debris is contained within Process Cans.~~ Fuel debris mixed with organic material ~~is~~*was* processed, as part of the ~~f~~*F*uel ~~d~~*D*ebris ~~p~~*P*rocessing ~~p~~*P*roject, to destroy the organic material and seal the debris in a fuel debris Process Can Capsule before being placed in a Failed Fuel Can. *The Process Can Capsules were vacuumed, purged and backfilled with helium, and then seal-welded closed.* Fuel debris in Process Cans without organic material does not require processing prior to placing in a Failed Fuel Can. Failed Fuel Cans may be placed in any one of the four oversized storage cells in the ~~PWR Basket~~ *MPC*.

### 5.1.1.2 ~~PWR Basket~~ *MPC* Loading and Sealing Operations

This section describes a general sequence of operations and controls necessary to load, seal, test and unload, if necessary, a*n* ~~PWR Basket~~ *MPC* in the Fuel Building and to control transfer operations to the ISFSI Storage Pad. The major components described in Chapter 4 are further defined with design and operating characteristics. Test and/or inspection methods demonstrate compliance with design requirements. The ~~final~~ sequence of operations to be used will be ~~established~~ *finalized* during ~~P~~*p*re-operational ~~T~~*t*esting and will be controlled by specific procedures.

Each ~~PWR Basket~~ *MPC* and the Transfer Cask are brought into the Fuel Building through the crane bay door. *After examination and any needed cleaning,* ~~T~~*t*he Transfer Cask is moved by use of the Fuel Building overhead crane ~~(independent dual hook design)~~ and Transfer Cask Lifting Yoke to the Cask Wash Pit area. The *empty* ~~PWR Basket~~ *MPC* is then moved by the same crane and placed into the Transfer Cask. After installation of ~~radiation shielding shims~~ *an annulus seal* in the gap between the Transfer Cask and ~~PWR Basket~~ *MPC*, ~~a cask lid assembly is bolted onto the top of the Transfer Cask. The cask lid assembly, which is a steel ring, ensures that the PWR Basket cannot be inadvertently lifted out of the Transfer Cask while loaded. The cask lid assembly also contains shield lid retainers which prevent the PWR Basket shield lid from coming out of the PWR Basket in the unlikely event of a tipover.~~ *the system is prepared for loading.*

The Transfer Cask (with ~~PWR Basket~~ *MPC*) is then moved by the Fuel Building overhead crane and suspended over the Cask Loading Pit immediately adjacent to the Spent Fuel Pool.

~~Borated~~*Filtered Spent Fuel Pool* water is flushed through the ~~PWR Basket~~ *MPC*/Transfer Cask gap to minimize unnecessary contamination of the ~~PWR Basket~~ *MPC* external surface while the Transfer Cask is in the Cask Loading Pit. While suspended in the Cask Loading Pit, the ~~PWR Basket~~ *MPC* is then filled with borated water. After the Transfer Cask is lowered to the Cask Loading Pit impact limiter, and the Cask Loading Pit is filled with borated water, the specified ~~PWR Basket~~ *MPC* contents are loaded. Operations will be conducted in accordance with approved Trojan Nuclear Plant fuel handling procedures.

~~The PWR Basket shield lid is placed into the PWR Basket while in the Cask Loading Pit.~~ *After completion of loading the contents into the MPC, the MPC lid is leveled and lowered onto the MPC in the Cask Loading Pit. The Lifting Yoke is attached to the Transfer Cask lifting trunnions and* ~~T~~*t*he loaded Transfer Cask is lifted ~~from~~ *to the top of* the *Cask* ~~L~~*L*oading ~~p~~*P*it~~,~~ *S*tanding water above the ~~shield~~ *MPC* lid is removed and the *Transfer* ~~c~~*C*ask is washed on the exterior to remove potential *loose* contamination. *The lid retention system is then engaged. The lid retention system is used to secure the MPC lid on the MPC during lifting of the Transfer Cask (containing the loaded MPC) out of the Cask Loading Pit and movement to the Decontamination and Assembly Station (DAS).* The *loaded* Transfer Cask is *removed from the Cask Loading Pit and* ~~returned~~ *moved by use of the Fuel Building overhead crane to the* ~~93' area~~ *DAS for decontamination and* ~~welding~~ *MPC preparation. The Lifting Yoke and MPC lid retention system are then disconnected from the Transfer Cask and MPC.*

The ~~cask preparation area~~ *DAS* on the ~~93'~~93-*foot* Fuel Building level will be curbed and sealed to collect drainage and rinse water, and *liquid waste will be* routed *to the* plant liquid radwaste system~~s~~.

Decay heat could eventually cause boiling in the ~~PWR Basket~~ *MPC* after it is removed from the Cask Loading Pit. As a precaution, lid sealing, hydrostatic testing, *MPC lid-to-shell weld leakage testing,* and draining must be completed within an administratively controlled period of time beginning when the ~~PWR Basket top is lifted from~~ *MPC lid is lowered onto the loaded MPC in* the Cask Loading Pit, *segregating the water inside the MPC from the rest of the water in the Cask Loading Pit.* Should ~~these~~ *this* time limit~~s~~ be ~~exceeded~~ *approached*~~, or PWR Basket temperature increase~~ such that ~~PWR Basket~~ *MPC* cooling is required, procedures will address methods to establish required ~~PWR Basket~~ *MPC* cooling. The method of cooling available will be dependent upon *the* status of ~~PWR Basket~~ *MPC* preparation and is discussed in the following paragraphs.

The methodology for the development of the administratively controlled "time-to-boil" is described in *Section 4.5.1.1.5* of Reference ~~5.~~*1*. The heat output of the contents (fuel and other materials) of the ~~PWR Basket~~ *MPC* is first calculated and a heat-up rate established. The heat-up rate is then used to determine the administratively controlled limit (in hours) to ensure that the water ~~contents of~~ in the ~~PWR Basket~~ *MPC* do*es* not boil. *However,* ~~E~~even if localized boiling were to occur within the ~~PWR Basket~~ *MPC*, $K_{eff}$ will remain less than 0.95 (due to the use of Boral in the design).

The ~~radiation shielding shims are~~ *annulus seal is* removed from the top of the ~~PWR Basket~~ *MPC* area to allow for completion of *MPC lid welding and* decontamination activities~~ and then reinstalled~~. ~~The exterior of the PWR Basket will be checked for loose surface contamination~~*The MPC lid top surfaces and accessible areas of the MPC shell are surveyed* ~~(~~to the extent possible ~~while in the Transfer Cask)~~ *for loose surface contamination* to determine if decontamination of the ~~PWR Basket~~ *MPC* is required. Contamination limits are ~~described~~ *established* in *Section 7.2.2.*

The water level in the ~~PWR Basket~~ *MPC* is lowered by approximately ~~75~~*50 to 120* gallons. This ensures that ~~the shield~~ *MPC* lid weld*ing* is not affected by water percolation. ~~The~~ *S*hims ~~are~~ *may be* installed between the ~~shield~~ *MPC* lid and the ~~PWR Basket~~ *MPC* shell *to optimize welding conditions, as necessary. Shim details may vary from MPC to MPC due to fabrication tolerances.* The automated welding system (AWS) is normally used to perform the ~~shield~~ *MPC* lid weld~~s~~; however, ~~the~~ manual welding ~~system~~ may be used as ~~a backup in the event that the AWS is unavailable, provided an ALARA review is conducted~~*desired or necessary.*~~.~~ *Root, final, and intermediate (at each approximately 3/8-inch of weld depth) dye-penetrant examinations are performed on the MPC lid-to-shell weld.*

Once the ~~PWR Basket shield~~ *MPC* lid weld is completed, the ~~PWR Basket~~ *MPC* is refilled with borated water and hydrostatically tested to ~~approximately 15 psig which exceeds 1.5 times~~ *at least 1.25 times* the *maximum* normal *design* ~~operating~~ pressure *of 100 psig*. This pressure is held for 10 minutes with no observable leakage *as the acceptance criterion.* After ~~test acceptance~~*successful hydrotesting, the MPC lid weld is dye penetrant examined. After successful dye penetrant testing*, approximately ~~75~~*20* gallons of water ~~is again~~ *are* removed from the ~~PWR Basket~~ *MPC*, ~~ensuring that~~ *with* the water ~~is~~*being* replaced with helium during the draining. The helium is pressurized to ~~approximately~~ *a nominal test pressure of* ~~15~~*90* psig and the ~~shield~~ *MPC* lid weld is helium leak tested. The maximum permissible ~~PWR Basket~~ *MPC* leak rate is ~~$\leq 1\times10^{-4}$ standard~~ $\leq 5\times10^{-6}$ *atmosphere*-cubic centimeters per second ~~(scc/sec) at ≥ 13 psig~~ (atm-cc/sec) *based on a pressure differential of one atmosphere across the confinement boundary. The measured leak rate at the test pressure is correlated to a pressure differential of one atmosphere for comparison against the acceptance limit.* Procedures for leak testing will be prepared using the guidance in ANSI N14.5. ~~After test acceptance, the shield lid weld is dye penetrant examined. After all three tests have been successfully completed, the structural lid and backing ring are installed. The automated welding system is normally used to perform the structural lid welds; however, the manual welding system may be used as a backup in the event that the AWS is unavailable, provided an ALARA review is conducted. At least three weld layers (each layer consists of approximately ¼" of weld metal) are used to attach the structural lid to the PWR Basket shell. A dye penetrant examination is performed on each ¼" of weld deposit. The fillet weld between the two lids, at the bottom of the valve access port, is completed and dye penetrant examined.~~

If ~~PWR Basket~~ *MPC* cooling is required at any time during the ~~PWR Basket~~ *MPC* preparation process, cooling can be provided by ~~the VDS. The VDS would be connected to the PWR Basket~~

~~and~~ *recirculating helium or* borated water ~~recirculated~~ through the ~~PWR Basket~~ *MPC cavity*, as required, to maintain desired ~~PWR Basket~~ *MPC* temperature.

~~Evacuation~~ *Blowdown* of the ~~PWR Basket~~ *MPC* is initiated by ~~pumping the liquid contents~~ *injecting pressurized helium into the vent port and directing the resulting water through the drain line* back into the Spent Fuel Pool or a suitable holding tank. To aid in removing residual moisture, ~~nitrogen or other inert gas~~ *helium* is blown through the ~~PWR Basket~~ *MPC* vent line (maximum pressure will be controlled to less than ~~15~~ *75* psig) and out the drain line until no water is visible coming from the drain line. The outlet of the drain line will be discharged to the ~~SFP~~ *Spent Fuel Pool or other appropriate location.* ~~The VDS discharge will be directed to a suitable filtration system to minimize the possibility of particulate airborne contamination. The VDS is used to perform multiple pump downs to achieve a stable internal PWR Basket vacuum pressure at or below 3 mm Hg for a minimum of 30 minutes. The PWR Basket is then flushed with helium and the evacuation/vacuum process is repeated.~~

*Two methods of drying the MPC cavity are available. The first is through the use of a vacuum drying system (VDS). The VDS is used to pump down the MPC cavity to achieve a stable internal vacuum pressure at or below 3 torr for a minimum of 30 minutes. The MPC cavity gas discharge is directed to a suitable filtration system to minimize the possibility of particulate airborne contamination. The second method utilizes a forced helium recirculation/moisture removal system where warm dry helium is introduced to the MPC cavity to absorb residual moisture. The helium is cooled and the absorbed moisture is removed. The helium may be heated prior to its return to the MPC in a closed loop to begin the moisture removal process again. The process is continued until the cooled MPC gas temperature measured at the demoisturizer exit reaches ≤21°F, a value equivalent to the saturation temperature of water at a pressure of 3 torr, or less. This temperature is maintained for at least 30 minutes. The helium recirculation/moisture removal system can also be used to cool the MPC in the event that the MPC needs to be unloaded.*

~~An analysis of the vacuum drying process as controlled by Technical Specifications has shown that the maximum fuel clad temperature will not exceed the short term storage temperature limit of Table 4.2-12. The maximum steady state fuel clad temperature that can occur during the vacuum drying process is 888°F which is 170°F below the short term limit. If the vacuum drying process can not be completed within Technical Specification time limits, then either a helium atmosphere will be established in the PWR Basket or cooling will be established using the VDS as previously described. With either a helium atmosphere or VDS cooling established, additional time is available to perform repairs and successfully achieve required vacuum drying.~~

Following completion of the ~~vacuum~~ drying process, the sealed ~~PWR Basket~~ *MPC* is backfilled with 99.~~999~~ *995* ~~%~~ *percent* pure helium to ~~14.5±0-0.5 psia~~ *≥29.3 psig and ≤33.3 psig at a reference temperature of 70°F.* The *helium backfill* ~~vacuum drying~~ system is used to ~~regulate~~ *establish the* internal *MPC* pressure *consistent with the assumptions in the thermal analysis.*

*After helium backfilling is complete, the MPC vent and drain port cover plates are welded in place over the vent and drain ports and dye-penetrant examinations are performed on each weld pass. Helium is then injected into the vent and drain port cavities to displace the air. Small set screws are immediately installed to help maintain the helium atmosphere in the vent and drain ports. The set screw holes are plug welded and dye penetrant examinations are performed on the plug welds. Finally, the vent and drain port cover plate welds are helium leak tested.*
*The MPC closure ring may be provided in one piece or in segments. The closure ring is installed on the MPC lid over the vent and drain port cover plates to form a complete ring and is welded in place in accordance with the design drawings to provide the redundant welded boundary to the MPC lid weld and the vent and drain port cover plate welds. Dye penetrant examinations are performed on the final closure ring weld surface.*

*The top lid is bolted onto the Transfer Cask. The Transfer Cask top lid provides shielding and ensures that the loaded, sealed MPC cannot be inadvertently lifted out of the Transfer Cask.* ~~The two penetration cover plates are welded in place and dye penetrant checked.~~The Transfer Cask containing the sealed ~~PWR Basket~~ *MPC* is ~~transported~~ *moved* by the Fuel Building overhead crane ~~back~~ to the ~~area inside the crane bay door~~*location* where a Concrete Cask has been ~~placed and~~ prepared for acceptance of the *loaded* ~~PWR Basket~~ *MPC*.

Plastic sheeting is placed on top of the Concrete Cask walls to minimize contamination from the bottom of the Transfer Cask. After installation of the Transfer Cask ~~shield~~ *bottom* door hydraulic cylinders, the Transfer Cask is placed on top of the Concrete Cask and correctly positioned by the use of holes located on each side of the Transfer Cask. Using ~~PWR Basket~~ *MPC* lift~~ing rings~~ *cleats* and slings, the ~~PWR Basket~~ *MPC* is slightly elevated to eliminate weight on the Transfer Cask bottom doors. The *Transfer Cask* bottom doors are opened, and the ~~PWR Basket~~ *MPC* is lowered into the Concrete Cask. Ceramic tiles in the bottom of the Concrete Cask prevent the stainless steel ~~PWR Basket~~ *MPC* from resting directly upon the carbon steel liner of the Concrete Cask. When the ~~PWR Basket~~ *MPC* is firmly resting on the ceramic tiles at the bottom of the Concrete Cask, the ~~PWR Basket~~ *MPC* lifting slings are ~~removed with the aid of an extension device and the Transfer Cask bottom doors are closed~~ *lowered onto the MPC lid. An indirect check of the MPC external contamination levels is performed by surveying the internal wall surfaces of the Transfer Cask. If contamination levels exceed the established limits, then the need to decontaminate the MPC will be evaluated.* The Transfer Cask is lifted from the Concrete Cask, *the Transfer Cask bottom doors are closed, and* the hydraulic ~~system is~~ *cylinders are* removed ~~and the interior of the Transfer Cask is checked for loose surface contamination~~ prior to ~~transporting~~ *moving the Transfer Cask* to the Cask Wash Pit ~~area to provide a second check of the surface contamination levels on the exterior of the PWR Basket that was just removed from the Transfer Cask. If contamination levels exceed the limits established in Chapter 7, then the need to decontaminate the PWR Basket previously placed in a Concrete Cask will be evaluated.~~. The Transfer Cask is inspected and cleaned for continued use or storage. The shield ring is installed on top of the Concrete Cask, the ~~PWR Basket hoist rings~~ *MPC lift cleats and slings* are removed, and *threaded inserts are installed in the empty holes of the MPC lid where the lift cleats were attached.* The Concrete Cask cover plate is bolted into position. A tamper indicating wire is threaded through at least two of the cover bolts.

The loaded Concrete Cask exterior ~~will be~~*is* surveyed for contamination and radiation levels ~~will be~~*are* measured before transporting and ~~placement~~*placing the loaded Concrete Cask* on the ISFSI pad. If measured radiation levels are significantly higher than expected (i.e., on the order of 1½ times the calculated value for the top of the Concrete Cask and 2 times the calculated value for the sides of the Concrete Cask), an investigation will be performed. If the measured Concrete Cask surface radiation levels exceed the design values in Table 7.4-1, the Concrete Cask will not be moved to the Storage Pad without performing an appropriate evaluation to verify compliance with 10 CFR 20 ~~regulations~~*and 10 CFR 72.104*.

After the ~~PWR Basket~~*MPC* is loaded into the Concrete Cask and radiation levels are verified to be within design limits, the Concrete Cask is moved to the ISFSI Pad. *At this point in the loading sequence,* ~~T~~*t*here are no conditions that would require the ~~PWR Basket~~*MPC* to be returned to the Spent Fuel Pool for unloading. ~~In the unlikely event of a PWR Basket leak, a Basket Overpack would be used as discussed in Section 5.1.1.5.~~

### *5.1.1.3 MPC Contingency Unloading Operations*

Although unlikely, ~~PWR Basket~~*MPC* unloading could be required if ~~the shield lid weld helium leak rate limit or vacuum drying pressure limit could not be met within Technical Specification time limits. In addition, PWR Basket unloading may be required if Concrete Cask surface radiation levels exceeded allowable values when measured in the Fuel Building~~ *certain acceptance criteria, such as helium leakage or dose rate, are not met in preparing the MPC or the Concrete Cask for storage at the ISFSI. Recognition of the failure to meet such acceptance criteria would occur at the time of MPC loading. Therefore, the Trojan Spent Fuel Pool would still be available for unloading of an MPC.*

If it is necessary to unload a*n* ~~PWR Basket~~*MPC*, the ~~initial steps of unloading will be performed in the DAS on the 93' elevation of the Fuel Building. The~~ Transfer Cask, *MPC lift cleats, and slings* would be used to return the ~~PWR Basket~~*MPC to the DAS*~~, if necessary~~. *These steps are performed in the reverse order of the loading operations.* With the ~~PWR Basket~~*MPC* in the Transfer Cask, the first step in the unloading process is the removal of the ~~vent and drain connection access cover plates (2 plates) for an initial sample~~*Transfer Cask top lid and removal of the MPC lift cleats.*

*The vent and drain ports are accessed by removing portions of the closure ring to the extent needed to expose the vent and drain port cover plates. Similarly, portions of the vent and drain port cover plates are removed to expose the vent and drain connections.* Once the vent connection is accessible, a sample of the ~~PWR Basket~~*MPC* internal atmosphere is obtained by connecting a sample rig and opening the installed vent connection valve. ~~A vacuum pump may be used to draw the sample since the PWR Baskset atmosphere may be at a slight negative pressure.~~ Based on the sample results, the ~~PWR Basket~~*MPC* atmosphere may be purged *and cooled* with helium ~~or nitrogen~~ by connecting the ~~VDS~~*helium recirculation cooling system* to the ~~drain connection~~*MPC and circulating cooled gas through the MPC cavity.* ~~The vent~~

~~connection would be directed to the off gas system.~~*Cooling continues until the MPC gas exit temperature is ≤200°F.* Once ~~purging~~ cooling is complete, *the helium recirculation cooling system is disconnected and* the vent is connected to a pipe ~~fitted with temperature and pressure monitors. The vent path is~~ routed to below the surface of the Spent Fuel Pool and connected to a diffuser. *The MPC is refilled with borated water through the drain port.* The drain connection of the ~~PWR Basket~~ *MPC* is connected to the ~~VDS~~ *borated water supply* via ~~a connection fitted with~~ a flow-limiting device. To begin *filling and* cooldown of the ~~PWR Basket~~ *MPC*, the installed vent connection valve is opened, and ~~the VDS is used to~~ *borated water is* slowly injected ~~water~~ *into the MPC cavity* via the drain ~~path~~ *line*. The drain ~~path~~ *line* directs water to the base of the ~~PWR Basket~~ *MPC*. ~~Analyses have been performed which demonstrate that injection flow rates less than 8 gpm do not result in fuel clad stress, or PWR Basket pressures in excess of design limits. The vent path pressure is monitored until the PWR Basket is reflooded. Once the PWR Basket is reflooded, water is recirculated to the Spent Fuel Pool until the PWR Basket water outlet temperature is less than 150°F.*~~*The reflooding flow rate is MPC-specific, and will be performed using approved procedures.*

With cooldown complete, ~~the VDS may be used to remove water from the shield lid weld area~~ *approximately 50 to 120 gallons of water may be removed from the top of the MPC cavity,* if desired. ~~This is accomplished by pumping approximately 75 gallons from the PWR Basket. The VDS is disconnected, and the vent path remains open. A cutter is installed on the PWR Basket and used to remove the structural lid weld. Once the structural lid weld is removed, the structural lid is removed, followed by removal of the shield lid weld. The cutter is removed in preparation for returning the PWR Basket to the Cask Loading Pit. At any time during the cutting of the structural or shield lid welds, if PWR Basket cooling is required, the VDS is connected and used to recirculate Spent Fuel Pool water through the PWR Basket to achieve the required cooldown. Once the desired PWR Basket temperature is achieved, the VDS is disconnected and the cutter reinstalled.*~~ *All fluid flow components are disconnected from the MPC, and the vent path remains open. A cutting system is used to remove the weld between the MPC lid and closure ring and the MPC shell and the closure ring. Then, the MPC lid-to-shell weld is removed.*

With the ~~structural lid removed and the shield lid prepared for removal, the PWR Basket is returned to the Cask Loading Pit.~~ *MPC closure ring and lid-to-shell welds removed, the lid retention system and annulus seal are installed in preparation for MPC lid removal.* Prior to placing the Transfer Cask in the Cask Loading Pit, *the* gap flush*ing unit* ~~water~~ is connected *and placed into service*. The Transfer Cask is lowered into the Cask Loading Pit *and the MPC lid retention system fasteners are removed.* The Cask Loading Pit is reflooded. After the Cask Loading Pit is reflooded, the ~~shield~~ *MPC* lid can be removed. The fuel is then removed from the ~~PWR Basket~~ *MPC* and returned to the Spent Fuel Pool.

### ~~5.1.1.3~~*5.1.1.4* Transfer to Storage Area Operations

The air pad system described in Section 5.2.1.1.~~7~~*8* is inserted under the Concrete Cask in the openings provided and inflated by standard service air compressors. The air pads, once inflated,

allow the Concrete Cask to be moved along a smooth concrete path connecting the Fuel Building area, where Concrete Cask loading with a*n* ~~PWR Basket~~ *MPC* was completed, to the ISFSI Storage Pad. Administrative controls will limit the transport speed to less than 2 ft/sec. Use of the air pad system results in blocking of the Concrete Cask air inlet flow path~~,~~*;* therefore, ~~procedure~~ *procedural* controls will be implemented to limit the time the air pads may be inserted. In the event the Concrete Cask cannot be transferred to the ISFSI storage location within the time limit for air pad system insertion, the air pads will be deflated and removed allowing restoration of air flow to the Concrete Cask.

~~A startup test~~ *Testing* to ~~confirm proper operation of the storage system~~ *measure external radiation dose and to confirm estimated personnel exposures* will be performed before the Concrete Cask is placed in service. The air pads will be placed under the Concrete Cask ~~which~~ *that* then can be moved (by use of a transport vehicle to push or pull the Concrete Cask) to the storage area. *As indicated in Figure 7.3-1,* ~~T~~*t*he loaded Concrete Casks are placed on the ISFSI Storage Pad on approximately 15-ft. center-to-center spacing~~s~~, *except for the 30-foot ±4-inch center-to-center gap in the middle of the ISFSI Storage Pad,* and/or in a configuration that supports the assumptions made in calculating the direct radiation rates of the array. Air is released from the pads so that the Concrete Cask is resting on the concrete Storage Pad surface. A loaded Concrete Cask is considered in service once it *has been placed in its designated storage location on the ISFSI pad and* has successfully passed ~~the startup test and is placed on the ISFSI pad~~*its required testing.*

### ~~5.1.1.4~~*5.1.1.5* Maintenance Operations

The Trojan ISFSI is designed to be a passive system and does not require specified maintenance tasks. ~~Recommended~~ *Certain* inspection and surveillance activities are required by the Technical Specifications. *Otherwise, maintenance is limited to minor activities.*

### ~~5.1.1.5~~*5.1.1.6* Off-Normal Event Recovery Operations

The analysis of normal and off-normal events and accident design events identified by ANSI/ANS 57.9, as applicable to the Trojan ISFSI, ~~are~~ *is* presented in Chapter 8. Each postulated event analyzed addresses both event detection and required corrective actions. Additionally, should an off-normal event occur, an inspection for possible damage will be completed within 24 hours. An engineering evaluation will also be required to establish that a component may safely continue to perform the required function.

~~Although shown by analysis not to be a credible design event, a method for recovery from a leaking PWR Basket has been developed. Recovery can be accomplished at the storage site without the benefit of a Spent Fuel Pool by implementing one of the following actions. Method a) is limited to use where the leaking weldment is accessible on top of the Concrete Cask. If the leak is not in the top weld or the leak cannot be repaired, then method b) is used.~~

~~a) After establishing the necessary radiation shielding, remove the Concrete Cask lid and locate the leaking weld area by use of a helium sniffer. The defective weld area is removed, verified by appropriate NDE, and rewelded. The finished weld area is dye penetrant checked and the PWR Basket is then purged and refilled with helium as described in the original PWR Basket loading sequence. Access to the filling and venting valves must be gained by removing the welded valve covers in the structural lid. These valve covers will be reinstalled (or replaced) per the original requirements for welding and testing.~~

~~b) Place the empty Transfer Cask into Transfer Station using a mobile crane. Install the Transfer Station collar and side members. Install the Transfer Cask door hydraulic system, and open the bottom doors. After establishing the necessary radiation shielding, remove the defective PWR Basket's cask lid, install the PWR Basket lifting rings and slings, and remove the cask shield ring. Move the Concrete Cask with defective PWR Basket into position within the Transfer Station. Lift the defective PWR Basket into the Transfer Cask, close the bottom doors, and insert locking bolts. Remove the empty Concrete Cask from the Transfer Station. Insert a Basket Overpack into the now empty Concrete Cask. Relocate the Concrete Cask containing the Basket Overpack under the Transfer Cask in the Transfer Station. Lift the defective PWR Basket slightly to remove weight from the bottom doors, remove the locking bolts, open the bottom doors, and lower the defective PWR Basket into the Basket Overpack. Remove the Concrete Cask from the Transfer Station and close the Transfer Cask bottom doors. Establish suitable radiological containment controls and remove the valve covers from the defective PWR Basket structural lid, and open the shield lid valve. Weld the Basket Overpack structural lid to the Basket Overpack shell. Install the quick connect and perform vacuum drying and helium backfill/leak testing per original loading requirements identified in Section 5.1.1.2. Weld the quick connect cover in place and perform dye penetrant examination. Reinstall the Concrete Cask shield ring and lid and reinstall the tamper wire. The Concrete Cask containing the inserted Basket Overpack is utilized for continued storage.~~

### ~~5.1.1.6~~*5.1.1.7* Off-Site Transfer Operations

The 10 CFR 71-*certified HI-STAR 100 Transport* ~~Shipping~~ Cask *(Docket 71-9261)* will be available to transport the ~~PWR Baskets~~ *MPC*s off-site to a DOE high-level waste repository or interim storage facility in the future. Transfer operations will utilize the Transfer Station described in Chapter 4 for ~~PWR Basket~~ *MPC* removal from the Concrete Cask and reinstallation in ~~a 10 CFR 71 approved Shipping~~ *the Transport* Cask.
~~Any PWR Basket sealed in a Basket Overpack must be removed from the Basket Overpack before being placed in a Shipping Cask. The procedural methods outlined in Section 5.1.1.5 b) will be utilized to position the Concrete Cask containing a Basket Overpack in the Transfer Station and to lift the PWR Basket from the Basket Overpack into the Transfer Cask. The Concrete Cask with the Basket Overpack will be removed and an empty Shipping Cask will then be positioned beneath the Transfer Cask. The PWR Basket will be lowered from the Transfer Cask into the Shipping Cask.~~

~~Note that the Basket Overpack OD (68.5") is greater than the Transfer Cask ID (67.25"+0.25", -0) and that the lifting lugs in a Basket Overpack are analyzed for only the weight of the empty Basket Overpack. Therefore, it cannot be loaded into or moved in the Transfer Cask.~~

### 5.1.2 FLOWCHARTS

Figure 5.1-1 contains an operation sequence flowchart of ~~PWR Basket~~ *MPC* loading, sealing, testing, and storage operations~~ including anticipated task completion times~~.

Figure 5.1-2 ~~contains a flowchart detailing actions and estimated task completion times for a leaking PWR Basket recovery operation utilizing the Basket Overpack~~*has been deleted.*

Figure 5.1-3 contains a flowchart detailing the actions ~~and estimated task completion times~~ for transferring a*n* ~~PWR Basket~~ *MPC* to a ~~Shipping~~ *Transport* Cask for off-site transport.

Figure 5.1-4 contains a flowchart detailing the actions ~~and estimated completion times~~ for *contingency* unloading a*n* ~~PWR Basket~~ *MPC*.

The sequence of operations is the basis for the collective dose assessment discussed in detail in Section 7.4. *The estimated completion times for the significant tasks depicted in these Figures 5.1-1, 5.1-3, and 5.1-4 are provided in Table 7.4-3.*

### 5.1.3 IDENTIFICATION OF SUBJECTS FOR SAFETY ANALYSIS

#### 5.1.3.1 Criticality Prevention

Specific techniques or operational procedures are not relied upon to assure criticality prevention. Geometrical spacing of the 24 fuel assemblies *and the use of Boral neutron absorbing plates affixed to the fuel storage cell walls* maintain~~s~~ subcritical conditions during dry storage conditions. While control assemblies may be stored integral with the fuel assemblies, they are not credited for criticality control.

#### 5.1.3.2 Chemical Safety

The Trojan ISFSI Concrete Cask system does not employ any hazardous chemicals that would require special precautions or procedures.

#### 5.1.3.3 Operation Shutdown Modes

Because the Trojan ISFSI Concrete Cask system relies on natural air circulation, it does not have any shutdown modes.

5.1.3.4 Instrumentation

The Trojan ISFSI is passive by design and requires no instrumentation to operate. The following chart lists the measuring and test equipment necessary to monitor the Trojan ISFSI for compliance to design requirements. Measuring and test equipment is not classified as important to safety. The instruments are commercially available, standard products and will be calibrated in accordance with Quality Assurance requirements.

Measuring and Test Equipment

| **Instrument** | **Function** |
|---|---|
| 1. Hand-held survey equipment (Gamma, neutron, and surface contamination) | Measures dose rates on Concrete Cask surface and contamination levels. |
| 2. Pressure and Vacuum Gauges | Measures helium, ~~nitrogen,~~ water, and vacuum pressures inside the ~~PWR Basket~~ *MPC*. |
| 3. Helium Leak Detector | Detects the presence of helium. |
| 4. Temperature monitoring devices | Measures temperature~~s~~ *rise between the ambient air and the outlet air ducts of the Concrete Cask to verify proper heat removal system operation.*<br><br>*Measures MPC gas exit temperature at the demoisturizer exit during use of the forced helium recirculation moisture removal system.* |
| 5. Seismic monitoring instrument | Measures earthquake intensity for comparison with design basis per 10 CFR 72.122(b)(3) |

5.1.3.5 Maintenance Techniques

The Trojan ISFSI does not require specified maintenance tasks. ~~Recommended~~ *Required* inspection and surveillance activities are described in the Technical Specifications. The measuring and test equipment identified in Section 5.1.3.4 will be maintained in accordance with ~~site~~ *approved* procedures which consider equipment manufacturer recommendations, as appropriate.

5.1.3.6 Heavy Loads Procedures

The handling of heavy loads will be addressed in a NUREG-0612 evaluation and in heavy loads procedures. Tests and certifications (including cranes, hooks, slings, trunnions, straps, cables, etc.) will be completed before fuel handling activities begin. Additionally, the evaluation and procedures will assure that the Trojan Fuel Building can withstand the loads from postulated

drops and that the ~~PWR Basket~~ *MPC* design ~~ac~~*de*celerations are not exceeded. Impact limiters will be used to mitigate the effects of a drop accident. Lifting within the Fuel Building is governed by the ~~Regulatory~~ *regulatory* requirements of 10 CFR 50 as defined in the Trojan Defueled Safety Analysis Report and the Decommissioning Plan. Chapter 8, Accident Analysis, also addresses drops at the ISFSI Storage Pad during handling operations.

## 5.2 SPENT FUEL HANDLING OPERATIONS

### 5.2.1 SPENT FUEL HANDLING AND TRANSFER

Spent fuel handling and transfer operations, including removal from the *Spent* ~~f~~*F*uel ~~p~~*P*ool, ~~PWR Basket~~ *MPC* loading and sealing, transfer to the ISFSI Storage Pad, and eventual transfer to an off-site location, are described in Sections 5.1.1.1 through 5.1.1.~~6~~*7*. Chapter 4 provides a description of the components and the applicable design basis utilized for safely maintaining the fuel/debris in a safe storage configuration. Specific equipment function is described in Sections 5.2.1.1.1 through 5.2.1.1.~~8~~*9*.

#### 5.2.1.1 Functional Description

The Transfer Cask, Lifting Yoke, gap flushing unit, ~~vacuum drying~~ *moisture removal* system, *helium backfill system, automated* welding system, hydraulic system, ~~and~~ air pad system, *and Transfer Station* are ~~necessary~~ *used* to facilitate ~~PWR Basket~~ *MPC* loading, storage, and eventual off-site ~~shipping~~ *transport* activities. ~~Additionally, a Transfer Station is utilized for off-normal transfer of a TranStor™ PWR Basket into a Basket Overpack or for off-site shipping.~~

##### 5.2.1.1.1 Transfer Cask

The Transfer Cask is a *welded steel cask constructed in accordance with ASME Section III, Subsection NF. Considered* special lifting device*s, the Transfer Cask lifting trunnions are* designed ~~and fabricated~~ *in accordance with the guidance* ~~to the requirements~~ of NUREG-0612 and ANSI N14.6 to be used during transfer operations with the Trojan Fuel Building overhead crane. *Transfer of the loaded MPC from the Transfer Cask into the Concrete Cask for storage occurs in the Fuel Building.* The Transfer Cask is installed into the Transfer Station at the ISFSI for ~~PWR Basket~~ *MPC* transfers *out of the Concrete Cask and into the Transport Cask in preparation for off-site transport of the MPC.*

The Transfer Cask consists of a cylinder with a steel-lead-~~neutron~~*steel gamma* shielding~~-steel~~ sandwich wall. *A water jacket used for neutron shielding when the MPC is dewatered is attached to the outer shell wall of the Transfer Cask.* The thick-walled cylinder *and water jacket* reduce~~s~~ *gamma and neutron* dose rates to an acceptable level as shown in Chapter 7. The *Transfer* ~~c~~*C*ask *top* lid extends over the ~~PWR Basket~~ *MPC* to prevent it from being inadvertently lifted out of the Transfer Cask during ~~PWR Basket~~ *MPC* transfer operations. At the bottom of the Transfer Cask are doors that slide in rails along each side of the Transfer Cask that open ~~upon hydraulic actuation~~ to allow lowering of the ~~PWR Basket~~ *MPC* into the Concrete or ~~Shipping~~ *Transport* Cask. *Mechanical stops* ~~Two steel pins~~ are used to prevent accidental opening of the *Transfer Cask bottom* doors.

5.2.1.1.2 Cask Lifting Yoke

The Transfer Cask Lifting Yoke is used for Transfer Cask handling operations in the Fuel Building. It is designed to interface with the Fuel Building overhead crane hook and is fabricated from high strength carbon steel. The Lifting Yoke may also be used to lift the empty Transfer Cask into place in the ISFSI Transfer Station. *The Lifting Yoke is considered a special lifting device in accordance with ANSI N14.6.*

5.2.1.1.3 Gap Flushing System

A gap flushing unit is used to flush filtered ~~borated~~ *Spent Fuel Pool* water through the ~~PWR Basket~~ *MPC*/Transfer Cask *annular* gap to minimize *the potential for* ~~unnecessary~~ contamination of the ~~PWR Basket~~ *MPC* external surface while the Transfer Cask is in the Cask Loading Pit.

5.2.1.1.4 ~~Vacuum Drying~~*Moisture Removal* System

Following fuel loading ~~and~~, *MPC lid* welding ~~of the shield and structural lids, a skid-mounted vacuum drying system is used to remove the water from the PWR Basket, dry the interior, and backfill it with helium. The vacuum drying system is designed to evacuate the PWR Basket to ≤ 3 mm Hg, flush the PWR Basket with helium, and repeat the process. During evacuation, the decay heat from the fuel helps to remove any residual moisture from the PWR Basket. Valves and gauges are located on the vacuum skid to control and monitor the performance of the system. The vacuum drying~~, *successful testing, and draining of the water from the MPC cavity, the residual moisture is removed from the cavity using either of two systems. A vacuum drying system may be used to reduce the pressure in the MPC cavity to a pre-determined pressure indicative of a sufficiently dry MPC to prevent corrosion of the fuel assemblies.*

*In lieu of vacuum drying, a helium recirculation moisture removal system may be used to remove residual moisture from the MPC cavity. This system is a closed loop system with appropriate piping, valves, power supplies and other components necessary to recirculate helium through the MPC via the vent and drain ports and process the helium as it absorbs the moisture in the MPC cavity. The helium recirculation* system may also be used for cooling the spent fuel inside the ~~PWR Basket~~ *MPC* as discussed in Section 5.1.1.2.

5.2.1.1.5 *Helium Backfill System*

*The helium backfill system (HBS) is used to inject helium of the required purity into the MPC cavity until the required pressure range is achieved.*

*5.2.1.1.6* ~~5.2.1.1.5~~*Automated* ~~Semi-automatic~~ Welding System

A*n automated* ~~semi-automatic~~ welding system is the preferred equipment used for ~~PWR Basket~~ *MPC* ~~and Basket Overpack~~ closure based on design, operation, and ALARA considerations. It

includes a customized welding system and adapter for mounting the system on top of the Transfer Cask.

~~5.2.1.1.6~~*5.2.1.1.7* Hydraulic System for Operation of Transfer Cask *Bottom* Doors

~~Two h~~*H*ydraulic cylinders are bolted to the outer Transfer Cask ~~wall after the Transfer Cask is placed in the position~~*when necessary* to ~~perform~~transfer *the loaded MPC* in or out of the Concrete Cask. The cylinders open the bottom doors of the Transfer Cask to allow ~~PWR Basket~~ *MPC* transfer, and *close* ~~closure of~~ the doors after completion of the operation. The doors may be opened or closed by other manual means in the event of failure of the hydraulic cylinders.

~~5.2.1.1.7~~*5.2.1.1.8* Air Pad System

The air pad system will be used to transport a loaded Concrete Cask from the Fuel Building to *its* storage location at the ISFSI Storage Pad and to the Transfer Station for ~~any required Basket Overpack installation and~~ *transfer of the MPC into the Transport Cask for* off-site shipping. The air pads are commercially available lifting devices which, when inflated, lift the cask a few inches with air pressure. The system uses *air compressor(s) and* four standard air lifting pads ~~and air compressors~~. The air pads lift the Concrete Cask approximately 3 inches and allow it to ride on a thin cushion of air. The air pad system can only be used on a smooth surface since cracks or open joints in the transport surface would result in excessive air loss and deflation of the air pads thereby preventing Concrete Cask movement. With the air pads inflated the Concrete Cask can then be moved by use of a transport vehicle to push or pull the Concrete Cask. Air pads minimize the need for handling room around the Concrete Cask, thus minimizing the size of the Storage Pad.

The air pad design limits the overall lift height of the Concrete Casks to approximately 3 inches. In the event the Concrete Cask deviates from the smooth transport pathway, air will escape from the air pads preventing movement of the Concrete Cask. As shown on Figure 8.2-3, the Concrete Cask must be inclined such that one edge of the Concrete Cask must be raised or lowered over 5 feet to result in a Concrete Cask tipover. Since the air pad design limits the lift height of the Concrete Cask to approximately 3 inches, a tipover event due to air pad failure is not considered credible.

~~5.2.1.1.8~~*5.2.1.1.9* Transfer Station

The Transfer Station is a structural steel frame designed as a stationary lateral and vertical restraint for the Transfer Cask during ~~PWR Basket~~ *MPC* transfers ~~on the ISFSI Transfer and Storage Pad~~. The transfer ~~could~~ *would* be from a Concrete Cask ~~to a Basket Overpack inside a Concrete Cask or from a Concrete Cask~~ to a ~~Shipping~~ *Transport* Cask.

The Transfer Cask is placed on a platform within the Transfer Station, with the Transfer Cask resting on the door rails. Lateral frames are then installed ~~at the approximate center of gravity of the loaded Transfer Cask~~ *to secure the Transfer Cask in the Transfer Station,* and hydraulic

cylinders and hardware for opening the *Transfer Cask* bottom doors are installed. The Transfer Cask *bottom* doors are then opened.

A Concrete Cask is then prepared for placement in the Transfer Station. The *Concrete* ~~c~~Cask lid ~~and shield ring are~~*is* removed and ~~PWR Basket~~ *MPC* rigging hardware and slings are attached to the *MPC* ~~structural~~ lid. The Concrete Cask *shield ring is removed and the Concrete Cask* is then moved under the Transfer Cask using air pads. Shielding is placed in any gaps or potential streaming paths utilizing both installed shield support frames and temporary hangers or wires.

Rigging (previously attached to the ~~PWR Basket~~ *MPC* ~~structural~~ lid) is then raised through the Transfer Cask (using a long pole) and attached to the qualified mobile crane hook. Remote engagement tools will be used where feasible to reduce exposure.

The ~~PWR Basket~~ *MPC* is raised in*to* the Transfer Cask sufficiently to clear the *Transfer Cask bottom* doors. A load cell installed on the Mobile Crane will be used *in conjunction with the Transfer Cask top lid* to prevent the ~~PWR Basket~~ *MPC* from being raised higher than the top of the Transfer Cask. The doors are closed, *mechanical stops* ~~locking bolts~~ installed, and the ~~PWR Basket~~ *MPC* lowered to rest on the *Transfer Cask bottom* doors.

Sufficient shielding is *put in place* ~~moved~~ to remove the Concrete Cask from the Transfer Station. The destination cask is then moved into position below the Transfer Cask. Shielding is *installed*~~placed~~. The ~~PWR Basket~~ *MPC* is then raised above the *Transfer Cask bottom* doors, *mechanical stops* ~~locking bolts~~ *are* removed, the *bottom* doors *are* opened, and the ~~PWR Basket~~ *MPC is* lowered into the destination cask.

Basket rigging is disengaged from the mobile crane hook and lowered to the ~~PWR Basket~~ *MPC* ~~structural~~ lid. The *loaded* cask is then moved from the Transfer Station, the ~~PWR Basket~~ *MPC* rigging is removed, and cask lid(s) or other closures are placed.

### 5.2.1.2 Safety Features

Spent fuel handling and transfer operations utilize both component designed features and administrative controls to assure that required actions are safely accomplished. The ~~PWR Basket~~ *MPC* is sheltered in the cavity of the Transfer Cask during spent fuel handling. Flushing of the ~~PWR Basket~~ *MPC*/Transfer Cask gap reduces potential contamination from the Cask Loading Pit. *A lid retention system prevents loss of the MPC lid during movement of the unsealed MPC from the Cask Loading Pit to the DAS.* The Transfer Cask *top* lid ~~with lid retainer is used to~~ prevent*s* accidental lifting of the *sealed* ~~PWR Basket~~ *MPC* out of the Transfer Cask.

Additional shielding is placed over the ~~PWR Basket~~ *MPC*/Transfer Cask gap and temporary shielding is used, *as required,* to minimize worker exposure. The ~~PWR Basket~~ *MPC* storage sleeves are designed to withstand a postulated drop accident from the Transfer Cask without significant damage to the stored fuel (~~Ref:~~ Section 8.2.13.3.~~2~~*3*). The Transfer Cask has *mechanical stops* ~~steel pins~~ in the *Transfer Cask* bottom doors to prevent them from being

inadvertently opened. A load cell will be used to *prevent* ~~limit~~ the mobile crane from raising the ~~PWR Basket~~ *MPC* higher than that analyzed by stopping the crane when the ~~top of the~~ Transfer Cask *top lid* is impacted by the ~~PWR Basket~~ *MPC*.

## 5.2.2 SPENT FUEL STORAGE

The major components of the Trojan ISFSI system used for spent fuel storage are the ~~PWR Basket~~ *MPC* and Concrete Cask. Design characteristics for each are described in Chapter 4. Operational performance is described in Sections 5.1.1.1 and 5.1.1.2. Removal of stored spent nuclear fuel from the Trojan ISFSI for off-site shipment is described in Section 5.1.1.~~6~~7.

### 5.2.2.1 Inspection and Surveillance Program

The inspections and surveillances required for the Trojan ISFSI are contained in the Technical Specifications.

### 5.2.2.2 Safety Features

Spent fuel storage at the Trojan ISFSI utilizes the inherent safety features of a passive dry cask design as well as additional administrative controls. Fuel assemblies with higher radiological source terms will be loaded toward the center of each ~~PWR Basket~~ *MPC* and lower radiological source term fuel will be loaded near the outer periphery in order to minimize dose rates. In storage, the ~~PWR Basket~~ *MPC* is sheltered in the cavity of the Concrete Cask that reduces the surface dose to ~~well~~ within allowable limits as demonstrated in Chapter 7. Additional shielding, in the form of a ring, is placed over the gap between the Concrete Cask and ~~PWR Basket~~ *MPC* after the transfer is complete. Although no credible event can overturn the Concrete Cask, the ~~PWR Basket~~ *MPC* and Concrete Cask are designed to withstand a postulated tipover without damaging stored fuel or breaching the confinement boundary. Other safety features (as related to off-normal events) are discussed in Chapter 8, Accident ~~Analyses~~*Analysis*.

## 5.3 OTHER OPERATING SYSTEMS

### 5.3.1 SYSTEM OPERATIONS

The Trojan ISFSI storage system is passive and does not require any systems for its operation once it is placed into storage. The only equipment required for this storage system, besides the storage structures, is for fuel loading and movement of the casks to the ISFSI as described in Section 5.2.1.

### 5.3.2 COMPONENT/EQUIPMENT SPARES

The Trojan ISFSI system is designed to withstand postulated design basis events as described in Chapter 8~~,~~; therefore, no equipment spares are needed.

## 5.4 SUPPORT SYSTEM OPERATION

### 5.4.1 INSTRUMENTATION AND CONTROLS

The operation of the Trojan ISFSI is passive and self-contained and, therefore, does not need any control systems. Temperature monitoring devices are used for measuring *the temperature rise between ambient and Concrete* ~~c~~Cask air outlet ~~temperature~~*to verify adequate heat removal is occurring in the cask system.*

### 5.4.2 SPARES

Other than the temperature monitoring devices identified in Section 5.4.1, no instrumentation or control systems are required for operation of the Trojan ISFSI. Since the devices are readily available commercial components, equipment spares are not required.

### *5.4.3* ~~5.5~~ CONTROL ROOM AND CONTROL AREAS

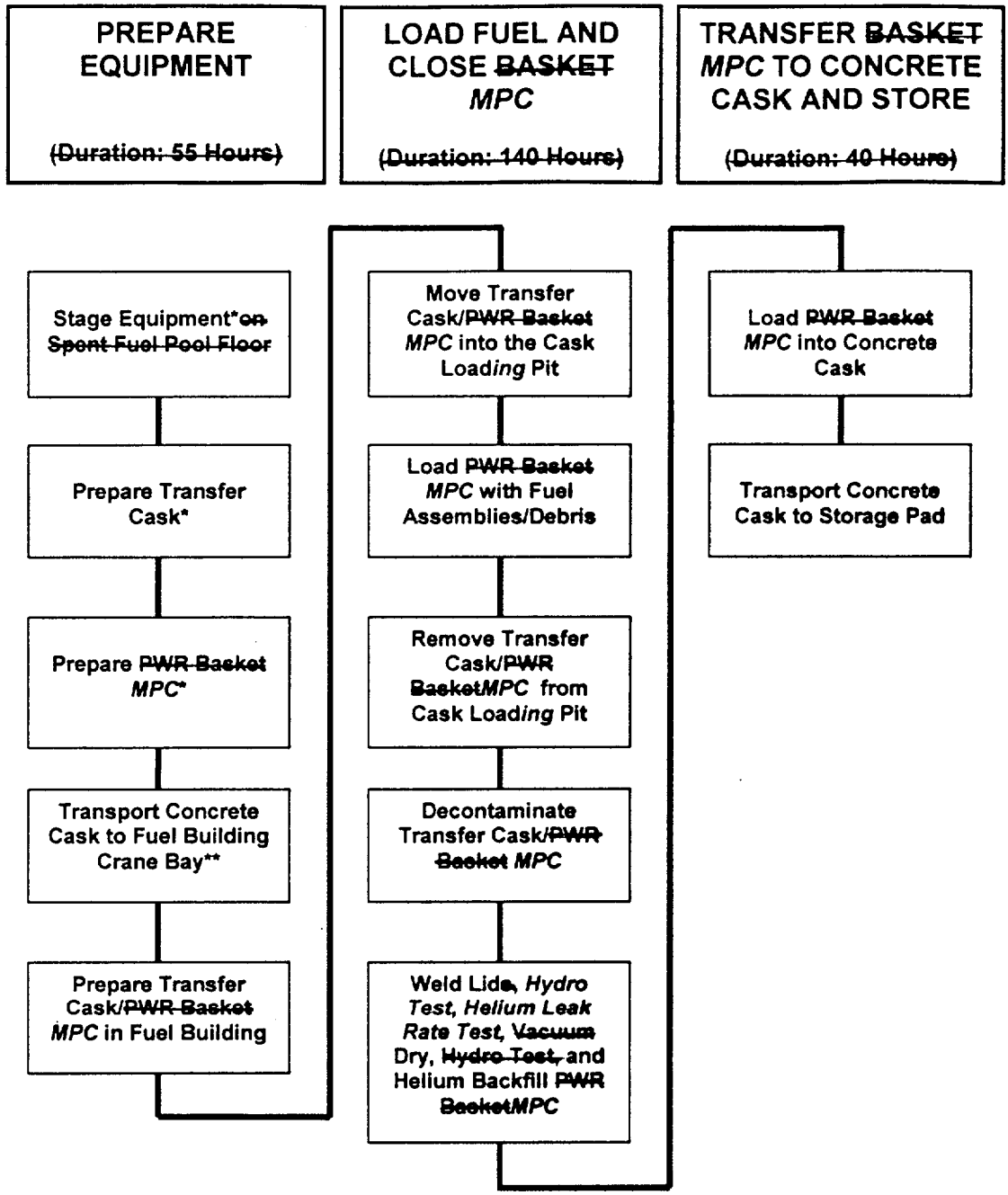
No control room or control areas are needed for the Trojan ISFSI.

### *5.4.4* ~~5.6~~ ANALYTICAL SAMPLING

No sampling or analysis is necessary to ensure safe operation of the Trojan ISFSI.

## *5.5* ~~5.7~~ REFERENCES

*1.* ~~BNFL Fuel Solutions Calculation PGE01-10.2.04-08, Revision 4, "TranStor™ Transfer Cask Water Heat-up Analysis." Calculation Number TI-035, Revision 4.~~ *Final Safety Analysis Report for the Holtec International HI-STORM 100 Dry Cask Storage System, Holtec Report No. HI-2002444, Revision 0, July 2000, through Proposed Revision 1F.*




* Activities may be performed in parallel

** Must be completed prior to "Load ~~PWR Basket~~ *MPC* into Concrete Cask"

**TROJAN ISFSI**
**SAFETY ANALYSIS REPORT**

**FIGURE 5.1-1**
**OPERATIONS SEQUENCE (TYPICAL)**
**Revision 2**

Figure 5.1-2 Deleted

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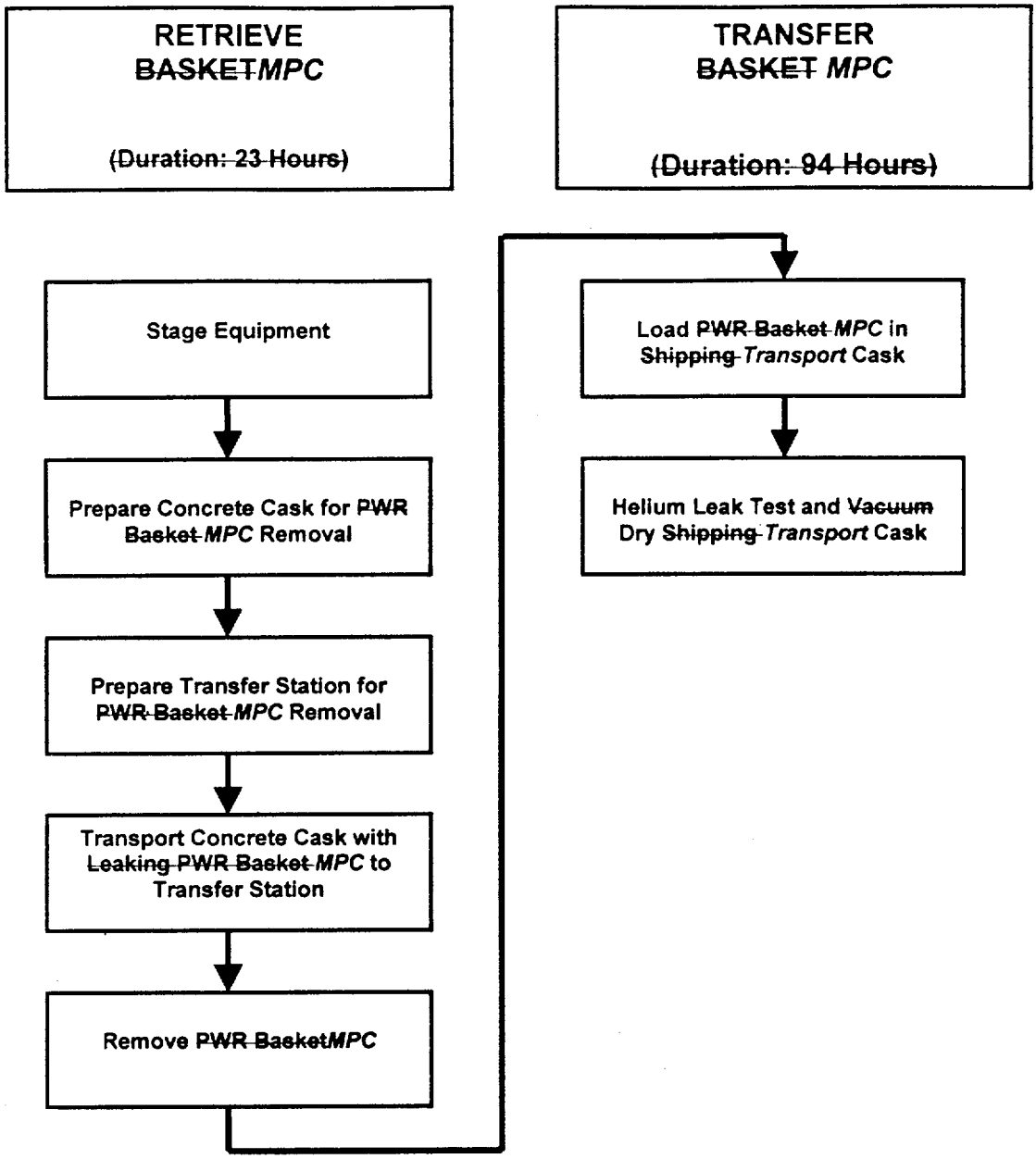
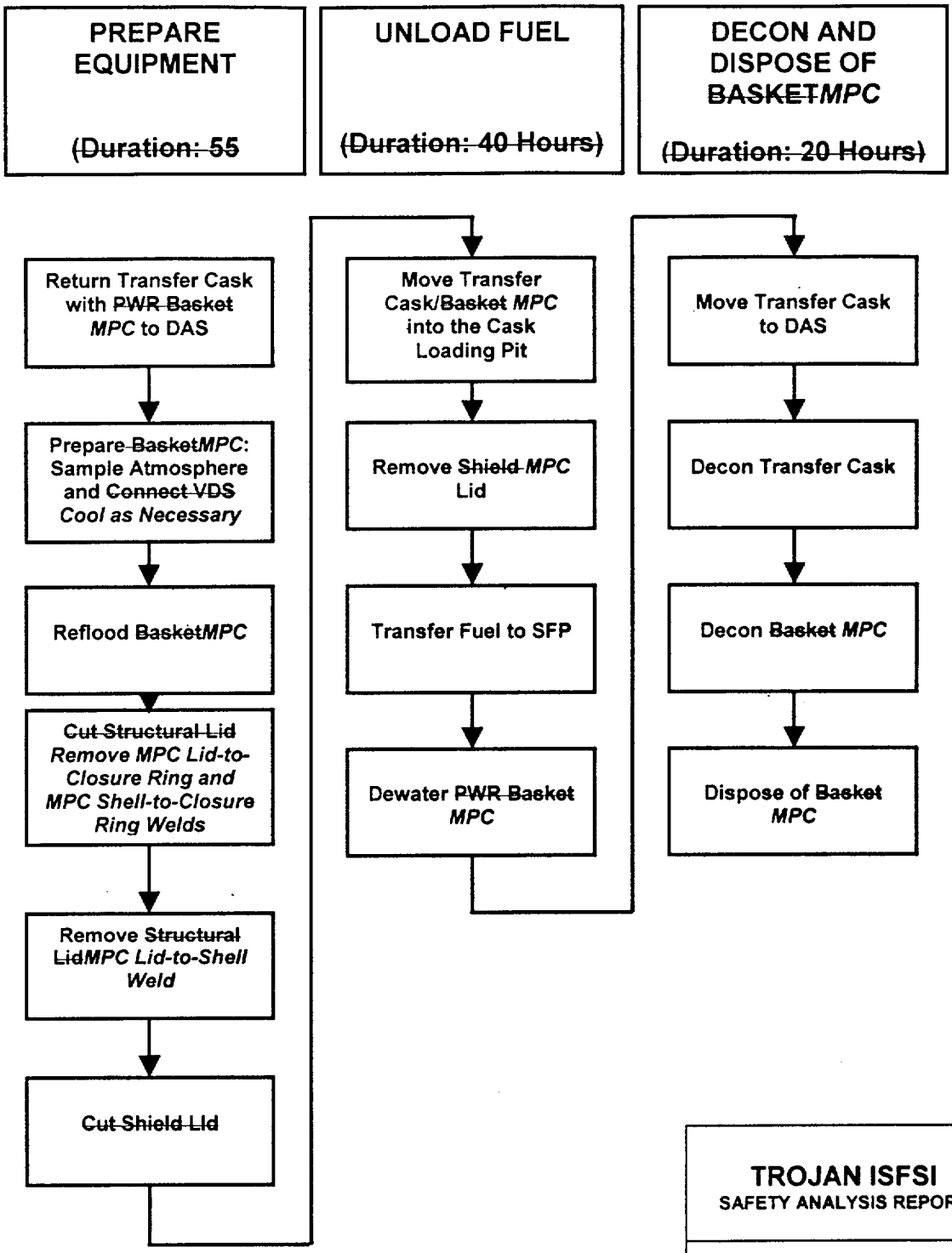



**TROJAN ISFSI**
SAFETY ANALYSIS REPORT

**FIGURE 5.1-3**
~~SHIPPING~~ *TRANSPORT* CASK OPERATION SEQUENCE (TYPICAL)

Revision 2




**TROJAN ISFSI**
SAFETY ANALYSIS REPORT

**FIGURE 5.1-4**
PWR BASKET *MPC* *CONTINGENCY* UNLOADING SEQUENCE (TYPICAL)

Revision 2

# Chapter 6

# 6.0 SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT

## 6.1 ONSITE WASTE SOURCES

Fuel loading and closure of the ~~PWR Basket~~*MPC* will be accomplished under the existing Trojan Nuclear Plant 10 CFR 50 license and the provisions of the ISFSI Technical Specifications. The radioactive waste generated during fuel loading and closure of each ~~PWR Basket~~*MPC* ~~would~~ *will* consist mainly of potentially contaminated water used for decontamination of the Transfer Cask and ~~PWR Basket~~*MPC lid,* and a small amount of solid waste, (e.g., absorbent rags, gap flush resin) used for decontamination. These wastes ~~would~~ *will* be processed using existing Trojan Plant procedures and systems. Gaseous waste is not anticipated, but the system used to ~~pump~~ *blow* down and ~~vacuum dry~~*remove the residual moisture from* the ~~PWR Basket~~*MPC* will be designed to filter ~~or capture~~ gaseous waste as required.

Normal operation of the Trojan ISFSI will not generate radioactive waste. The spent nuclear fuel is ~~sealed~~ *contained* inside a leak-tight stainless steel ~~PWR Basket~~*MPC* and the normal surveillance and inspection activities do not affect the confinement capability of the ~~PWR Basket~~*MPC*.

## 6.2 OFFGAS TREATMENT AND VENTILATION

During the fuel loading and ~~PWR Basket~~*MPC* closure process, the ~~PWR Basket~~*MPC* that contains the spent nuclear fuel is ~~vacuum~~ dried and backfilled with helium while located inside the Trojan Fuel Building. Any radioactive gas that is drawn off the ~~PWR Basket~~*MPC* during the ~~vacuum~~ drying process is passed through a system designed to filter ~~or capture~~ gaseous waste as required.

Gaseous radioactive waste is not generated by normal operation of the Trojan ISFSI. Any radioactive gases that are released from the spent nuclear fuel during the storage period will remain inside the leak-tight ~~PWR Basket~~*MPC*. The ~~PWR Basket~~*MPC* is designed to remain sealed while stored at the ISFSI, but if opening the ~~PWR Basket~~*MPC* is required during the storage period, a venting system ~~would~~ *will* be used that ~~would~~ filter*s* ~~or capture~~ radioactive gases as required.

## 6.3 LIQUID WASTE TREATMENT AND RETENTION

During the fuel loading and ~~PWR Basket~~*MPC* closure process, potentially contaminated liquid will be generated while decontaminating the Transfer Cask and ~~PWR Basket~~*MPC lid*. This potentially contaminated liquid will be collected and processed by existing Trojan Nuclear Plant procedures and systems under the 10 CFR 50 license.

Liquid radioactive waste is not generated by normal operation of the Trojan ISFSI. The ~~PWR Basket~~*MPC* is ~~vacuum-~~dried during the closure process. Therefore, there is no liquid in the leak-tight ~~PWR Basket~~*MPC* during the storage period.

## 6.4 SOLID WASTES

During the fuel loading and ~~PWR Basket~~*MPC* closure process, a small amount of solid low level waste, (e.g., absorbent rags, gap flush resin) will be generated while decontaminating the Transfer Cask and ~~PWR Basket~~*MPC lid.* This solid waste will be collected and processed using existing Trojan Nuclear Plant procedures under the 10 CFR 50 license. Solid waste generated during the ~~PWR Basket~~*MPC* ~~pump down and vacuum drying~~*closure* process~~,~~ will be captured and processed by existing Trojan Nuclear Plant procedures and systems under the 10 CFR 50 license.

Solid radioactive waste is not generated by normal operation of the Trojan ISFSI because any solid waste will remain inside the leak-tight ~~PWR Basket~~*MPC* during the storage period. No contamination is anticipated at the ISFSI, but periodic surveys are performed during normal operation to confirm that there is no contamination. In the unlikely event that contamination is discovered, a small amount of radioactive waste, e.g., swipes and absorbent rags, would be collected and processed as low level waste.

## 6.5 RADIOLOGICAL IMPACT OF NORMAL OPERATIONS - SUMMARY

Gaseous, liquid or solid radioactive waste ~~are~~ *is* not generated during normal operation of the Trojan ISFSI. Although no leakage from the ~~PWR Baskets~~*MPCs* is expected during normal operations, bounding calculations have been performed to assess the impact of the maximum hypothetical ~~PWR Basket~~*MPC* leakage rates. This analysis is described in Section 7.2.2.

The radioactive waste generated during the loading and closure of the ~~Baskets~~ *MPCs* is collected and processed under the 10 CFR 50 license using existing Trojan Nuclear Plant procedures and systems.

# Chapter 7

# 7.0 RADIATION PROTECTION

## 7.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

### 7.1.1 POLICY CONSIDERATIONS

The Radiation Protection Program used for operating the ISFSI implements the regulatory requirements of 10 CFR 20, "Standards for Protection Against Radiation," 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," and Oregon Administrative Rule (OAR) 345-26-390, through approved procedures.

The primary objective of the Radiation Protection Program is to maintain exposures to workers, visitors, and the general public As Low As Reasonably Achievable (ALARA). The dry storage system that will be used at the Trojan ISFSI has been designed with the principles of ALARA considered for the operation, inspection, maintenance, and repair of the system. PGE provides or will contract qualified staff, facilities, and equipment to ensure that exposures are ALARA during operation of the ISFSI. The ISFSI will be monitored and evaluated on a routine basis to maintain radiation exposures to unrestricted areas ALARA.

Specific design and operation oriented ALARA considerations are described in the following sections.

### 7.1.2 DESIGN CONSIDERATIONS

The Concrete Cask system is designed to ensure that occupational radiation exposures are ALARA as defined in 10 CFR 20. As such, special design considerations have been taken to ensure exposure rates are ALARA. These considerations include:

1. Thick walls and lids that contain various shielding materials;
2. Totally passive system that requires minimum maintenance;
3. Welded closures that provide redundant radioactive material confinement;
4. Non-planar paths for the air inlets and outlets to minimize radiation streaming; and
5. Spacing of the Concrete Casks on the Storage Pad to provide self-shielding for interior Concrete Casks.

Detailed descriptions of the Concrete Cask system components are included in Chapter 4.

### 7.1.3 OPERATIONAL CONSIDERATIONS

ISFSI operational details including ~~PWR Basket~~*MPC* loading, closure, transfer, storage, and off-site shipping are included in Chapter 5. Loading and transfer operations have been determined following ALARA guidelines. ISFSI personnel will follow site procedures consistent with Regulatory Guide 8.8 and Regulatory Guide 8.10. Personnel radiation exposure during handling and closure of the ~~PWR Basket~~*MPC* is minimized by the following steps.

1. Fuel loading procedures that follow accepted practice and build on existing experience;

2. Loading spent nuclear fuel in the ~~PWR Basket~~*MPC* within the controlled environment of the Fuel Building to prevent the spread of contamination;

3. Loading the most radioactive fuel in interior ~~PWR Basket~~*MPC* positions;

4. Injecting filtered~~, borated~~ water into the annulus between the Transfer Cask and ~~PWR Basket~~*MPC* to minimize contamination of the ~~PWR Basket~~*MPC* external surface;

5. Placing the ~~shielding~~ lid on the ~~PWR Basket~~*MPC* while the Transfer Cask and ~~PWR Basket~~*MPC* remain in the Cask Loading Pit;

6. Decontaminating the exterior of the Transfer Cask and welding the ~~PWR Basket~~*MPC* lid while the ~~PWR Basket~~*MPC* is still filled with water;

7. Draining the ~~PWR Basket~~*MPC* while still housed in the Transfer Cask;

8. Using portable shielding as necessary;

9. Using the shielded Transfer Cask that is remotely operated to transfer the ~~PWR Basket~~*MPC* to the Concrete Cask;

10. Placing a shielding ring over the annular gap between the Concrete Cask and ~~PWR Basket~~*MPC*;

11. Swiping the Concrete Cask exterior for contamination prior to leaving the Fuel Building;

12. Storing higher dose rate Concrete Casks toward the northeast section of *the* ISFSI Storage Pad to minimize dose rates at the normally occupied areas to the south and west of the Storage Pad;

13. Using ALARA pre-job briefings prior to fuel movement and Concrete Cask loading sequence; and

14. Use of mock-ups for training.

## 7.2 RADIATION SOURCES

### 7.2.1 CHARACTERIZATION OF SOURCES

The *design basis radiation source terms for the shielding analysis of* ~~general criteria for the radioactive material to be stored in~~ the Trojan ISFSI are provided in Table 3.1-2. *The burnup of 42,000 MWD/MTU bounds all fuel assemblies to be stored in the Trojan ISFSI.* ~~Shielding analyses were performed for two design basis cases, 40,000 MWd - 5 year cooled fuel, and 45,000 MWd - 6 year cooled fuel. The entire Trojan spent fuel inventory is bounded by these two cases with respect to burnup level and cooling time (Reference 4). For each case,~~ *Since the final power operation occurred in November 1992, a cooling time of nine years bounds the Trojan spent fuel inventory for storage after November 2001. The entire Trojan spent fuel inventory is therefore bounded by the 42,000 MWD/MTU burnup and the nine-year cooled case. A* lower bound initial enrichment ~~levels are~~ *of 3.09 wt% U-235 is* assumed, since this will yield the maximum gamma and neutron source terms for ~~fuel of a given~~ *the 42,000 MWD/MTU* burnup level. *This enrichment is a conservatively low enrichment that bounds the Trojan spent fuel inventory at this burnup level.*

*The values in Table 3.1-2 and the analysis reported in this chapter are based on a Westinghouse 17x17 fuel assembly. Table 3.1-1 indicates that the Westinghouse 17x17 has a higher Uranium mass compared to the B&W 17x17 assembly. Therefore, the Westinghouse 17x17 will have a higher radiation source term than the B&W 17x17 for the same enrichment, burnup, and cooling time combinations.*
~~Five distinct radiation sources are modeled by the shielding analyses. The active fuel region of the assembly contains two sources, the fuel region gamma source and the fuel region neutron source. In addition, three non-fuel region gamma sources are modeled, the bottom nozzle region gamma source, the gas plenum region gamma source, and the top nozzle region gamma source. Each of these three gamma sources is almost entirely due to Co-60 activity from activated steel. Each of the above sources is described in the following subsections.~~

~~The radiation source data used in the shielding analyses are taken from the Office of Civilian Radioactive Waste Management (OCRWM) spent fuel computer database (Reference 6). This database gives radiation source strengths and spectra for spent fuel (on a per MTU of fuel basis) as a function of burnup level, cooling time, and initial enrichment. The database also gives sufficient additional data to calculate the gamma source terms for the non-fuel assembly regions.~~

*The principal sources of direct radiation from the Concrete Cask are:*

1. *Gamma radiation originating from the following sources*
   - *Decay of radioactive fission products*
   - *Secondary photons from neutron capture in fissile and non-fissile nuclides*
   - *Hardware activation products generated during power operations*

2. *Neutron radiation originating from the following sources*
   - *Spontaneous fission*
   - *Alpha,neutron ($\alpha$,n) reactions in fuel materials*
   - *Secondary neutrons produced by fission from subcritical multiplication*
   - *Gamma,neutron ($\gamma$,n) reactions (this source is negligible)*

*The foregoing can be grouped into three distinct sources, each of which is discussed below: fuel gamma source, fuel neutron source, and non-fuel hardware Co-60 source. The source terms for the analyses presented in this SAR were calculated using the same methods described in the HI-STORM 100 System FSAR. The neutron and gamma source terms, along with the quantities of radionuclides available for release, were calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.4 system (References 1 and 2) (the HI-STORM FSAR used the SCALE 4.3 system).*

### 7.2.1.1 Fuel Gamma Source

*Table 7.2-2 presents the gamma source terms that were used for the active fuel portion of the design basis fuel assembly for the shielding analyses. The source is presented in both MeV/sec and photons/sec for an energy range of 0.45 MeV to 3.0 MeV. The HI-STORM 100 System FSAR (Reference 3) states: "Photons with energies below 0.45 MeV are too weak to penetrate the HI-STORM overpack or HI-TRAC, and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose." Since the Concrete Cask and HI-STORM 100 overpack are similar in their shielding characteristics (concrete and steel shielding) this conclusion is also valid for the Concrete Cask. Since the Trojan Transfer Cask design is similar to the Holtec HI-TRAC 100, this conclusion is also valid for the Trojan Transfer Cask.*~~The fuel gamma source is comprised of 163 principle fission product radionuclides, several activation products, and actinide radionuclides present within the $UO_2$ fuel. The radionuclides that are the major contributors to the fuel source term are listed in Table 7.2-1.~~

~~The fuel region gamma source is modeled as a homogenous volumetric source which completely fills the PWR Basket interior over the axial span that contains active fuel. The gamma source is evenly distributed throughout this defined volume.~~ The fuel region gamma source is based upon 24 complete *design basis Westinghouse 17x17* PWR fuel assemblies~~, each with the maximum (initial) uranium loading of 0.469 MTU~~. Using the same term for complete fuel assemblies in the model bounds the source term for partial fuel assemblies (~~failed~~ *damaged* fuel) and fuel debris because a complete fuel assembly has a higher source term.

~~The fuel region gamma sources for each design basis case are listed in Table 7.2-2. Gamma source strengths are shown for both of the bounding burnup and cooling time cases. For each of the two cases, minimum initial enrichment levels are conservatively assumed. An initial enrichment level of 3.02% U-235 is assumed for the 40,000 MWd case, and an initial enrichment~~

~~of 3.3% U-235 is assumed for the 45,000 MWd case. These are conservative lower bound enrichment levels which bound the entire Trojan spent fuel inventory (i.e., Trojan fuel of a similar burnup level has a higher enrichment level).~~

~~The gamma source strength (in gammas/sec-cask) in Table 7.2-2 is given for each gamma energy line. The gamma source strengths from the OCRWM computer database for each energy line, in units of gammas/sec-MTU, are multiplied by the fuel loading of 0.469 MTU/assembly and the number of assemblies (24) to determine the source strength for each energy line in units of gammas/sec-cask. The gamma source strengths for each energy line are added together to yield the total gamma source strengths per cask shown in Table 7.2-2.~~

~~The 1.25 MeV gamma source strengths shown in Table 7.2-2 include the additional gamma source from activated control components which may be inserted into the assemblies. The additional gamma source comes from activated control component stainless steel cladding. The OCRWM database gives the active fuel region Co-60 activity (in Ci) per MTU of fuel as a function of burnup, cooling time, and enrichment. It also gives the total amount of cobalt initially present in the assembly per MTU of fuel. The active fuel region cobalt activation level (Ci of Co-60 per gram of initial cobalt) can be determined from this data. The total amount of cobalt present in the control component is then determined from the total stainless steel mass in the control component cladding and the maximum cobalt concentration in stainless steel. This cobalt mass is multiplied by the cobalt activation level (described above) to yield a total active fuel region Co-60 activity due to the control component cladding. This Co-60 gamma source is then collapsed onto the nearest fuel source gamma energy line (the 1.25 MeV line) and added to the source strength output for the spent fuel. The resulting 1.25 MeV gamma source strength is that shown in Table 7.2-2.~~

The axial burnup profile present in the active fuel (Reference ~~8~~*4*) is modeled in the shielding analyses. This profile, which has a peak to average ratio of 1.1, is shown in Figure 7.2-1. The peak to average ratio of the gamma source strength profile is also 1.1 because the gamma source strength is roughly directly proportional to the fuel burnup level. The gamma shielding analyses account for the peak axial profile in the dose rate calculations~~ on the Concrete Cask and Transfer Cask sides. The dose rates calculated on the cask ends are based upon the flat source distribution, and conservatively neglect the effects of the axial burnup profile~~.

### 7.2.1.2 Fuel Neutron Source

Neutron sources are based on spontaneous fission sources from various actinides and alpha, neutron (α,n) reactions. The primary neutron source is the spontaneous fission of $Cm^{244}$-*244*. ~~The total~~*Table 7.2-3 presents the* neutron source *term used* ~~strength for each of the two burnup cases (per MTU of fuel) is taken from the OCRWM computer database. These sources are based on the same initial enrichments assumed~~ for the active fuel~~ region gamma source~~. ~~These neutron sources are multiplied by the assembly uranium loading of 0.469 MTU and the cask capacity of 24 (assemblies) to yield the per cask total neutron source strengths shown in Table 7.2-3.~~

~~The neutron source strengths shown for each neutron energy group were obtained by multiplying the total neutron source strengths by the normalized neutron spectrum shown in the right column of Table 7.2-3. This normalized spectrum is assumed in all of the shielding analyses. The spectrum, which is similar to the spontaneous fission spectrum of $Cf^{252}$, is taken from the Multipurpose Canister Sub-System Design Procurement Specification (Reference 9).~~

The neutron source strength has highly non-linear dependence upon fuel burnup level~~. Surveys of OCRWM computer database neutron source strength outputs show that the neutron source strength~~ *and* is roughly proportional to the burnup level raised to the 4.2 power. Due to the strong non-linear dependence of the neutron source strength on burnup level, the axial burnup profile of the fuel causes the total neutron source strength to increase by 13.875~~%~~ *percent* over the neutron source strength calculated at the assembly average burnup level presented in Table 7.2-3. *This increase was accounted for in the neutron shielding analysis.* Sub-critical neutron multiplication causes a further ~~30.8%~~ increase in the total neutron source strength presented in Table 7.2-3. ~~The increases in total neutron source strength were taken into account in the neutron shielding analyses.~~*This increase was automatically accounted for in the Monte Carlo N-Particle (MCNP) calculations.*

Due to the non-linear dependence, the 10~~%~~ *percent* axial burnup peaking factor yields an axial peaking factor of ~~~~*approximately* 50~~%~~ *percent* for the neutron source strength. The neutron shielding analyses explicitly modeled the axial neutron source profile which results from the axial burnup profile. This was done by dividing the active fuel region into several small axial subsections, each with a different neutron source strength. These axial subsections are described in Table 7.2-4. The axial span (measured from the bottom of the active fuel region) is shown for each subsection, along with the relative fuel burnup level and the relative neutron source strength. These values are relative to the assembly average burnup level and the neutron source strength calculated for the assembly average burnup level.

### 7.2.1.3 Non-Fuel Region Gamma Sources

*As mentioned above, the non-fuel hardware of a fuel assembly (e.g., steel and inconel in the end fittings) become activated during in-core operations to produce a radiation source. The primary radiation from these portions of the fuel assembly is Co-60 activity. Radiation from other isotopes within the steel and inconel has a negligible impact on the radiation dose rate compared with the Co-60 activity. Therefore, Co-60 was the only isotope considered in the analysis. The method used to calculate the activity in the non-fueled regions of the assembly is fully described in Section 5.2.1 of the HI-STORM 100 System FSAR. Consistent with the analysis in the HI-STORM FSAR, the Co-59 impurity level assumed in the steel and inconel of the fuel assembly was 1.0 gm/kg or 1000 ppm. It was also assumed for this analysis that the fuel assemblies contained non-zircaloy grid spacers with a Co-59 impurity level of 1.0 gm/kg.*

*Table 7.2-5 presents the mass data for the non-fuel parts of the Westinghouse 17x17 fuel assembly that were used in the source term calculations. The gas plenum region was split into two separate regions for modeling considerations. As stated above, a Co-59 impurity level of 1.0 gm/kg (0.1 wt%) was used for both inconel and stainless steel. Therefore, there is little distinction between stainless steel and inconel in the source term generation and since the shielding characteristics are similar, stainless steel was used in the shielding calculations instead of inconel.*
~~The gamma sources for the three non-fuel assembly regions (the bottom nozzle region, the gas plenum region, and the top nozzle region) are almost entirely due to Co-60 in activated metal components. The Co-60 activity for each of these non-fuel assembly regions can be determined, as a function of assembly burnup level, cooling time, and initial enrichment, from data available in the OCRWM spent fuel computer database (Reference 6).~~

~~As discussed in Section 7.2.1.1, the OCRWM database gives the active fuel region Co-60 activity (in Ci) and the total fuel region initial cobalt inventory (in grams) per MTU of fuel. This allows the core region cobalt activation level (Ci Co-60 per gram initial cobalt) to be determined as a function of burnup level, initial enrichment, and cooling time.~~ During reactor operation, neutron flux levels in the non-fuel regions of the assembly (above and below the core) are much lower than those present in the core region. Thus, the Co-60 activation level is expected to be much lower for the non-fuel regions. Cobalt activation level adjustment factors, *consistent with those used in the HI-STORM FSAR,* have been determined for ~~each of the three~~ non-fuel *material*~~assembly regions (Reference 10)~~, *and are presented in Table 7.2-6. The Co-60 activity levels in the non-fuel material of the fuel assembly calculated using these flux adjustment factors are presented in Table 7.2-7.*

*In addition to the non-fuel material on the fuel assembly, Burnable Poison Rod Assemblies (BPRAs), Rod Cluster Control Assemblies (RCCAs), and Thimble Plug Devices (TPDs) were considered in the analysis. BPRAs were only used in the first cycle of operation at TNP and analysis has demonstrated that the radiation source from these devices do not contribute noticeably to the radiation dose rates outside the Concrete Cask. As a result these devices were not considered in the analyses reported in this chapter. The TNP typically operated with RCCAs fully withdrawn. Therefore, the activation of these devices was limited to the extreme lower portion of the devices. Analysis has determined that the radiation source from these devices does not contribute noticeably to the radiation dose rates outside the Concrete Cask. Therefore, these devices were not considered in the analysis in this chapter.*

*The TPDs were used at TNP for all cycles except the last. These devices, which are positioned in the top of the fuel assembly during normal in-core operations and during normal storage conditions, are significantly activated. Since these devices are positioned in the upper portion of the fuel assembly during storage, the radiation source from these devices contributes noticeably to the dose rates at the top of the Concrete Cask. Therefore, these devices were considered in the shielding analysis. SAS2H and ORIGEN-S were used to calculate the Co-60 activity from the activation of the steel and inconel in these devices. The HI-STORM FSAR used a Co-59 impurity*

*level of 0.8 gm/kg and 4.7 gm/kg for the steel and inconel, respectively, in the analysis of TPDs. The same impurity levels were used in this analysis. The accumulated burnup for these devices was 118,674 MWD/MTU. Based on the end date for Cycle 13, the minimum cooling time for these devices is 11 years. The downtime between cycles was accounted for in the calculation of source term for these devices . Table 7.2-8 shows the mass of the materials that were used for the TPDs. Table 7.2-9 shows the calculated Co-60 source term for the TPDs.*

~~The initial grams of cobalt in each assembly non-fuel region is determined using the OCRWM database. The core region Co-60 activity per initial grams of cobalt is multiplied by the activation level adjustment factor, for the appropriate region, to yield the Co-60 activity per initial grams of cobalt for the assembly non-fuel region.~~

~~The Co-60 activity for each region is obtained by multiplying the Co-60 activity per initial grams of cobalt for the region by the initial grams of cobalt in that region.~~

~~These per assembly Co-60 activities are multiplied by 24 to yield cask total Co-60 activities for each non-fuel region of the PWR Basket. These Co-60 activities are then converted into 1.173 MeV and 1.333 MeV gamma source strengths (a Co-60 decay emits one 1.173 MeV gamma and one 1.333 MeV gamma). These total (per cask) non-fuel region gamma source strengths are shown in Table 7.2-5 for each of the three non-fuel regions and for each of the two fuel burnup cases. In the shielding models, each of these non-fuel region gamma sources is evenly distributed throughout the corresponding axial subsection of the PWR Basket interior.~~

~~7.2.1.4Greater Than Class C Waste~~

~~This section has been deleted.~~

~~7.2.1.5~~*7.2.1.4* Fuel Debris

~~The PGE fuel debris consists of individual fuel pellets and fragments from damaged fuel rods~~. For the shielding analysis, fuel debris source terms *from individual fuel pellets and fragments from damaged fuel rods* are conservatively assumed to be the same as for intact fuel. This assumption is conservative because the fuel debris will be stored in Failed Fuel Cans, separate from intact fuel, and the total quantity of fuel debris is only a few kilograms, as compared to an intact fuel assembly with several hundred kilograms of fuel material.

~~7.2.1.6~~*7.2.1.5* Non-Fuel Bearing Components

~~In addition to failed fuel, f~~*F*uel assembly hardware, non-fuel bearing components, and one fuel skeleton will also be stored. These components are made of 304 stainless steel, zirconium IV, and Inconel. The source terms from these additional components were not independently considered in the shielding calculations, but the fuel source terms would bound this additional waste.

## 7.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

Loading of spent nuclear fuel and other wastes into the ~~PWR Basket~~*MPC* is carried out under water in the Spent Fuel Pool Cask Loading Pit which prevents the spread of contamination. The ~~PWR Baskets~~*MPCs* are dried and sealed within the controlled environment of the Fuel Building. The gaseous waste from the ~~PWR Baskets~~*MPCs* will be passed through a local HEPA filter.

Once the ~~PWR Basket~~*MPC* is dried and seal welded, there are no credible off normal events or accidents that will cause breaching of the ~~PWR Basket~~*MPC* and subsequent release of airborne radioactivity. Therefore, no airborne releases to the environment from the spent nuclear fuel assemblies are expected to occur during loading~~, and~~ handling, *or storage* operations. *This discussion notwithstanding, analyses have been performed to calculate the dose to an individual at the Trojan ISFSI Controlled Area boundary due to an effluent release under off-normal and hypothetical accident conditions. These analyses are presented in Sections 8.1.3, 8.1.4, and 8.2.1.*

### *7.2.2.1 MPC Surface Contamination*

During normal operation of the ISFSI, a potential source of airborne radioactivity is from surface contamination on the ~~PWR Basket~~*MPC* exterior, which would be deposited there from the Spent Fuel Pool water. As discussed in Chapter 5, filtered~~, borated~~ water is injected into the Transfer Cask/~~PWR Basket~~*MPC* annulus to prevent contamination of the outside surface of the ~~PWR Basket~~*MPC* when it is submerged in the Cask Loading Pit and the Transfer Cask is washed down after being removed from the Spent Fuel Pool to remove potential contamination. The exterior of the ~~PWR Basket~~*MPC* will be checked for loose surface contamination while the ~~PWR Basket~~*MPC* is in the Transfer Cask to the extent possible because the ~~PWR Basket~~*MPC* surface is not readily accessible while the ~~PWR Basket~~*MPC* is in the Transfer Cask. As a second check, the interior of the Transfer Cask will be checked for loose surface contamination after the ~~PWR Basket~~*MPC* is removed because the interior surface of the Transfer Cask would be representative of the loose surface contamination on the exterior surface of the ~~PWR Basket~~*MPC* that was just inside the Transfer Cask. A limit of $10^{-4}$ μCi /cm$^2$ beta-gamma and $10^{-5}$ μCi /cm$^2$ alpha will be used and if loose surface contamination is above this limit, the need to decontaminate the ~~PWR Basket~~*MPC* will be evaluated. These limits were used in an analysis of a release of radioactive particulates from the ~~PWR Basket~~*MPC* surface while in storage on the Storage Pad. This analysis, which is described in ~~Chapter 8~~*Section 8.1.3*, shows that the consequences of the postulated release are negligible.

*As stated above,* Chapter 8 also describes the postulated leakage from ~~PWR Baskets~~*MPC* ~~and~~ ground level releases. Postulated ~~PWR Basket~~*MPC* leakage is evaluated for ~~normal,~~ off-normal~~,~~ and hypothetical accident conditions. The doses resulting from ~~this~~ postulated failure*s* are within regulatory limits, which further demonstrates the conservative design of the storage system when its confinement capabilities are considered.

### 7.2.2.2 Confinement Vessel Releasable Source Term

*The inventory of isotopes other than Co-60 is calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.4 system. The isotopic inventory for PWR fuel in the MPC was based on the design basis fuel assembly with a burnup of 42,000 MWD/MTU, nine years cooling time, and an enrichment of 3.09 wt% U-235. This assumed burnup and the cooling time were chosen to conservatively bound the actual burnup and cooling times for all spent fuel at the Trojan Nuclear Plant.*

*All isotopes that contribute greater than 0.1 percent to the total curie inventory for the fuel assembly are considered in the evaluation as fines. This analysis also includes those actinides that contribute greater than 0.01 percent to the total curie inventory as the dose conversion factors for these isotopes are in general greater than other isotopes (e.g., isotopes of plutonium, americium, curium, and neptunium). This approach is in accordance with ISG-5 (Reference 5). A summary of the isotopes available for release is provided in Table 7.2-1.*

### 7.2.2.3 Crud Radionuclides

*The majority of the activity associated with crud is due to Co-60 (Reference 6). The inventory for Co-60 was determined by using the crud surface activity for PWR rods ($140 \times 10^{-6}$ Ci/cm$^2$) provided in NUREG/CR-6487, multiplied by the surface area per assembly ($3 \times 10^5$ cm$^2$ for PWR fuel, also provided in NUREG/CR-6487). The source terms were then decay corrected nine years using the basic radioactive decay equation:*

$$A(t) = A_0 e^{-\lambda t}$$

*where:*

- $A(t)$ = *activity at time t (Ci),*
- $A_0$ = *the initial activity (Ci),*
- $\lambda$ = *the* $\ln 2/t_{1/2}$ *(where* $t_{1/2}$ *= 5.272 years for Co-60 [Reference 7]), and*
- $t$ = *the time in years (9 years).*

*A summary of the Co-60 inventory available for release is provided in Table 7.2-1.*

## 7.3 RADIATION PROTECTION DESIGN FEATURES

### 7.3.1 INSTALLATION DESIGN FEATURES

Section 7.1.2 describes the design features of the installation and equipment that ensure exposures to radiation are ALARA. The ISFSI is a passive outdoor storage system. Each Concrete Cask located at the ISFSI has sufficient natural circulation to ensure adequate air cooling of the ~~PWR Basket~~*MPC*. All radiation sources are confined within the sealed ~~PWR Basket~~*MPC*, which serves as a confinement boundary and shielding. The ~~PWR Basket~~*MPC* is inside the Concrete Cask, which provides further shielding.

Concrete Cask loading and decontamination will be performed within the Trojan Fuel Building. Operation of Fuel Building systems for Concrete Cask loading, control of filtered ventilation, and radioactive waste treatment are covered under the existing 10 CFR 50 license.

A detailed description of the ISFSI is included in Chapter 4. The ISFSI layout is described in Chapter 2. Concrete Cask operational details are included in Chapter 5. The criteria for the design of the installation features and systems are provided in Chapter 3.

Specific shielding design features are described in the next section. These features are consistent with guidance provided in Regulatory Guide 8.8 *(Reference 11)*.

Applicable portions of Regulatory Guide 8.8, position C.2 have been used as guidance as follows:

1. Access to the ISFSI is controlled in accordance with 10 CFR 72. Normal access to the ISFSI is through a single access point.

2. Radiation shielding is provided by the ~~PWR Basket~~*MPC* and Concrete Cask and constitutes the primary method of reducing personnel exposure to radiation.

3. The ISFSI is a passive installation that has no operations to control. Therefore, no process instrumentation or controls are necessary during storage.

4. Airborne contaminants and gaseous radiation sources are confined by the seal-welded ~~PWR Basket~~*MPC*.

5. No crud is produced by the ~~PWR Basket~~*MPC* or Concrete Cask.

6. Concrete Cask decontamination is performed, *as necessary,* prior to placement on the Storage Pad. Once the Concrete Casks are in place, there are no credible mechanisms that could result in contamination of the ISFSI components.

7. Area radiation monitoring instrumentation consists of thermoluminescent devices (TLDs) posted at the perimeter of and in the Controlled Area near the Concrete Casks.

8. No resin or sludge is produced from the ~~PWR Basket~~*MPC* or Concrete Casks.

### 7.3.2 SHIELDING

The Concrete Cask system is designed to maintain radiation exposure As Low As Reasonably Achievable (ALARA). The Concrete Cask ~~is~~ design~~ed to provide~~ *results in* an average external dose rate (gamma and neutron) of less than 100 mrem/hr on the sides and ~~250~~ *300* mrem/hr on top and at the air inlets and outlets. The maximum dose rate at the top of the ~~PWR Basket~~*MPC* ~~structural~~ lid ~~will~~ *may* be limited by temporary shielding to allow limited personnel access during ~~PWR Basket~~*MPC* closure operations. This design satisfies the requirements of 10 CFR 72.104, 10 CFR 72.106, and OAR 345-26-390, which establish dose limits for members of the public in unrestricted areas (i.e., at or beyond the Controlled Area Boundary of ~~325~~ *225* meters).

*Besides the Concrete Cask, MPC, and the Transfer Cask, no other radiation shielding features are required for the TNP ISFSI. However, the ISFSI location has natural earth berms located on the North and East sides. The terrain in the other directions is not flat but there are no earth berms immediately surrounding the ISFSI. Conservatively, the analysis in this SAR does not take any shielding credit for earth berms or physical structures that exist between the ISFSI and the Controlled Area Boundary. The terrain was assumed to be flat ground.*

#### 7.3.2.1 Radial and Axial Shielding Configurations

The radiation shielding for the stored spent nuclear fuel assemblies is provided by a variety of shielding materials~~. Figures 7.3-1 and 7.3-2 show shielding model geometries (described below) for~~ *in* the ~~PWR Basket~~*MPC,* ~~in the~~ Concrete Cask, *and Transfer Cask.* Figures ~~7.3-5~~*4.2-4* and ~~7.3-6~~*4.7-1 depict the Concrete Cask and the Transfer Cask.* ~~show shielding model geometries (described below) for the PWR Basket in the Transfer Cask. Also,~~ *The shielding models were created in full three-dimensional detail and accurately represent the configurations shown in those figures (minor exceptions include the hole for the gap flush system and inflatable seal details for the Transfer Cask). The top lid on the Transfer Cask was conservatively not modeled.* ~~t~~*T*he densities for constituent ~~nuclides~~*elements* of all shielding materials used in the calculational models are given in Table~~s~~ 7.3-1 ~~and 7.3-2~~.

The ~~PWR Basket~~*MPC* contains a~~n~~ ~~eight~~ *9.5*-inch thick *stainless* steel ~~shield~~ lid *and a 2.5-inch thick baseplate, both of which connect to the 0.5-inch thick MPC shell.* ~~This shield~~*The MPC* lid provides radiation protection for workers engaged in the ~~PWR Basket~~*MPC* closure and transfer operations as well as the largest majority of the shielding in the top axial direction during storage. Additional shielding in the top axial direction *of the Concrete Cask* is provided by the ~~3~~ *0.75*-inch thick steel ~~structural~~ lid on the ~~PWR Basket top and the steel lid on the~~ Concrete Cask.

In addition, a steel shield ring, *6 inches tall and 4 inches thick with an inner diameter of 64 inches,* immediately above the ~~PWR Basket~~*MPC*/Concrete Cask inner liner annulus adds protection from radiation streaming up this annulus. Shielding located axially beneath the ~~PWR Basket~~*MPC* consists of the steel ~~PWR Basket~~*MPC* ~~bottom~~*baseplate*, the steel Concrete Cask liner bottom, and a thick section of concrete. ~~These shielding materials are listed in Tables 7.3-1 and 7.3-2.~~

Radiation shielding *for the Concrete Cask* in the radial direction during storage is provided primarily by the *MPC fuel basket internals and the* steel ~~PWR Basket~~*MPC* shell, followed consecutively by the *2-inch thick* steel Concrete Cask inner liner and a *29-inch* thick concrete wall. Cooling air penetrations are from the Concrete Cask sides, and contain at least two sharp bends to minimize radiation streaming. The four sets of air inlet and exhaust ducts in the Concrete Cask are fabricated with 0.5"~~-~~*-inch* thick steel walls.

*The Transfer Cask is comprised of multiple layers of shielding. The shielding in the radial direction consists of a 0.75-inch steel shell, 3 5/8 inches of lead, 1 inch of steel, 5 inches of water, and 0.5 inch of steel. The lead and steel shielding is primarily designed to attenuate gamma radiation produced by the contained fuel and activated hardware. Attached to the outer steel shell is a water jacket. The water jacket is designed to moderate and absorb the neutrons produced by the contained fuel. Eight 2-inch thick steel heat transfer fins are also present in the water jacket region that run radially out from the Transfer Cask outer steel shell to the outer wall of the water jacket. In the shielding models, these steel fins are modeled explicitly to account for the reduction in neutron shielding at these points. The water jacket extends from the bottom of the Transfer Cask to just below the lifting trunnions. In the axial direction, the Transfer Cask bottom doors positioned below the MPC are 7.25-inch solid steel, and a Transfer Cask top lid is positioned above the MPC. The Transfer Cask top lid was conservatively not modeled in this analysis. During normal operations, no personnel contact is required with the bottom of the Transfer Cask bottom doors.*

*Figures 4.2-1a and 4.2-1b depict the MPC and its internal configuration.* In the shielding models, the ~~fuel PWR Basket~~*MPC basket, Boral neutron poison plates, and sheathing are modeled explicitly, while the fuel assembly is modeled as a homogenized mass. The active fuel region and the non-fuel regions above and below the fuel assembly were modeled separately. The fuel assembly* ~~interior~~ is sub-divided into ~~four~~ *six* axial subregions. *The detail dimensions of these sub-regions are listed in Table 7.3-2. When the TPDs were considered to be present in the fuel assembly, the mass from these devices was included in the appropriate region in the models.* ~~In each subregion, the complicated PWR Basket internal structures are mixed into a single homogenous material that fills the subregion.~~

In the active fuel region, the homogenous material includes spent fuel material (modeled as pure $UO_2$), *and* the Zircaloy cladding and guide tube ~~material (modeled as pure zirconium), the carbon steel fuel support sleeve material (modeled as pure iron), and the stainless steel control component cladding material. Several other PWR Basket internal components, such as the~~

~~structural support tubes, which do not necessarily surround the radiation sources, are conservatively neglected in the homogenous material density calculations. The materials from the neutron poison sheets are included in the fuel region homogenous material for neutron calculations only.~~

The ~~gas plenum region homogenized material includes the~~ Zircaloy cladding material *that may be present in the homogenized regions above and below the active fuel zone is conservatively neglected.* ~~and the iron and nickel materials in the plenum springs (which are the source of the gamma radiation). The top and bottom nozzle region homogenized materials consist of the metal materials which make up the top and bottom nozzles, and which are the source of the gamma radiation. The homogenized materials also include the Zircaloy present in the solid end caps of the fuel rods. Also, the bottom nozzle region material description includes the iron from the carbon steel support sleeves which extend into that region. The fuel support sleeves do not always extend into the top nozzle region, so the top nozzle region homogenized material does not include the support sleeve material.~~

~~The homogenized material description for the four axial subregions of the PWR Basket interior are shown in Tables 7.3-1 and 7.3-2.~~

~~The Transfer Cask shielding design reduces the dose from the loaded PWR Basket. Shielding at the top is provided by the PWR Basket shield and structural lids. Radially, the Transfer Cask is composed of a steel shell filled with two separate shielding materials. The first is a lead shield designed to attenuate gamma radiation. The second is a strong neutron absorber designed to moderate and absorb the neutrons. Steel heat transfer fins are also present in the neutron shield region. In the shielding models, these steel fins are mixed with the neutron material to create the homogenous material listed in Tables 7.3-1 and 7.3-2. The Transfer Cask bottom doors are thick steel which reduces the contact dose rate. However, no personnel contact with the bottom doors is required during normal operation.~~

~~During off-site transfer operations, the Shipping Cask lid is removed and the Transfer Station shield ring is positioned to provide additional shielding when the PWR Basket is transferred from the Transfer Cask to the Shipping Cask.~~

### 7.3.2.2 Shielding Evaluation

*The fully three-dimensional, continuous energy, coupled neutron-gamma Monte Carlo code Monte Carlo N-Particle Version 4A (MCNP-4A) (Reference 8) was used for all shielding analyses.* ~~A combination of computer codes and manual calculations are used for radiation shielding analyses of both the Concrete Cask and Transfer Cask containing design basis fuel.~~ *The MCNP shielding code was used to calculate neutron and gamma dose rates along the surface and 1 meter from an individual Concrete Cask and the Transfer Cask. MCNP is capable of modeling three-dimensional geometries using explicit elemental material descriptions for each material zone. MCNP uses point-wise cross section data; therefore, no group structures are*

*used. MCNP gives average flux levels (and spectra) over each user-defined surface area detector. The gamma and neutron flux-to-dose conversion factors shown in Tables 7.3-3 and 7.3-4 are used to convert the MCNP flux output into dose rates (in mrem/hr).* ~~The results from these analyses are summarized in Figures 7.3-9 and 7.3-11.~~

*The individual Concrete Cask and the Transfer Cask are analyzed containing 42,000 MWD/MTU fuel with the cooling time of nine years. The analysis of operational exposure was also performed for a burnup of 42,000 MWD/MTU and nine-year cooling. The analysis of the dose rate-versus-distance from the 34 Concrete Casks located on the ISFSI Storage Pad accounted for different burnup and cooling times as discussed below. The results of the Concrete Cask and Transfer Cask analyses are summarized in Table 7.4-1 and Figures 7.3-2 and 7.3-3.*

*As indicated above, the dose rate-versus-distance from the TNP ISFSI was calculated using MCNP-4A. Figure 7.3-1 provides a pictorial representation of the ISFSI with all thirty-four Concrete Casks. The two empty locations were originally designated for Greater Than Class C (GTCC) waste; however, these locations will not be used. Based on the storage capacity of the ISFSI (34 casks), it is not practical to try to model the entire ISFSI in MCNP or any other computer code. Therefore, a methodology similar to that described in Section 5.4 of the HI-STORM 100 System FSAR was used in the calculation of the dose rate-versus-distance from the ISFSI. The dose rate-versus-distance was calculated first for a single Concrete Cask. Then numerous MCNP calculations, using relatively small models, were performed to develop ratios for the dose rate contribution from casks situated behind other casks. These ratios were used in conjunction with the dose rate-versus-distance from a single Concrete Cask to estimate the dose rate from the entire ISFSI storage area.*

*The dose rate from the radiation source was separated into two components. For the purposes of this discussion, the first is referred to as the top-dose. This was the dose rate from radiation that leaves the top of the Concrete Casks. The second component is referred to as side-dose. This was the dose rate from radiation that leaves the sides of the Concrete Casks. In both cases, top-dose and side-dose, in-air scattering of radiation (skyshine) was accounted for in the dose calculations.*

*In calculating the dose rate from the entire ISFSI storage area, the geometry impacted each of the dose components (top and side) in a different fashion. The total top-dose rate was a summation of the top-dose rates from all 34 casks, where the actual distance from the dose location to the individual cask was accounted for.*

*The total side-dose rate was a summation of the side-dose rates from all 34 casks where the distances within the facility and the self-shielding of one row of casks to another row were accounted for. Since the side-dose rate is from radiation leaving the side of the Concrete Cask, this dose contribution is greatly reduced if the cask is situated behind another cask. The front cask blocks some but not all of the radiation from the back cask from reaching the Controlled*

*Area boundary. The fraction of radiation blocked was therefore calculated with MCNP, as mentioned above, and used in the determination of the total side-dose.*

*As mentioned earlier, the models assumed a flat terrain surrounding the Concrete Cask and the ISFSI storage area. The MCNP models consisted of the Concrete Cask surrounded by 600 meters of air in the radial direction and 700 meters of air in altitude. The cask was assumed to be sitting on an infinite slab of soil.*

*As mentioned earlier, a different burnup and cooling time was used for each cask location in the ISFSI. The burnup and cooling times for each cask were chosen to bound all of the assemblies in that particular cask and it was assumed that all casks were completely filled with 24 intact fuel assemblies. Since a different burnup and cooling time is being used for each cask, the dose versus distance was calculated perpendicular to the center of each side of the ISFSI (i.e., the dose was calculated separately from all four sides of the ISFSI). Conservatively, it was assumed that all six casks along the front of the ISFSI in the direction being analyzed contained TPDs. The number of TPDs in the inventory is less than 144 (6 casks x 24 assemblies/cask) and it is conservative to place them closest to the dose location. The results of the dose rate calculations are discussed in Sections 7.4 and 7.6.*

~~**7.3.2.2.1**Direct Dose Evaluation~~

~~QAD-CGGP (Reference 3), a three-dimensional point-kernel code, is used for gamma dose rate calculations at the Transfer Cask and Concrete Cask ends, as well as for the Transfer Cask radial gamma dose rate calculations. The QAD-CGGP shielding model geometries are shown in Figures 7.3-1, 7.3-2, 7.3-5, and 7.3-6. The material descriptions for each shielding model material region are shown in Tables 7.3-1 and 7.3-2. QAD-CGGP calculates gamma flux levels at each detector location specified by the user.~~

~~The MCNP Monte Carlo shielding code is used to calculate neutron dose rates for the Transfer Cask and Concrete Cask ends, as well as for the Transfer Cask side. MCNP is capable of modeling three-dimensional geometries using explicit elemental material descriptions for each material zone. MCNP uses point-wise cross section data, so no group structures are used. MCNP gives average flux levels (and spectra) over each user-defined surface area detector.~~

~~The gamma and neutron flux-to-dose conversion factors shown in Tables 7.3-3 and 7.3-4 are used to convert the QAD-CGGP and MCNP flux output into dose rates (in mrem/hr).~~
~~The radial shielding configuration for the Trojan Concrete Cask system is identical to that of the SNC VSC-24 storage cask system, with equal thicknesses of steel and concrete. Also, the Trojan PWR Basket has an internal structure which provides shielding equal to or greater than that present in the VSC-24 system basket. The Trojan PWR Basket employs fuel support sleeves with thicker walls than those of the VSC-24 basket. The Trojan PWR Basket also contains additional internal structural materials not present in the VSC-24 basket.~~

~~For these reasons, a simple source ratio technique is used to calculate the gamma and neutron dose rates on the Trojan Concrete Cask side using the gamma and neutron dose rates previously calculated for the VSC-24 storage cask side (Reference 7).~~

~~The neutron dose rates on the Concrete Cask side are assumed to be directly proportional to the total neutron source strength. The VSC-24 system calculated neutron dose rate (before the adjustments to account for axial peaking were made) is multiplied by the ratio of the Trojan Concrete Cask total neutron dose rate over the VSC-24 cask total neutron dose rate. Then, the resulting dose rate is increased by 50% to account for the 10% peaking in the fuel burnup level and 30.8% to account for sub-critical neutron multiplication (as discussed in Section 7.2.1.2). Note that the neutron source spectrum is virtually identical for the Trojan and VSC-24 systems.~~

~~For the gamma dose rates, the process is somewhat more complicated because the gamma spectrum is different for the two cask systems. To fully account for spectral effects, the total cask gamma source strengths for the two systems are compared for each gamma energy line. It was found that the Trojan system gamma source strength was greatest relative to the VSC-24 system gamma source strength for the 1.25 MeV gamma energy line, the energy line which contributes the great majority of the cask external gamma dose rate. Therefore, the source strength ratio for the 1.25 MeV gamma energy line is multiplied by the VSC-24 system cask side gamma dose rate (before peaking effects are considered) to yield the Trojan Concrete Cask side gamma dose rates. These resulting dose rates are then increased by 10% (as discussed in Section 7.2.1.1) to account for axial burnup peaking.~~

~~7.3.2.2.2Scattered Dose Evaluation~~

~~Measurable scattered dose rates at cask surfaces are possible. This is due to the presence of the air annulus between the PWR Basket and the inner Concrete Cask or Transfer Cask surface causing scattering in axial directions, and the air inlets and outlets on the Concrete Cask causing scattering primarily in the radial direction. Neutron and gamma dose rates at the Concrete Cask air inlet are directly calculated using the MCNP Monte Carlo code. The dose rates calculated by MCNP automatically include both the direct and scattered flux components. Dose rate contributions from the fuel region neutron and gamma sources and from the bottom nozzle region gamma source are separately calculated and summed to yield total Concrete Cask air inlet dose rates. Two-dimensional MCNP analyses model the cask air inlet structure with a high degree of accuracy (with material conservatively removed in some areas). Flux levels at the air inlet entry points are directly calculated. These are converted to dose rates using the flux-to-dose conversion factors shown in Tables 7.3-3 and 7.3-4.~~

~~Gamma and neutron dose rates at the air outlets were calculated for the VSC-24 system using shielding codes along with complex manual albedo calculations. Air outlet dose rates for the Trojan storage system are scaled from those calculated for the VSC-24 system. The QAD-CGGP code is used to model both the VSC-24 and the Trojan storage systems.~~

~~The gamma flux levels are calculated by gamma energy line at the inner end of the outlet duct (i.e., on the inner surface of the cask liner). For both systems, the fuel region and non-fuel region gamma sources are considered. The flux levels at each energy level for the two systems are compared. As with the Concrete Cask side, the Trojan storage system gamma flux was highest relative to that of the VSC-24 system for the 1.25 MeV energy line, the line which is the dominant contributor to the gamma dose rates. Therefore, the air outlet gamma dose rate calculated for the VSC-24 system is multiplied by the ratio of the 1.25 MeV flux levels (at the inside of the outlet duct) to yield an air outlet gamma dose rate for the Trojan storage system.~~

~~Since the neutron source spectrum is similar for the two cask systems, and since the system geometries are quite similar, it is assumed that the air outlet neutron dose rate is proportional to the total cask neutron source strength. Therefore, the VSC-24 air outlet neutron dose rate is multiplied by the ratio of the Trojan Concrete Cask neutron source strength over the VSC-24 neutron source strength. The total Trojan Concrete Cask inlet and outlet dose rates are:~~

| | ~~Combined Dose Rate Fuel Concrete Cask (mrem/hr)~~ |
|---|---|
| ~~Air Inlet~~ | ~~18.77~~ |
| ~~Air Outlet~~ | ~~6.76~~ |

### 7.3.3 VENTILATION

The Concrete Cask is designed for passive, natural convection cooling of the ~~PWR Basket~~*MPC*. The air flow path is formed by the channels at the bottom (air entrance), the air inlet ducts, the annulus between the ~~PWR Basket~~*MPC* exterior and the Concrete Cask interior, and the air outlet ducts.

The air inlets and outlets are steel lined penetrations that take non-planar paths to minimize radiation streaming.

The Concrete Cask system is designed to prevent the release of radioactive material during normal storage conditions. However, the potential effects of postulated ~~PWR Basket~~*MPC* leakage is evaluated in Chapter 8. Evaluations of partial and full blockage of the air inlets are *also* presented in Chapter 8.

There are no off-gas systems required for normal operation of the ISFSI because the ~~PWR Basket~~*MPC* is sealed.

### 7.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

During fuel loading the existing Trojan plant instrumentation is utilized as described in the Trojan Nuclear Plant Defueled Safety Analysis Report in Chapter 5. During storage, area radiation monitoring will consist of TLDs posted at the perimeter of and in the Controlled Area near the Concrete Casks. The TLDs will be used to monitor operation of the ISFSI for the Radioactive Effluent and Environmental Monitoring Program described in Section 7.6.

## 7.4 ESTIMATED ON-SITE COLLECTIVE DOSE ASSESSMENT

The Transfer Cask and Concrete Casks are designed to limit dose rates to minimal levels for operators, inspectors, maintenance, and radiation protection personnel when the casks are being loaded, moved, and stored. Table 7.4-1 contains the maximum calculated design basis dose rates and the calculated working dose rates for loading and handling the Transfer Cask and Concrete Casks under normal conditions. All values for dose rates include both gamma and neutron flux components.

Working dose rates and personnel requirements for the cask loading cycle, move-to-storage, and loading for off-site shipping ~~cycle~~ are shown in Table 7.4-3. The operational sequence for these activities is shown in more detail in Figures 5.1-1 and 5.1-3. Based on the estimates shown in Table 7.4-3, *the following table reflects* the collective dose for loading, moving to storage, and loading for off-site shipping for the 34 fuel *storage* casks:

| Dose (person-rem) | |
|---|---|
| | 34 Casks |
| Load Cask | ~~97.4~~*84.6* |
| Move to Storage | ~~2.04~~*1.3* |
| Load for Shipping | ~~17.1~~*28.6* |
| Total | ~~116.54~~*114.5* |

From Table 7.4-3, conservative estimates of the periodic inspection, ~~and~~ surveillance, *and maintenance* requirements result in a collective dose of about ~~5~~ *1.1* rem/yr while the Concrete Casks are being stored. This dose is based on a ~~daily~~ *weekly* visual inspection of each stored Concrete Cask, ~~twice a day temperature readings of the air outlet temperature of each Concrete Cask,~~ quarterly radiation protection surveys, and annual inspections of the Concrete Cask concrete and concrete Storage Pad. The annual collective dose will decrease with the age of the Concrete Cask because the estimate is based on a freshly loaded Concrete Cask *and ISFSI.*

The annual dose estimate for surveillance is considered very conservative based on operating experience from Consumers Power (Palisades) which shows that the annual dose for daily temperature and screen checks, monthly and annual radiological surveys, security surveillances, snow removal, and pad surveillances is about 120 mrem per year for thirteen casks.

## 7.5 RADIATION PROTECTION PROGRAM

### 7.5.1 ORGANIZATION

The PGE organization that will implement the Radiation Protection Program during ISFSI construction and fuel loading is described in Section 9.1.1 and shown in Figure 9.1-1. The PGE organization that will implement the Radiation Protection Program during long term spent nuclear fuel storage at the ISFSI is described in Section 9.1.2 and shown in Figure 9.1-2.

### 7.5.2 RADIATION PROTECTION EQUIPMENT, INSTRUMENTATION, AND FACILITIES

The various equipment and instrumentation for performing radiation surveys and measuring and minimizing personnel exposure, and the facilities for radiation protection activities are summarized in this section. The radiation protection equipment, instrumentation, and facilities are highly simplified because the Concrete Cask introduces limited radiological hazards.

#### 7.5.2.1 Radiation Protection Instrumentation

Following termination of Trojan's Part 50 license, radiation protection instrumentation, including radiation detection, airborne monitoring, and personnel monitoring instrumentation will not normally be located or maintained at the Trojan ISFSI. This instrumentation will be owned, operated, maintained, and calibrated by and at an off-site facility. The Trojan ISFSI will maintain an agreement or contract with the off-site facility to provide radiation protection services which are mainly anticipated to entail direct radiation surveys, contamination surveys, and personnel monitoring device reading and calibration. The off-site facility will provide their own instruments and personnel and will be responsible for their own training and qualification.

#### 7.5.2.2 Area Radiation Monitoring Instrumentation

Area radiation monitoring instrumentation consists of TLDs posted at the perimeter of and in the Controlled Area near the Concrete Casks. TLDs will be read quarterly to monitor direct radiation from the ISFSI.

#### 7.5.2.3 Radiation Protection Facilities

Due to the minimal radiological hazards introduced by ISFSI operation, the ISFSI will not have radiation protection facilities onsite. Decontamination services, bioassay services, protective clothing, respirators, and additional instrumentation are available, if required, although no situations are anticipated which would necessitate use of these services or equipment.

## 7.5.3 RADIATION PROTECTION PROCEDURES

The purpose of this section is to summarize how ISFSI procedures implement the Radiation Protection Program to maintain radiation exposure ALARA while spent nuclear fuel is stored in the ISFSI.

### 7.5.3.1 Control of Radiation Exposure to the Public

Monitoring, analyzing, and reporting radiation levels in the environment is performed in accordance with the Radioactive Effluent and Environmental Monitoring Program to demonstrate that the dose to the public is below regulatory limits and ALARA.

Radiation monitoring will be accomplished by posting TLDs at the perimeter of and in the Controlled Area near the Concrete Casks and reading the TLDs quarterly.

No gaseous, liquid, or solid radioactive effluents are produced by the storage system because of its sealed design. Therefore, routine monitoring for effluents is not performed.

### 7.5.3.2 Control of Personnel Radiation Exposure (Occupational)

Personnel radiation exposure is maintained ALARA by a combination of shielding, access control, contamination control, surveys and monitoring, work planning, training, and sound radiation protection practices implemented by procedures. The procedures for personnel radiation protection are prepared consistent with the requirements of 10 CFR 20 and are approved, maintained, and adhered to for activities involving personnel radiation exposure.

#### 7.5.3.2.1 Shielding

The objective of radiation shielding is to reduce external doses to personnel, in conjunction with a program for controlling personnel access and occupancy in radiation areas, to levels which are both ALARA and within the regulations defined in 10 CFR 20. Radiation protection implementing procedures provide for evaluation of the use of temporary shielding for activities involving high dose rates.

#### 7.5.3.2.2 Access Control and Area Designations

The Restricted Area, as defined in 10 CFR 20, has the same boundaries as the ~~access C~~controlled *access* ~~A~~*a*rea that surrounds the ISFSI Protected Area. Physical access to the ~~access C~~controlled *access* ~~A~~*a*rea is restricted by the ~~access C~~controlled *access* ~~A~~*a*rea fence. Access into the Protected Area is controlled as described in the ISFSI Security Plan.

A Radiologically Controlled Area (RCA) is an area where access is controlled for the purpose of protecting individuals from exposure to radiation. RCAs are determined by the radiation level, contamination level, or the presence of radioactive materials. Procedures describe the requirements for radiological postings advising workers of potential radiological hazards at the entrance and boundaries of RCAs.

7.5.3.2.3 Facility Contamination Control

Radioactive contamination of the ISFSI is not anticipated because *accessible portions of* the external surface of the ~~PWR Basket~~*MPC* ~~is~~ *are* checked for loose surface contamination before the Concrete Cask is moved to the Storage Pad as described in Section 7.2.2.*1*. In addition, the spent nuclear fuel is inside the seal-welded ~~PWR Basket~~*MPC* and there are no credible accidents that would cause a gaseous, liquid, or solid release of radioactivity. However, procedures direct the use of various practices and equipment to ensure that the potential for the spread of contamination is controlled at the source to the greatest extent possible.

7.5.3.2.4 Personnel Contamination Control

As stated above, the ~~PWR Basket~~*MPC* is checked for loose surface contamination prior to being placed in the Concrete Cask. However, surveys for contamination at the Concrete Cask air inlets and outlets are routinely performed to confirm that contamination is not present. If contamination was discovered by a survey, protective clothing is available to prevent contamination of personnel who would need to enter the contaminated area. Similarly, respiratory protective equipment is not required because airborne radioactivity is not credible either. However, the potential for airborne radioactivity would be considered if surface contamination were discovered. Respiratory protective equipment is available if by evaluation it was determined that use of respiratory equipment would result in exposures that are ALARA.

7.5.3.2.5 Area Surveys

Quarterly surveys are performed in the accessible areas of the ISFSI. These surveys consist of contamination surveys and external radiation measurements in appropriate areas. Additionally, specific surveys are performed as needed for operational and maintenance functions involving potential exposure of personnel to radiation or radioactive materials.

7.5.3.2.6 Personnel Monitoring

TLDs are worn by personnel within RCAs ~~to measure radiation dose~~*when radiation levels are greater than 0.25 mrem/hr and as required by applicable Radiation Work Permits (RWPs).* Monitoring for internal deposition of radioactive materials is not required because surface and airborne radioactivity is not anticipated and there is no credible accident that would result in

gaseous, liquid, or solid release of radioactivity. However, monitoring for internal contamination could be performed by contracted offsite facilities if desired.

7.5.3.2.7 Work Planning

Work in RCAs is planned prior to performance. Consideration is given to dosimetry requirements, personnel protective equipment, monitoring requirements, and special cautions pertinent to the work. The planning also considers the maximum radiation level that will be encountered.

7.5.3.2.8 Training

Individuals requiring unescorted access to the ISFSI receive training which includes radiological protection fundamentals. Individuals who require access to RCAs will receive radiation protection training commensurate with their responsibilities in accordance with 10 CFR 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations."

The need for specialized ALARA training is evaluated during work planning in accordance with radiation protection implementing procedures. Specialized ALARA training may include dry runs, pre-job briefings, and other special training classes.

7.5.3.2.9 Controls, Practices, and Special Techniques

Radiation protection implementing procedures specify that during the planning phase for activities in high dose rate areas, various engineering controls to minimize exposures should be evaluated and/or implemented. These engineering controls and practices include, but are not limited to, temporary shielding; remote surveillance equipment; multi-discipline input regarding ALARA goals; pre-job, in-progress, and post-job briefings; and adequate lighting, ventilation, work space, and work area accessibility.

7.5.3.3 Records and Reports

PGE will maintain records of the radiation protection program, surveys, and individual monitoring results, as well as records that show compliance with the dose limits for individual members of the public. PGE will submit reports of individual monitoring as required by *10 CFR 19 and* 10 CFR 20.

## 7.6 ESTIMATED OFF-SITE COLLECTIVE DOSE ASSESSMENT

### 7.6.1 RADIOACTIVE EFFLUENT AND ENVIRONMENTAL MONITORING PROGRAM

No radioactive gas, liquid, or solid waste effluents are expected during operation. Therefore, a radioactive effluent monitoring system is not required and routine monitoring for effluents is not performed. The radioactive effluents released during fuel loading operations, which is a 10 CFR 50 licensed activity, are monitored and controlled by existing plant systems as explained in Chapter 6.

The ISFSI will emit direct radiation that will be monitored in the environment. The Radioactive Effluent and Environmental Monitoring Program will be implemented by posting TLDs at the perimeter of and in the Controlled Area near the Concrete Casks. TLDs will be read quarterly to monitor radiation levels in the nearby vicinity of the ISFSI.

### 7.6.2 ANALYSIS OF MULTIPLE CONTRIBUTION

Once the ISFSI is completed and the Trojan Nuclear Plant is decommissioned, the only significant radiation will come from the storage installation. No other nuclear facility is projected for the vicinity of the ISFSI (i.e., within a 5-mile radius).

The incremental contribution of the ISFSI to the total dose of a member of the general public has been estimated by calculation. The dose is estimated as ~~8.5~~*23.52* mrem per year at a distance of ~~1000 feet~~*225 meters* ~~(approximate distance~~ to the Controlled Area boundary ~~which is 325 meters or 1066 feet)~~ from the edge of the ISFSI Storage Pad, *based on a 2080 hour per year occupancy.* This *estimated* dose satisfies the *whole body dose* requirements of 10 CFR 72.104~~ for members of the general public based on an occupancy of 8760 hours per year at the Controlled Area boundary~~. *The 23.52 mrem per year Controlled Area boundary dose includes a maximum effluent contribution of 2.52 mrem per year, which is within the limit of*~~ An occupancy factor of 2000 hours per year is used for the purposes of compliance with~~ OAR 345-26-0390(4)(f). The *2080 hour per year occupancy factor is used, in accordance with Interim Staff Guidance Document 13 (Reference 9), to represent a conservative maximum estimate of a real individual's occupancy time at the Controlled Area boundary (40 hours per week for 52 weeks).* ~~corresponding dose is 1.95 mrem per year. This~~ *The 2080 hour per year* occupancy factor is conservative considering the land usage patterns in the vicinity of the ISFSI.

~~The calculated dose at 1000 feet from the edge of the ISFSI Storage Pad (approximate distance to the Controlled Area boundary) compares with a background dose of 62 mrem per year (based on a 8760 hour occupancy) at the Trojan site.~~

### 7.6.3 ESTIMATED DOSE EQUIVALENTS

The sum of the maximum doses from normal and off-normal releases and direct radiation are given in Table 8.2-2.

### 7.6.4 LIQUID RELEASE

There are no radioactive liquids to be released from the Trojan ISFSI.

## 7.7 REFERENCES


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**Table 7.2-1**
**~~Major~~ Isotopic ~~Contributors to Fuel Source Term~~*Inventory***
***(Curies per assembly)***

Security-Related Information Table Withheld Under 10 CFR 2.390.

# Security-Related Information Table Withheld Under 10 CFR 2.390.

## Table 7.2-2

### Fuel Region Gamma Source Strength ~~(gammas/sec-cask)~~ ~~for Various Burnup and Cooling Time Combinations~~ ~~(including control component source)~~

| ~~Line Energy (MeV)~~ | ~~Gamma Source Strength 40,000 MWd - 5 year~~ | ~~Gamma Source Strength 45,000 MWd - 6 year~~ |
|---|---|---|
| ~~0.01~~ | ~~4.022E+16~~ | ~~3.685E+16~~ |
| ~~0.025~~ | ~~9.631E+15~~ | ~~8.553E+15~~ |
| ~~0.0375~~ | ~~1.044E+16~~ | ~~9.847E+15~~ |
| ~~0.0575~~ | ~~8.000E+15~~ | ~~7.249E+15~~ |
| ~~0.085~~ | ~~5.243E+15~~ | ~~4.664E+15~~ |
| ~~0.125~~ | ~~5.431E+15~~ | ~~4.870E+15~~ |
| ~~0.225~~ | ~~4.365E+15~~ | ~~3.810E+15~~ |
| ~~0.375~~ | ~~2.665E+15~~ | ~~2.211E+15~~ |
| ~~0.575~~ | ~~7.260E+16~~ | ~~7.190E+16~~ |
| ~~0.85~~ | ~~1.807E+16~~ | ~~1.585E+16~~ |
| ~~1.25*~~ | ~~8.860E+15~~ | ~~8.433E+15~~ |
| ~~1.75~~ | ~~1.142E+14~~ | ~~9.709E+13~~ |
| ~~2.25~~ | ~~4.993E+13~~ | ~~2.275E+13~~ |
| ~~2.75~~ | ~~1.875E+12~~ | ~~1.024E+12~~ |
| ~~3.5~~ | ~~2.409E+11~~ | ~~1.321E+11~~ |
| ~~5.0~~ | ~~3.961E+08~~ | ~~5.134E+08~~ |
| ~~7.0~~ | ~~4.568E+07~~ | ~~5.921E+07~~ |
| ~~9.5~~ | ~~5.248E+06~~ | ~~6.801E+06~~ |
| ~~Total~~ | ~~1.856E+17~~ | ~~1.743E+17~~ |

~~* 1.25 MeV source strength includes $Co^{60}$ gammas from activated control components.~~

| Gamma Energy Range (MeV) | Gamma Source Strength 42,000 MWD/MTU 9 year cooling (photons/sec/assembly) | Gamma Source Strength 42,000 MWD/MTU 9 year cooling (MeV/sec/assembly) |
|---|---|---|
| 0.45-0.7 | 2.17E+15 | 1.25E+15 |
| 0.7-1.0 | 2.16E+14 | 1.83E+14 |
| 1.0-1.5 | 5.23E+13 | 6.54E+13 |
| 1.5-2.0 | 1.81E+12 | 3.17E+12 |
| 2.0-2.5 | 1.17E+11 | 2.64E+11 |
| 2.5-3.0 | 6.70E+09 | 1.84E+10 |
| Total | 2.44E+15 | 1.50E+15 |

**Table 7.2-3**

**FUEL REGION NEUTRON SOURCE STRENGTHS**
**~~(Neutrons/sec-cask)~~**

| ~~Neutron Energy Range (MeV)~~ | ~~Neutron Source Strength 40,000 MWd - 5 yr. Cool~~ | ~~Neutron Source-Strength 45,000 MWd - 6 yr. Cool~~ | ~~Normalized Neutron Source Spectrum Fraction~~ |
|---|---|---|---|
| ~~6.43 - 20.0~~ | ~~1.695E+08~~ | ~~2.197E+08~~ | ~~0.0185~~ |
| ~~3.0 - 6.43~~ | ~~1.925E+09~~ | ~~2.494E+09~~ | ~~0.21~~ |
| ~~1.85 - 3.0~~ | ~~2.126E+09~~ | ~~2.755E+09~~ | ~~0.232~~ |
| ~~1.4 - 1.85~~ | ~~1.201E+09~~ | ~~1.556E+09~~ | ~~0.131~~ |
| ~~0.9 - 1.4~~ | ~~1.622E+09~~ | ~~2.102E+09~~ | ~~0.177~~ |
| ~~0.4 - 0.9~~ | ~~1.769E0+09~~ | ~~2.292E+09~~ | ~~0.193~~ |
| ~~0.1 - 0.4~~ | ~~3.464E+08~~ | ~~4.489E+08~~ | ~~0.0378~~ |
| ~~Total~~ | ~~9.159E+09~~ | ~~1.187E+10~~ | ~~1.0~~ |

| *Neutron Energy Range (MeV)* | *Neutron Source-Strength 42,000 MWD/MTU 9 year cooling (neutrons/sec/assembly)* |
|---|---|
| *0.1 - 0.4* | *1.24E+07* |
| *0.4 - 0.9* | *6.34E+07* |
| *0.9 - 1.4* | *5.81E+07* |
| *1.4 - 1.85* | *4.29E+07* |
| *1.85 - 3.0* | *7.58E+07* |
| *3.0 - 6.43* | *6.87E+07* |
| *6.43 - 20.0* | *6.08E+06* |
| *Total* | *3.28E+08* |

**Table 7.2-4**

**Relative Burnup Level and Source Strengths for PWR Assembly Axial Sub-Sections**

| Axial Span (inches from fuel bottom | Relative Burnup Level | Relative Neutron Source Strength |
|---|---|---|
| 0 - 7.2 | 0.59 | 0.109 |
| 7.2 - 14.4 | 0.89 | 0.613 |
| 14.4 - 21.6 | 1.03 | 1.132 |
| 21.6 - 28.8 | 1.07 | 1.329 |
| 28.8 - 36.0 | 1.09 | 1.436 |
| 36.0 - 64.8 | 1.1 | 1.492 |
| 64.8 - 100.8 | 1.09 | 1.436 |
| 100.8 - 108.0 | 1.07 | 1.329 |
| 108.0- 115.2 | 1.05 | 1.227 |
| 115.2 - 122.4 | 1.02 | 1.087 |
| 122.4 - 129.6 | 0.96 | 0.842 |
| 129.6 - 136.8 | 0.82 | 0.435 |
| 136.8 - 144.0 | 0.56 | 0.088 |

**Table 7.2-5**

**~~NON-FUEL REGION Co$^{60}$~~**
**~~GAMMA SOURCE STRENGTHS~~**
**~~(γ/sec-cask)~~*MASS OF NON-FUEL MATERIAL IN THE WESTINGHOUSE 17x17 FUEL ASSEMBLY***

| ~~Non-Fuel Region~~ | ~~Fuel Burnup Level (MWd/MTU)~~ | ~~Cooling Time (years)~~ | ~~1.173 MeV Gamma Source Strength~~ | ~~1.333 MeV Gamma Source Strength~~ |
|---|---|---|---|---|
| ~~Bottom Nozzle~~ | ~~40,000~~ | ~~5~~ | ~~5.021E+12~~ | ~~5.021E+12~~ |
| ~~Gas Plenum~~ | ~~40,000~~ | ~~5~~ | ~~7.347E+12~~ | ~~7.347E+12~~ |
| ~~Top Nozzle~~ | ~~40,000~~ | ~~5~~ | ~~1.538E+13~~ | ~~1.538E+13~~ |
| ~~Bottom Nozzle~~ | ~~45,000~~ | ~~6~~ | ~~4.718E+12~~ | ~~4.718E+12~~ |
| ~~Gas Plenum~~ | ~~45,000~~ | ~~6~~ | ~~6.904E+12~~ | ~~6.904E+12~~ |
| ~~Top Nozzle~~ | ~~45,000~~ | ~~6~~ | ~~1.445E+13~~ | ~~1.445E+13~~ |

| ***Region*** | ***Mass of Stainless Steel (kg)*** | ***Mass of Inconel (kg)*** |
|---|---|---|
| *Bottom end fitting* | *5.9* | -- |
| *In-core grid spacers* | -- | *4.9* |
| *Plenum region 1* | *1.15* | -- |
| *Plenum region 2* | *0.84* | *0.79* |
| *Top end fitting* | *6.89* | *0.96* |

*Table 7.2-6*

***SCALING FACTORS USED IN CALCULATING THE Co-60 SOURCE***

| *Region* | *Scaling Factor* |
|---|---|
| *Bottom end fitting* | *0.2* |
| *In-core grid spacers* | *1.0* |
| *Plenum region 1* | *0.2* |
| *Plenum region 2* | *0.2* |
| *Top end fitting* | *0.1* |

*Table 7.2-7*

***Co-60 GAMMA SOURCE STRENGTHS***
***(curies Co-60/assembly)***

| ***Region*** | ***Co-60 Source Term 42,000 MWD/MTU 9 years of cooling*** |
|---|---|
| *Bottom end fitting* | *60.7* |
| *In-core grid spacers* | *251.9* |
| *Plenum region 1* | *11.8* |
| *Plenum region 2* | *16.8* |
| *Top end fitting* | *40.4* |

**Table 7.2-8**

**MASS OF MATERIAL IN THE THIMBLE PLUG DEVICES**

| Non-Fuel Region | Mass of Stainless Steel (kg) | Mass of Inconel (kg) |
|---|---|---|
| Plenum region 1 | 1.6 | -- |
| Plenum region 2 | 1.6 | -- |
| Top end fitting | 2.31 | 0.42 |

*Table 7.2-9*

***THIMBLE PLUG DEVICE Co-60 GAMMA SOURCE STRENGTHS (curies Co-60/device )***

| ***Non-Fuel Region*** | ***Co-60 Source Term 118,674 MWD/MTU 11 years of cooling*** |
|---|---|
| *Plenum region 1* | *12.8* |
| *Plenum region 2* | *12.8* |
| *Top end fitting* | *18.9* |

## Table 7.3-1

## *MPC AND CONCRETE CASK* ELEMENTAL DENSITIES (gm/cc)

| Elements | Atomic # | Concrete | SS 304 | Carbon Steel | Lead | Air | Neutron Shield | Fuel | Bottom Nozzle | Plenum | Top Nozzle |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Hydrogen | 1 | 0.013 | | | | | 4.10E-2 | | | | |
| Boron-10 | 5 | | | | | | 8.000E-3 | 0.003 | | | |
| Boron-11 | 5 | | | | | | 3.800E-2 | 0.013 | | | |
| Carbon | 6 | | | | | | 2.000E-2 | 0.004 | | | |
| Nitrogen | 7 | | | | | 9.765E-4 | | | | | |
| Oxygen | 8 | 1.165 | | | | 2.997E-4 | 9.59E-1 | 0.198 | | | |
| Sodium | 11 | 0.040 | | | | | 1.000E-3 | | | | |
| Mg | 12 | 0.006 | | | | | | | | | |
| Aluminum | 13 | 0.107 | | | | | 2.610E-1 | 0.044 | | | |
| Silicon | 14 | 0.737 | 0.059 | | | | 1.000E-2 | | 0.008 | | 0.006 |
| Sulfur | 16 | 0.003 | | | | | 1.050E-1 | | | | |
| Argon | 18 | | | | | 1.665E-5 | | | | | |
| Potassium | 19 | 0.045 | | | | | | | | | |
| Calcium | 20 | 0.194 | | | | | 1.310E-1 | | | | |
| Chromium | 24 | | 1.505 | | | | | | 0.207 | | 0.144 |
| Mang. | 25 | | 0.159 | | | | 1.000E-3 | | 0.022 | | 0.015 |
| Iron | 26 | 0.029 | 5.465 | 7.821 | | | 6.900E-1 | 0.511 | 1.189 | 0.504 | 0.525 |
| Cobalt | 27 | | | | | | | | | | |
| Nickel | 28 | | 0.733 | | | | | | 0.101 | 0.025 | 0.070 |
| Zirc. | 40 | | | | | | | 0.319 | 0.227 | 0.319 | 0.031 |
| Niobium | 41 | | | | | | | | | | |
| Moly | 42 | | | | | | | | | | |
| Lead | 82 | | | | 11.34 | | | | | | |
| U-238 | 92 | | | | | | | 1.476 | | | |
| Total | | 2.339 | 7.921 | 7.821 | 11.34 | 1.293E-3 | 2.265 | 2.568 | 1.754 | 0.848 | 0.791 |

| Elements | Atomic # | Concrete | SS 304 | Carbon Steel | Boral | Air | Soil | Active Fuel | Lead | Water |
|---|---|---|---|---|---|---|---|---|---|---|
| Hydrogen | 1 | 0.013 | | | | | 0.01635 | | | 0.10229 |
| Boron-10 | 5 | | | | 0.11934 | | | | | |
| Boron-11 | 5 | | | | 0.54405 | | | | | |
| Carbon | 6 | | | 0.0391 | 0.18414 | | | | | |
| Nitrogen | 7 | | | | | 8.849E-4 | | | | |
| Oxygen | 8 | 1.165 | | | | 2.851E-4 | 0.92414 | 0.37423 | | 0.81171 |
| Sodium | 11 | 0.040 | | | | | | | | |
| Mg | 12 | 0.006 | | | | | | | | |
| Aluminum | 13 | 0.107 | | | 1.85247 | | 0.21860 | | | |
| Silicon | 14 | 0.737 | | | | | 0.54091 | | | |
| Sulfur | 16 | 0.003 | | | | | | | | |
| Potassium | 19 | 0.045 | | | | | | | | |
| Calcium | 20 | 0.194 | | | | | | | | |
| Chromium | 24 | | 1.5048 | | | | | 0.00084 | | |
| Mang. | 25 | | 0.1584 | | | | | | | |
| Iron | 26 | 0.029 | 5.5044 | 7.7809 | | | | 0.00156 | | |
| Nickel | 28 | | 0.7524 | | | | | | | |
| Copper | 29 | | | | | | | | 0.0090 | |
| Zirc. | 40 | | | | | | | 0.64462 | | |
| Silver | 47 | | | | | | | | 0.0023 | |
| Tin | 50 | | | | | | | 0.01118 | | |
| Lead | 82 | | | | | | | | 11.2887 | |
| U-235 | 92 | | | | | | | 0.09018 | | |
| U-238 | 92 | | | | | | | 2.69369 | | |
| Total | | 2.339 | 7.92 | 7.82 | 2.7 | 1.17E-3 | 1.7 | 3.8163 | 11.3 | 0.914 |

*Different densities were used for stainless steel to represent the fuel assembly hardware as shown in Table 7.3-2.*

## Table 7.3-2

## *AXIAL CONFIGURATION OF THE MODELED FUEL ASSEMBLY*

## ~~ELEMENTAL DENSITIES (Atoms/barn-cm)~~

| ~~Elements~~ | ~~Atomic #~~ | ~~Concrete~~ | ~~SS304~~ | ~~Carbon Steel~~ | ~~Lead~~ | ~~Air~~ | ~~Neutron Shield~~ | ~~Fuel~~ | ~~Bottom Nozzle~~ | ~~Plenum~~ | ~~Top Nozzle~~ |
|---|---|---|---|---|---|---|---|---|---|---|---|
| ~~Hydrogen~~ | ~~1~~ | ~~7.770E-3~~ | | | | | ~~2.472E-2~~ | | | | |
| ~~Boron-10~~ | ~~5~~ | | | | | | ~~5.113E-4~~ | ~~1.720E-4~~ | | | |
| ~~Boron-11~~ | ~~5~~ | | | | | | ~~2.058E-3~~ | ~~7.060E-4~~ | | | |
| ~~Carbon~~ | ~~6~~ | | | | | | ~~1.005E-3~~ | ~~2.180E-4~~ | | | |
| ~~Nitrogen~~ | ~~7~~ | | | | | ~~4.199E-5~~ | | | | | |
| ~~Oxygen~~ | ~~8~~ | ~~4.390E-2~~ | | | | ~~1.128E-5~~ | ~~3.608E-2~~ | ~~7.472E-3~~ | | | |
| ~~Sodium~~ | ~~11~~ | ~~1.050E-3~~ | | | | | ~~2.580E-5~~ | | | | |
| ~~Mg~~ | ~~12~~ | ~~1.490E-4~~ | | | | | | | | | |
| ~~Aluminum~~ | ~~13~~ | ~~2.390E-3~~ | | | | | ~~5.829E-3~~ | ~~9.830E-4~~ | | | |
| ~~Silicon~~ | ~~14~~ | ~~1.580E-2~~ | ~~1.270E-3~~ | | | | ~~2.253E-4~~ | | ~~1.750E-4~~ | | ~~1.220E-4~~ |
| ~~Sulfur~~ | ~~16~~ | ~~5.640E-5~~ | | | | | ~~1.967E-3~~ | | | | |
| ~~Argon~~ | ~~18~~ | | | | | ~~2.510E-7~~ | | | | | |
| ~~Potassium~~ | ~~19~~ | ~~6.960E-4~~ | | | | | | | | | |
| ~~Calcium~~ | ~~20~~ | ~~2.920E-3~~ | | | | | ~~1.968E-3~~ | | | | |
| ~~Chromium~~ | ~~24~~ | | ~~1.743E-2~~ | | | | | | ~~2.397E-3~~ | | ~~1.673E-3~~ |
| ~~Mang.~~ | ~~25~~ | | ~~1.740E-3~~ | | | | ~~8.997E-6~~ | | ~~2.400E-4~~ | | ~~1.670E-4~~ |
| ~~Iron~~ | ~~26~~ | ~~3.130E-4~~ | ~~5.894E-2~~ | ~~8.435E-2~~ | | | ~~7.443E-3~~ | ~~5.516E-3~~ | ~~1.282E-2~~ | ~~5.436E-3~~ | ~~5.658E-3~~ |
| ~~Cobalt~~ | ~~27~~ | | | | | | | | | | |
| ~~Nickel~~ | ~~28~~ | | ~~7.520E-3~~ | | | | | | ~~1.034E-3~~ | ~~2.515E-4~~ | ~~7.220E-4~~ |
| ~~Zirc.~~ | ~~40~~ | | | | | | | ~~2.103E-3~~ | ~~1.502E-3~~ | ~~2.103E-3~~ | ~~2.030E-4~~ |
| ~~Niobium~~ | ~~41~~ | | | | | | | | | | |
| ~~Moly~~ | ~~42~~ | | | | | | | | | | |
| ~~Lead~~ | ~~82~~ | | | | ~~3.296E-2~~ | | | | | | |
| ~~U-238~~ | ~~92~~ | | | | | | | ~~3.736E-3~~ | | | |
| ~~Total~~ | | ~~7.504E-2~~ | ~~8.690E-2~~ | ~~8.435E-2~~ | ~~3.296E-2~~ | ~~5.352E-5~~ | ~~8.184E-2~~ | ~~2.091E-2~~ | ~~1.817E-2~~ | ~~7.791E-3~~ | ~~8.545E-3~~ |

| ***Region*** | ***Start (in.)*** | ***Finish (in.)*** | ***Length (in.)*** | ***Material Modeled*** | ***Density (gm/cc)*** |
|---|---|---|---|---|---|
| *Bottom end fitting* | *0.0* | *2.738* | *2.738* | *Steel* | *1.850* |
| *Space* | *2.738* | *3.738* | *1.0* | *Void* | *--* |
| *Active Fuel* | *3.738* | *147.738* | *144.0* | *Fuel* | *3.816* |
| *Plenum region 1* | *147.738* | *151.916* | *4.178* | *Steel* | *0.236* |
| *Plenum region 2* | *151.916* | *156.095* | *4.179* | *Steel* | *0.336* |
| *Top end fitting* | *156.095* | *159.765* | *3.67* | *Steel* | *1.836* |

**Table 7.3-3**

**NEUTRON ENERGY GROUP FLUX-TO-DOSE CONVERSION FACTORS**

**(mrem/hr per neutron/cm$^2$-sec)**

***(from Reference 10)***

| Neutron Energy (MeV) | Conversion Factor |
|---|---|
| *2.5E-08*~~2.12E-07~~ | *3.67E-03*~~3.78E-03~~ |
| *1.0E-07*~~7.67E-07~~ | *3.67E-03*~~3.96E-03~~ |
| *1.0E-06*~~2.09E-06~~ | *4.46E-03*~~4.14E-03~~ |
| *1.0E-05*~~6.58E-06~~ | *4.54E-03*~~4.32E-03~~ |
| *1.0E-04*~~1.96E-05~~ | *4.18E-03*~~4.50E-03~~ |
| *1.0E-03*~~6.50E-05~~ | *3.76E-03*~~4.68E-03~~ |
| *1.0E-02*~~3.42E-04~~ | *3.56E-03*~~4.68E-03~~ |
| *0.1*~~1.97E-03~~ | *2.17E-02*~~4.32E-03~~ |
| *0.5*~~5.72E-02~~ | *9.26E-02*~~6.48E-03~~ |
| *1.0*~~0.331~~ | *1.32E-01*~~5.40E-02~~ |
| *2.5*~~0.83~~ | *1.25E-01*~~0.1188~~ |
| *5.0*~~1.47~~ | *1.56E-01*~~0.1332~~ |
| *7.0*~~2.09~~ | *1.47E-01*~~0.1296~~ |
| *10.0*~~2.41~~ | *1.47E-01*~~0.1260~~ |
| *14.0*~~2.74~~ | *2.08E-01*~~0.1260~~ |
| *20.0*~~3.54~~ | *2.27E-01*~~0.1296~~ |
| ~~4.51~~ | ~~0.1332~~ |
| ~~5.66~~ | ~~0.1404~~ |
| ~~7.27~~ | ~~0.1476~~ |
| ~~9.09~~ | ~~0.1476~~ |
| ~~11.1~~ | ~~0.1656~~ |
| ~~13.56~~ | ~~0.2088~~ |

## Table 7.3-4

## Gamma Flux-to- Dose Conversion Factors

## (mrem/hr per $\gamma/cm^2$-sec)

***(from Reference 10)***

| Gamma Energy (MeV) | Conversion Factor |
|---|---|
| 0.01 | 3.96E-03 |
| 0.03 | 5.82E-04 |
| 0.05 | 2.90E-04 |
| 0.07 | 2.58E-04 |
| 0.1 | 2.83E-04 |
| 0.15 | 3.79E-04 |
| 0.2 | 5.01E-04 |
| 0.25 | 6.31E-04 |
| 0.3 | 7.59E-04 |
| 0.35 | 8.78E-04 |
| 0.4 | 9.85E-04 |
| 0.45 | 1.08E-03 |
| 0.5 | 1.17E-03 |
| 0.55 | 1.27E-03 |
| 0.6 | 1.36E-03 |
| 0.65 | 1.44E-03 |
| 0.7 | 1.52E-03 |
| 0.8 | 1.68E-03 |
| 1.0 | 1.98E-03 |
| 1.4 | 2.51E-03 |
| 1.8 | 2.99E-03 |
| 2.2 | 3.42E-03 |
| 2.6 | 3.82E-03 |
| 2.8 | 4.01E-03 |
| 3.25 | 4.41E-03 |
| 3.75 | 4.83E-03 |
| 4.25 | 5.23E-03 |
| 4.75 | 5.60E-03 |
| 5.0 | 5.80E-03 |
| 5.25 | 6.01E-03 |
| 5.75 | 6.37E-03 |
| 6.25 | 6.74E-03 |
| 6.75 | 7.11E-03 |
| 7.5 | 7.66E-03 |
| 9.0 | 8.77E-03 |
| 11.0 | 1.03E-02 |
| 13.0 | 1.18E-02 |
| 15.0 | 1.33E-02 |

**Table 7.4-1**

**Maximum Expected Dose Rates**
**for the Storage Cask System**
***(42,000 MWD/MTU and 9-Year Cooling)***

| | Dose Rate (mrem/hr ) | | |
|---|---|---|---|
| Location | Design | Surface[a] | Working[b] |
| Transfer Cask Side | NA[c] | ~~545.8~~ *199.5* | ~~193.2~~ *78.2* |
| ~~Basket~~ Top (~~Outside Surface of Structural~~ *MPC* Lid) | NA[c] | ~~213.4~~ *484.7* | ~~180~~*211.6* (estimated) |
| Concrete Cask Top | ~~250~~*300*[d] | ~~158.5~~ *275.3* | ~~133.9~~ *111.1* |
| Concrete Cask Side | 100[d] | ~~19.1~~*13.7* | ~~10.2~~*6.5* |

a Surface dose rate is calculated on the surface, i.e., on contact.

b Working dose rate is the calculated dose rate one meter from the surface.

c NA = Not Applicable: Storage System Design Limits are not given for the Transfer Cask and ~~PWR Basket~~*MPC* Top. The Transfer Cask and ~~PWR Basket~~*MPC* Top present radiation dose rate concerns only during handling and transfer activities (e.g., short-term, temporary). Expected occupational doses for these activities are addressed in Tables 7.4-3~~ and 7.3-4~~.

d *The as-configured Trojan Storage System results in average external dose rates that are less than these values.*

## Table 7.4-3

## Estimated Personnel Doses while Operating the Cask System

### *(42,000 MWD/MTU and 9-Year Cooling)*

| Activity | Personnel Work Groups | Exposure Time (hrs) | Working Dose Rate (mrem/hr) | Exposure* (person-mrem) |
|---|---|---|---|---|
| Load Transfer Cask | 2 Operators | 8.0 | 0.2[a] | 3 |
| Monitor | 1 R.P. | 5.5 | 0.2[a] | 1 |
| Decontaminate Cask | 2 R.P. | 4.0 | ~~19~~*78.3*[b] | ~~1576~~*626* |
| Monitor | 1 R.P. | 1.0 | ~~19~~*78.3*[b] | ~~19~~*78* |
| Weld ~~Shield~~ *MPC* Lid | 2 Welders/Operators<br>1 Inspector (at ~~basket~~ *MPC*) | 1.33 | ~~70~~[d]*152.5* | ~~279~~*608* |
| ~~Weld Structural Lid~~ | ~~2 Welders/~~<br>~~1 Inspector (at basket)~~ | ~~2.5~~ | ~~49~~[e] | ~~368~~ |
| Weld ~~Shield and Structural~~ *MPC* Lid | 2 Welders/<br>1 Inspector (on platform) | 1.75 | ~~14~~*26.8* | ~~7~~*4141* |
| Weld ~~Shield and Structural~~ *MPC* Lid | ~~3~~*2* Welders/<br>1 Inspector (at system control) | ~~32.8~~*17.0* | ~~1.~~*42.7* | ~~18~~*4138* |
| ~~Vacuum~~ *MPC* Dry/*Backfill* | 1 Technician | 8.0 | ~~14~~*26.8* | ~~112~~*214* |
| *Weld Closure Ring* | *2 Welders/*<br>*1 Inspector (at MPC)* | *1.0* | *152.5* | *458* |
| *Weld Closure Ring* | *2 Welders/*<br>*1 Inspector (on Platform)* | *1.0* | *26.8* | *80* |
| *Weld Closure Ring* | *2 Welders/*<br>*1 Inspector (at system control)* | *1.0* | *2.7* | *8* |
| Monitor | 1 R.P. | 1.0 | ~~14~~*26.8* | ~~14~~*27* |
| Load Storage Cask | 2 Operators | 1.5 | ~~14~~*26.8* | ~~42~~*80* |
| Monitor | 1 R.P. | 1.0 | ~~14~~*26.8* | ~~14~~*27* |
| Totals | | ~~68.38~~ | ---- | ~~286~~*42,489* |
| Move to Storage | 2 Operators | 2.0 | ~~10~~*6.5* | ~~40~~*26* |
| Monitor | 1 R.P. | 2.0 | ~~10~~*6.5* | ~~20~~*13* |
| Totals | | ~~4.0~~ | ---- | ~~60~~*39* |
| Load ~~Shipping~~ *Transport* Cask | 2 Operators | 20.0 | ~~12~~*20.0* | ~~480~~*800* |
| Monitor | 1 R.P. | 2.0 | ~~12~~*20.0* | ~~24~~*40* |
| Totals | | ~~22.0~~ | ---- | ~~504~~*840* |
| ~~Annual~~ *Weekly* Surveillances of ~~Fuel Casks in Storage~~*ISFSI* | 1 Specialist | ~~424.7~~[c]*8.7*[c] | ~~10~~*27.4* | ~~4247~~[c]*238*[c] |
| ~~5-yr~~ *Annual* Inspections ~~of Concrete Cask~~ *and Quarterly Surveys* | 1 Specialist | ~~1.5~~*4.0*[c] | ~~10~~*27.4* | ~~15~~*110*[c] |
| *Regular ISFSI Maintenance* | *2 Specialists* | *12.0*[c] | *31.6* | *758*[c] |
| *Totals* | | | | *1106*[c] |

* Calculated exposures have been rounded to the nearest whole number.

a Radiation reading in Spent Fuel Pool area.

b Assumes worst case of dry ~~basket~~*MPC*. If water is left in ~~basket~~ *MPC* as planned, dose rate will be less.

c ~~Includes daily visual inspection of air inlets/outlets, twice daily temperature reading, and annual concrete inspection of 34 fuel Concrete Casks. Also, includes 25 mrem/year for quarterly radiation protection surveys and annual concrete Storage Pad inspection for entire ISFSI.~~*These values reflect amounts accumulated in one year.*

~~d Temporary shielding on shield lid during welding and NDE. Expected average dose rate.~~

~~e Temporary shielding on structural lid during welding and NDE. Expected average dose rate.~~

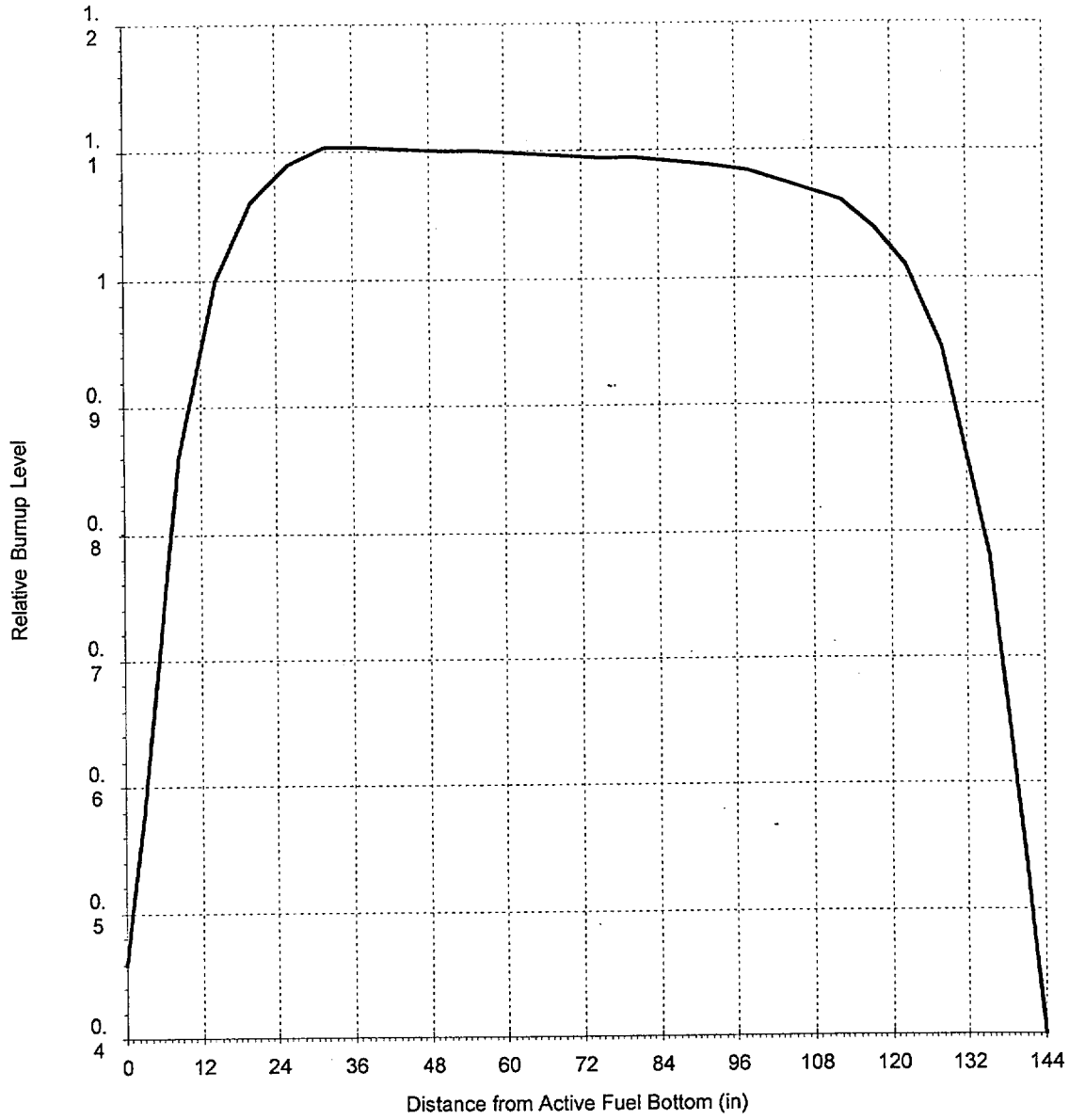


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**FIGURE 7.2-1**
**PWR FUEL ASSEMBLY AXIAL BURNUP PROFILE**

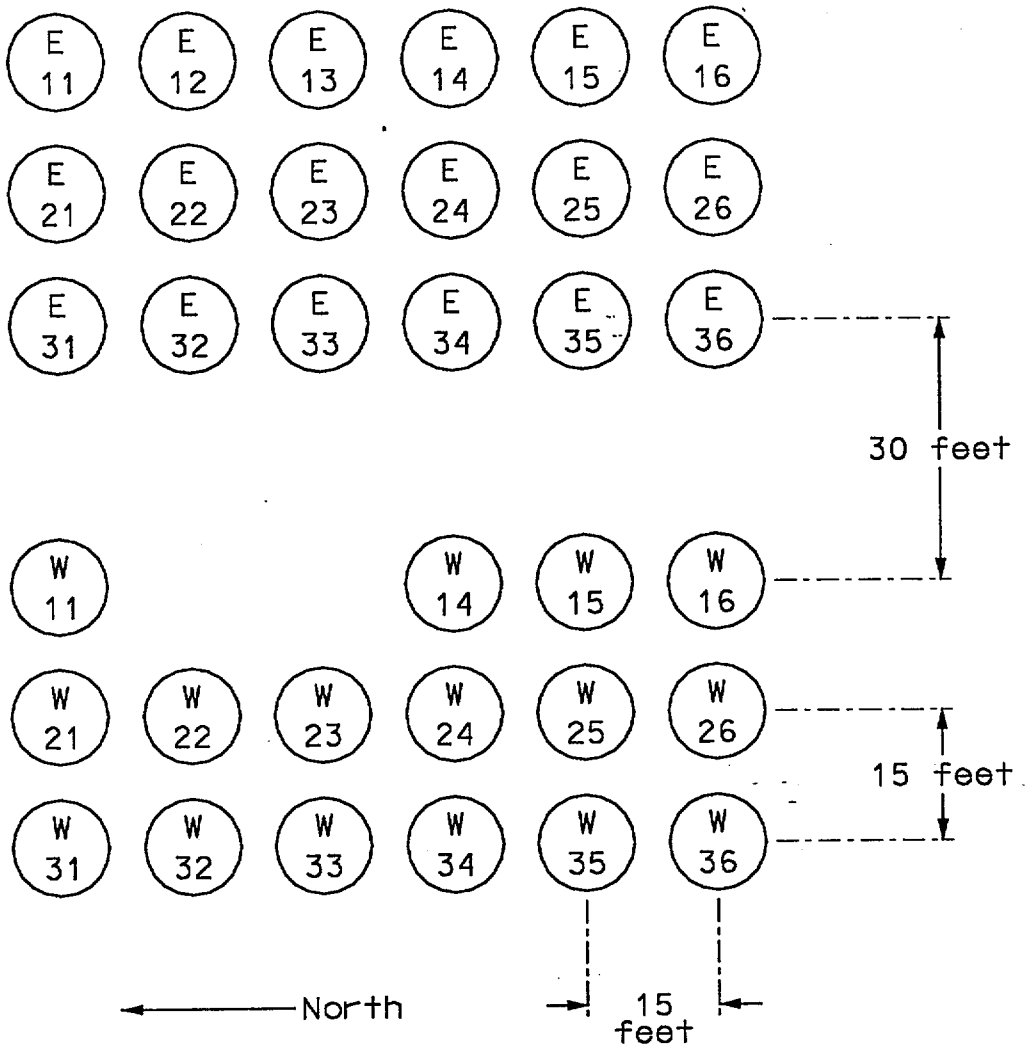


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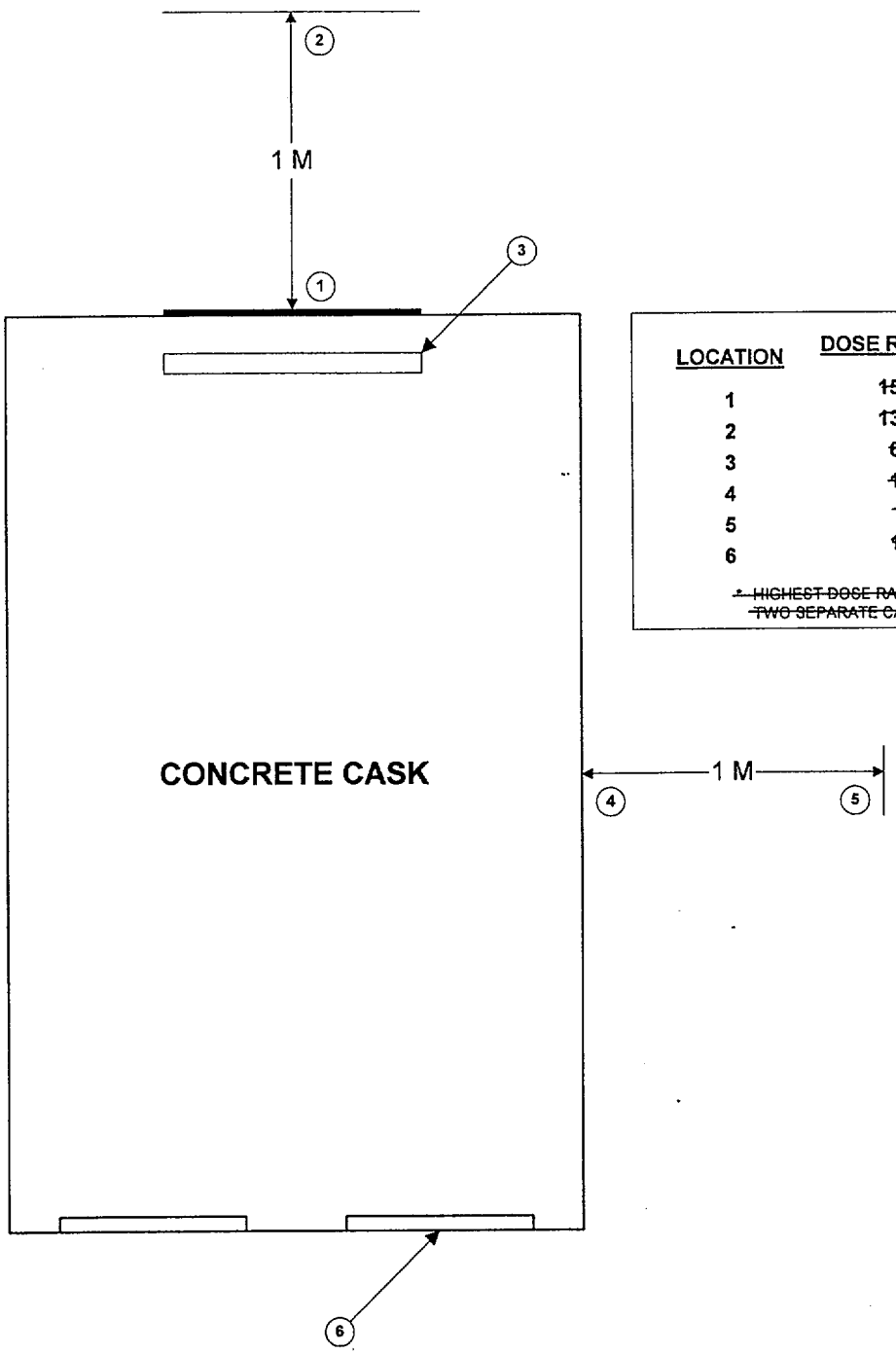
FIGURE 7.3-1
ISFSI PAD CONFIGURATION

REVISION 2

g:/newtroj/common/e66497/isfsi/sarfig7 3-1.dgn




| LOCATION | DOSE RATE (mrem/hr) |
|---|---|
| 1 | ~~156.5~~ 275.3 |
| 2 | ~~133.9~~ 111.1 |
| 3 | ~~6.76~~ 38.7 |
| 4 | ~~19.1~~ 13.7 |
| 5 | ~~10.2~~ 6.5 |
| 6 | ~~18.77~~ 3.9 |

~~* HIGHEST DOSE RATE FROM TWO SEPARATE CASES~~

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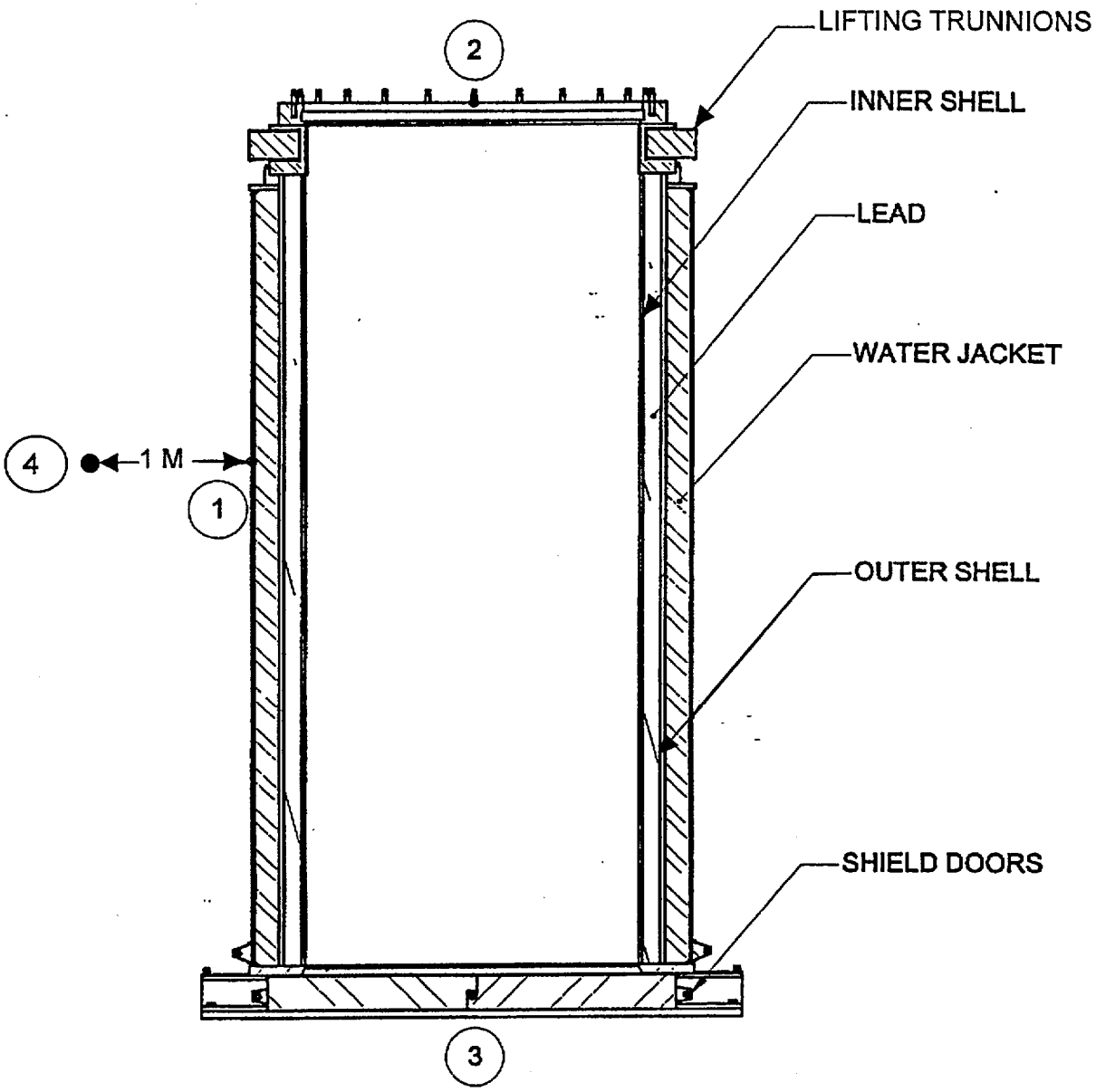
**FIGURE 7.3-2**
SUMMARY OF DOSE RATES FOR CONCRETE CASK ~~(PWR)~~

Revision 2




| LOCATION | DOSE RATE (mremlhr) |
|---|---|
| 1 | 199.5 |
| 2 | 484.7 |
| 3 | 563.0 |
| 4 | 78.2 |

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FIGURE 7.3-3
SUMMARY OF DOSE RATES FOR TRANSFER CASK

Revision 2

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