Chapter 8

8.0 ACCIDENT ANALYSIS

The analyses of normal and off-normal events and accident design events identified by ANSI/ANS 57.9, as applicable to the ISFSI, are presented in this Chapter. Regulatory Guide 3.48 specifies that the four event types in ANSI/ANS 57.9 be addressed. Design Events I and II consist of normal and off-normal events that are expected to occur routinely or to occur with a frequency of approximately once per year. Design Events III and IV consist of infrequent events and postulated accidents that might occur over the lifetime of the ISFSI or hypothetical events that are postulated because their consequences may result in the maximum potential impact on the immediate environment. Section 8.1 addresses the normal and off-normal events, and Table 8.0-1 lists these events and the ISFSI components evaluated for each of them. Section 8.2 addresses the infrequent events, and Table 8.0-2 lists these events and the components evaluated for each of them. In addition, this Chapter identifies the accident conditions considered in the design of the ISFSI in accordance with State of Oregon OAR 345-26-390(4)(a).

The evaluations of normal, off-normal, and postulated accident conditions assure that the ISFSI components that are classified as important to safety are capable of performing their required functions. The required functions of the ~~PWR Basket,~~*MPC* ~~Basket Overpack,~~ and Concrete Cask are identified in Section 3.3.1.

The fuel handling components that are part of the Trojan ISFSI are identified in Section 4.7. Fuel handling components classified as important to safety are the Transfer Cask and the Transfer Station. They are relied upon to safely handle the ~~PWR Baskets~~*MPCs* containing high-level radioactive materials and minimize the potential for their drop. The other components perform no functions that are important to safety for ISFSI operations.

The analyses in this chapter reflect normal, off-normal, and infrequent events that are postulated to occur while the loaded Concrete Cask is handled and stored on the ISFSI Storage Pad. Events that could occur during loading of the Concrete Cask in the Fuel Building and transport to the Storage Pad are addressed in the 10 CFR 50 license. Events that could occur after a*n* ~~PWR Basket~~ *MPC* has been loaded into a ~~Shipping~~ *Transport* Cask for transport and moved out of the ISFSI ~~will be~~*are* addressed in ~~conjunction with~~ the ~~Shipping~~ *HI-STAR 100 Transport* Cask ~~s~~*S*afety ~~a~~*A*nalysis *Report (Docket 71-9261; Reference 4)* ~~submitted pursuant to 10 CFR 71~~.

8.1 NORMAL AND OFF-NORMAL EVENTS

This section covers Design Events I and II: events that would be expected to occur during normal operations and those that might occur with moderate frequency on the order of once during any calendar year of operations.

Normal operation of the ISFSI equipment and appurtenances has been described in Chapter 5. These operations include:

1. Normal operational handling, lifting, loading, and transporting the ~~PWR Basket~~*MPC* and Transfer Cask within the Fuel Building.

2. Loading the ~~PWR Basket~~*MPC* into the Concrete Cask.

3. Moving and locating the loaded Concrete Cask onto the ISFSI Storage Pad.

4. Storage of the loaded Concrete Cask on the ISFSI Storage Pad.

5. Retrieval of the ~~PWR Basket~~*MPC* from the Concrete Cask and loading it into a ~~Shipping~~ *Transport* Cask for off-site disposal.

The structural analysis of the ISFSI components for normal operations includes consideration of anticipated loads on the Concrete Cask, ~~PWR Basket~~*MPC*, and ~~PWR Basket internals~~*MPC fuel basket* during storage and handling operations. The structural analysis methodology and results have been presented in Section 4.2.5. Chapter 4 also includes ~~a structural analysis of the PWR Basket Overpack and~~ analyses of Concrete Cask and ~~PWR Basket~~*MPC* thermal-hydraulic *performance,* and criticality performance *of the MPC,* under normal storage conditions.

8.1.1 OFF-NORMAL STRUCTURAL ANALYSIS

8.1.1.1 ~~PWR Basket~~*MPC* Off-Normal Handling Load

This event consists of a lateral impact of the ~~PWR Basket~~*MPC* against the inside of the Concrete Cask.

8.1.1.1.1 Postulated Cause of Event

During transfer of the Concrete Cask to the ISFSI Storage Pad, an inadvertent movement may cause lateral impact of the ~~PWR Basket~~*MPC* against the inside of the Concrete Cask. Additionally, during ~~placement~~ *removal* of the ~~PWR Basket~~*MPC* ~~into~~ *from* the Concrete Cask *for transport,* crane operation may cause a lateral impact against the inside of the Concrete Cask.

8.1.1.1.2 Detection of Event

This event may be detected by observation of personnel monitoring Concrete Cask movement operations or ~~Concrete Cask loading~~*MPC handling* operations.

8.1.1.1.3 Analysis of Effects and Consequences

~~The off-normal handling load was analyzed for the Concrete Cask assuming 2 ft/sec Concrete Cask speed.~~ *The approach used to evaluate the effect of the off-normal handling load with a Concrete Cask containing a Holtec MPC is to compare the load imparted on the PWR Basket in this event with the load for which the MPC is qualified in the HI-STAR 100 generic licensing basis (Docket 72-1008).* The Concrete Cask speed will be limited by administrative procedures to no greater than 2 ft/sec. This is equivalent to a drop from a height of :

$$h = v^2/2g = 0.062 \text{ ft} = 0.75 \text{ in}$$

where: v = velocity (ft/sec) = 2 ft/sec
g = acceleration of gravity = 32.2 ft/sec^2

The deceleration applied to the PWR Basket during such impact ~~can be~~*was* found using the following formula (Reference ~~8.~~1, Chapter 15).

$$a = g[1+\sqrt{1+2h/\delta_{st}}]$$

where: δ_{st} = deflection due to the dead weight load = 0.0055 in
h = drop height = 0.75 in
g = acceleration of gravity

The dead weight deflection is calculated (by the factoring of maximum deflection in the horizontal drop analysis) as 0.0055 in. Thus, the equivalent static deceleration level is:

$$a = g[1+\sqrt{1+(2 \times 0.75 / 5.5\text{E-}3)}] = 17.5g$$

~~The methodology applied in the analysis is a scaling of the results of the Shipping Cask drop analyses. The highest drop stress (horizontal or vertical) is multiplied by the ratio of handling acceleration (17.5g) to the drop acceleration. The resulting handling stresses are identified in Table 8.1-1.~~

The *calculated* stresses *in the PWR Basket* due to *this* off-normal handling impact ~~have been~~*were* combined with the stresses due to other loads and *then compared with ASME Code allowables*

for Service Level C loadings. ~~evaluated in Table 8.1-1. It can be seen that primary membrane stress intensity (P_m) and the maximum local primary membrane plus primary bending stress intensity ($P_L + P_b$) values are within ASME code allowables for Service Level C loadings. In addition, the stresses on the PWR Basket internals are within the ASME code allowables for Service Level C loadings.~~

The design basis handling accident for the Holtec MPC is a 60g lateral side drop, which is considered an ASME Service Level D loading. The minor physical differences between the Trojan-specific MPC-24E/EF and the generically certified Holtec MPC-24E/EF have a negligible effect on the structural strength of the MPC. The calculated stresses in the MPC due to this accident condition are presented in Table 8.1-1. Since the design basis impact load applied to the MPC is more than 3 times greater than the load applied to the PWR Basket (i.e., 60g vs. 17.5g) and the stress allowables are only approximately 60 percent greater (i.e., Level D vs. Level C), it is proven by comparison that the MPC can withstand the off-normal handling load described above.

8.1.1.1.4 Corrective Actions

The ~~PWR Basket~~*MPC* is designed to withstand acceleration loads which bound this handling load. No corrective actions are required.

8.1.2 OFF-NORMAL THERMAL ANALYSIS

8.1.2.1 Severe Environmental Condition

This event involves severe environmental conditions consisting of sustained high temperature and low temperature cases.

8.1.2.1.1 Postulated Cause of Event

Although sustained temperature extremes of the magnitude analyzed are not expected, it is assumed that the loaded Concrete Casks on the Storage Pad are subjected to sustained high and low ambient temperatures. Analyses were performed to calculate the steady state Concrete Cask, ~~PWR Basket~~*MPC*, and fuel cladding temperatures for sustained 100 °F ambient conditions with 24 hour average solar loads and for -40 °F ambient conditions with no solar load. These analyses assume that the Concrete Cask has reached a steady state condition relative to the assumed ambient temperature. The maximum thermal payload of ~~26~~ *17.4* kWt was also used for this analysis.

The maximum anticipated heat load analysis in Section 8.2.2 evaluates an ambient temperature of 125°F which envelopes the maximum historical ambient temperature of 107°F experienced in the region.

While the temperatures for the -40°F case do not represent the absolute lowest component temperatures that could occur for a -40°F condition (as a result of using the maximum thermal load), the -40°F condition with the maximum thermal load results in the highest thermal gradients in the ~~PWR Basket~~*MPC* structure.

8.1.2.1.2 Detection of Event

This event may be detected by the observation by personnel and confirmed by ambient temperature monitoring.

8.1.2.1.3 Analysis of Effects and Consequences

The analysis of off-normal ambient temperature uses the thermal models described in Section 4.2.6. The same models and calculations used for the normal conditions were used to model the -40°F and 100°F ambient conditions. The maximum steady state temperatures for the 100°F case and the -40°F case are provided in Table 4.2-12.

Figures 8.1-1 and 8.1-2 provide details of the temperature distributions. As these figures and Table 4.2-12 show, the component temperatures are within the acceptance criteria. The thermal gradients across the ~~PWR Basket~~*MPC* interior were higher for the -40°F case than for other cases. The stress analysis for this case is described in Section 4.2.5.3.1.

8.1.2.1.4 Corrective Actions

The Concrete Cask system is designed to accommodate steady state 100°F (with the design basis solar loads) or -40°F (with no solar loads). No corrective actions are required.

8.1.2.2 <u>Blockage of One-Half of the Air Inlets</u>

This event postulates obstructed air flow because of blockage of one-half of the air inlets.

8.1.2.2.1 Postulated Cause of Event

This event would be caused by partial air flow blockage of the air pad channel screens or air inlet area. The Concrete Cask has four wire mesh screen covered openings which permit air entry into the air pad channels. There are two horizontal air pad channels located parallel to each other on the bottom of the Concrete Cask. The air inlets are openings oriented perpendicular to the air pad channels and permit a continuous flow of air through the bottom of the Concrete Cask, up the annulus, and out the air outlet vents located on the sides of the Concrete Cask near the top. Figure 4.2-4 shows the configuration of the Concrete Cask including the orientation of the air inlet/outlet vents.

8.1.2.2.2 Detection of Event

This event may be detected by the ISFSI facility staff as they perform their required visual surveillance or by the monitoring of Concrete Cask air outlet temperatures.

8.1.2.2.3 Analysis of Effects and Consequences

The analysis of this event uses the air flow model described in Section 4.2.6. Blocking ~~two~~ *50 percent* of ~~the air inlets reduces~~ the inlet area ~~by a factor of two which~~ increases the ~~loss coefficient (k/A^2) for the entrance~~*hydraulic resistance at this area* by *approximately* a factor of four. However, the Concrete Cask flow system is designed so that the inlet losses are a relatively small portion of the total pressure drop due to the air flow. Hence, the increase in the total ~~$\Sigma k_i/A_i^2$ is about 60%~~*hydraulic resistance to air flow is significantly less than the increase at the inlets*. This reduces the air mass flow rate by ~~16%~~*34 percent*. ~~These combined effects (one increasing the pressure loss and one decreasing the pressure loss) increase the overall pressure loss due to the air flow by 15%.~~ The reduced air flow creates a higher ΔT between the air inlets and outlets to balance the higher flow pressure losses resulting in an increase in the air outlet temperature. ~~When these values are input to the ANSYS finite element thermal model of the Concrete Cask, t~~*T*he resulting concrete temperatures remain below the temperature limits as shown in Table 4.2-12.

There are no radiological releases or adverse radiological consequences from this event.

8.1.2.2.4 Corrective Actions

The required action when a vent or vents are found to be blocked is to remove the foreign material blocking the air intakes. Since screens are provided for the vents, most blocking material will be on the outside and easily removed. Materials that may be located inside the screens may be removed by hand-held tools after the screen is removed.

8.1.3 OFF-NORMAL CONTAMINATION RELEASE

8.1.3.1 <u>Small Release of Radioactive Particulates from Exterior of ~~Baskets~~*MPCs*</u>

This event involves the release of surface contamination on the exterior of the ISFSI ~~PWR Baskets~~ *MPCs* to the environment.

8.1.3.1.1 Postulated Cause of Event

A*n* ~~PWR Basket~~*MPC*, when submerged in the Spent Fuel Pool, may become slightly contaminated. If this surface contamination is not detected and removed prior to placement of

the loaded Concrete Cask on the Storage Pad, the particulate material could eventually be released to the environment.

8.1.3.1.2 Detection of Event

Release of radioactive particulate material from the ~~PWR Basket~~*MPC*'s exterior surfaces could be identified during radiological contamination surveys. As described in Section 7.5, these surveys will be conducted quarterly.

8.1.3.1.3 Analysis of Effects and Consequences

This analysis was performed to demonstrate that the proposed contamination limits would not result in a radiological concern at a distance of 100 meters from the ISFSI. If the surface contamination were not detected, the worst case scenario would be its release after the ~~PWR Basket~~ *MPC* is placed in the Concrete Cask and the Cask is placed on the Storage Pad. For such an atmospheric release, it is assumed that the release consists of a plume of ^{60}Co particulates. The off-site dose can then be calculated using the general methods described in Regulatory Guide 1.25. The release parameters are a wind speed of 1 m/sec, an atmospheric dispersion factor, and a two-hour period consistent with a short duration release assumed in Regulatory Guide 1.25. The methodology applied involves assuming an allowable surface contamination on the exterior of the ~~PWR Basket~~*MPC* that, if released, would result in a Committed Effective Dose Equivalent (CEDE) for ^{60}Co inhalation of ~~2.4~~*2.50* mrem to a "reference man" standing 100 meters from the point of release. The equation used to determine dose for the event is:

$$\text{CEDE (mrem)} = \text{DCF(mrem/}\mu\text{Ci)} \bullet \chi/\text{Q(sec/m}^3) \bullet \text{t(sec)} \bullet \text{BR(m}^3\text{/sec)} \bullet \text{Q(}\mu\text{Ci/sec)}$$

where:

DCF	=	Dose conversion factor = 2.187×10^2 mrem/µCi
χ/Q	=	Atmospheric dispersion factor = 0.035 sec/m^3
t	=	Reference Man's Exposure Time = 7200 sec
BR	=	Reference Man's Breathing Rate = $3.3\cancel{3} \times 10^{-4}$ m^3/sec
Q	=	Release rate (Ci/sec)

Solving this equation to determine the quantity of ^{60}Co that corresponds to the assumed CEDE yields:

$$Activity_{^{60}Co} = Qt = \frac{CEDE}{DCF(\frac{\chi}{Q})BR}$$

Based on this equation, the activity that would result in a CEDE of ~~2.4~~*2.50* mrem to a "reference man" standing at the plume centerline during the entire two-hour passage of the release at a distance of 100 meters from the release point is ~~941.6~~*989.8* µCi of ^{60}Co. If this activity were in the form of particulate contamination evenly distributed on the top and side external surfaces of ~~the~~ *up to* 36 ~~PWR Baskets~~*MPCs* (~~181.25~~*181.3*-inch height; ~~66~~ *68-3/8*-inch diameter) stored at the ISFSI, the allowable surface contamination will be 1.0 x 10^{-4} µCi/cm^2.

This value (1.0 x 10^{-4} µCi/cm^2) represents the allowable beta-gamma contamination on the exterior of a*n* ~~PWR Basket~~*MPC*.

The corresponding limit for α contamination is 1.0 x 10^{-5} µCi/cm^2 in accordance with the convention used in 10 CFR 71.87 for allowable surface contamination on transportation packages.

8.1.3.1.4 Corrective Actions

No corrective action is required. Compliance with the limit assures that the Total Effective Dose Equivalent requirements of 10 CFR 20.1301 and State of Oregon OAR 345-26-0390(4)(f) are met.

8.1.3.2 Radiological Impact from Off-Normal Operations

The specific radiological impact of the above off-normal operations is discussed within the applicable sections. A brief summary is included in Table 8.1-3.

8.1.4 OFF-NORMAL ~~BASKET~~ *MPC* LEAKAGE

8.1.4.1 Postulated Leakage of Radioactive Gases, Fuel Fines, and Cladding Surface Contamination

The ~~PWR Basket~~*MPC* confinement boundary is designed, fabricated, and tested to ensure that there will be no leakage of radioactive materials during normal operation. However, in order to conservatively assess the maximum potential radiological consequences of potential ~~PWR Basket~~ *MPC* leakage, this evaluation assesses the postulated effects of bounding off-normal fuel pin failure conditions and ~~PWR Basket~~*MPC* leakage.

8.1.4.1.1 Postulated Cause of Event

There is no known mechanism that would initiate this event. The analysis is performed to bound any potential off-normal radiological releases. Leakage is assumed to occur from all 34 ~~PWR Baskets~~*MPCs* assuming the failure of the cladding integrity of 1~~%~~ *percent* of the fuel pins in each ~~PWR Basket~~*MPC*. ~~An additional PWR Basket~~ *One MPC* is assumed to contain

10~~%~~ *percent* failed fuel pins. The failure of the cladding integrity of 1~~%~~ *percent* of the fuel pins in a*n* ~~PWR Basket~~*MPC* is a normal operating condition, and the failure of 10~~%~~ *percent* of the fuel pins in a*n* ~~PWR Basket~~*MPC* is an off-normal condition.

8.1.4.1.2 Detection of Event

The potential release of radioactive material from the ~~PWR Baskets~~*MPCs* could be identified during radiological contamination surveys. As described in Section 7.5, these surveys will be conducted quarterly.

8.1.4.1.3 Analysis of Effects and Consequences

The calculations were completed using guidance contained in NRC Spent Fuel Project Office Interim Staff Guidance 5 *(Reference 15)*. Annual doses *due to inhalation and submersion in the effluent plume* were calculated at ~~the Controlled Area boundary (325 m)~~*a distance of 150 meters* using *the* 50 percentile dispersion factor. ~~The annual dose at 325 m is then divided over the 16 direction sectors based on the wind direction frequency determined during Trojan preoperational data collection. The doses in the highest sector (North 19.7%) are~~*The doses are calculated for* normal releases consisting of leakage ~~at the Technical Specification rate for~~*from* all 34 ~~PWR Baskets~~*MPCs, and*~~. T~~ *t*he off-normal release is based on leakage from 1 ~~PWR Basket~~*MPC* ~~at the Technical Specification leakage rate~~. The release calculations include doses from gaseous *isotopes*, volatile fission products, fuel fines, and external fuel crud radioactivity. The projected dose is a sum of the normal, off-normal, and direct radiation dose projected for 1 year ~~in the worst sector~~.

The equation*s* used to estimate the doses are:

~~Fission Gas~~

~~Release Rate (Q_i) =~~

$$\frac{A_i(Ci) \times f_b \times f_g \times L_{ts}\left(\frac{cm^3}{sec}\right)}{V_b(cm^3)}$$

Volatiles/Fines/*Gases*/Crud

Release Rate (Q_i) =

$$\frac{A_i(Ci) \times f_b \times [f_v \text{ or } f_f \text{ or } f_c \text{ or } f_g] \times L_{ts}\left(\frac{cm^3}{sec}\right)}{V_b(cm^3)}$$

A_i = ~~PWR Basket~~*MPC* activity of nuclide $_i$ *(see Table 7.2-1)*

L_{ts} = ~~PWR Basket~~*MPC* ~~Technical Specification~~ leak rate *under off-normal conditions* = ~~1.4X10^{-4}~~ *7.37 x 10^{-6}* (cm^3/sec)

V_b = ~~PWR Basket~~*MPC* internal flow volume = *5.96 x 10⁶* (cm³)
f_b, f_v, f_f, f_c, f_g See Table 8.1-4 for definitions

~~Gas Dose~~

$$\cancel{CDE_{i,j}\ (mrem/yr) =} \quad \cancel{Q_i(\frac{Ci}{sec}) x \frac{\chi}{Q}(\frac{sec}{m^3}) DCF_i(\frac{mrem - m^3}{hr - Ci}) x\ 8760 \frac{hr}{yr}}$$

~~χ/Q = dispersion factor = 1.6E-5 sec/m³~~
~~$DCF_{i,j}$ = dose conversion factor for nuclide $_i$, for organ $_j$~~

Inhalation: Volatiles/Fines/*Gases*/Crud Dose

$CDE_{i,j}$ *or* $CEDE_i$, (mrem/yr) =

$$Q_i(\frac{Ci}{sec}) x \frac{\chi}{Q}(\frac{sec}{m^3}) x\ B_r(\frac{m^3}{sec}) x\ DCF(I)_{i,j}(\frac{mrem}{\mu Ci}) x\ 1E6\ (\frac{\mu Ci}{Ci}) x\ \cancel{3.15}\ 7.49E\cancel{7}6(\frac{sec}{yr})$$

B_r = breathing rates (Reference Man) = 3.3~~3~~ x 10^{-4} m³/sec
χ/Q = *dispersion factor = 6.0E-5 sec/m³*
$DCF(I)_{i,j}$ = *inhalation dose conversion factor for nuclide i, for organ j*
$CDE_{i,j}$ = *committed dose equivalent for internal organ dose from nuclide i, for organ j*
$CEDE_i$ = *committed effective dose equivalent from nuclide i*

Submersion: Volatiles/Fines/Gases/Crud Dose

$DDE_{i,j}$ *or* SDE_i *(mrem/yr) =*

$$Q_i(\frac{Ci}{sec}) x \frac{\chi}{Q}(\frac{sec}{m^3}) x\ DCF(S)_{i,j}(\frac{mrem}{\mu Ci} * \frac{m^3}{sec}) x\ 1E6\ (\frac{\mu Ci}{Ci}) x\ 7.49E6(\frac{sec}{yr})$$

$DCF(S)_{i,j}$ = *submersion dose conversion factor from nuclide i, for organ j*
$DDE_{i,j}$ = *deep dose equivalent from nuclide i, for organ j*
SDE_i = *skin dose equivalent for organ j*

The total effective dose equivalent (TEDE) to the whole body is then

$$TEDE = \sum_i CEDE_i + \sum_i DDE_i$$

$TEDE$	=	*total effective dose equivalent for the whole body*
$CEDE_i$	=	*committed effective dose equivalent from nuclide i*
DDE_i	=	*deep dose equivalent from nuclide i*

For a given organ the total organ dose equivalent ($TODE_j$) is

$$TODE_j = \sum_i CDE_{i,j} + \sum_i DDE_{i,j}$$

$TODE_j$	=	*total organ dose equivalent for organ j*
$CDE_{i,j}$	=	*committed dose equivalent from nuclide i, for organ j*
$DDE_{i,j}$	=	*deep dose equivalent from nuclide i, for organ j*

The projected doses *to the whole body and critical organs due to an effluent release from the MPC* are contained in Table 8.2-2.

8.1.4.1.4 Corrective Actions

No corrective action is required to mitigate the radiological impact since the Total Effective Dose Equivalent requirements of 10 CFR 20.1301 and State of Oregon OAR 345-26-0390(4)(f) are not exceeded. The doses shown in Table 8.2-2 are based on an occupancy of ~~8760~~ *2080* hours per year. For the purposes of compliance with OAR 345-26-0390(4)(f), an occupancy factor of ~~2000~~ *2080* hours per year is used as discussed in Section 7.6.2.

~~As discussed in Section in Section 5.1.1.5, "Off-Normal Event Recovery Operations," a leaking PWR Basket can be corrected by repair of a leaking weld or by use of a Basket Overpack to preserve the inert storage environment of the PWR Basket.~~

8.1.4.2 <u>Radiological Impact from Off-Normal Operations</u>

The specific radiological impact of the above off-normal operations is discussed within the applicable sections. A brief summary is included in Table 8.1-3.

8.2 ACCIDENTS

This section provides the results of analyses of the Design Events III and IV ~~events~~ from ANSI/ANS 57.9 and of several beyond design basis accidents. The results show that the Concrete Cask system provides an adequate margin of safety for the protection of the public, facility personnel, and the environment. In addition to these accidents, this section also provides the results of analyses of bounding natural phenomena.

8.2.1 FAILURE OF FUEL PINS WITH SUBSEQUENT BREACH OF ~~PWR BASKET~~*MPC CONFINEMENT BOUNDARY*

This accident involves the failure of the fuel rods in 24 fuel assemblies in a*n* ~~PWR Basket~~*MPC*. Thirty percent of the available fission product gas inventory is released to the environment at ground level.

8.2.1.1 Cause of Accident

This accident is considered the hypothetical accident, since there is no known causal factor which results in 100~~%~~ *percent* fuel rod failure and the breach of ~~PWR Basket~~*MPC* integrity.

The off-site radiological consequences of this hypothetical accident are used to show that the Controlled Area boundary doses are within the 10 CFR 72.106 limits.

8.2.1.2 Accident Analysis

This analysis assumes that 100~~%~~ *percent* of the fuel rods in a*n* ~~PWR Basket~~*MPC* with 24 fuel assemblies fail and release 30~~%~~ *percent* of the available fission product gases. Volatile and fission product fines are assumed to be released to the ~~PWR Basket~~*MPC* as a result of fuel damage. The release fractions are listed in Table 8.1-4. The fission products and crud from the fuel rods are released at ground level to the environment. This release is then used to determine the whole body and organ doses to a reference person located at ~~the Controlled Area boundary~~*a distance of 150 meters* for the duration of the release.

The release fractions are consistent with the guidance presented in Interim Staff Guidance 5. The dose from the released activity is dominated by ~~volatile and fission product fines~~*the dose due to inhalation of the effluent release*. Assuming the ~~PWR Basket~~*MPC* fails and releases the available amount of radioactivity, the off-site doses can be calculated by using the methods described in Interim Staff Guidance 5~~ and RG 1.109~~. The important parameters for these calculations are~~:~~

~~D~~*d*ispersion~~ - per Trojan FSAR~~
And ~~R~~*r*elease ~~T~~*t*ime~~ - 30 days~~.

The assumed ~~PWR Basket~~*MPC* contains *24 spent nuclear fuel assemblies with an assumed burnup of 42 GWD/MTU, an enrichment of 3.09 wt% ^{235}U, and nine years of cooling time.*~~the following:~~

~~-A loading of 0.466 MTU/assembly (this bounds 5-year cooled assemblies),~~
~~-Three 40 Gwd/MTU assemblies with 3.42% ^{235}U, and~~
~~-Twenty One 35 Gwd/MTU assemblies with 3.56% ^{235}U.~~

This assumed combination of burnup, cooling time, and enrichment is chosen to conservatively bound the actual burnups and cooling times for all fuel at the TNP site.

The equations and constants needed to estimate the dose are the same as those presented in Section 8.1.4.1.3 with the exception that

χ/Q = *accident condition dispersion factor* = 5.20×10^{-4} sec/m^3;
L = *MPC leak rate during accident conditions* = 1.28×10^{-5} (cm^3/sec); *and*
t = *release duration = 30 days* = 2.59×10^{6} sec.

~~There will be enough five-year old fuel in 1998 to fill nine Concrete Casks. A total of 25 of the five-year old assemblies have burnup levels between 35 and 40 GWd/MTU, 39 five-year old assemblies have burnup levels between 30 and 35 GWd/MTU, and 129 five-year old assemblies have burnup levels less than 30 GWd/MTU. Some assemblies have burnup levels greater than 40 GWd/MTU, but these assemblies will have at least nine years of cooling at time of loading in the PWR Basket. The 40 GWd/MTU, five-year old fuel is bounding with respect to fission gas inventory. The release calculations assume the above 25 assemblies (burnup levels between 35 and 40 GWd/MTU) are evenly distributed among the nine Concrete Casks that are assumed to contain five year cooled fuel (three per cask, at most). The other 21 five-year old assemblies in each of the nine Concrete Casks have a burnup level of 35 GWd/MTU or less (35 GWd/MTU conservatively assumed).~~

~~Release Rate (Q_i)~~ ~~=~~ $$\frac{A_i(Ci) \times f_b \times (f_g \text{ or } f_v \text{ or } f_f \text{ or } f_c) \times L_{ts}(\frac{cm^3}{sec})}{V_b\, cm^3}$$

~~A_i = PWR Basket activity of nuclide~~;
~~L_{ts} = PWR Basket Technical Specification leak rate = 1.4×10^{-4}(cm^3/sec)~~
~~V_b = PWR Basket internal free volume (cm^3)~~
~~f_b, f_v, f_f, f_c See Table 8.1-4 for definitions~~

~~Fission Gas Dose (CDE) =~~ $\cancel{Q_i(\frac{Ci}{\sec}) x \frac{\chi}{Q}(\frac{\sec}{m^3}) x DCF_{ij}(\frac{mrem - m^3}{hr - Ci}) x 720(\frac{hrs}{30d})}$

$$\cancel{Q_i(\frac{Ci}{\sec}) x \frac{\chi}{Q}(\frac{\sec}{m^3}) x B_r(\frac{m^3}{\sec}) x DCF_{ij}(\frac{mrem}{\mu Ci}) x 1.6E6(\frac{\mu Ci}{Ci}) x 2.59E6(\frac{\sec}{30d})}$$

~~Volatile/Fines/Crud Dose CDE) =~~

~~where:~~	~~$DCF_{i,j}$~~	~~=~~	~~dose conversion Factor for nuclide $_i$, organ $_j$~~
	~~χ/Q~~	~~=~~	~~atmospheric dispersion~~
	~~t~~	~~=~~	~~release duration 30 days~~
	~~$CDE_{i,j}$~~	~~=~~	~~Committed Dose Equivalent nuclide $_i$, organ $_j$~~
	~~B_r~~	~~=~~	~~breathing rate (Reference Man) = 3.33×10^{-4} m^3/sec~~

8.2.1.3 Accident Dose Calculations

The results of the calculation for ~~the Controlled Area boundary~~*a distance of 150 meters* are shown ~~on~~ *in* Table 8.2-2. ~~The requirements of 10 CFR 72.106 for design basis accidents applicable to any individual located on or beyond the nearest Controlled Area boundary are 5,000 mrem to the whole body or any organ.~~ As shown in Table 8.2-2, all doses at ~~325~~ *150* meters are <5,000 mrem. Thus, the *Trojan ISFSI* Controlled Area boundary established at ~~325~~ *225* meters meets the requirements of 10 CFR 72.106.

This event has been postulated to define the bounding consequences of postulated ~~PWR Basket~~*MPC* leakage. It was not used in establishing the boundary pursuant to Oregon Administrative Rule OAR 345-26-0390(4)(c). ~~For the purpose of operational convenience that boundary has also been conservatively established at 325 meters.~~ As shown in the analyses and evaluations of the design basis off-normal and infrequent (accident) events, there are no design basis accidents that will result in a loss of ~~PWR Basket~~*MPC* confinement barrier or the release of significant quantities of radiological material. The limiting event with regard to OAR 345-26-0390(4)(c) is discussed in Section 8.2.4.

8.2.2 MAXIMUM ANTICIPATED HEAT LOAD

This hypothetical event involves severe high temperature conditions.

8.2.2.1 Cause of Accident

This hypothetical event results from severe environmental conditions, an assumed 125°F ambient temperature and 12 hours of insolation, occurring when a design basis thermally loaded Concrete Cask is first placed in service. These parameters are beyond the anticipated range of conditions expected at the ISFSI. The temperature assumed for this event envelopes the maximum historical ambient temperature experienced in the region (107°F).

8.2.2.2 Accident Analysis

This event was analyzed to show that under extreme heat load conditions, the accident fuel cladding temperature limit of 1,058°F (570°C) and the concrete temperature limit of 350°F (177°C) are not violated.

This analysis uses the thermal models described in Section 4.2.6. The analysis assumes an ambient temperature of 125°F with 12 hours of insolation. Table 4.2-12 provides a summary of the analysis results showing that the components remain within the acceptance criteria. Figure 8.2-1 provides details of the temperature distribution.

~~As a result of the higher temperature increase (due to full solar loads) on the Concrete Cask surface, t~~*T*he thermal gradient across the concrete wall (and, hence, the stress) is ~~lower~~ *higher* for this accident condition than for the normal (75°F ambient) case discussed in Section 4.2. *The thermal stress analysis for the Concrete Cask is described in Section 4.2.5.4.3.*

8.2.2.3 Accident Dose Calculation

There are no radiological releases or adverse radiological consequences from this event.

8.2.3 CONCRETE CASK OVERTURNING EVENT

This event involves overturning a loaded Concrete Cask on the Storage Pad.

8.2.3.1 Cause of Accident

This accident is considered a beyond design basis accident, since there is no known causal factor which results in the Concrete Cask overturning. As shown in the evaluation of tornado*e*s, earthquakes, floods, and explosions, there are no known events at the ISFSI site that would result in overturning of a Concrete Cask on the ISFSI Storage Pad.

8.2.3.2 Accident Analysis

The following parameters are important to the overturning event:

1. Cask overturning energy dissipation through deformation of the Concrete Cask, ISFSI Storage Pad, and engineered fill;

2. Structural integrity of the Concrete Cask; and

3. Structural integrity of the ~~PWR Basket~~*MPC* confinement boundary as determined by deceleration loads on the ~~PWR Basket~~*MPC*.

The final design configuration of the ISFSI Storage Pad is an 18 inch thick reinforced concrete slab with 24-inch thick engineered fill over site foundation competent rock. This configuration is based on the objective of providing sufficient energy dissipation, in addition to the energy dissipation in the Concrete Cask deformations, to limit the stored fuel decelerations to within acceptable limits during an overturning event (58.9 inch equivalent flat drop height).

The slab-engineered fill design was developed from detailed calculations consistent with the methodology used in EPRI-7551, "Structural Design of Concrete Storage Pads for Spent-Fuel Casks," April 1993, but with parameters specific to the Trojan ISFSI site. The calculations use static, ultimate capacity analyses to determine the limit on steady deceleration of the Concrete Cask during impact. The static calculations are carried out by incrementally increasing the gravity multiplier for the Concrete Cask and internals. A plane strain finite element model is taken as a sufficiently close representation of the impact event. The finite element model used is a half-symmetric plane strain slice. The 2D slice is taken to be 12 inches thick and uses 8-node, bi-quadratic displacement interpolation quadrilaterals to model the Concrete Cask, the slab, the engineered fill layer, and the underlying bedrock. Gap elements between the Concrete Cask and slab account for the changing contact area and transmit the Concrete Cask load to the slab. The calculations allow energy absorption due to material limits in the Concrete Cask as well as in the slab. For conservatism, the engineered fill and bedrock are modeled as elastic in compression but with no tensile capacity. As concrete material limit states are reached, the loads are redistributed through equilibrium iteration until the ultimate capacity of the slab structural system is reached. The total energy absorbed by the system is calculated by accumulating the increments in the elastic and plastic strain energy for each load increment.

By equating this internal energy to the potential energy of the Concrete Cask prior to a drop, the drop height can be calculated and plotted versus the steady deceleration. A slab and subgrade design is then qualified by finding the steady deceleration for the drop height of interest and multiplying by a factor of 1.5 to determine the peak deceleration of the Concrete Cask. The factor of 1.5 is the upper bound based on drop tests of a heavy steel Concrete Cask on a 3-foot

slab and should be smaller for the Concrete Cask which absorbs energy during the impact. ~~The limiting peak deceleration for the stored fuel during an overturning event is 44g.~~

The load carrying capability of the engineered fill is characterized by a single property, the subgrade reaction coefficient, k, which has the units of pressure per unit of displacement. This property is usually determined by conducting field bearing tests using a standard size 1-foot diameter "rigid" plate. Generally, a k value above 250 psi/in characterizes a competent subgrade well suited for foundation design. Subgrades that are characterized through their k value are usually engineered deep fills, unlike the Trojan site which consists of a 24-inch thick layer of engineered fill on deep bedrock. In order to design the Storage Pad-engineered fill system that meets the Concrete Cask deceleration limit for the Concrete Cask, a structurally competent fill layer must be carefully designed. This is accomplished by an analysis of a 12-inch rigid plate on a soil/bedrock structure. A finite element grid for the fill bedrock substrata and the 12-inch diameter plate is generated consistent with that used in the drop analysis. The analysis results show that an in-situ k value for the Trojan site is in the range of 340 psi/in for a 24-inch fill. Field testing was performed to verify that the structural properties of the engineered fill material for the Storage Pad are consistent with the calculation described above. The coefficient of subgrade reaction (k) of the fill varied somewhat among the specific test locations, as expected. However, the test report concludes that the engineered fill material substantially met the target value of 340 psi/in for the coefficient of subgrade reaction.

In order to provide further assurance that the fill material was consistent with the Concrete Cask overturning analysis, additional dynamic analyses were performed to confirm that the results of the overturning analysis continued to be acceptable given the measured values of subgrade reaction.

The interaction between the Concrete Cask, the strength of the slab, and the support provided by the subgrade is highly nonlinear. The stiffer subgrade and higher strength concrete slab cause the Concrete Cask to have a larger role in dissipating energy than is the case with softer soil. As further confirmation of this nonlinear interaction between the Concrete Cask, slab and subgrade the dynamic calculation was undertaken to simulate the Concrete Cask impacting the slab. By summing the forces at a section cut along the contact interface in the model, the force history was extracted to calculate the peak deceleration force applied to the Concrete Cask. A second objective of this calculation was to determine a more realistic amplification factor to convert the steady deceleration to peak deceleration in the static calculations. Previously, a factor of 1.5 was used as an upper bound on the dynamic amplification considering massive steel casks under end drop conditions. The dynamic calculations showed a peak deceleration of about 32g, which implies that the dynamic amplification factor for the static calculation is about 1.1 rather than 1.5.

Based on both the static simulation and the dynamic time history analysis, the subgrade is adequate for deceleration limitations on the Concrete Cask internals during a hypothetical Concrete Cask overturning event.

The analyses described above are based on a total system weight of 290,000 lbs, which includes the TranStor™ Concrete Cask and a fully loaded PWR Basket. The Trojan Storage System, which consists of a Holtec MPC-24E or -24EF stored in a TranStor™ Concrete Cask, weighs slightly more at 292,700 lbs (Table 4.2-4). The slightly increased weight of the system has a beneficial effect on the structural margin of safety under the non-mechanistic tip-over scenario. This fact can be deduced from two factors. First, since the center of gravity of the Trojan Storage System containing an MPC is lower in height than that of the TranStor™ Concrete Cask containing a PWR Basket, the impact velocity is less for the Trojan Storage System with the MPC. The second factor is indicated by the linear impact model, which shows that the maximum deceleration experienced by a falling object on a stationary target is approximately proportional to the square root of the ratio of the impact patch stiffness, K, to the mass of the striking body, m. Because the stiffness of the contact patch is governed by the TranStor™ Concrete Cask and ISFSI pad stiffnesses, K remains unchanged for the hybrid system tip-over scenario. However, the mass, m, is slightly increased due to the increased MPC mass. Therefore, the maximum deceleration due to tip-over will also decrease slightly over the computed value for the Concrete Cask containing a PWR Basket.

Cask Overturning Overall Damage Assessment

From the static finite element analysis, cracking patterns that develop show a shear cone in the slab underneath the impact, and some "crushing" that is localized near the impact. Bending cracks on the bottom of the slab under the impact area and on the top of the slab away from the impact are also in evidence. The impact area is about 10 inches on either side of the impact symmetry line.

A shear cone develops in the Concrete Cask under the impact area and bending cracks develop due to the ovalization of the Concrete Cask. Some crushing in the cask concrete local to the point of impact also occurs. Some local spalling is likely, but should be minimal. The change across the diameter of the Concrete Cask inner steel shell is calculated at 0.4 inch to 0.5 inch for the 58.9-inch equivalent flat drop height. Thus, the Concrete Cask will sustain cracks and localized damage, but its structural integrity would be maintained. Righting the Concrete Cask and extracting the ~~fuel PWR Basket~~*MPC* after Concrete Cask overturning would be achievable.

Results

Based on the analysis, it has been concluded that only minor damage to the Concrete Cask would occur due to impact with the pad. The Concrete Cask and contained ~~PWR Basket~~*MPC* with internals would experience a *bounding* deceleration of 32g to 38g due to impact with the pad.

This deceleration value is much less than the design basis deceleration value of 60g for the Holtec MPC. As discussed in the Holtec HI-STAR FSAR, ~~T~~*t*his deceleration load would not result in damage to the spent fuel. ~~Lawrence Livermore National Laboratory Report UCID-21246, "Dynamic Impact Effects on Spent Fuel Assemblies," evaluated the capability of fuel assemblies to withstand various drop orientations. The conservative evaluation contained in UCID-21246 concluded that typical fuel assemblies can withstand loading equivalent to 63g without exceeding the yield strength of the cladding.~~ Therefore, it was concluded that these components are capable of withstanding the impact loads and would remain intact.

8.2.3.3 Accident Dose Calculation

This event is a beyond design basis event. There are no rädiological releases or adverse radiological consequences from this event.

8.2.4 TORNADO

This event involves the potential effects of a tornado on the ISFSI.

8.2.4.1 Cause of Accident

This event would be the result of a tornado generated at or near the ISFSI.

8.2.4.2 Accident Analysis

The Trojan ISFSI is located in an area classified by Regulatory Guide 1.76 (Design Basis Tornado) as a Region III. The Trojan ISFSI Concrete Cask is designed for a Regulatory Guide 1.76 area classified as Region I which requires the Concrete Cask to withstand loads associated with the most severe meteorological conditions including extreme wind and tornado. Tornado design parameters used to evaluate the suitability of the Concrete Cask include tornado winds, wind generated pressure differentials, and tornado generated missiles. A comparison of design requirements is shown in Table 8.2-3.

The methods used to convert the tornado and wind loadings into forces on the Concrete Cask are based on NUREG-0800, Section 3.3.1 -Wind Loadings, and Section 3.3.2 -Tornado Loadings. Loads due to tornado generated missiles are based on NUREG-0800, Section 3.5.3 -Barrier Design Procedures.

8.2.4.2.1 Wind Loads

The tornado wind velocity is transformed into an effective pressure applied to the Concrete Cask using procedures delineated in ANSI A58.1, "Building Code Requirements for Minimum Design

Loads in Buildings and Other Structures." The maximum velocity pressure, p, is determined from the maximum tornado wind velocity as follows:

$$p = (0.00256)\ V^2 \text{ psf} \qquad = 331.8 \text{ psf} = 2.3 \text{ psi}$$

where:

V = Maximum tornado wind speed = 360 mph

The above effective velocity pressure is assumed constant with height and, since the Concrete Cask is small in relation to the radius of the tornado, is assumed to be uniform over the projected area of the Concrete Cask. Gust factors are taken as unity in evaluating effects of velocity pressures on Concrete Cask surfaces.

The total tornado wind loading on the projected area of the Concrete Cask, W_w, is then computed as follows:

$$W_w = p(C_f)(A_p)$$

where:

p = Effective velocity pressure (psf) = 331.8 psf = 2.3 psi

C_f = Net pressure coefficient = 0.52 (Reference ANSI-A58.1, Table 12)

A_p = Projected area of Concrete Cask normal to wind
= 136 x 211.5/144 = 199.75 ft^2 = 28,764 in^2

then,

$$W_w = (331.8)\ (0.52)(199.75) = 34{,}464 \text{ lbs}$$

The overturning moment (M_w) acting on the Concrete Cask is:

$$M_w = (34{,}464)(211.5/2) = 3.6 \times 10^6 \text{ lb-in.}$$

The calculated force and moment are insufficient to rotate or slide a Concrete Cask since the resisting moment of the Concrete Cask is (~~290,000~~*292,700*) 58.5 = ~~16.9~~*17.1* x 10^6 lb_f-in; and the ratio of wind force to the normal force, i.e., (34,464/~~290,000~~*292,700*) or ~~0.119~~*0.118*, is much less than the typical value of the coefficient of friction for steel on concrete (0.3) which precludes sliding. (Note: The shortest base dimension of 58.5 in., which considers the air inlet channels at the base, is used.)

Stresses in the Concrete Cask are computed using the following cross-section properties:

Cross Sectional Area $(A_x) = (\pi/4)(136^2 - 78^2) = 9{,}748$ in.^{2}

Moment of Inertia $(I_y) = (\pi/64)(136^4 - 78^4) = 1.5 \times 10^7$ in.^{4}

Distance to Extreme Fiber $(C_x) = 68$ in.

The critical section for the Concrete Cask is at the bottom of the cavity. The shear stress in the Concrete Cask, conservatively ignoring the Concrete Cask liner, is $W_w/A_x = 3.5$ psi. The Concrete Cask is assumed cantilevered at the base of the liner bottom where the moment due to the wind loading is $(W_w)(211.5-19.5)/2 = 3.31 \times 10^6$ in.-lb. Then, the bending stress is $M/(I_y/C_x) = 15.0$ psi. Both normal and shear stresses are included in load combinations and evaluated in Table 4.2-10.

8.2.4.2.2 Tornado Missiles

The Concrete Cask is designed to withstand the effects of impacts associated with postulated tornado generated missiles as identified in NUREG-0800, Section 3.5.1.4.III.4. These missiles consist of a massive high kinetic energy missile which deforms on impact, a rigid missile to test penetration resistance, and a small rigid missile of a size sufficient to just pass through any openings in protective barriers. Missiles are assumed to impact in a manner that produces the maximum damage to the Concrete Cask.

The Concrete Cask body and closure elements have been analyzed for penetration resistance to an armor piercing shell missile. Results confirm that sufficient thickness of concrete and steel is available to prevent perforation, spalling or scabbing of the various Concrete Cask boundary elements. Overall response of the Concrete Cask has been evaluated for impacts associated with the high energy deformable missile. Such analyses indicate that the Concrete Cask will remain upright following the event, and that loads associated with this impact do not compromise the integrity of the Concrete Cask. The analyses which have been conducted are summarized below.

The potential effects of the small rigid missile (one-inch diameter steel sphere) were not analyzed because (1) the effects on the Concrete Cask concrete and steel cover are bounded by the effects of the armor piercing artillery shell, and (2) the air passages in the Concrete Cask do not present a direct path for the sphere to enter and strike a*n* ~~PWR Basket or Basket Overpack~~*MPC*.

8.2.4.2.3 Local Damage Prediction - Cask Body

Local damage of the Concrete Cask body has been assessed using the National Defense Research Committee (NDRC) formula. This formula has been selected as the basis for predicting depth of

penetration and minimum thickness of concrete to prevent spalling and scabbing. Penetration depths computed by this method have been shown to provide reasonable correlation with test results. (Ref*erences*~~:~~ ~~8.~~8 and ~~8.~~9, EPRI Reports *NP-1217 and* NP-440~~ and NP-1217~~). The depth of penetration, X, as predicted using this approach may be expressed as follows:

For $X/2d \leq 2.0$:

$$X = [4KNWd^{-0.8} (V/1000)^{1.8}]^{0.5} = 5.69 \text{ inches}$$

where:

f_c = Design compressive strength of concrete

d = Diameter of missile (8 inches)

K = Coefficient depending on the concrete strength
$= 180/(f_c)^{0.5}$
= 2.85 assuming 4000 psi concrete

N = Missile shape factor
= 1.14 for sharp nosed missiles (Ref: EPRI NP-1217)

W = Missile Weight (275 lbs)

V = Velocity (184.8 ft/sec)

The minimum depth of concrete necessary to preclude spalling and scabbing is then selected as three (3) times the depth of penetration predicted using the NDRC formula, or 17.1 inches. Since the minimum thickness of concrete in the Concrete Cask body is well in excess of 17.1 inches, it is concluded that adequate protection is provided for local damage due to tornado missiles.

8.2.4.2.4 Local Damage Prediction - Cask Closure Plate

The Concrete Cask is closed with a 0.75 inch thick steel plate bolted in place. By calculating the perforation thickness of a 126 mph, 275 lb., 8 inch diameter artillery shell impacting a steel plate, the ability of the closure plate to adequately withstand tornado generated missiles is established.

The perforation thickness in a steel plate is given in Reference ~~8.~~10, Topical Report BC-TOP-9A, Revision 2, "Design of Structures for Missile Impact," by Bechtel Power Corporation.

$T = [(0.5)(M_m)(V_s)^2]^{2/3}/672d_m = 0.52$ in

where:

T = Perforation thickness (in)

M_m = Missile mass (slugs) = W/g = 275/32.2 = 8.54 slugs

W = Missile weight = 275 lbs

g = Acceleration due to gravity = 32.2 ft/sec^2

V_s = Missile striking velocity (ft/sec) = 184.8 ft/sec

d_m = Missile diameter (in) = 8 in.

Therefore, the Concrete Cask closure plate is adequate to withstand local impingement damage due to tornado generated missiles.

8.2.4.2.5 Overall Damage Prediction

Since the Concrete Cask is a freestanding structure, the principal consideration in overall damage response is the likelihood of upsetting or overturning of the Concrete Cask as a result of high energy missile impacts. Such assessments have been conducted using the principles of conservation of momentum during the impact event. The analyses, which are summarized below, indicate that the Concrete Cask will remain upright.

The force developed by the missile has been calculated using methodology presented in Reference 8.10, Topical Report, BC-TOP-9A, Revision 2, "Design of Structures for Missile Impact," Bechtel Power Corporation, 1974. The maximum force, F, is:

$$F = (0.625)(v)(W) = (0.625)(184.8)(3{,}960) = 457.4 \text{ kips}$$

From the principles of conservation of momentum, the impulse of the force from the missile impact on the Concrete Cask must equal the change in angular momentum of the Concrete Cask. Likewise, the impulse force due to the impact of the missile must equal the change in linear momentum of the missile. With reference to Figure 8.2-2, these relationships may be expressed as follows:

During the deformation phase, the change in momentum of the missile becomes:

$$\int_{t_1}^{t_2} (F)\,(dt) = M\,(v_2 - v_1)$$

where:

F = Impact impulse force on missile

M = Mass of missile
= 3960 lbs/g
= 123 slugs

t_1 = Time at impact

t_2 = Time at conclusion of deformation phase

v_1 = Velocity of missile at impact
= 184.8 ft/sec (126 mph)

v_2 = Velocity of missile at t_2

The change in angular momentum of the Concrete Cask about a point on the bottom rim becomes:

$$\int_{t_1}^{t_2} (M_c)\,dt = \int_{t_1}^{t_2} (211.5 \cdot F)\,dt = I_c\,(w_1 - w_2)$$

where,

M_c = Moment of the impact impulse force on the Concrete Cask

I_c = Cask mass moment of inertia about a point on the bottom rim
$= \cancel{1.75}\,\mathit{1.77} \times 10^8$ slug-in^2

w_1 = Angular velocity at time t_1

w_2 = Angular velocity at time t_2

Equating the impulse of the impact force on the missile to the impulse of the force on the Concrete Cask yields (See Figure 8.2-2):

$$-123[v_2 - (184.8 \text{ ft/sec})(12 \text{ in/ft})] = (\cancel{1.75}\,\mathit{1.77} \times 10^8/211.5)(w_2)$$

where:

$$v_2 = (246.4)w_2$$

then,

$$w_2 = \cancel{0.32}\,\mathit{0.31} \text{ rad/sec, and}$$

$$v_2 = \cancel{78.8}\,\mathit{76.4} \text{ in/sec}$$

During the restitution phase, the final velocity of the missile will depend upon the coefficient of restitution of the missile, the geometry of the missile and target, the angle of incidence, and upon the amount of energy dissipated in deforming the missile and target. It is assumed, based upon tests conducted by EPRI, (Reference ~~8~~9, EPRI Report NP-440, Tests 6 and 7) that the final velocity of the missile, v_f, following the impact is zero.

Equating the impulse of the force on the missile during restitution to the impulse of the force on the Concrete Cask yields:

$$-[m(v_f - v_2)] = I_c/211.5(w_f - w_2)$$

then:

$$w_f = 0.32 \text{ rad/sec}$$

The final kinetic energy of the Concrete Cask following the impact, E_k, is then determined as:

$$E_k = (I_c)(w_f)^2/2 = [(\cancel{1.75}\,\mathit{1.77} \times 10^8)(0.32)^2/2](1/12) = \cancel{7.47}\,\mathit{7.55} \times 10^5 \text{ in-lb}_f$$

And the energy required to overturn the Concrete Cask, E_p, is:

$$E_p = (W_c)(h) = (\cancel{290{,}000}\,\mathit{292{,}700})(\cancel{16.9}\,\mathit{17.1}) = \cancel{4.9}\,\mathit{5.01} \times 10^6 \text{ in-lb}_f$$

where:

$$W_c = \cancel{290{,}000}\,\mathit{292{,}700} \text{ lbs}$$

$h = \cancel{16.9}\,\mathit{17.1}$ in. (accounting for chamfer of the Concrete Cask bottom rim)

Hence, by comparison, overturning of the Concrete Cask is not postulated to occur as a result of impact from tornado generated missiles. The above analysis is conservative since it assumes direct in-line impact of the missile with the Concrete Cask.

The shear capacity at the location of the Concrete Cask outlets also has been calculated to evaluate resistance of the Concrete Cask to tornado generated missiles. The capacity of the concrete section is calculated using shear-friction formula (ACI-349, Section 11):

where: $U_s = \phi V_n = 0.85\ A_{vf} f_y \mu = 1{,}106$ kips,

$\phi =$ strength reduction factor = 0.85

$A_{vf} =$ (32) (0.44) = 14.1 in^2 - total area of reinforcement perpendicular to the shear plane

$f_y =$ $1.1 \cdot 60{,}000 = 66{,}000$ psi - reinforcement yield strength increased by 10% for dynamic loading (ACI-349, Appendix C)

$\mu =$ 1.4 - for monolithically placed concrete.

The maximum moment due to the impact exists in the Concrete Cask section adjacent to the bottom:

$$M = Fl = 457.4 \cdot (211.5 - 19.5) = 87{,}820 \text{ kips} \cdot \text{in.}$$

Section capacity of the concrete section has been conservatively calculated per Section 9.5.2.3 of ACI-349 code.

$$U_m = \phi(f_r I_g / y_t) = 94{,}170 \text{ kips} \cdot \text{in}$$

where:

$f_r =$ $7.5\sqrt{f'_c} = 474.34$ (concrete modulus of rupture)

where:

$f'_c =$ 4000 psi (concrete compressive strength)

$I_g =$ $1/4\ \pi(R^4 - r^4) = 1.5 \times 10^7$ in.^{4} (gross moment of inertia of concrete section)

where:

$R =$ 68 in.

$r =$ 39 in.

y_t = 68 in. (distance from centroidal axis of gross section, neglecting reinforcement, to extreme fiber in tension)

ϕ = 0.9 (strength reduction factor -Section 9.3.2, ACI-349)

These results are evaluated in combination with other loads in Table 4.2-10.

8.2.4.2.6 Combined Tornado Wind and Missile Loading

The effects of tornado winds and missiles have been considered both separately and combined in accordance with NUREG-0800, Section 3.3.2.II.3.d. For the case of tornado wind plus missile loading, the stability of the Concrete Cask has been assessed and found to be acceptable. Equating the kinetic energy of the Concrete Cask following missile impact to the potential energy yields a maximum postulated rotation of the Concrete Cask as a result of the impact of 2.6 degrees. Applying the total tornado wind load to the Concrete Cask in this configuration results in an overturning moment of 3.8×10^6 in-lbs with the restoring moment on the Concrete Cask calculated to be ~~1.54~~*1.57* x 10^7 in-lbs. Hence, overturning of the Concrete Cask under the combined effects of tornado winds plus tornado-generated missiles will not occur.

8.2.4.2.7 ~~Baskets/Basket Overpack~~*MPC* Under Tornado Loadings

Since the postulated tornado missile and wind loadings are not capable of overturning the Concrete Cask they have no effect on the ~~Basket or Basket Overpack~~*MPC*. The Concrete Cask protects the ~~PWR Basket~~*MPC* from direct impact of missiles. The atmospheric pressure drop caused by the tornado (-3 psid) is less than the internal pressure capacity of ~~either~~ the ~~PWR Baskets or Basket Overpack~~*MPC;*~~,~~ therefore, tornado effects will have no adverse impact on the ~~PWR Baskets or Basket Overpack~~*MPC*.

8.2.4.2.8 Transfer Station Tornado Loadings

Tornado winds (240 mph maximum), on the Transfer Station are bounded by lateral seismic acceleration loads as discussed in Section 3.2.5.6.

With regard to tornado missiles, EPRI reports NP-768 and NP-769, "Tornado Missile Risk Analysis" provide the basis to determine the risk to facilities due to tornado missiles. Using the data presented in the EPRI reports from numerous tornado missile computer simulations, the probability of a missile hitting any square foot of area per missile per tornado can be determined (based on a given tornado severity and missile population within 2000 feet of the site). The probabilities of missile hit per square foot per missile per tornado for each tornado intensity F′1 to F′4 are shown in Table 8.2-4. The tornado intensity regions referred to in the EPRI reports correspond to the regions defined in NRC Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," 1974.

To develop an estimate of the number of tornado missiles available on the Trojan plant site, a survey of the plant site and surrounding area to a radius of 2,000 feet was conducted in about 1989. The 2000-foot radius was based on conclusions in the EPRI report that this is the maximum transport distance of tornado missiles. This survey consisted of a thorough walkdown of all areas of the plant site. When potential missiles were identified in the survey, conservative estimates of the number were made for use in the analysis. The survey indicated a total of approximately 17,000 missiles. This is conservative by a factor of three (3) relative to the 5,000 - 6,000 value used for the plants evaluated by EPRI. The results of the 1989 survey are considered conservative for the ISFSI due to the limited staffing and scope of activities.

The combination of the missile survey results with the missile strike data yields the probability per square foot, per tornado, of a missile hit on Trojan structures or equipment. Multiplying these probabilities by the total exposed area of the Transfer Station main structure, including the Transfer Cask (~3800 ft^2), gives the probability of a missile strike on the Transfer Station for a given tornado.

The final step in determination of tornado missile strike probabilities is to combine the strike probability per tornado with the annual frequency of occurrence of the different intensity tornados. As shown in Table 8.2-4, the tornado missile strike probabilities per year for the Transfer Station are very low; well below the threshold of significant risk, typically considered to be in the range of 10^{-6} to 10^{-7} per year. Also, these values represent probabilities of missile strikes, and not necessarily unacceptable damage. Most of the representative potential missiles would not be expected to cause significant damage to the Concrete Cask.

Tornado missiles are shown to not represent a significant risk particularly during the short period of time (small fraction of a year) that the station is used to transfer spent fuel assemblies.

In summary, the missile strike probabilities are concluded to remain below the range considered to be of significant risk. Although the Transfer Station is expected to be in use for relatively short periods of time (a small fraction of each year), the risk reduction based on the time of use is not included in the probability estimates presented above.

8.2.4.3 Accident Dose Calculations

As previously described, this event does not result in the release of radiological material to the environment. However, the worst tornado missile impact calculation shows that 5.69 inches of the Concrete Cask side wall was dislodged. The dose rate at the surface of the Concrete Cask resulting from the loss of 5.69 inches of concrete shield is ~~286.5~~*66.5* mrem/hr for a Concrete Cask with *an* ~~PWR Basket~~*MPC* compared to the ~~original~~ *initial dose* rate of ~~19.1~~*13.7* mrem/hr. *These values are based on the design basis burnup and cooling time of 42,000 MWD/MTU and nine-year cooling.*

The Concrete Cask would be repaired by filling the damaged area with grout. It is presumed that some period of time will be required to obtain the materials needed to repair the Concrete Cask surface. Shielding materials will be maintained on site for use in mitigating the consequences of this event until such time as a repair to the Concrete Cask surface can be completed. It is estimated that shielding materials can be in place within 12 hours of the event. It is estimated that once the necessary materials are obtained two technicians would be able to complete the repair in approximately 30 minutes. The collective dose to the repair crew would be less than or equal to approximately ~~0.826~~*0.155* person-rem (~~413~~ *77.5* mrem to each technician).

This Design Basis event was considered in the establishment of the appropriate Controlled Area boundary pursuant to 10 CFR 72.106 and Oregon Administrative Rule OAR 345-26-0390(4)(c). The direct radiation levels at *the Controlled Area boundary, 225* ~~100~~ meters, as a result of this event are minimal for the expected duration of the event. ~~Therefore, the Controlled Area boundary will be conservatively established at 325 meters based on the beyond design basis event described in Section 8.2.1.~~

8.2.5 EARTHQUAKE EVENT

This event is a Seismic Margin Earthquake (SME)

8.2.5.1 Cause of Accident

An earthquake that affects the ISFSI initiates this event. The Seismic Margin Earthquake (SME) is described and discussed in Section 2.6.2.4.

8.2.5.2 Accident Analysis

The loaded Concrete Cask has been analyzed for the Seismic Margin Earthquake (SME). The SME, which has a peak horizontal ground acceleration of 0.38g and a peak vertical ground acceleration of 0.25g, has been used as the design basis earthquake for the Concrete Cask. The analysis of this event is summarized below. Use of the SME in accordance with Oregon Administrative Rule OAR 345-26-390(4)(c) is also described in Section 2.6.2.4.

The Concrete Cask is a very stiff structure. Its lowest natural frequencies are well beyond the Zero Period Acceleration (ZPA) threshold of the site spectra. No dynamic amplification of the ground motion is expected from the Concrete Cask. Although free-standing, it has been analyzed as a cantilever fixed at the base (Roark and Young, "Formulas for Stress and Strain," 5th Edition, Table 36, Case 3b). For the purpose of calculating seismic loads, the Concrete Cask is treated as a rigid body attached to the ground. Equivalent static analysis methods were used to calculate loads, stresses, and overturning moments.

The fundamental natural frequency of vibration for the Concrete Cask was determined as shown below (Reference ~~8.~~1):

f_n = $[(K_n)/2\pi]\ [(E)(I)(g)/(w)(L^4)]^{0.5}$ = ~~48.8~~*48.5* cycles per second

where: f_n = Frequency of the n^{th} mode

K_n = 3.52 for first mode of vibration

E = Modulus of Elasticity = $57{,}000\ (f_c)^{0.5} = 57{,}000\ (4{,}000\ \text{psi})^{0.5}$
= 3,604,996 psi

I = $1.49 \times 10^7\ \text{in}^4$

g = 386.4 in/sec/sec

L = Height of Concrete Cask = 211.5 in

w = Uniform weight density of cantilever = ~~290,000~~*292,700*/211.5
= ~~1371.2~~*1,383.9* lb/in

It can be seen from Regulatory Guide 1.60 that the dynamic amplification factor for this frequency is unity and the loads can be treated as static.

The Concrete Cask has been evaluated statically for overturning by conservatively applying equivalent static loads to the Concrete Cask in each of two orthogonal horizontal directions simultaneously with an upward vertical component. Combination of the three components is performed in accordance with Reference ~~8.~~12. The reference recognizes that the maximum accelerations from the three directions cannot occur at the same time. It suggests that when one of the components is at its maximum value, the other two can be taken as 40% of their corresponding peaks. Although the SME peak horizontal ground acceleration was developed from the geometric mean of the two peak horizontal ground accelerations, for conservatism in the Concrete Cask design an orthogonal horizontal ground acceleration component of 40% of the peak horizontal ground acceleration (0.40 x 0.38g) will be conservatively considered to occur in combination with the peak horizontal ground acceleration. The total seismic uplift potential has been calculated by multiplying the ground acceleration in each of the two horizontal directions by the weight of the fully loaded Concrete Cask and combining the resulting component forces on the basis of the square root of the sum of the squares. For the SME, the resulting factor of safety against overturning is:

F.S. = Restoring Moment/Overturning Moment

$$\text{Horizontal seismic acceleration} = [(0.38g)^2 + (0.40 \cdot 0.38g)^2]^{0.5} = 0.41g$$

$$\text{Horizontal seismic load} = 0.41\ g\ W_c$$
$$\text{Vertical seismic acceleration} = (0.40 \cdot 0.25g) = 0.10g$$

$$\text{Vertical Seismic load} = 0.10\ g\ W_c$$

W_c = ~~290,000~~*292,700* lbs

then:

F.S. = $[(W_c)(1.0\text{-}0.1)(58.5)] / [(W_c)(0.41)($~~109.5~~*107.8*$)]$ = ~~1.17~~*1.19* > 1.10

Therefore, the SME criteria are satisfied and no uplift of the edge of the Concrete Cask will occur.

Furthermore, as Figure 8.2-3 shows, a vertical ground displacement of approximately 5.3 feet would be required to move the center of gravity over the edge of the Concrete Cask so that overturning could occur. This type of ground displacement and/or failure of the foundation is considered to be unrealistic; hence, it is concluded that in addition to not overturning due to the SME inertia loads, the Concrete Cask will also not overturn due to vertical movement of the foundation. Therefore, based on this analysis, it can be concluded that the Concrete Cask will not overturn or fall during an SME.

The ~~PWR Basket~~*MPC* and Concrete Cask are rugged and, since overturning is precluded, their stresses due to SME are negligible. The ~~PWR Basket~~*MPC* stresses are bounded by the much higher drop accelerations, while the Concrete Cask seismic demands can be calculated as follows:

Shear: $V = 0.41W_c$ = 0.41 (~~290~~*292.7*) = ~~118.9~~*120.0* kips

Moment: $M = V \cdot l$ =(~~118.9~~*120.0*) (211.5-19.5) = ~~22,829~~*23,040* kip-in

Where l equals the height of the Concrete Cask above the air inlet ducts, in inches.

Both of these values are lower than the section capacities calculated in Section 8.2.4.2.5. The seismic shear and moment are included in the structural evaluation Table 4.2-10 in Chapter 4.0.

To address the potential for Concrete Cask sliding under SME ground motions, a conservative energy balance relationship can be used. The instantaneous maximum kinetic energy imparted to the Concrete Cask is considered to be the kinetic energy corresponding the peak SME ground motion velocity.

$$KE = \frac{W}{2g}(V)^2$$ where V = the SME peak ground velocity

The energy dissipated during Concrete Cask sliding is the sliding friction force acting through the displacement of the Concrete Cask.

$$S.E. = W\mu\delta$$

where S.E.= sliding energy, μ = the coefficient of sliding friction, and δ = the Concrete Cask displacement corresponding to the velocity impulse.

The energy balance relationship is then;

S.E. = K.E, or

$$\delta = \frac{V^2}{2g\mu}$$

The peak SME ground motion velocity is 23.67 cm/sec = 9.32 in/sec. Including a simultaneously occurring orthogonal velocity component of 0.40 (9.32) = 3.73 in/sec, the velocity vector becomes,

$$V = [(9.32)^2 + (3.73)^2]^{1/2} = 10.04 \text{ in/sec}$$

If a conservative value of the coefficient of friction during sliding of 0.25 is selected for steel (Concrete Cask steel base) on smooth concrete (Storage Pad), the Concrete Cask displacement becomes,

$$\delta = \frac{(10.04)^2}{2(386)(0.25)} = 0.52in$$

For 10 SME velocity impulses (NUREG-0800 Rev.1, 1981, Section 3.7.3) and no consideration of ground motion reversal, the Concrete Cask sliding displacement would become,

$$\delta = 10\ (0.52) = 5.2 \text{ in}$$

Thus, it is concluded that Concrete Cask sliding displacements under SME ground motions are small, and no Concrete Cask impacts would result.

8.2.5.3 Accident Dose Calculations

The seismic event will not topple or damage the Concrete Cask, thus, there are no radiological consequences. The Concrete Casks are relatively undisturbed by the earthquake.

8.2.6 PRESSURIZATION

This event produces maximum internal ~~PWR Basket~~ *MPC* pressure and establishes the values used in stress calculations applicable to the ~~PWR Basket~~*MPC* ~~containment/~~confinement boundary. These values conservatively bound pressurization events that could occur in postulated accident scenarios.

8.2.6.1 Cause of Accident

This event is considered a beyond Design Basis event. There is no known causal factor for this event. The event would result from the breach of fuel rods in a*n* ~~PWR Basket~~*MPC* with release of 30~~%~~ *percent* of the fission gases combining with 100~~%~~ *percent* of the fill gases (helium). This would pressurize the ~~PWR Basket~~*MPC* shell and ~~structural~~ lids.

8.2.6.2 Accident Analysis

The analysis of this accident entails calculation of the free volume in the ~~PWR Basket~~*MPC*, calculation of the quantity of helium fill gas, Burnable Poison Rod Assembly (BPRA) gas, and fission gas from 24 fuel assemblies, and the subsequent calculation of the pressure in the ~~PWR Basket~~ *MPC* if these gases are added to the ~~He~~ *helium* backfill gas ~~(at 1 atm)~~ already present in the ~~PWR Basket~~*MPC*. The initial internal gas *pressure and* temperature ~~is~~ *are* assumed to be *46 psia and* ~~150~~*70*°F, *respectively*. The free volume of the ~~PWR Basket~~*MPC* and fuel rod voids was calculated to be ~~352,101~~*363,946* cubic inches.

The fuel rods were assumed to be backfilled during manufacture with helium gas at 35 atm which resulted in a calculated 151 moles of available backfill gas. The quantity of fission gases was conservatively estimated assuming that 30~~%~~ *percent* of the total fission gases present are released from the fuel. The total quantity of available fission gas was calculated to be 689 moles assuming 0.303 atoms of gas/fission and a burnup of 45,000 MWd/MTU. The quantity of helium backfill gas in the ~~PWR Basket~~*MPC* was calculated to be ~~207~~ *774* moles (based on a *reference backfill of 46 psia at* ~~final average PWR Basket gas temperature of 150~~*70*°F ~~at backfill~~).

Total moles of gas in the ~~PWR Basket~~*MPC* (same backfill conditions as maximum normal):

$$N_{total} = N_{basket} + (N_{rod\ fill}) + (0.3)(N_{fission\ gas}) + (0.3)(N_{BPRAHeGas})$$

N_{total} = ~~207~~ *774* moles + 151 moles + (0.3) 689 moles + (0.3) 130 moles = ~~604~~ *1,171* moles

Maximum ~~PWR Basket~~*MPC* pressure under accident conditions with helium at a bulk temperature of ~~503~~*374*°F (~~535~~ *463* °K):

$$P_{total} = \frac{\cancel{604}\ 1{,}171 \text{ moles} \times 0.082\cancel{056}\ \frac{\text{atm x Liter}}{\text{mole x K}} \times \cancel{535}\ 463 \text{ K}}{\cancel{352,101}\ 363{,}946 \text{ in}^3 \times \frac{1 \text{ Liter}}{61.02 \text{ in}^3}} \cancel{= 4.60 \text{ atm} = 67.6 \text{ psia } (52.9 \text{ psig})}$$

= 7.45 atm = 109.5 psia (94.8 psig)

This hypothetical accident pressure loading is evaluated in tandem with the normally occurring dead weight and handling loading per Section III, Division 1, Class ~~2~~ *1* of the ASME Code. The resulting stresses (in combination with other loads) and corresponding acceptance criteria are summarized in Table 8.2-1. As shown in this table, stresses are within the Code allowables for the Service Level D loadings. ~~The stresses in the Basket Overpack resulting from the hypothetical accident pressure were also determined to meet Code allowable limits.~~

8.2.6.3 Accident Dose Calculations

There are no radiological releases or adverse radiological consequences from this event.

8.2.7 FULL BLOCKAGE OF AIR ~~FLOW~~*INLETS*

This hypothetical accident postulates sudden ~~loss of air flow through~~ *blockage of* the Concrete Cask *air inlets.*

8.2.7.1 Cause of Accident

This hypothetical accident would be caused by complete air flow blockage of all air inlets ~~and all outlets~~. The Concrete Cask has four wire mesh screen--covered openings ~~which~~ *that* permit air entry into the air pad channels. There are two horizontal air pad channels located parallel to each other on the bottom of the Concrete Cask. The air inlets are openings oriented perpendicular to the air pad channels and permit a continuous flow of air through the bottom of the Concrete Cask, up the annulus, and out the air outlet vents located on the sides of the Concrete Cask near the top. Figure 4.2-4 shows the configuration of the Concrete Cask including the orientation of the air inlet/outlet vents.

8.2.7.2 Accident Analysis

The analysis of this event uses a steady state condition with the ambient temperature equal to ~~75~~*100*°F, ~~(no solar load)~~*solar insolation,* and air flow in the annulus (between the ~~PWR Basket~~*MPC* shell and the Concrete Cask liner) as the initial condition. The analysis is then changed to a transient condition with sudden ~~removal~~ *blockage* of *the* air ~~flow in the annulus (simulating a full blockage condition with no internal convective air heat removal and an adiabatic heatup of the Concrete Cask)~~*inlets disrupting the normal buoyancy-driven air flow.* The transient analysis is performed for ~~seven days (~~*approximately* ~~168~~*58* hours~~)~~. This analysis utilizes the same *Concrete Cask body and MPC exterior* thermal model described in Section 4.2.6.*4.*

Calculations show that the Concrete Cask inner concrete temperature limit is reached well before the short-term fuel cladding or other component temperature limits are reached. As shown in Figure 8.2-4, the maximum *inner* concrete allowable temperature of 350°F for short-term conditions is reached at approximately ~~31.5~~*57.1* hours after all air *inlet* flow is blocked. ~~The corresponding fuel cladding temperature at 31.5 hours after all air flow is blocked is approximately 826°F.~~ The approximate value~~s~~ for *the* ~~maximum~~ concrete outer surface temperature~~, maximum PWR Basket shell temperature and maximum fuel temperature are~~*at 57.1 hours (i.e., when the concrete inner surface temperature reaches its limit) is* shown in Table 4.2-12.

Potential blockage of the air inlets would be detected by the ISFSI facility staff as they perform required ~~visual~~ surveillances. The required action when a vent or vents are found to be blocked is to remove the foreign material blocking the air inlets. Since screens are provided for the vents, blocking material will likely be on the outside and easily removed. Materials that may be located inside the screens may be removed by hand-held tools after the screen is removed.

8.2.7.3 Accident Dose Calculations

There are no radiological releases or adverse radiological consequences from this event.

8.2.8 EXPLOSIONS OF CHEMICALS, FLAMMABLE GASES, AND MUNITIONS

This analysis addresses the hazards posed by potential explosions on transportation routes and in the vicinity of the ISFSI. The effects of accidents related to operation of a natural gas turbine combined cycle plant at the Trojan site are addressed in Section 8.2.14.

8.2.8.1 Cause of Accident

As presented in Section 2.2.3.1, the only source of potential explosions near the Trojan site that could affect safety related structures is shipment of commercial explosive cargo near the plant.

Trojan plant structures and the ISFSI site itself contain no explosive materials. The small quantities of gasoline or fuel oil that may be contained in the fuel tanks of vehicles (e.g., forklifts and mobile cranes) or standby power supply engines near the ISFSI present an insignificant explosion hazard. Explosions unrelated to transportation are not considered significant. Refer to Section 2.2.3.1 for additional information on potential sources of explosions in the vicinity of the site.

In addition, the Trojan DSAR analysis calculated the probability of a disabling accident based on a 2.2 psi overpressure recognizing that the actual overpressure could be 4.4 psi as a result of reflected waves. The probabilities for a disabling accident from rail and barge shipments were each less than 10^{-6} per year and would be similar for the ISFSI.

8.2.8.2 Accident Analysis

As noted in Section 2.2.3.1, the maximum anticipated transportation-related explosion overpressure at the plant site is 2.2 psi. Considering reflected shock waves from a detonation on a nearby transportation route, the resulting overpressure may increase by a factor approaching two. An overpressure of 4.4 psi is conservative for an analysis of the ISFSI Concrete Casks. As noted above, the explosion hazard from activities at the ISFSI site is insignificant.

The Concrete Casks have been shown in Section 8.2.4 to withstand a tornado wind pressure of 331.8 psf (or 2.3 psi) and missile impacts without sliding or overturning. The magnitude of explosion that would result in overturning or sliding of a Concrete Cask was determined as follows:

The force required to slide a Concrete Cask is:

F_{slide} = W_{cask} x 0.3 = ~~290,000~~*292,700* lbs x 0.3 = ~~87,000~~*87,810* lbs

where:

0.3 is the friction coefficient between the Concrete Cask and the Storage Pad

W_{cask} = Loaded Concrete Cask weight

The initial moment required to uplift a Concrete Cask is:

M = ~~290,000~~*292,700* lbs x 58.5 in = ~~16.9~~*17.1* x 10^6 lbs-in

where:

58.5 in*ches* is the moment arm

The force required to develop the above moment:

$$F_{overturn} = M / (L/2) = \cancel{16.9}\mathit{17.1} \times 10^6 \text{ lbs-in} / (211.5 \text{ in}/2) = \cancel{159,811}\mathit{161,702} \text{ lbs}$$

where:

L = Length of the Concrete Cask

The force required to slide the Concrete Cask is smaller and, therefore, is controlling. The minimum pressure on the Concrete Cask to result in this force is:

$$p = F_{slide} / (C_f A_p) = \cancel{87,000}\mathit{87,810} / (0.52)(199.75)\mathit{(144)} = 5.87 \text{ psi}$$

where:

C_f = Net pressure coefficient = 0.52 (Reference ANSI A58.1, Table 12)

A_p = Projected area of Concrete Cask normal to wind
= 136 x 211.5/144 = 199.75 ft^2

Therefore, the pressure required to cause Concrete Cask sliding is 5.87 psi which is much greater than the assumed 4.4 psi pressure that could be caused by explosions in the vicinity of the ISFSI. An even greater pressure would be required to overturn a Concrete Cask. Based on the foregoing, the integrity of the ISFSI Concrete Casks~~, PWR Baskets,~~ and ~~Basket Overpacks~~*MPCs* would not be adversely affected by postulated explosions near the site.

8.2.8.3 Accident Dose Calculations

There are no radiological consequences from this accident.

8.2.9 FIRES

Section 2.2.3.3 provides information regarding the hazard to the ISFSI presented by fires. No significant fires are expected at the ISFSI. The major transient combustible used within the ISFSI would be the gasoline, propane, or diesel fuel oil used in ~~forklifts~~ *transport vehicles* and the mobile crane. Normally, these vehicles would not be located at the ISFSI. ~~Forklifts~~ *Transport vehicles* would be used during the initial movement of the Concrete Casks to the pad, and a mobile crane would be used for the infrequent event of ~~installing a Basket Overpack and for~~ loading of ~~Shipping~~ *Transport* Casks. When these vehicles are in use, they will be accompanied by personnel who would detect and suppress the small fires associated with fuel leaks.

The plant is protected from industrial and forest fires by natural barriers and by the distance between combustibles and the ISFSI Concrete Casks. Additional protection is provided by the paved open areas surrounding the ISFSI. In addition, the massive concrete walls of the Concrete Casks provide shielding from the effects of thermal flux generated by nearby fires. Therefore, fires pose an insignificant hazard to the ISFSI.

8.2.10 COOLING TOWER COLLAPSE

The potential hazard associated with a postulated collapse of the ~~492~~ *499*-foot high Trojan Plant cooling tower was identified in Section 2.2.3.5. The cooling tower has been determined to pose no threat to the ISFSI owing to the distance between the tower and the ISFSI. The shortest distance between the cooling tower and the ISFSI is over 800 feet. This distance is sufficiently large that the unlikely collapse of the tower would not have any adverse impact on the ISFSI.

8.2.11 VOLCANISM

8.2.11.1 Cause of Accident

Volcanic activity near the Trojan Plant ~~are~~ *is* addressed in Section 2.5.6 of the TNP DSAR. Four volcanoes are located in the general area, the closest one being 34 miles from the site.

8.2.11.2 Accident Analysis

The discussion of volcanoes in DSAR Section 2.5.6 concludes that the potential eruptions pose a minimal risk to the plant. Nevertheless, the effects that are believed to be of concern are ash fall, mud flow, and flooding. The Trojan Nuclear Plant DSAR discusses the maximum ash thickness that would occur from an eruption similar to the May 18, 1980 paroxysmal eruption of Mt. St. Helens if the ash clouds were directed at the Trojan Nuclear Plant. An ash fall depth at the ISFSI site was determined to be about 1.8 inches. This would not be sufficient ash to block the Concrete Cask air inlets. However, the full blockage of the air inlets has been analyzed in Section 8.2.7 and shown not to result in exceeding temperature limits or to adversely affect the safe storage of spent fuel.

The results of analyses performed to determine the effects of volcanically generated mud flow and flooding, including the effects of volcanically-induced dam failures, are provided in the DSAR. These analyses show that the effects at the ISFSI site are minimal and would not pose a hazard to the ISFSI.

8.2.11.3 Accident Dose Calculations

The effects of volcanically-induced hazards pose a negligible risk to the ISFSI, and no radiological consequences are anticipated from this event.

8.2.12 LIGHTNING

8.2.12.1 Cause of Accident

This event would be caused by meteorological conditions at the site. Lightning striking one of the Concrete Casks is not a likely event, because the ISFSI Storage Pad is surrounded on two sides by an earthen berm and some lightning protection will be afforded by the lighting towers that will be located around the Storage Pad (Reference ~~8.~~7). In addition, the Trojan Nuclear Plant is in a low isokeraunic level area; the mean annual days of thunderstorm activity at the ISFSI will be less than ten.

8.2.12.2 Accident Analysis

Even if the Concrete Cask were to be hit by lightning, the likely path to ground would be from the steel Concrete Cask lid to the steel base plate via the steel Concrete Cask liner and the steel air inlet ducts. The ~~PWR Basket~~*MPC* is surrounded by these steel structures and would not provide a likely ground path. Therefore, a lightning strike would not affect ~~PWR Basket~~*MPC* integrity. The heat absorbed would be insignificant from the standpoint of ~~PWR Basket~~*MPC* cooling due to a very short duration of the event. If the lightning entered or exited the Concrete Cask via the concrete shell, some local spalling of concrete may occur. A significant loss of concrete shielding would not be expected. Concrete Cask operation would not be adversely affected.

8.2.12.3 Accident Dose Calculations

Based on the evaluation above, the radiological consequences of this accident would be similar or less than those discussed in Section 8.2.4 for a localized loss of concrete shielding following a tornado missile strike.

8.2.13 ~~OVERPACK OPERATIONS AND~~ OFF-SITE SHIPPING EVENTS

At the Transfer Station, positioning stops are attached to the structure to accurately position a Concrete Cask or ~~Shipping~~ *Transport* Cask relative to the Transfer Cask ~~which~~ *that* is rigidly restrained within the Transfer Station.

Adapter plates are used in the top of the Concrete Cask and ~~Shipping~~ *Transport* Cask ~~which~~ *that* mate with the Transfer Station stops for accurate horizontal positioning. The Transfer Station shield ring, when lowered into position for ~~PWR Basket~~*MPC* transfer, mates with the Concrete Cask or ~~Shipping~~ *Transport* Cask adapter plate such that the inside diameter of the Concrete Cask or ~~Shipping~~ *Transport* Cask, the inside diameter of the shield ring, and the inside diameter ~~or~~ *of* the Transfer Cask are aligned. With these alignment design features, interferences during

~~PWR Basket~~*MPC* movements are not likely to occur. The following events are analyzed, however, in order to bound any similar events.

8.2.13.1 Interference During Raising or Lowering the ~~Basket~~ *MPC*

The ~~PWR Basket~~*MPC* catches on the Transfer Cask while being moved. While proper procedures to ensure alignment of the components should prevent this condition from occurring, it is analyzed nevertheless to bound similar occurrences.

8.2.13.1.1 Cause of Accident

The cause is operator error for failing to assure adequate clearance and/or alignment.

This event may be detected by audible noise emitted by the ~~PWR Basket~~*MPC* as it contacts the Transfer Cask or *visually* by upward movement of the Transfer Cask.

8.2.13.1.2 Accident Analysis

The locations where the ~~PWR Basket~~*MPC* is moved relative to the Transfer Cask are the Fuel Building, at elevation 45 ft., while loading the ~~PRW Basket~~*MPC* into the Concrete Cask, or at the Transfer Station during movements between a Concrete Cask or ~~Shipping~~ *Transport* Cask.

At both locations, an impact limiter which is designed to preclude unacceptable damage to the fuel is located beneath the receiving ~~Concrete C~~cask. No damage to the fuel would occur in the unlikely event of a failure in the lifting system.

8.2.13.1.3 Accident Dose Calculation

There are no radiological releases or adverse radiological consequences from this event.

8.2.13.2 Interference During ~~Basket~~ *MPC* Lowering into a Concrete Cask *or Transport Cask*

The ~~Basket~~ *MPC* catches on the Concrete Cask *or Transport Cask* edge or side while being lowered into a Concrete Cask *or Transport Cask.*

While proper procedures to ensure alignment of the components should prevent this condition from occurring, it is analyzed nevertheless to bound similar occurrences.

8.2.13.2.1 Cause of Accident

The cause is operator error for failing to assure adequate clearance and/or alignment.

This event may be detected by audible noises emitted from the ~~PWR Basket~~*MPC* sliding on the Transfer Cask, Concrete Cask, *or* ~~Shipping~~ *Transport* Cask, or ~~Basket Overpack~~ *visually* by a slackening of the ~~wire~~ slings which connect the ~~PWR Basket~~*MPC* to the crane hook.

8.2.13.2.2 Accident Analysis

Since the only force acting on the ~~PWR Basket~~*MPC* during lowering is gravity, the ~~worse~~ *worst* case condition would be a load of 1g on the ~~PWR Basket~~*MPC* bottom or side if it were to be completely supported from its interference. The stresses applied to the ~~PWR Basket~~*MPC* in this scenario are bounded by those analyzed in Sections 8.2.13.3 and 8.2.13.4. The Transfer Cask and Concrete Cask are both analyzed to support the weight of the ~~PWR Basket~~*MPC*.

To recover from this event the operator would immediately halt lowering the ~~PWR Basket~~*MPC*, inspect the area for interference, and then raise the ~~PWR Basket~~*MPC* back into the Transfer Cask. If interference still existed after another attempt to lower the ~~PWR Basket~~*MPC*, the operator would raise the ~~PWR Basket~~*MPC* back into the Transfer Cask and investigate the Concrete Cask, ~~Basket Overpack, Shipping~~ *Transport* Cask, or the Transfer Cask for obstructions or foreign objects.

8.2.13.2.3 Accident Dose Calculation

There are no radiological releases or adverse radiological consequences from this event.

8.2.13.3 ~~Basket~~ *MPC* Drop into Concrete Cask *or Transport Cask*

The ~~PWR Basket~~*MPC* is dropped vertically into the Concrete Cask or ~~Shipping~~ *Transport* Cask during ~~Basket Overpack operations or during PWR Basket~~*MPC* handling for transfer off site.

8.2.13.3.1 Cause of Accident

Postulated crane failure.

8.2.13.3.2 Accident Analysis

~~Basket~~ *MPC* lifting operations may be performed on the ISFSI reinforced Transfer Pad in order to transfer a loaded and sealed ~~PWR Basket~~*MPC* from a Concrete Cask to the ~~Shipping~~ *Transport* Cask ~~or to allow installation in a Basket Overpack~~. During these operations, the loaded ~~Basket~~ *MPC* will be raised ~~and subsequently lowered vertically~~*from* within the Concrete Cask~~s~~ *and subsequently lowered vertically into the HI-STAR 100 Transport Cask.*

For the postulated scenario (drop of a*n* ~~PWR Basket~~*MPC* into the Concrete Cask or ~~Shipping~~ *Transport* Cask), only an end drop is applicable because the ~~PWR Basket~~*MPC* is ~~dropped~~

~~straight into a~~*transferred* vertical*ly into a* Concrete Cask *or Transport Cask.* The end drop height is assumed to be the distance ~~from the top of the Transfer Cask doors to~~*between the bottom of a raised MPC in the Transfer Cask and* the bottom of the Concrete Cask *or Transport Cask* cavity. The distance of ~~241~~ *249* inches for the ~~Shipping~~ *Transport* Cask is bounding *(see Section 8.2.13.3.3).*

The Transfer Station design utilizes an impact limiter embedded in the Transfer Station foundation mat to mitigate the consequences of a hypothetical ~~PWR Basket~~*MPC* drop during ~~PWR Basket~~*MPC* transfer from the Concrete Cask to a ~~Shipping~~ *Transport* Cask ~~(or Concrete Cask with an Overpack)~~. The ~~revised~~ design concept of the Transfer Station is summarized below.

The Transfer Station ~~will be~~*is* a fixed structure supported on a reinforced concrete mat foundation placed directly on sound rock. An impact limiter ~~will be placed~~*is located* directly below the ~~PWR Basket~~*MPC* transfer position and flush with the top of the Transfer Station mat. The impact limiter is designed to ~~maintain~~ *provide defense-in-depth to ensure* acceptable ~~PWR Basket~~ *MPC* ~~decelerations~~ *confinement boundary stresses* under hypothetical drops into either a Concrete Cask or a ~~Shipping~~ *Transport* Cask.

Prior to ~~PWR Basket~~*MPC* transfer operations, an empty Transfer Cask will be placed into position in the Transfer Station (directly above the impact limiter) using a mobile crane, aligned into its proper position, and restrained in place. At this point and during subsequent ~~PWR Basket~~*MPC* transfers, the Transfer Cask will remain~~ed~~ fixed in place and vertical and horizontal design load reactions will be resisted by the Transfer Station structure.

In preparation for a*n* ~~PWR Basket~~*MPC* transfer, a loaded Concrete Cask will be transported on air pads from its storage position to the vicinity of the Transfer Station. After being prepared (lid removed, etc.), the Concrete Cask will be moved ~~onto~~ *into* position beneath the Transfer Cask and directly over the impact limiter embedded in the Transfer Station foundation mat. Shielding will then be placed in position to cover the gap between the Transfer Cask and Concrete Cask. The shielding is an integral part of the Transfer Station design and is positioned using hydraulic jacks.

After preparing the Transfer Cask and attaching appropriate rigging to the ~~PWR Basket~~*MPC*, the ~~PWR Basket~~*MPC* will be lifted by the mobile crane from the Concrete Cask into the Transfer Cask *(see Figure 8.2-6)*. The Transfer Cask shield doors will be closed, and the ~~PWR Basket~~*MPC* will be lowered to rest. The shielding and Concrete Cask will then be removed, and the prepared ~~Shipping~~ *Transport* Cask will be transported on air pads to the loading position beneath the Transfer Cask. The shielding will be replaced and the ~~PWR Basket~~*MPC* subsequently lowered from the Transfer Cask into the ~~Shipping~~ *Transport* Cask ~~(or Concrete Cask with Basket Overpack)~~.

As indicated above, ~~I~~*i*n overall concept, the Transfer Station is essentially a passive support structure for the Transfer Cask during vertical ~~spent fuel PWR Basket~~*MPC* movements between the Concrete Cask and the ~~Shipping~~ *Transport* Cask.

~~An additional evaluation was performed to assess the structural integrity of the fuel during this postulated drop. For the postulated drop of a PWR Basket dropped into an empty Concrete Cask or Shipping Cask, the resulting deceleration load was calculated to be less than approximately 75g. This is well below the 124g acceptance criteria for a vertical drop used for the PWR Basket structural design. To assess the affects on the spent fuel, the methodology of Lawrence Livermore National Laboratory Report UCID-21246, "Dynamic Impact Effects on Spent Fuel Assemblies," was used.~~

~~The methodology of UCID-21246 provides a reasonable assessment of the lower bound threshold of gross damage to spent fuel rods in a structurally stable PWR Basket under impact conditions. In a vertical orientation, the deceleration limit for an individual spent fuel rod is based on elastic (Euler) buckling. For the Trojan 17 x 17 eight grid Westinghouse fuel assemblies, the deceleration corresponding to fuel rod cladding elastic buckling, is 105g for uncorroded fuel rod cladding.~~

~~To assess the impact of potential fuel cladding corrosion on the evaluation of fuel integrity, an upper bound cladding oxide thickness was applied between the bottom two fuel assembly grids (nearest the impact surface) but not above. This assumption minimizes the cladding cross section in the area of potential bucking and maximizes the impact inertial loads. The resulting deceleration limit was determined to be approximately 100g, well above the 75g loading calculated for the postulated PWR Basket drop.~~

~~The results of the fuel cladding integrity evaluation is consistent with other data in the technical literature. A paper presented at the 10th International Conference on Structural Mechanics in Reactor Technology (SMIRT) summarized an analysis of the effects of dropping a typical PWR fuel assembly in a vertical orientation 30 ft. onto a rigid surface using a complex three-dimensional finite element model (Reference 8.13). The analysis found that only the bottom three fuel pellets in a fuel rod would be fractured over 1.325" from the bottom of the fuel rods (less than 1% of the total height of the fuel pellets).~~

~~A second paper presented at the SMIRT conference (Reference 8.14) describes the results of the analysis and verification mockup testing of a PWR spent fuel assembly to assess the structural integrity of the fuel during drops in various orientations. The analysis and testing used a Westinghouse 17 x 17 nine grid fuel assembly. The analysis and testing of a vertical drop was from a height of 19.68 ft. which is approximately the same as the maximum drop height in the Trojan ISFSI Transfer Station (19.81 ft.). The mockup vertical drop test resulted in no failed fuel rods that would lead to fission gas release (i.e., clad rupture). The conservative analysis had~~

~~predicted the failure of one fuel rod in each outside corner of the assembly (approximately 1.5% of the fuel rods in the assembly).~~

~~Minor fuel damage such as that experienced in the referenced testing and analysis would not result in an inadvertent criticality. Previous analysis of the potential criticality of fuel debris concluded that the fuel pellets contained in up to 25 Westinghouse 17 x 17 fuel rods can not be arranged in any configuration that is as reactive as the configuration assumed in the criticality analysis of the fuel debris canisters. The most reactive configuration for the number of pellets described above resulted in a~~ $\cancel{k_{eff}}$ ~~of 0.8738 (water moderated).~~

8.2.13.3.3 MPC Drop Analysis

A postulated MPC drop accident during inter-cask transfer operations at a generic and bounding Transfer Station was analyzed to demonstrate that the confinement boundary will not breach as a result of this event (i.e., through-wall cracking will not occur in the MPC confinement boundary). The MPC was assumed to drop vertically from a height of 25 feet onto a concrete target. This conservatively bounds the aforementioned 249 inches. The target concrete pad was assumed to have a compressive strength of 6,000 psi, and a thickness of 22 feet. This bounds the actual stiffness of the MPC pedestal in the Concrete Cask, the Transfer Station embedded impact limiter, and foundation below.

The postulated MPC drop event was analyzed as a transient, nonlinear problem involving a number of structural components in dynamic contact. The finite-element method was used to conduct the numerical simulation for the drop event to obtain a conservative damage estimation. The impact damage is evaluated in terms of stress, strain and deformation in the structural members involved in the impact.

A commercial computer code developed by the Livermore Software Technology Corporation and QA validated by Holtec International, LS-DYNA, was used to numerically model the impact problem. LS-DYNA is the same computer code used for dynamic analyses described in the HI-STORM 100 System FSAR.

The acceptance criterion taken from Section 3.5.2.1.4 of Appendix B to the HI-STORM CoC is that MPC confinement integrity must be maintained after the drop accident. Namely, the MPC confinement boundary must not be breached in the accident. A breach is defined as the strain calculated at any location exceeding the specified failure strain for the material.

The stainless steel used to manufacture the MPC is considered as a bi-linear elasto-plastic material; the failure strain is established at 0.38 in/in (38 percent). This is a conservative assumption since a material subjected to an instantaneous deformation usually has increased yield and ultimate strains.

To bound the physical problem, the following assumptions were made:

1. *The loaded MPC is assumed to weigh 90,000 lbs in the analysis, which conservatively envelops the actual weight of the heaviest Trojan MPC-24E or MPC-24EF, provided in Table 4.2-4 of this SAR.*

2. *A hypothetical postulated failure in any of the components in the load path supporting the MPC could cause it to fall vertically and impact the base pedestal of the Concrete Cask. The impact load is ultimately borne by the impact limiter and Transfer Station pad with the intervening Concrete Cask structural members serving as limited dissipaters of energy and providing added compliance to the collision interface. For conservatism, it was assumed that the MPC free-falls and collides with the target pad without any dissipative contribution from embedded impact limiter or Concrete Cask structural members.*

3. *The MPC contents, namely, the fuel basket and the stored spent fuel, are metallic components that, under impactive loading, would absorb a certain amount of energy. For conservatism, the contents of the MPC are represented as a rigid mass (i.e., no energy dissipation).*

4. *The MPC top lid-to-shell weld joints are explicitly modeled with full recognition of the discontinuity stresses that are expected to develop at these locations. The material behavior of the weld joints is assumed to be the same as the MPC shell material.*

5. *The compressive strength of the target concrete pad employed in the analysis is assumed to be 6,000 psi. The concrete pad thickness is assumed to be 22 feet. The combined effect of the above assumptions makes the impact target significantly stiffer than the actual embedded impact limiter and Transfer Station foundation.*

The finite element model was developed based on the actual configuration of the heaviest Holtec International MPC model (MPC-32). This bounds the configuration for all other MPCs. The MPC lid was modeled with solid elements and has a $^1/_{16}$-inch radial gap with the MPC shell. Shell elements were used to model the MPC shell and MPC baseplate. The MPC shell was discretized using very fine grids, especially at the connection regions with the lid and the baseplate. The stored fuel assemblies as well as the fuel basket, basket supports and fuel spacers were represented as a solid sitting on the baseplate and with a small gap to the shell. The weld between the lid and the shell was modeled by solid elements along the periphery.

The concrete model spanned sufficiently far away from the impacted region. Non-reflecting boundary conditions were applied at the appropriate boundary surfaces to simulate the infinite

half space of the target. The grids in the impacted region are finer than that of the remote region to simulate the expected large gradients.

The maximum Von Mises stress in the shell was calculated to be 43,588 psi, indicating that the shell is plastically deformed. However, this calculated stress is smaller than the ultimate stress of the material (64,000 psi). The maximum plastic strain is less than 15.5 percent compared with the failure strain limit of 38 percent. Therefore, the confinement boundary is not breached as a result of this non-mechanistic accident. Note that the MPC shell is deformed most at the lower end because of the impact-induced local bending. The maximum compressive stress in the concrete pad is 9,253 psi along the periphery of the directly impacted area, which is greater than the compressive strength of the concrete.

It is concluded that the postulated free-fall drop accident of a loaded MPC from a HI-TRAC transfer cask into a HI-STAR 100 Transport Cask or Concrete Cask at the Transfer Station will not result in the breach of MPC confinement, since the maximum plastic strain is only 41 percent of the failure strain. The results of this evaluation show that no through-wall cracking of the MPC confinement boundary occurs and confinement is not breached.

~~8.2.13.3.3~~*8.2.13.3.4* Accident Dose Calculation

There are no radiological releases or adverse radiological consequences from this event.

8.2.13.4 Loaded ~~Shipping~~ *Transport* Cask Drop

A vertical or horizontal drop of a loaded ~~Shipping~~ *Transport* Cask is speculated to occur during transfer to a heavy-haul trailer or rail car prior to the installation of transportation packaging impact limiters.

Section 9.7.6 establishes that a program provide the requirements governing handling or lifting fuel bearing components including ~~Shipping~~ *Transport* Casks. Handling/lifting of spent fuel or handling/lifting of loads over spent fuel are performed only in accordance with approved lift plans. An evaluation of consequences of a drop or handling accident shall be performed prior to initiating the handling/lifting activities.

In accordance with the program described in Section 9.7.6, an evaluation to criteria equivalent to ~~that~~ *those* specified in NUREG--0612 will be performed of the entire fuel transfer and loading process. Handling of the ~~Shipping~~ *Transport* Cask at the ISFSI could utilize increased safety factors in the rigging to preclude drops or impact limiters to mitigate the effects of drops prior to installation of the transportation packaging.

8.2.14 NATURAL GAS TURBINE COMBINED CYCLE POWER PLANT EVENTS

PGE has evaluated repowering the Trojan Plant with a Natural Gas-fired Turbine Combined Cycle (NGTCC) power plant. This modification to the plant and site would involve locating one or two NGTCC facilities within the existing Turbine Building. A 16-inch diameter pipeline carrying natural gas would be routed onto the site generally from the south and west. The pipeline would enter a metering station from which an 8-inch diameter pipeline will be routed to each gas-fired turbine in the Turbine Building.

PGE will not implement the modifications required for repowering until the spent nuclear fuel currently in the Spent Fuel Pool has been transferred to the ISFSI. This analysis has been performed to show that, when the spent fuel has been stored within the ISFSI, the hazards associated with the operation of a NGTCC facility at TNP would have no adverse impact on the ISFSI.

8.2.14.1 Cause of Accident

The introduction of a NGTCC power plant at the TNP site introduces several potential accidents that have been evaluated in this section. The NGTCC design would include both natural gas and conventional steam-cycle turbine generators that would be located in the Turbine Building. Supply lines for natural gas would be routed to the plant. Tanks for alternative fuel supply (fuel oil) would be constructed. An exhaust stack would be required for each NGTCC unit.

The use of turbine generators introduces the possibility of turbine missiles generated by postulated turbine blade failures. These missiles potentially could strike the Concrete Casks.

Use of natural gas on the site requires the potential effects of fires and explosions to be considered. The fire hazard posed by the alternative fuel tanks is evaluated.

The exhaust stack location is evaluated to assure that a stack that failed during a seismic event or tornado would not adversely affect the ISFSI.

8.2.14.2 Accident Analysis

8.2.14.2.1 Turbine Missiles

The orientations of the gas and steam turbine generators would place the ISFSI outside of the low trajectory missile zone described in Regulatory Guide 1.115 which is 25 degrees to either side of a line perpendicular to the turbine shaft. The low trajectory missile zone is applicable to low-pressure stage shrunk-on wheels of the 1800-rpm turbines generally used with light-water-cooled reactors. Since the proposed NGTCC is a 3600-rpm gas turbine, the appropriate missile strike zone is expected to be similar, however the missile hazard associated

with the high speed turbines would be a high-trajectory missile. An evaluation of the probability of a high trajectory gas turbine missile strike within the ISFSI area has been performed. The probability of a strike upon the ISFSI Storage Pad was determined to be approximately 2E-5.

Assuming that the probability of missile generation due to turbine wheel breakup is approximately the same as that of a steam turbine (about 1E-5/yr), the probability of a missile being generated and striking within the ISFSI area would be less than approximately 2E-10/yr which is below the threshold value of 1E-7/yr for which a consequence analysis is required. In addition, the proposed location of the steam turbine is on El. 45-ft of the Turbine Building. On that location, intervening structures (floor slabs, structural steel and equipment) provide the necessary shielding to prevent high trajectory missiles from leaving the Turbine Building. Therefore, neither gas turbine or steam turbine missiles represent a significant hazard to the ISFSI.

8.2.14.2.2 Fire/Heat Flux Effects

To determine the effects of the radiant heat flux on the ISFSI Concrete Casks, a conservative steady state Concrete Cask surface temperature was calculated, by considering radiation and forced convection heat transfer. This temperature provides an upper limit for the temperature rise of Concrete Cask cooling air due to a postulated fire fed by a natural gas line break at a point closest to the ISFSI. The temperature rise was calculated to be less than 4°F, without taking credit for the substantial heat sink available from the cask concrete. Also, conservative absorptance and emittance values for concrete were used; a 5 mph wind was modeled even though the reference thermal flux calculation used a 24 mph wind; and the transient time to reach steady state, which would be quite long, was not credited. The 4°F air temperature rise is less than the accident case margin between the normal steady-state concrete temperature and the limiting concrete temperature for the design basis heat load.

The postulated fire would tend to increase ambient air temperature in the vicinity of the fire. However, this would have minimal effect on ambient temperature at the Concrete Casks due to the large standoff distance, combined with the buoyancy of the heated air and the location of the Concrete Cask inlet air ducts at the base of the Concrete Cask. Even if ambient air temperature were raised near the Concrete Casks, this would have little or no effect on the ΔT through the Concrete Cask itself. The temperature dependence of the Reynolds number, through the kinematic viscosity of air, is minor when factored into the heat transfer balance.

The thermal flux effects from oil tank fires are bounded by those due to natural gas line breaks.

8.2.14.2.3 Partially Confined Explosion at the ISFSI

The partially confined explosion within the ISFSI Concrete Cask array bounds the consequences of any unconfined explosion that may result from a natural gas deflagration.

A partially confined explosion presents a potentially greater hazard than an explosion in the free field due to flame acceleration in the vicinity of the ISFSI Concrete Casks. The Concrete Casks are approximately 11.3 feet in diameter and 17.6 feet high. The center-to-center spacing of the Concrete Casks is 15 feet. In an unconfined deflagration, the flame speed is typically low and flame acceleration does not occur or is minimal. However, partial confinement of the burning gas or the presence of obstacles in the flame path causes turbulence. Turbulence can accelerate the flame front. The effects of fuel reactivity, obstacle density, and confinement can be correlated to the flame speed. Based on the low reactivity of natural gas, and the medium density and open arrangement of Concrete Casks on the pad, a flame speed of approximately 112 ft/sec was selected for flame fronts within the ISFSI array.

An explosion overpressure calculation was performed assuming that the ISFSI Storage Pad was completely enveloped by a cloud of natural gas at the upper flammability limit of 15~~%~~ *percent* by volume, and ignition occurred at the center of a four Concrete Cask array. The calculated overpressure generated by the explosion was 1.47 psi. This overpressure is bounded by the explosion analysis in Section 8.2.8.

The potential for the explosion to overturn a Concrete Cask or cause it to slide was calculated. The peak overpressure would induce a sliding force on the Concrete Cask of 41.3 kips. Based on the weight of the Concrete Cask (~~290,000~~*292,700* lb*s*), the coefficient of static friction would have to be less than 0.14 to permit sliding. The coefficient of friction for the Concrete Cask sitting on a Storage Pad is much greater than this value, typically greater than 0.3. Therefore, sliding of the Concrete Cask is not a concern.

The overturning moment induced by the pressure of the explosion would have to overcome the resisting moment resulting from the weight of the Concrete Cask and the distance from its center of gravity to the edge of the Concrete Cask base. The overturning moment on a Concrete Cask generated by the blast pressure is less than the resisting moment and, thus, the partially confined explosion would not overturn a Concrete Cask. (Refer to the discussion of sliding and overturning in Section 8.2.8.)

The partially confined explosion would have no adverse effects on operability of the ISFSI.

8.2.14.2.4 Confined Explosions Inside Plant Structures

Confined explosions within plant structures were investigated to determine if the effects of those explosions could generate missiles that could impact and cause damage to the ISFSI Concrete Casks. The worst case confined explosions were those in the Containment and in the Auxiliary/Fuel Building. Other structures were found to be of a lighter construction than those structures and, therefore, the blast effects and ability to generate missiles was determined to be less than for those structures.

The debris that could reach the ISFSI site as a result of explosions in the Containment and Auxiliary/Fuel Building were calculated to be small (approximately 20 lb) and with impact energies insufficient to overturn or cause significant damage to the Concrete Casks. In addition, the debris would be unlikely to block more than half the Concrete Cask air inlets which is a condition analyzed in Section 8.1.2.2. Therefore, the effects of confined explosions in plant structures was found to pose an insignificant risk to ISFSI operability.

8.2.14.2.5 Seismic Event or Tornado

The proposed installation of the NGTCC plant would involve the construction of structures and systems designed to conventional Uniform Building Code requirements for seismic events and high wind exposures. Only the stack of the NGTCC unit that extends to the north of the Turbine Building could hypothetically impact the ISFSI in the event of extreme seismic ground motions or tornado winds and missiles. However, the stack is typically 210 feet in height, and it would be located approximately 280 feet from the ISFSI. Therefore, the stack would pose no hazard to the ISFSI should it fail during a seismic event or tornado.

Based on the evaluations described above, the following may be concluded:

1. A NGTCC plant turbine missile impact on an ISFSI Concrete Cask was determined to be not credible.

2. Overpressures and thermal flux from hypothetical bounding case unconfined or partially confined natural gas-air mixture deflagrations or oil tank fire would not result in effects beyond the design basis of the ISFSI.

3. Bounding case hypothetical natural gas-air mixture confined explosions inside plant structures would not generate debris missiles capable of causing unacceptable damage to the ISFSI.

4. Extreme seismic event or tornado effects on the NGTCC plant would not result in damage to the ISFSI.

Based on the above, it has been concluded that the proposed NGTCC plant at the Trojan site would not adversely affect the ISFSI.

8.2.14.3 Accident Dose Calculations

With the exception of missiles generated by confined explosions, no event associated with the use of a NGTCC facility presents a significant potential to cause increased offsite or occupational doses. The Concrete Cask would be repaired following a missile impact by filling the damaged

area with grout. It is presumed that some period of time will be required to obtain the materials needed to repair the Concrete Cask surface. Shielding materials will be maintained on site for use in mitigating the consequences of this event until such time as a repair to the Concrete Cask surface can be completed. It is estimated that shielding materials can be in place within 12 hours of the event. It is estimated that once the necessary materials are obtained two technicians would be able to complete the repair in approximately 30 minutes. The collective dose to the repair crew would be less than or equal to approximately ~~0.826~~*0.155* person-rem (~~413~~ *77.5* mrem to each technician). Direct radiation levels at the Controlled Area boundary as a result of this event are minimal for the expected duration of the event.

8.3 SITE CHARACTERISTICS AFFECTING SAFETY ANALYSIS

The ISFSI site is located as depicted in Figures 2.1-1 and 2.1-2. The installation is designed for storing 36 Concrete Casks and its layout is shown in Figure 2.1-3. The *loaded* Concrete Casks reside on a thick concrete slab with fifteen feet center-to-center spacing and an aisle through the middle of the array. The Controlled Area for the ISFSI site is shown on Figure 2.1-2. The ISFSI site is well shielded by an embankment on the north and east sides. Figure 1.1-2 shows the accessibility of the site to truck, rail, and barge transportation. Section 2.2.3 notes that the nearest natural gas line is approximately 1.5 miles from the site; operation of this gas line will not present a hazard to the ISFSI from explosion because of the distance from the site. Also, the hazards arising from the planned use of natural gas to fuel a gas-turbine generator on site are addressed in Section 8.2.14.

Site characteristics that affect the safety analysis are summarized in Table 8.3-1.

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Table 8.0-1

Design Basis Normal and Off-Normal Events

Sheet 1 of 1

Normal and Off-Normal Events (expected frequency up to one/year)	**~~PWR Basket~~*MPC* Press. Boundary**	**~~PWR Basket Internals~~*MPC Basket***	**Concrete Cask**
For analyses of Normal Events, refer to Chapter 4	-	-	-
1. Off-Normal Structural Analysis			
a. ~~PWR Basket~~*MPC* Off-Normal Handling Load	X	X	-
2. Off-Normal Thermal Analysis			
a. Severe Environmental Conditions	X	X	X
b. Blockage of One-Half of the Air Inlets	X	X	X
3. Off-Normal Contamination Release			
a. Small Release of Potential ~~Basket~~ *MPC* Surface Contamination	Radiological Consequences Only		

Table 8.0-2

Design Basis and Beyond Design Basis Infrequent (Accident) Events

Design Basis and Beyond Design Basis Infrequent (Accident) Events		~~PWR Basket~~*MPC* Press. Boundary	~~PWR Basket Internals~~*MPC Basket*	Concrete Cask
1.	Failure of Fuel Pins with Subsequent Breach of ~~PWR Basket~~*MPC*	Radiological Consequences Only		
2.	Maximum Anticipated Heat Load - 125°F Ambient Temperature and Full Solar Load	X	X	X
3.	Concrete Cask Overturning	X	X	X
4.	Tornado	X	-	X
5.	Earthquake	X	-	X
6.	Pressurization	X	-	-
7.	Full Blockage of Air Inlets	X	X	X
8.	Explosions of Chemicals, Flammable Gases, and Munitions	X	-	X
9.	Fires	-	-	X

Table 8.0-2

Design Basis and Beyond Design Basis Infrequent (Accident) Events

Design Basis and Beyond Design Basis Infrequent (Accident) Events			**~~PWR Basket~~*MPC* Press. Boundary**	**~~PWR Basket Internals~~*MPC Basket***	**Concrete Cask**
10.	Cooling Tower Collapse		-	-	X
11.	Volcanism		-	-	X
12.	Lightning		-	-	X
13.	~~Overpack Operations and~~ Off-Site Shipping Events			~~Radiological Consequences Only~~	
	a.	Interference During Raising the ~~Basket~~ *MPC* from Concrete Cask into Transfer Cask	*X*	-	*X*
	b.	Interference During ~~Basket~~ *MPC* Lowering into a Concrete Cask *or Transport Cask.*		~~Radiological Consequences Only~~	
	c.	~~PWR Basket~~*MPC* Drop Into Concrete or ~~Shipping~~ *Transport* Cask	X	X	-
	d.	Loaded ~~Shipping~~ *Transport* Cask Drop			

Table 8.0-2

Design Basis and Beyond Design Basis Infrequent (Accident) Events

Design Basis and Beyond Design Basis Infrequent (Accident) Events	**~~PWR Basket~~*MPC* Press. Boundary**	**~~PWR Basket Internals~~*MPC Basket***	**Concrete Cask**
14. Natural Gas Turbine Combined Cycle Power Plant Events	X	-	X

Table 8.1-1

~~PWR Basket~~*MPC* Stresses (ksi) Resulting From Off-Normal Handling Event

Component		~~Off-Normal Handling~~	~~Dead Weight~~	~~Thermal~~	~~Pressure~~	~~Total~~ *Maximum Stress*	ASME Allowable Service Level ~~C~~*D*
Fuel Basket	P_m $P_L + P_b$					*13.72* *47.06*	*36.95* *55.45*
Shell	P_m $P_L + P_b$	~~9.9~~ ~~18.3~~	~~0.1~~ ~~0.2~~	~~N/A~~	~~0.5~~ ~~6.9~~	~~10.5~~*6.35* ~~25.4~~*24.65*	~~18.2~~*43.45* ~~27.3~~*65.20*
~~Bottom Plate~~ *Baseplate*	~~P_m~~ $P_L + P_b$	~~13.3~~ ~~13.2~~	~~0.0~~ ~~0.2~~	~~N/A~~	~~0.2~~ ~~4.6~~	~~13.5~~ ~~18.0~~*35.93*	~~18.2~~ ~~27.3~~*67.32*
~~Structural~~ Lid	~~P_m~~ $P_L + P_b$	~~6.2~~ ~~6.2~~	~~0.0~~ ~~0.1~~	~~N/A~~	~~0.0~~ ~~1.4~~	~~6.2~~ ~~7.7~~*21.83*	~~18.2~~ ~~27.3~~*61.05*
~~Top~~ *Lid* Weld[a]	~~P_m~~ ~~$P_L + P_b$~~ *Total Shear*	~~2.0~~ ~~12.0~~	~~0.1~~ ~~0.2~~	~~N/A~~	~~0.9~~ ~~1.2~~	~~3.0~~ ~~13.4~~*4.72*	~~14.6~~ ~~21.8~~*11.16[a]*
~~Sleeve Assembly~~	~~P_m~~ ~~$P_L + P_b$~~	~~15.4~~ ~~24.0~~	~~0.1~~ ~~0.1~~	~~N/A~~	~~0.0~~ ~~0.0~~	~~15.5~~ ~~24.1~~	~~27.4~~ ~~41.2~~
~~Shield Lid Ring Weld~~	~~P_m~~ ~~$P_L + P_b$~~	~~1.8~~ ~~4.0~~	~~0.1~~ ~~0.2~~	~~N/A~~	~~0.0~~ ~~0.0~~	~~1.9~~ ~~4.2~~	~~13.7~~ ~~20.5~~
~~Shield Lid Weld~~	~~P_m~~ ~~$P_L + P_b$~~	~~1.9~~ ~~4.3~~	~~0.1~~ ~~0.2~~	~~N/A~~	~~0.7~~ ~~0.9~~	~~2.6~~ ~~5.4~~	~~18.2~~ ~~27.3~~

[a] Weld *stress* allowable~~s~~ *is* reduced by ~~20~~*55* percent ~~consistent with the calculation of critical flaw size for NDE examinations~~*per ASME Table NG-3352-1 for a single groove weld that is examined by root and final PT.*

Table 8.1-2 Deleted
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Table 8.1-3

Summary of Impact from Off-Normal Operations

OFF-NORMAL EVENT	DOSE IMPACT AT CONTROLLED AREA BOUNDARY	DETECTION	CAUSES	CORRECTIVE ACTIONS	EFFECTS AND CONSEQUENCES
~~PWR Basket~~*MPC* Off-Normal Handling	None	Observation	Inadvertent motion	None	None
Severe Environment	None	Observation	Weather	None	None
Half Air Inlet Blockage	None	Concrete Cask outlet temperature and Surveillance	Debris	Manual action to clear debris	Concrete Cask outlet temp increase
Particulate Release	Negligible	Radiological Survey	Release of external contamination	None	Negligible radiological release
Fission Gas and Particulate Release	Negligible	Radiological Survey	~~PWR Basket~~*MPC* leakage	Repair ~~PWR Basked~~*MPC* weld ~~or install Basket Overpack~~	Negligible radiological release

Table 8.1-4

Release Fractions

Variable	Normal Operation	Off-Normal Operation	Hypothetical Accident
Fraction of fuel rods that develop cladding breaches, ~~f (b)~~ f_b	0.01	0.1	1.0
Fraction of gases that are released due to fuel cladding breach, ~~f (g)~~ f_g	0.3	0.3	0.3
Fraction of volatiles that are released due to a cladding breach, ~~f (v)~~ f_v	2E-4	2E-4	2E-4
Mass fraction of fuel that is released as fines due to a cladding breach, ~~f (f)~~ f_f	3E-5	3E-5	3E-5
Fraction of crud that spalls off of rods, ~~f (c)~~ f_c	0.15	~~1.0~~ *0.15*	1.0

Table 8.2-1

~~PWR Basket and Basket Overpack~~*MPC* Stresses/*Loads* ~~(ksi)~~ Resulting From Accident Pressurization

~~Component~~		~~Accident Pressure~~	~~Dead Weight~~	~~Thermal~~	~~Normal Handling~~	~~Total~~	~~ASME Allowable~~
~~Shell~~	~~P_m~~ ~~$P_L + P_b$~~	~~2.9~~ ~~41.5~~	~~0.1~~ ~~0.2~~	~~N/A~~	~~0.9~~ ~~1.7~~	~~3.9~~ ~~43.4~~	~~36.4~~ ~~54.6~~
~~Bottom Plate~~	~~P_m~~ ~~$P_L + P_b$~~	~~1.3~~ ~~27.8~~	~~0.0~~ ~~0.2~~	~~N/A~~	~~1.1~~ ~~1.2~~	~~2.4~~ ~~29.2~~	~~36.4~~ ~~54.6~~
~~Structural Lid~~	~~P_m~~ ~~$P_L + P_b$~~	~~0.1~~ ~~8.3~~	~~0.0~~ ~~0.1~~	~~N/A~~	~~0.5~~ ~~0.6~~	~~0.6~~ ~~9.0~~	~~36.4~~ ~~54.6~~
~~Top Weld[a]~~	~~P_m~~ ~~$P_L + P_b$~~	~~5.2~~ ~~7.3~~	~~0.1~~ ~~0.2~~	~~N/A~~	~~0.1~~ ~~1.1~~	~~5.4~~ ~~8.6~~	~~29.1~~ ~~43.7~~
~~Sleeve Assembly~~	~~P_m~~ ~~$P_L + P_b$~~	~~0.0~~ ~~0.0~~	~~0.1~~ ~~0.1~~	~~N/A~~	~~1.3~~ ~~2.0~~	~~1.4~~ ~~2.1~~	~~43.9~~ ~~65.6~~
~~Shield Lid Ring Weld~~	~~P_m~~ ~~$P_L + P_b$~~	~~0.0~~ ~~0.0~~	~~0.1~~ ~~0.2~~	~~N/A~~	~~0.1~~ ~~0.2~~	~~0.2~~ ~~0.4~~	~~27.3~~ ~~41.0~~
~~Shield Lid Weld~~	~~P_m~~ ~~$P_L + P_b$~~	~~4.3~~ ~~5.3~~	~~0.1~~ ~~0.2~~	~~N/A~~	~~0.1~~ ~~0.2~~	~~4.5~~ ~~5.7~~	~~36.4~~ ~~54.6~~
~~Overpack Shell~~	~~P_m~~ ~~$P_L + P_b$~~	~~3.4~~ ~~9.5~~	~~0.1~~ ~~0.2~~	~~N/A~~	~~N/A~~	~~3.5~~ ~~9.7~~	~~43.2~~ ~~64.8~~
~~Overpack Bottom Plate~~	~~P_m~~ ~~$P_L + P_b$~~	~~1.5~~ ~~50.0~~	~~0.0~~ ~~0.2~~	~~N/A~~	~~N/A~~	~~1.5~~ ~~50.2~~	~~43.2~~ ~~64.8~~
~~Overpack Structural Lid~~	~~P_m~~ ~~$P_L + P_b$~~	~~1.5~~ ~~50.0~~	~~0.0~~ ~~0.1~~	~~N/A~~	~~N/A~~	~~1.5~~ ~~50.1~~	~~43.2~~ ~~64.8~~
~~Overpack Top Weld[a]~~	~~P_m~~ ~~$P_L + P_b$~~	~~-~~ ~~1.5~~	~~0.1~~ ~~0.2~~	~~N/A~~	~~N/A~~	~~0.1~~ ~~1.7~~	~~34.6~~ ~~51.8~~

Component	*Maximum Stress/Load*[1]	*ASME Allowable Stress/Capacity*[1]	*Safety Factor*
Shell	8.67	36.15	4.17
Baseplate	30.46	54.23	1.78
Lid	1.99	54.23	27.2
Lid Weld	460.7[2]	1,477[2]	3.20

[1] *Units are ksi unless otherwise noted.*

[2] *Values are in kips.*

Table 8.2-2

Hypothetical Accident Dose Calculations

Dose Results (mrem/yr) ~~(worst sector)~~ (~~325~~ *150* meters)									
	Whole body	Thyroid	Red Bone Marrow	Lung	Bone Surface	Gonad	Breast	~~Remainder~~	Skin
Normal *(34 MPCs)*	~~0.67~~ *0.18*	~~0.14~~ *0.018*	~~0.37~~ *0.18*	~~3.4~~ *0.64*	~~2.7~~ *1.95*	~~0.08~~ *0.034*	~~0.16~~ *0.021*	~~0.40~~	~~0.14~~ *6.6E-4*
Off-Normal *(1 MPC)*	~~0.15~~ *0.037*	~~0.03~~ *7.1E-4*	~~0.10~~ *0.049*	~~0.71~~ *0.09*	~~0.77~~ *0.57*	~~0.02~~ *0.009*	~~0.03~~ *7.8E-4*	~~0.09~~	~~0.0001~~ *7.0E-5*
Direct dose	~~8.5~~ *21.0*	~~8.5~~ *21.0*	~~8.5~~ *21.0*	~~8.5~~ *21.0*	~~8.5~~ *21.0*	~~8.5~~ *21.0*	~~8.5~~ *21.0*	~~8.5~~	~~8.5~~ *21.0*
Total	~~9.3~~ *21.22*	~~8.7~~ *21.02*	~~9.0~~ *21.23*	~~12.6~~ *21.73*	~~12.0~~ *23.52*	~~8.6~~ *21.04*	~~8.7~~ *21.02*	~~9.0~~	~~8.6~~ *21.00*
Dose Results (mrem/30days) (1 sector) (~~325~~ *150* meters)									
Hypothetical accident *(1 MPC)*	~~22.5~~ *1.89*	~~1.2~~ *0.028*	~~26.6~~ *2.55*	~~72.2~~ *5.45*	~~288~~ *29.6*	~~4.74~~ *0.45*	~~1.36~~ *0.030*	~~12.7~~	~~0.05~~ *0.003*

Table 8.2-3

Regulatory Guide 1.76 Design Basis Comparison

Tornado Parameter	Trojan ISFSI Concrete Cask Design	RG 1.76 DBT (Region III) Requirement
Maximum Wind Speed	360 mph	240 mph
Rotational Speed	290 mph	190 mph
Translational Speed	70 mph	50 mph
Pressure Drop	3.0 psi	1.5 psi
Pressure Drop	2.0 psi/sec	0.6 psi/sec

Table 8.2-4

Tornado Missile Probability Data

Tornado Intensity	Missile Strike Probability Per Unit Area/Tornado/Missile	Annual Frequency of Occurrence (EPRI NP-768)(b)	Missile Strike Probability--Transfer Station with Cask
F′1	2.89×10^{-11}(a)	2.3×10^{-5}	4.3×10^{-8}
F′2	3.57×10^{-11}	1.6×10^{-5}	3.8×10^{-8}
F′3	9.06×10^{-11}	5.7×10^{-6}	3.2×10^{-8}
F′4	6.05×10^{-11}	1.9×10^{-6}	7.1×10^{-9}

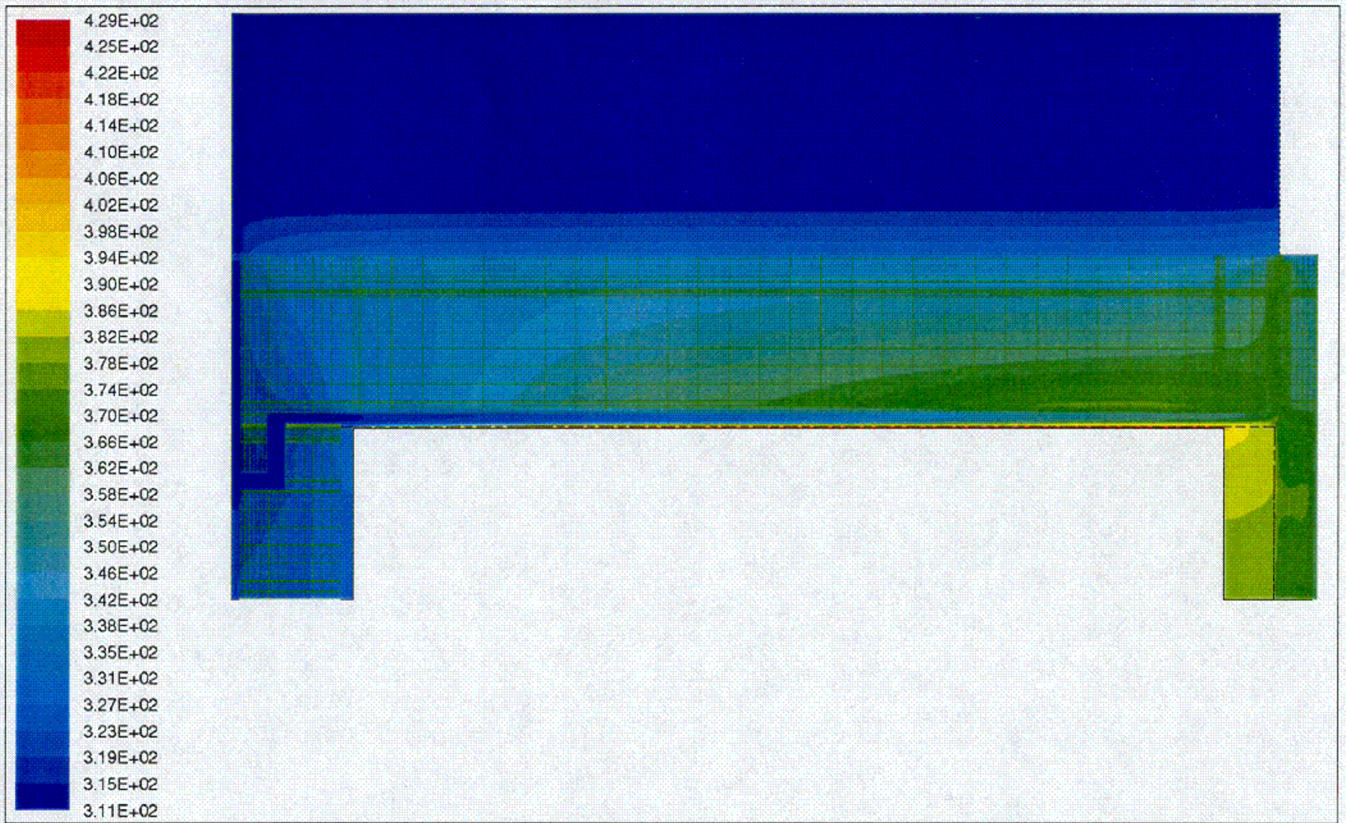
(a) A conservatively extrapolated value, as the EPRI reports assumes an F′1 tornado incapable of missile transport.

(b) Data for Region III as defined in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," 1974

Table 8.3-1

Summary of Site Characteristics Affecting the Safety Analysis

Site Characteristic	Effect on ISFSI Safety Analysis
Severe environmental conditions in summer and winter	Evaluation of steady state Concrete Cask, ~~PWR Basket~~*MPC*, and fuel temperatures for 100°F ambient temperature with 24 hour average solar loads and -40°F ambient temperature with no solar load
Tornadoes	Evaluation of possible Concrete Cask damage including overturning due to wind loading, failure of confinement due to pressure differential, and impact damage due to tornado generated missiles
Earthquakes	Evaluation of seismic motion including possible overturning
Explosion of Chemicals, Flammable Gases, and Munitions	Evaluation of effects on Concrete Cask including potential overturning and sliding
Fires	Evaluation of potential for fire hazard at the ISFSI site
Cooling Tower Collapse	Evaluation of impact of cooling tower collapse on ISFSI
Volcanism	Evaluation of the effects of potential ash, mud and flooding caused by a volcanic eruption
Lightning	Evaluation of the impact of a postulated lightning strike
Natural Gas Turbine Hazards	Evaluation of turbine missiles generated by the gas and steam turbines and fire/heat flux, explosions, and missiles caused by gas line ruptures.



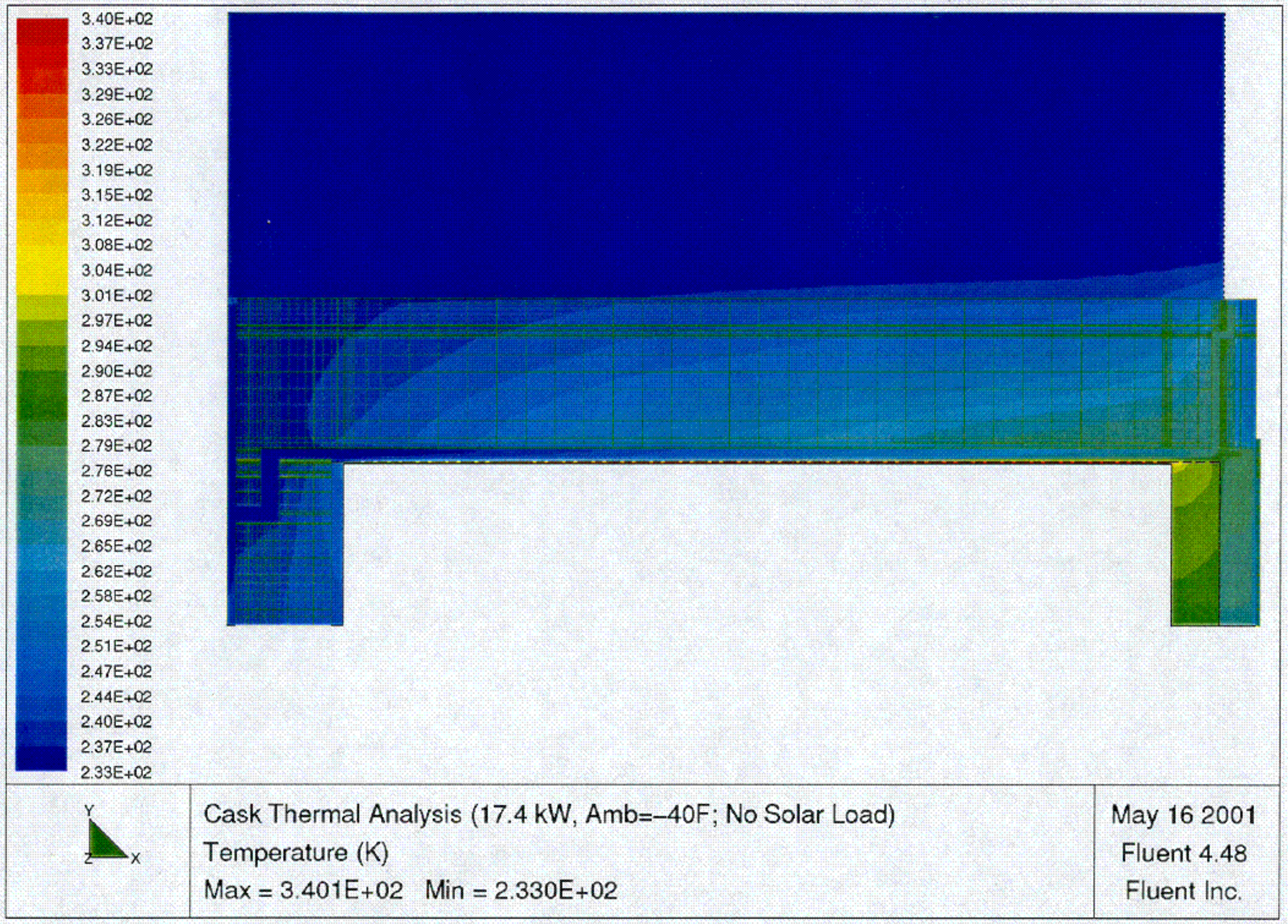
Cask Thermal Analysis (17.4 kW, Amb=100F; Solar Load)
Temperature (K)
Max = 4.294E+02   Min = 3.108E+02

May 16 2001
Fluent 4.48
Fluent Inc.

TROJAN ISFSI
SAFETY ANALYSIS REPORT

FIGURE 8.1-1
CONCRETE CASK TEMPERATURE DISTRIBUTION (100°F DAY)
Revision 2

C01



TROJAN ISFSI

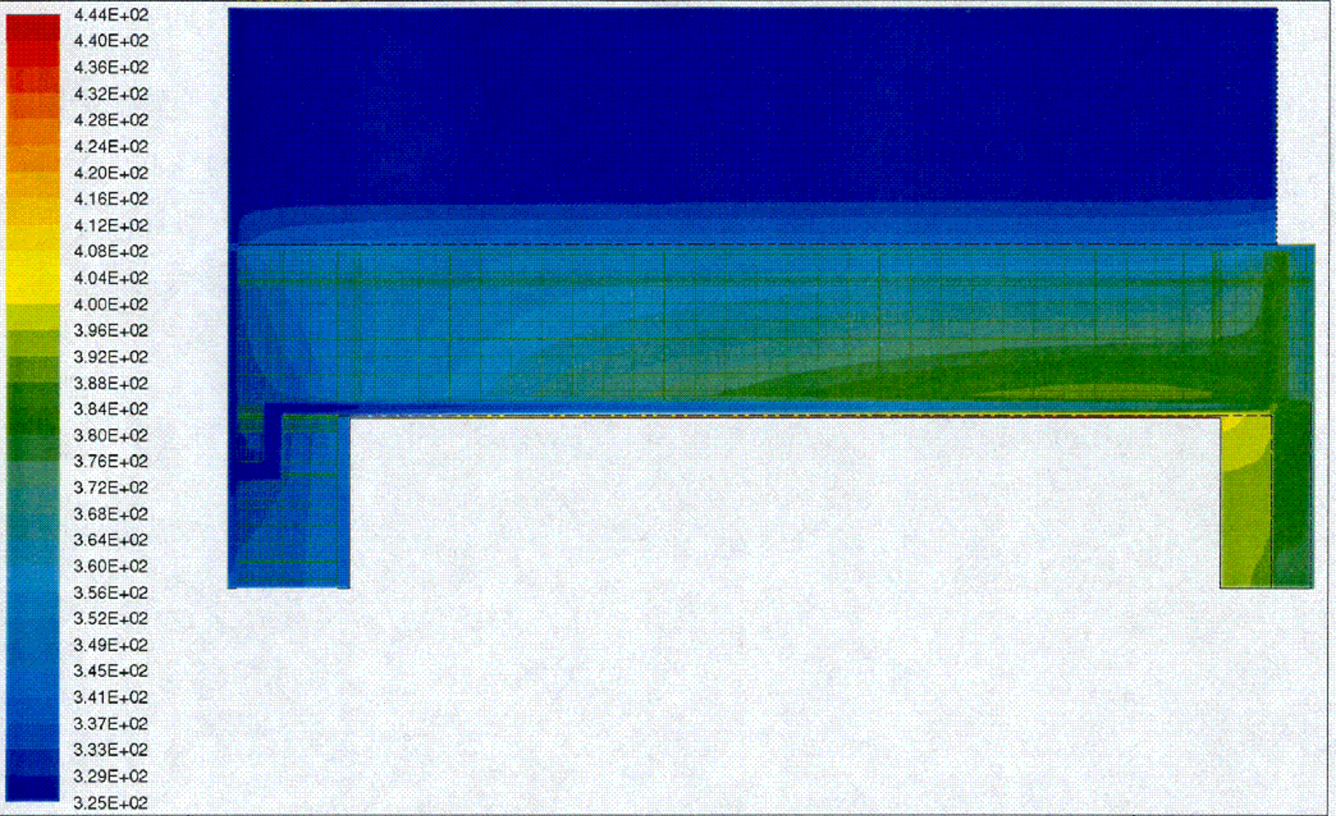
SAFETY ANALYSIS REPORT

FIGURE 8.1-2

CONCRETE CASK TEMPERATURE DISTRIBUTION (-40°F DAY)

Revision 2

C02



Cask Thermal Analysis (17.4 kW, Amb=125F; Solar Load)
Temperature (K)
Max = 4.439E+02 Min = 3.247E+02

May 16 2001
Fluent 4.48
Fluent Inc.

TROJAN ISFSI
SAFETY ANALYSIS REPORT

FIGURE 8.2-1
CONCRETE CASK TEMPERATURE DISTRIBUTION (125°F SHORT TERM)

Revision 2

C03

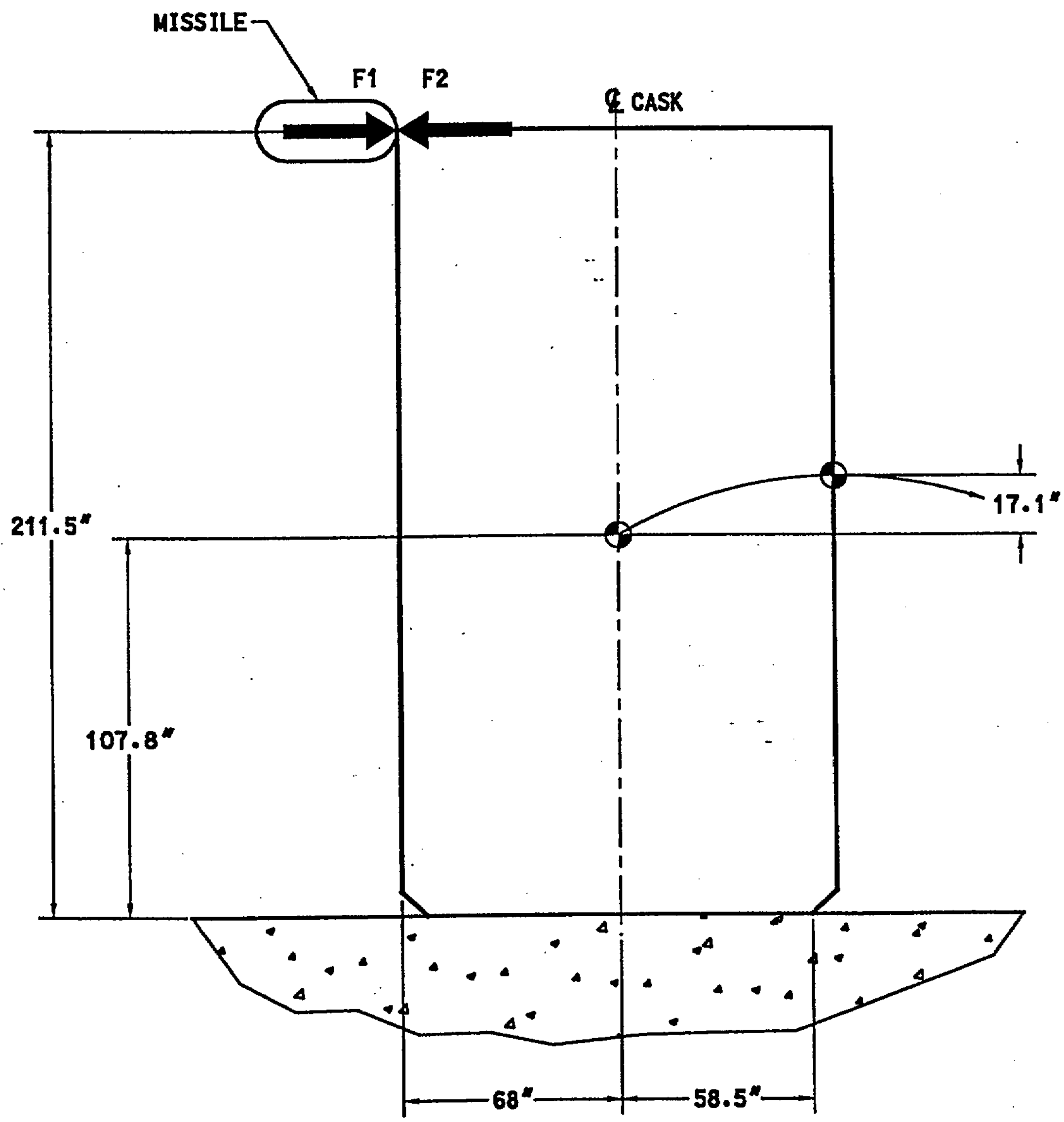


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FIGURE 8.2-2
MISSILE/CASK IMPACT GEOMETRY
REVISION 2

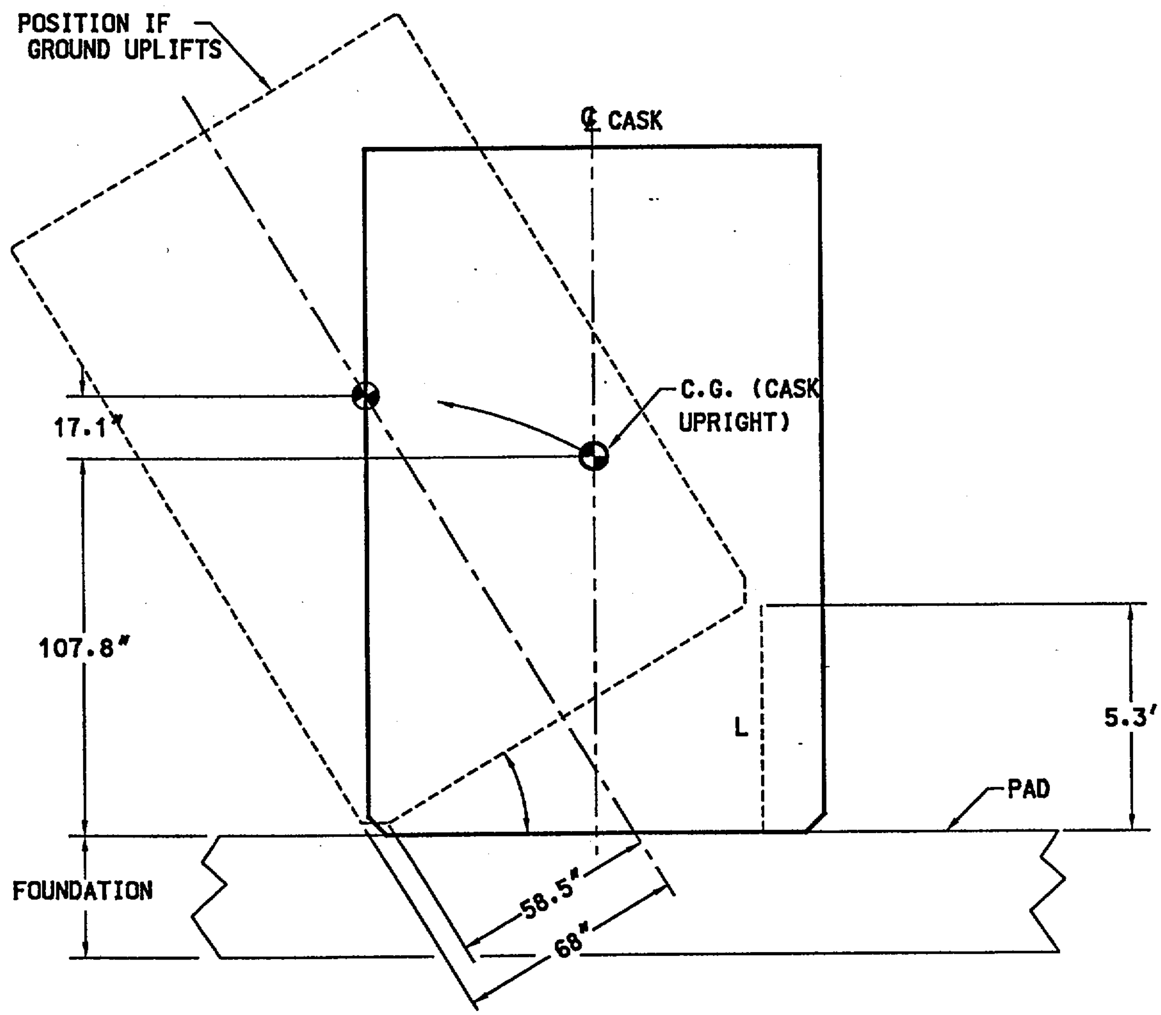

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FIGURE 8.2-3
CASK TIP-OVER GEOMETRY
REVISION 2

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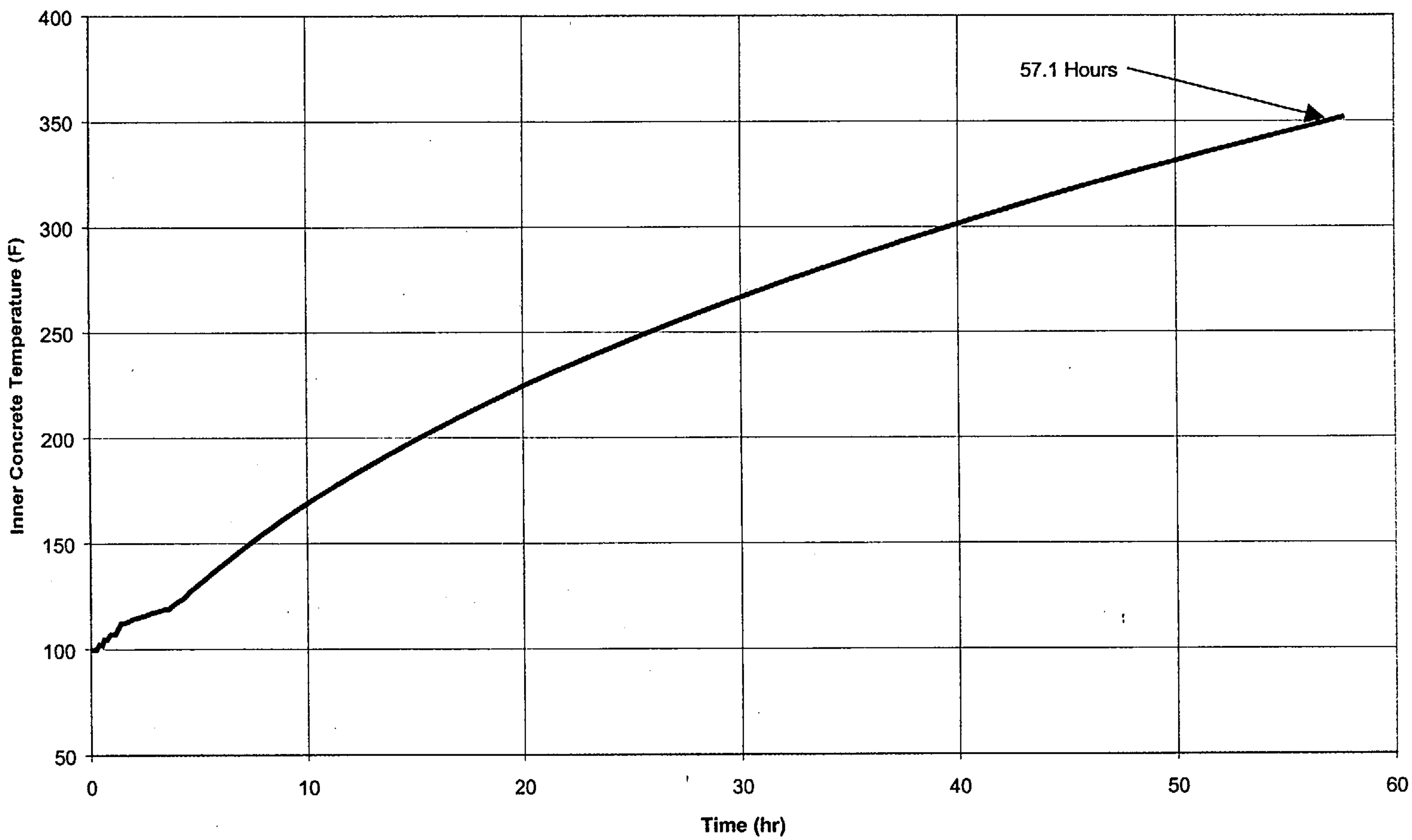

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SAFETY ANALYSIS REPORT

FIGURE 8.2-4

~~MAX.~~ *INNER* CONCRETE TEMPERATURE ALL INLETS BLOCKED

Revision 2

Figure 8.2-5 Deleted

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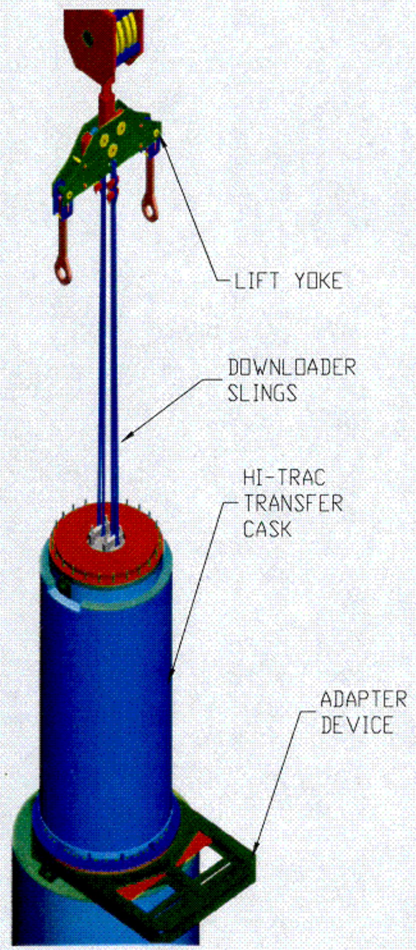
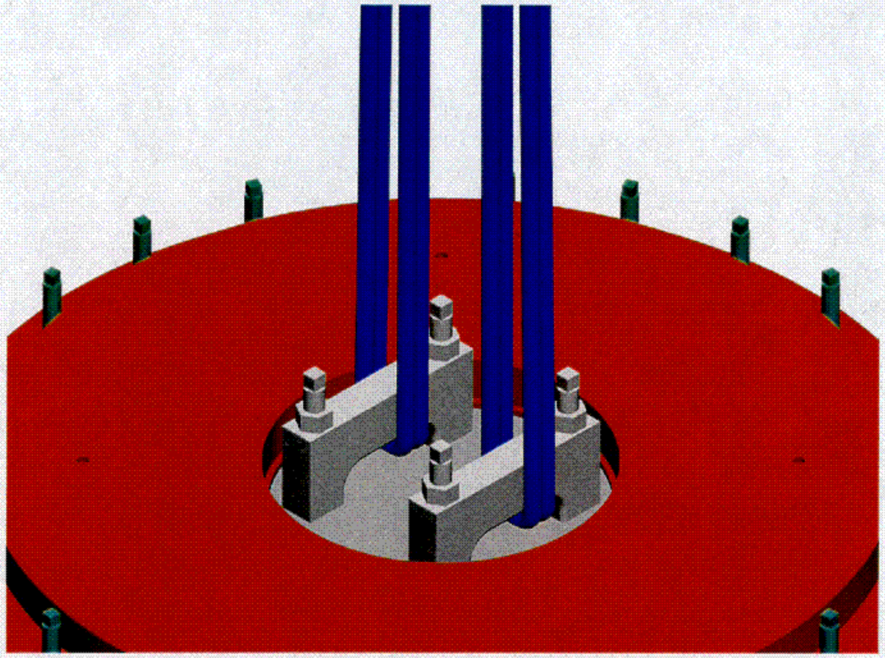




TROJAN ISFSI

SAFETY ANALYSIS REPORT

FIGURE 8.2-6

MPC LIFT CLEATS AND RIGGING

Revision 2

C04

Chapter 9

9.0 CONDUCT OF OPERATIONS

This chapter discusses the organization and procedures established by Portland General Electric (PGE) for the construction, operation, and decommissioning of an Independent Spent Fuel Storage Installation (ISFSI). Included are descriptions of organizational structure, testing, training programs, normal operations, emergency planning, decommissioning, and security.

The Trojan ISFSI is jointly owned by Portland General Electric (PGE), 67.5 percent; the City of Eugene, 30 percent through the Eugene Water and Electric Board (EWEB); and Pacific Power and Light/PacifiCorp (PP&L), 2.5 percent. PGE is the majority owner and has responsibility for operating and maintaining the ISFSI. The Bonneville Power Administration (BPA), a power marketing agency under the United States Department of Energy (DOE), is obligated through net billing agreements to pay costs associated with EWEB's 30% share of TNP operation including decommissioning and spent fuel management costs. The financial capabilities of the joint owners, for construction, operation, and decommissioning of the ISFSI, are presented in ~~the~~*PGE-1061,* "TNP Decommissioning Plan *and License Termination Plan* (PGE-*1078*~~1061~~)."

9.1 ORGANIZATIONAL STRUCTURE

Section 9.1.1 describes the organization that will be in place during ISFSI design, construction, pre-operational testing, fuel loading, startup testing, and initial operation. This organization is shown on Figure 9.1-1 and is referenced in the following sections as the construction and fuel loading organization.

Section 9.1.2 describes the organization that will be in place during long term operation of the ISFSI. This organization is shown on Figure 9.1-2 and is referenced in the following sections as the operation organization. PGE will progressively transition responsibility for ISFSI operation from the construction and fuel loading organization to the operation organization as each Concrete Cask enters the ISFSI controlled access area.

9.1.1 ISFSI CONSTRUCTION AND FUEL LOADING ORGANIZATION

This section describes the organization that will be in place prior to long-term operation of the ISFSI. This organization is responsible for pre-operational activities and loading of the ISFSI.

9.1.1.1 Corporate Organization

The Trojan Site Executive ~~and Plant General Manager~~ is the Corporate Executive with overall responsibility for the design, construction, pre-operational testing, fuel loading, startup testing, and operation of the ISFSI. ~~The Trojan Site Executive and Plant General Manager reports to the Senior Vice President, Power Supply.~~

9.1.1.1.1 Nuclear Oversight

The ~~General~~ Manager, Nuclear Oversight, reports to the Trojan Site Executive ~~and Plant General Manager~~ and is responsible for evaluating the effectiveness of the Quality Assurance (QA) Program, auditing vendor activities, coordinating the Corrective Action Program, providing quality control coverage for site activities, and maintaining the PGE Nuclear Quality Assurance Program (PGE-8010). The Nuclear Oversight Department has the authority and independence to identify quality problems and to initiate stop work orders for any condition adverse to quality.

9.1.1.1.2 Independent Review and Audit Committee

The Independent Review and Audit Committee (IRAC) is responsible for advising the Trojan Site Executive ~~and Plant General Manager~~ on matters relating to the safe storage of spent nuclear fuel. This review and audit function is independent of the organization responsible for operation or maintenance of the ISFSI.

IRAC is composed of a minimum of five regular or alternate members. The Trojan Site Executive ~~and Plant General Manager~~ *designates in writing the Chairman, members, and alternates. The Chairman does not have any direct responsibility for operation or maintenance of the ISFSI. The IRAC collectively has experience and knowledge in spent nuclear fuel handling and storage, chemistry and radiochemistry, engineering, radiation protection, and quality assurance.*

9.1.1.2 Site Organization

9.1.1.2.1 ~~Trojan Site Executive and Plant~~ General Manager, *Trojan*

The ~~Trojan Site Executive and Plant~~ General Manager, *Trojan, reports to the Trojan Site Executive,* is ~~also~~ the senior ~~site~~ *facility* organizational position and is responsible for ~~day-to-day~~ management of the ISFSI ~~and ensuring that the~~ design, construction, pre-operational testing, fuel loading, *and* startup testing~~, and initial operation of the ISFSI~~ *to ensure these functions* are safely ~~conducted~~*accomplished.*

~~9.1.1.2.2Nuclear Oversight~~

~~The General Manager, Nuclear Oversight reports to the Trojan Site Executive and Plant General Manager and is responsible for evaluating the effectiveness of the Quality Assurance (QA) Program, auditing vendor activities, coordinating the Corrective Action Program, providing quality control coverage for site activities, and maintaining the PGE Nuclear Quality Assurance Program (PGE-8010). The Nuclear Oversight Department has the authority and independence to identify quality problems and to initiate stop work orders for any condition adverse to quality.~~

~~9.1.1.2.3~~*9.1.1.2.2* Engineering ~~and Decommissioning~~

The ~~General~~ Manager, Engineering, ~~and Decommissioning~~ reports to the ~~Trojan Site Executive and Plant~~ General Manager, *Trojan,* and is responsible for overseeing the design of structures and systems, preparation of specifications, *and* procurement of materials and equipment~~, and construction of the ISFSI. The General Manager, Engineering and Decommissioning is also responsible for planning and scheduling of activities, design control, reviews of design and construction activities, and preparation of pre-operational and startup test procedures.~~

9.1.1.2.3 Decommissioning Projects

The Manager, Decommissioning Projects, reports to the General Manager, Trojan, and is responsible for planning and scheduling of activities, reviews of design and construction activities, preparation of pre-operational and startup test procedures for the ISFSI, fabrication of ISFSI components, and management of the ISFSI contract.

9.1.1.2.4 Plant Support~~ and Technical Functions~~

The ~~General~~ Manager, Plant Support, ~~and Technical Functions~~ reports to the ~~Trojan Site Executive and Plant~~ General Manager, *Trojan,* and is responsible for licensing activities including reviewing, responding *to*, and interpreting federal and state regulatory documents~~,~~*;* physical security including the administration of the security organization~~,~~ *and* implementation of the site security program~~,~~*; training; and* purchasing~~, and cost control~~.

~~9.1.1.2.5Independent Review and Audit Committee~~

~~The Independent Review and Audit Committee (IRAC) is responsible for advising the Trojan Site Executive and Plant General Manager on matters relating to the safe storage of spent nuclear fuel. This review and audit function is independent of the organization responsible for operation or maintenance of the ISFSI.~~

~~IRAC is composed of a minimum of five regular or alternate members. The Trojan Site Executive and Plant General Manager designates in writing the Chairman, members, and alternates. The Chairman does not have any direct responsibility for operation or maintenance of the ISFSI. The IRAC collectively has experience and knowledge in spent nuclear fuel handling and storage, chemistry and radiochemistry, engineering, radiation protection, and quality assurance.~~

~~9.1.1.2.6~~*9.1.1.2.5* Operations

The Manager, Operations reports to the ~~Trojan Site Executive and Plant~~ General Manager, *Trojan,* and is responsible for performance of pre-operational and startup testing, *ISFSI loading,* and maintaining personnel trained and qualified in accordance with the Certified Fuel Handler Training Program (PGE-1057) for fuel handling operations.

~~9.1.1.2.7~~*9.1.1.2.6* Personnel/Radiation Protection

The Manager, Personnel/Radiation Protection reports to the ~~Trojan Site Executive and Plant~~ General Manager, *Trojan,* and is responsible for emergency preparedness, chemistry, radiation protection, the ALARA program, and industrial safety program.

~~9.1.1.2.8~~*9.1.1.2.7* Maintenance

The Manager, Maintenance reports to the ~~Trojan Site Executive and Plant~~ General Manager, *Trojan,* and is responsible for developing and implementing predictive, preventive, and corrective maintenance for the ISFSI.

9.1.1.3 Interrelationships with Contractors and Suppliers

The development of the ISFSI, including design, construction, testing, and operation are managed by PGE. The contractor for the design, ~~and~~ safety analysis, *and loading* of the ~~ISFSI~~ *Trojan Storage System* is *Holtec International. Holtec has evaluated the* BNFL Fuel Solutions (BFS) *TranStor^{TM} cask analyses and performed new evaluations and analyses, where necessary, to assure that the design functions of the storage cask system will be met within the licensing basis established for the Trojan ISFSI.* ~~PGE reviews and approves BFS quality assurance procedures prior to their implementation, reviews and approves contractor and sub-contractor procedures prior to work at the Trojan ISFSI site, approves sub-tier suppliers performing quality related work prior to use, and inspects quality-related equipment prior to use at the Trojan ISFSI.~~

9.1.1.4 Technical Staff

The design for the ISFSI *Concrete Casks and Failed Fuel Cans was* ~~will be~~ primarily performed by ~~BNFL Fuel Solutions (~~BFS~~)~~. *The design of the MPCs and the Trojan Storage System is performed by Holtec International.* The design, calculations, and analyses will be reviewed and approved by PGE prior to construction. The qualifications of the PGE staff meet or exceed the requirements specified in Section 5.3.1 of the Trojan Permanently Defueled Technical Specifications.

9.1.2 ISFSI OPERATION ORGANIZATION

This section describes the ISFSI organization that will be in place during long term storage of spent nuclear fuel. The ISFSI operation organization is shown in Figure 9.1-2. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organizational charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation.

PGE will progressively transition responsibility for ISFSI operation from the construction and fuel loading organization described in Section 9.1.1 to the operation organization as each Concrete Cask enters the ISFSI controlled access area. The construction and fuel loading organization may remain in place after the transition to assist with ISFSI operation or to perform other non-ISFSI related functions. *Similarly, ISFSI operation organization positions (e.g., ISFSI Manager, ISFSI Safety Review Committee) may be filled prior to the start of ISFSI operations in preparation for personnel assuming their roles.*

9.1.2.1 Corporate Organization

The Corporate Executive responsible for Trojan has overall responsibility for safe operation of the ISFSI.

9.1.2.1.1 Corporate Support

Corporate support will be available by either corporate staff or contract personnel to provide support and expertise to the ISFSI Manager in the following areas: Quality Assurance, Engineering, Radiation Protection, Licensing, Maintenance, Security, Purchasing, and Environmental.

Quality Assurance audits and inspections will be performed by personnel independent of the ISFSI line organization. The results of the audits and recommendations for improvement will be provided directly to the ISFSI Manager and the Corporate Executive.

9.1.2.1.2 ISFSI Safety Review Committee

The ISFSI Safety Review Committee provides independent review of matters related to the safe storage of spent nuclear fuel. The ISFSI Safety Review Committee's responsibilities are described in Section 9.6.

9.1.2.2 Operating Organization

The ISFSI Manager reports to the Corporate Executive and is responsible for safe operation of the ISFSI, maintaining personnel trained and qualified in accordance with the Certified ISFSI Specialist Training Program (PGE-1072), and operation of ISFSI equipment that is important to safety. This position provides direction for the safe operation, maintenance, radiation protection, training and qualification, and security of the ISFSI and personnel.

ISFSI Specialists, who report to the ISFSI Manager, are responsible for day-to-day operation of the ISFSI.

9.1.2.3 Succession of Authority

In order to assure continuity of operation and organizational responsiveness to off-normal situations, a normal order of succession and delegation of authority will be established. The ISFSI Manager will designate in writing personnel who are qualified to act as the ISFSI Manager in his absence.

~~9.1.2.4Corporate Support~~

~~Corporate support will be available by either corporate staff or contract personnel to provide support and expertise to the ISFSI Manager in the following areas: Quality Assurance, Engineering, Radiation Protection, Licensing, Maintenance, Security, Purchasing, and Environmental.~~

~~Quality Assurance audits and inspections will be performed by personnel independent of the ISFSI line organization. The results of the audits and recommendations for improvement will be provided directly to the ISFSI Manager and the Corporate Executive.~~

~~9.1.2.5ISFSI Safety Review Committee~~

~~The ISFSI Safety Review Committee provides independent review of matters related to the safe storage of spent nuclear fuel. The ISFSI Safety Review Committee's responsibilities are described in Section 9.6.~~

9.1.3 PERSONNEL QUALIFICATION REQUIREMENTS

During loading of the ISFSI, each member of the construction and fuel loading organization shall meet or exceed the staff qualifications described in the Trojan Permanently Defueled Technical Specifications, Section 5.3.1. Fuel handling operations will be directly supervised by a Certified Fuel Handler as required by the Trojan Permanently Defueled Technical Specifications, Section 5.2.2.d.

The ISFSI Manager and ISFSI Specialists are qualified as described in Table 9.1-1. Subsequent to initial fuel loading and transfer to the ISFSI, operation of equipment and controls that are identified as important to safety for the ISFSI shall be limited to personnel who are trained and certified in accordance with the Certified ISFSI Specialist Training Program (PGE-1072) or personnel who are under the direct visual supervision of a person who is trained and certified in accordance with the Certified ISFSI Specialist Training Program (PGE-1072).

9.1.4 LIAISON WITH OUTSIDE ORGANIZATIONS

The ISFSI, including design, procurement, construction, pre-operational testing, startup testing, and operation, is managed by PGE. These activities will be performed in accordance with ~~the~~ approved procedures. ~~BNFL Fuel Solutions~~*Holtec International* provides engineering, technical support, and other services for the ISFSI project relating primarily to the design and construction of ~~structures and components~~*the MPCs and related ancillary systems and other services as required.* Other qualified vendors may be selected to provide services and/or equipment.

9.2 PRE-OPERATIONAL AND STARTUP TESTING

Prior to operation of the Trojan ISFSI, *a* preoperational test~~s~~, a startup test, and other tests and inspections will be performed. The preoperational test~~s~~ verif*ies*~~y~~ that the storage system functions as stated in the Safety Analysis Report. The startup test ~~ensures that each storage system operates properly and within the bounds of~~ *confirms that actual dose rates are less than the maximum expected dose rates of Table 7.4-1, such that estimated personnel exposures are bounded by the* applicable safety analyses. Tests and inspections ensure that the storage system and handling equipment satisfy the design criteria stated in Chapter 3.

Several of the tests and inspections of equipment involved with loading the storage system will be performed under the 10 CFR 50 license (e.g., load testing the Fuel Building crane). These tests and inspections are not pre-operational or startup tests of the storage system, but are discussed below due to their importance to the safe loading and operation of the storage system.

Some tests and inspections listed in Table 9.2-1 are performed on each Concrete Cask/~~PWR Basket~~*MPC*. Fabrication of additional ~~Concrete Casks and PWR Baskets~~*MPCs may* ~~will~~ continue well after the beginning of fuel loading operations. The results of such tests and inspections for Concrete Casks and ~~PWR Baskets~~*MPCs* will be available at the site following completion of the testing.

The startup test for each Concrete Cask will not be performed until actual loading of each Concrete Cask has occurred. Results of the startup tests will be available at the site following the completion of each startup test.

9.2.1 ADMINISTRATIVE PROCEDURES FOR CONDUCTING TEST PROGRAM

The development, approval, and performance of pre-operational and startup test procedures will be procedurally controlled. These procedural controls will specify how needed changes to test procedures are incorporated.

The procedure that governs testing will specify how the test results will be evaluated, documented, and approved. Test results will be within the acceptance criteria specified in test procedures.

The procedure that governs testing will specify the process for identifying needed system modifications that are recognized during testing. Also, the procedure will require evaluation of whether retesting is required after a needed modification has been implemented.

PGE or ~~BNFL Fuel Solutions~~ *Holtec International* will, as applicable, be responsible for developing test procedures, performing tests, and ensuring that test acceptance criteria are

satisfied for tests performed at the fabricators *and at the Trojan site*. PGE will review the tests performed by ~~BNFL Fuel Solutions~~ *Holtec International* and the test results for adequacy.

~~For tests performed at the Trojan site, PGE will be responsible for developing the test procedures, performing the tests, and ensuring that the test acceptance criteria are satisfied.~~

9.2.2 TEST PROGRAM DESCRIPTION

The test program is divided into two parts: preoperational testing and startup testing. Other tests and inspections which are not pre-operational or startup tests, are also briefly discussed in this section because of their importance to the proper operation and integrity of the storage system and handling equipment. The preoperational, startup, and other tests are described in this section and a summary is provided in Table 9.2-1.

The storage system uses passive cooling, and therefore has no "operating" systems, other than the air outlet temperature monitoring system, to test prior to the loading of spent nuclear fuel *(i.e., pre-operational testing)*. However, the other tests and inspections described below are performed to ensure the storage system will function in accordance with the design.

Startup testing is performed for each Concrete Cask after loading with spent nuclear fuel. Startup testing confirms that the ~~storage system is operating properly once loaded with spent nuclear~~ *actual dose rates are less than the maximum expected dose rates of Table 7.4-1, such that estimated personnel exposures are bounded* ~~fuel. Startup testing also ensures that the storage system loading is bounded~~ by the safety analyses.

In addition to the tests and inspections described in this section, the ~~PWR Baskets~~ *MPCs*, Transfer Cask, and Concrete Casks will be inspected prior to use to ensure that these components are fabricated in accordance with the design drawings. Materials used specifically for shielding will be tested for ~~density~~ *shielding effectiveness. The material used specifically as neutron poison/shielding in the MPC fuel basket – Boral – will be tested to provide the necessary assurances that it will perform its intended function.* Steel properties will be verified by review ~~and~~ *of* appropriate test reports. Structural adequacy of concrete will be determined by testing during construction.

9.2.3 TEST DISCUSSION

9.2.3.1 Physical Facilities

9.2.3.1.1 ~~PWR Basket~~ *MPC* and Associated Equipment

PGE is constructing ~~a full scale prototype of the PWR Baskets~~ *several MPC weld mock-ups and a full size, full weight dummy MPC.* ~~Construction of the prototype will serve to identify any~~

~~potential difficulties in the fabrication process and will also be used for testing. Although the final PWR Basket design may be modified as a result of the prototype fabrication experience, the prototype basket will be very similar to the final PWR Baskets.~~ The ~~prototype~~ *dummy*~~basket~~ *MPC* will be loaded into the Transfer Cask to verify fit and suitability of the ~~PWR Basket~~*MPC* lift rig*ging*. The ~~basket~~ lifting ~~rigs,~~ slings, ~~rings~~*cleats*, and crane~~(s)~~ used to lift the ~~basket~~ *MPC* will be load tested to demonstrate the ability to safely lift a fully loaded ~~PWR Basket~~*MPC*.

~~A~~ *The MPC weld* mock-up*s* ~~or the prototype~~ will be used to test the automated welding equipment, including actual welding of the lid~~s~~, ~~and valve access~~*the vent and drain* port cover plates, *and closure ring*. Emphasis will be placed on acceptability of the weld*s*, as well as compliance with approved ALARA practices.

The ~~prototype basket or~~*MPC weld* mock-up*s* will also be used to test the cutting equipment, including actual removal of the *closure ring, vent and drain port cover plates, and the MPC lid* ~~valve access lids from the basket~~ after they have been welded in place. This test demonstrates that the ~~basket valve access lids~~*vent and drain port cover plates and MPC lid* can be safely removed.

The ~~PWR Basket~~*MPC* ~~shield~~ lid ~~retainers~~ *retention system* will be tested to demonstrate *the capability to keep the* ~~that they function as designed to retain the shield~~ *MPC* lid *on the MPC when subjected to forces equivalent to those experienced* during and after a crane mishandling event or a *postulated* Transfer Cask tip over.

A*n* ~~PWR Basket~~*MPC* ~~internal assembly~~ will be loaded with a dummy fuel assembly and a Failed Fuel Can to check the fit up and satisfactory operation of associated handling tools and equipment.

The hydrostatic test and dewatering equipment will be tested to ensure that the hydrostatic testing and dewatering can be accomplished in the amount of time necessary to prevent boiling of the borated water in the ~~PWR Basket~~*MPC* as described in Chapter 5. The ~~vacuum~~ drying and helium backfill equipment will be tested to ensure that the ~~vacuum~~ drying and helium backfill can be accomplished as described in Chapter 5.

9.2.3.1.2 Transfer Cask and Associated Equipment

~~Load testing of t~~*T*he Transfer Cask ~~and~~ *lifting* trunnions will be ~~performed~~ *load tested at the fabrication shop* at 300~~%~~ *percent* of design load. The *remainder of the Transfer Cask structural load path and the* Lifting Yoke ~~has been~~*will be* load tested *at the fabrication shop* to ~~300%~~*150 percent* of design load, *demonstrating the structural capability of the Transfer Cask and the Transfer Cask bottom doors. The bottom doors will then be checked for proper operation after supporting the test load.* The crane~~(s)~~ that lift*s* the loaded Transfer Cask will also be load tested. *After a prolonged period out of service* ~~Testing will also be performed prior to~~

~~lifting a loaded PWR Basket if the load test has not been performed within the period of time specified in the test procedure~~ (e.g., ~~for~~ *at the point of* loading a *HI-STAR 100 Transport*~~Shipping~~ Cask several years after commencing ISFSI operation), *the applicable components will be load tested to the original requirements, as necessary, prior to lifting a loaded MPC.*

~~A test load equivalent to the heaviest fully loaded PWR Basket will be placed in the Transfer Cask to demonstrate the structural capability of the Transfer Cask bottom doors. The bottom doors will then be checked for proper operation after supporting the test load.~~

The system used to inject water into the annulus between the ~~PWR Basket~~*MPC* and the Transfer Cask will be tested to ensure that sufficient water is injected to minimize surface contamination of the ~~PWR Basket~~*MPC* external surface.

The load travel path at the site will be checked to ensure that the Transfer Cask can be safely moved from the Fuel Building *crane* bay to the Cask Wash Pit. The Transfer Cask with the ~~PWR Basket prototype or equivalent weight~~*dummy MPC* will be moved from the Cask Wash Pit to the Cask Loading Pit and lowered into the Cask Loading Pit to verify the load travel path and clearances.

The Transfer Cask will be moved from the Cask Loading Pit, along the safe load path, to the west side of the Decontamination and Assembly Station. The Transfer Cask will be moved into the Decontamination and Assembly Station, then moved out of the Decontamination and Assembly Station along the safe load path into the Fuel Building *crane* bay.

9.2.3.1.3 Concrete Cask

The ~~prototype basket~~*dummy MPC* will be placed in the Concrete Cask and the shield ring and lid will be installed to check the fitup of these components. The air outlet temperature monitoring system *components* will be tested and calibrated on each Concrete Cask prior to inserting a loaded ~~PWR Basket~~*MPC.*

9.2.3.1.4 Air Pad System

The air pad system will be tested by moving a ~~test load equivalent to~~*Concrete Cask containing the dummy MPC that weighs at least as much as* a fully *fuel*-loaded Concrete Cask from the Fuel Building to the Storage Pad and from the Storage Pad to the Transfer Station. This test will ensure that the air pad system can safely move the Concrete Cask and will verify the travel path.

~~9.2.3.1.5Overpack~~

~~The Basket Overpack will be tested by placing a Basket Overpack into a Concrete Cask, placing the prototype basket in the Basket Overpack, and simulating closure of the Basket Overpack including installation of the quick connect, quick connect cover, and shield ring.~~

~~9.2.3.1.6~~*9.2.3.1.5* Transfer Station

The Transfer Station will be tested prior to use by placing the Transfer Cask in the Transfer Station and moving the ~~prototype basket~~*dummy MPC* through a transfer sequence in the Transfer Station.

9.2.3.2 Operations

A startup test ~~will be performed for each Concrete Cask after it is loaded with spent nuclear fuel. The startup test~~ will *consist of the* measure*ment of* external radiation dose rates *for each Concrete Cask after it is loaded with spent nuclear fuel* to confirm that the ~~design~~*actual dose rates are less than the maximum expected dose rates of Table 7.4-1* ~~dose rates have been satisfied~~. This will confirm that the ~~PWR Basket loading and~~ estimates of personnel exposures are bounded by the safety analysis.

In addition, ~~the startup test~~*heat transfer validation testing of the Trojan Storage System (i.e., MPC inside the Concrete Cask) will be performed.* ~~will confirm that the heat generated by each spent nuclear fuel Concrete Cask is consistent with expected heat output of the spent nuclear fuel that is loaded in the PWR Basket.~~ *Following the loading of the first Concrete Cask placed in service (expected to be the lowest heat load Concrete Cask),* ~~T~~*t*he heat ~~generation~~ *transfer* will be confirmed by measuring the temperature difference between the Concrete Cask air inlets (ambient air temperature) and air outlets and comparing the *average* measured temperature difference against a calculated temperature difference that is based on the ~~PWR Basket~~*MPC* loading. This test will confirm *the analytic methods and predicted thermal behavior described in the storage system thermal evaluation*~~that the PWR Basket is loaded as designed and does not exceed the heat loading specified in the safety analyses~~. *The validation test will be repeated for the highest heat load Concrete Cask (expected to be the third Concrete Cask loaded).* Measured temperature differences that are higher ~~or lower~~ than the calculated difference by more than the uncertainty specified in the test procedure will be evaluated.

9.2.3.3 Test Response

The tests will be deemed successful if the acceptance criteria provided in the test procedures are achieved safely and without damage to any of the components or associated equipment.

9.2.3.4 Corrective Action

Modifications to equipment or components will be performed, should they become necessary, to ensure that the acceptance criteria are achieved. The modified equipment or components will be retested to confirm that the modification is sufficient. If required, pre-operational test procedure changes will be incorporated into the appropriate operating procedures.

9.3 TRAINING PROGRAMS

The main objective of the training program is to provide ISFSI staff personnel with the specialized training necessary to operate and maintain the ISFSI in a safe manner.

9.3.1 TRAINING PROGRAM DESCRIPTION

Individuals requiring unescorted access to the ISFSI will receive training in the following areas: Radiation Protection, Security, Radiological Emergency Plan, Quality Assurance, Fire Protection, Chemical Safety, *OSHA compliance,* and the Policy statement on worker responsibility for safe operation of the ISFSI. Individuals requiring continued unescorted access will receive refresher training on these topics annually.

Operation of equipment and controls that are identified as important to safety for the ISFSI shall be limited to personnel who are trained and certified in accordance with the Certified ISFSI Specialist Training Program (PGE-1072) or personnel who are under the direct visual supervision of a person who is trained and certified in accordance with the Certified ISFSI Specialist Training Program (PGE-1072).

Fuel Handlers certified in accordance with PGE-1057 (Certified Fuel Handler Training Program) will be ~~retained until initial~~*used for* fuel *handling*~~loading and testing have been completed~~. After fuel and debris are safely transferred to the ISFSI, Concrete Cask handling and transfer is effectively accomplished by Certified ISFSI Specialists who are trained and certified in accordance with PGE-1072, "Certified ISFSI Specialist Training Program."

Individuals who work in or frequent the *ISFSI* Restricted Area will receive radiation protection training commensurate with their responsibilities in accordance with 10 CFR 19, "Notices, Instructions and Reports to Workers: Inspection and Investigations."

Security training will be provided in accordance with the training and qualification requirements outlined in the Trojan ISFSI Security Plan (PGE-1073).

Records will be maintained on the status of trained personnel, training of new employees, and refresher training of present personnel.

9.4 NORMAL OPERATIONS

This section describes the administrative controls and conduct of operations associated with activities considered important to safety. Also described in this section is the management system for maintaining records related to the operation of the ISFSI.

9.4.1 PROCEDURES

Activities affecting quality are accomplished in accordance with approved and documented instructions, procedures, or drawings. Written procedures will be used for ISFSI operations, maintenance, and testing activities that are quality-related as defined in the PGE Nuclear Quality Assurance Program (PGE-8010). ~~In addition to the procedures stated in ISFSI Technical Specification 5.4.1, p~~*P*rocedures will be used to implement the Fire Protection Program, training and certification of ISFSI personnel, fuel/debris classification criteria, loading sequence, and individual ~~PWR Basket~~*MPC*/Concrete Cask inventory. The review and approval process for ISFSI procedures, and changes thereto, will be procedurally controlled. The ISFSI Manager or his designee will approve ISFSI procedures and changes prior to implementation.

Temporary changes to ISFSI procedures are allowed if the intent of the existing procedure is not altered and the change is approved by the ISFSI Manager or his designee.

ISFSI procedures will require that a change to the ISFSI or a change to an ISFSI procedure will be reviewed *for safety impact and* to ensure that the proposed change does not ~~constitute an unreviewed safety question as defined by~~*require prior NRC approval pursuant to* 10 CFR 72.48~~(a)(2)~~. If the Trojan Nuclear Plant possesses a 10 CFR 50 license at the time of the proposed change, then the proposed change will also be reviewed *for safety impact with respect to the Trojan Nuclear Plant, and to ensure that the proposed change does not require prior NRC approval pursuant to* ~~in accordance with~~ 10 CFR 50.59~~to ensure that the proposed change does not represent an unreviewed safety question for the Trojan Nuclear Plant~~.

9.4.2 RECORDS

Administrative procedures will be established and maintained to ensure quality assurance records are identifiable and retrievable. In addition to quality assurance records, the following records will also be maintained in accordance with 10 CFR 72.174:

1. Operating records, including maintenance and modifications.

2. Records of off-normal occurrences.

3. Events associated with radioactive releases.

4. Environmental survey records.

5. Personnel Training and Qualification Records.

6. Records of ISFSI design ~~and procedure~~ changes made pursuant to 10 CFR 72.48.

7. Records showing the receipt, inventory (including location), disposal, acquisition, and transfer of spent fuel and related nuclear material as required by 10 CFR 72.72(a).

8. Records of material control and inventory procedures to account for material in storage as required by 10 CFR 72.72.

Records of ISFSI procedure changes, and tests and experiments, conducted pursuant to 10 CFR 72.48 will be maintained in accordance with 10 CFR 72.48. Storage of the above records will be in accordance with the requirements of the PGE Nuclear Quality Assurance Program (PGE-8010).

Security records, including security training and qualification records, will be maintained in accordance with the Trojan ISFSI Security Plan (PGE-1073).

9.5 EMERGENCY PLANNING

The Trojan ISFSI Emergency Plan (PGE-1075) meets the requirements of 10 CFR 72.32(a) for the ISFSI.

Analysis of the potential radiological impact of off-normal events and postulated accidents associated with the ISFSI, including the beyond design basis accident used to determine the Controlled Area boundary, has been conducted. Based on this analysis, any potential radiological release beyond the ISFSI Controlled Area is not expected to exceed the U. S. Environmental Protection Agency (EPA) Protective Action Guide exposure levels, as detailed in EPA-400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents." For this reason, the actions in the Emergency Plan were designed to safeguard site personnel in the event of an off-normal condition or accident that potentially involves the release of radioactive materials. The actions in the Emergency Plan are based on the reduced likelihood of a radiological emergency and the reduced consequences associated with operation of the ISFSI.

9.6 REVIEWS AND AUDITS

9.6.1 INDEPENDENT SAFETY REVIEWS

Independent Safety Reviews will be performed by qualified Independent Safety Reviewers. These will be thorough reviews performed by persons knowledgeable in the subject area being reviewed. The Independent Safety Reviews will be completed prior to implementation of proposed activities.

The Independent Safety Reviewer will be an individual not having direct involvement in the performance of the activities under review, but who may be from the same functionally cognizant organization as the individuals performing the original work. The Independent Safety Reviewer will have five years of professional level experience and either a Bachelor's Degree in Engineering or the Physical Sciences or equivalent in accordance with ANSI/ANS-3.1-1981. The Chairman of the Safety Review Committee will designate the Independent Safety Reviewers in writing.

The following subjects will be independently reviewed by a qualified Independent Safety Reviewer:

1. ~~Safety~~ Evaluations for changes to the facility as described in the Safety Analysis Report, changes to procedures as described in the Safety Analysis Report, and tests or experiments not described in the Safety Analysis Report to verify that such actions *are safe and* do not ~~involve a change to the ISFSI Technical Specifications or constitute an unreviewed safety question as defined in~~*require prior NRC approval pursuant to* 10 CFR 72.48.

2. Proposed changes to the programs required by ISFSI Technical Specification 5.5 and those programs described in Section 9.7, to verify such changes *are safe and* do not ~~involve a change to ISFSI Technical Specifications and will not constitute an unreviewed safety question as defined in~~*require prior NRC approval pursuant to* 10 CFR 72.48.

9.6.2 ISFSI SAFETY REVIEW COMMITTEE

The ISFSI Safety Review Committee will be responsible for reviewing and advising the Corporate Executive responsible for Trojan on matters relating to the safe storage of spent nuclear fuel. This review function is independent of the line organization responsibilities.

The ISFSI Safety Review Committee will be composed of a minimum of a Chairman and three members. Alternates may be substituted for regular members. The Corporate Executive

responsible for Trojan will designate, in writing, the Chairman, members, and alternates for this committee.

The ISFSI Safety Review Committee will collectively have experience and knowledge in the following functional areas:

1. Spent Nuclear Fuel Handling and Storage
2. Engineering
3. Radiation Protection
4. Quality Assurance
5. Physical Security and Safeguards Information

The ISFSI Safety Review Committee will meet at least once prior to receipt of spent nuclear fuel for storage at the ISFSI and at least once prior to transporting the spent fuel off-site. The Committee will also meet at least once annually and at any time deemed necessary by the Corporate Executive responsible for Trojan. A quorum will consist of three regular members or duly appointed alternates. At least one member of the quorum will be the Chairman or the Chairman's designated alternate.

The ISFSI Safety Review Committee will, as a minimum, perform the following functions:

1. Advise the Corporate Executive responsible for Trojan on matters related to safe storage of spent nuclear fuel.
2. Advise the Manager of the audited organization and the Corporate Executive responsible for Trojan of the result of the audit.
3. Recommend to the Manager of the audited organization any corrective actions that will assist in the correction of the deficiency.
4. Notify the Corporate Executive responsible for Trojan of any safety significant disagreement between the ISFSI Safety Review Committee and the ISFSI Manager within 24 hours.

The ISFSI Safety Review Committee will be responsible for the review of:

1. ~~Safety e~~*E*valuations for procedures and changes thereto, completed under the provisions of 10 CFR 72.48, to verify that such actions *are safe and* do not

~~constitute an unreviewed safety question as defined in~~ *require prior NRC approval pursuant to* 10 CFR 72.48. This review may be completed after implementation of the affected procedure.

2. Changes to structures, systems, or components important to safety to verify that such changes *are safe and* do not ~~constitute an unreviewed safety question as defined in~~ *require prior NRC approval pursuant to* 10 CFR 72.48. The review may be completed after implementation of the change.

3. Tests or experiments involving the safe storage of spent nuclear fuel, which are not described in the Safety Analysis Report, to verify such tests or experiments *are safe and* do not ~~constitute an unreviewed safety question as defined in~~ *require prior NRC approval pursuant to* 10 CFR 72.48. This review may be completed after performance of the test or experiment.

4. Proposed changes to the Trojan ISFSI Technical Specifications or the License.

5. Violations of codes, regulations, orders, license requirements, or internal procedures/instructions ~~which~~ *that* are important to the safe storage of spent nuclear fuel.

6. Indications of unanticipated deficiencies in any aspect of design or operation of structures, systems, or components that could affect safe storage of spent nuclear fuel.

7. Significant accidental, unplanned, or uncontrolled radioactive releases, including corrective action to prevent recurrence.

8. Significant operating abnormalities or deviations from normal and expected performance of equipment that affects safe storage of spent nuclear fuel.

9. The performance of the corrective action system.

10. Internal and external experience information related to the safe storage of spent nuclear fuel that may indicate areas for improving facility safety.

Reports or records of these reviews will be forwarded to the Corporate Executive responsible for Trojan within 30 days following completion of the review.

The audit responsibilities will encompass:

1. The conformance to provisions contained within the Trojan ISFSI Technical Specifications and applicable license conditions.

2. The training and qualifications of facility staff.

3. The implementation of programs required by Technical Specification 5.5 and Safety Analysis Report, Section 9.7.

4. Actions taken to correct deficiencies occurring in equipment or controls important to safety.

5. Facility operations, modifications, maintenance, and surveillance related to equipment or controls important to safety to verify that these activities are performed in a safe manner.

6. Other activities and documents as requested by the Corporate Executive responsible for Trojan.

Reports or records of these audits, including any recommendations, will be forwarded to the Corporate Executive responsible for Trojan within 30 days following completion of the audit.

9.6.3 RECORDS

Written records of reviews and audits will be maintained. As a minimum these records will include:

1. Results of the activities conducted under the provisions of Sections 9.6.1 and 9.6.2.

2. Recommendations to the Manager of the organization being audited.

3. An assessment of the safety significance of the review or audit findings.

4. Documentation of the reviews conducted under Section 9.6.1.

5. Determination of whether each *of the first three* item*s of ISFSI Safety Review Committee review responsibility listed in*~~considered under~~ Section 9.6.2 *above* ~~for potentially constituting an unreviewed safety question (first three areas of ISFSI Safety Review Committee review responsibility listed in Section 9.6.2 above) constitutes an unreviewed safety question as defined in~~ *is safe and requires prior NRC approval pursuant to* 10 CFR 72.48.

9.7 PROGRAMS

The following programs will be established, implemented, and maintained in addition to the programs required by ISFSI Technical Specifications, Section 5.5.

9.7.1 THIS SECTION DELETED.

9.7.2 FIRE PROTECTION PROGRAM

The Fire Protection Program provides controls to prevent and protect the facility from fires and explosions which could impact the safe storage of spent nuclear fuel or cause the release of radioactive material.

9.7.3 TRAINING PROGRAM

The Training Program contains the training and certification requirements for the ISFSI Specialists.

9.7.4 TROJAN ISFSI EMERGENCY PLAN

The Trojan ISFSI Emergency Plan contains actions and responsibilities to be performed in response to a radiological emergency. This plan complies with the requirements of 10 CFR 72.32(a).

9.7.5 TROJAN ISFSI SECURITY PLAN

The Trojan ISFSI Security Plan (PGE-1073) contains a detailed plan for security measures for physical protection of the ISFSI. In addition, this plan contains contingencies for responding to threats and potential radiological sabotage. This plan complies with the requirements of 10 CFR 72, Subpart H, "Physical Protection."

9.7.6 TRANSFER CASK AND CONCRETE CASK HANDLING AND STORAGE PROGRAM

The Transfer Cask and Concrete Cask Handling and Storage Program places controls on the handling and storage of the Concrete Cask, Transfer Cask, and ~~PWR Basket~~*MPC*. This program will consist of elements that will: limit the heights at which the~~se~~ *loaded* Concrete Casks can be lifted, use of the Transfer Cask, and arrangement of Concrete Casks on the ISFSI Storage Pad. The program also provides requirements for use of a mobile crane for lifting ~~PWR Baskets~~*MPCs* in the Transfer Station and provides controls for movement and loading of ~~Shipping~~ *Transport* Casks.

9.7.7 STRUCTURAL INSPECTION PROGRAM

The Structural Inspection Program establishes periodic inspection of the concrete surface of the Concrete Casks and the Storage Pad. The inspections will ensure that the structural integrity of the concrete is maintained. The structural inspection program establishes periodic inspection of the Transfer Station foundation, anchorage and structural steel and fasteners. The program provides acceptance criteria, evaluation methods for degradation, and repair and restoration instructions.

9.7.8 CONCRETE CASK INTERIOR INSPECTION PROGRAM

The Concrete Cask Interior Inspection Program establishes periodic inspections of the first Concrete Cask placed in service, at the Trojan ISFSI, at five year intervals. The Concrete Cask interior annulus area and the interior areas of vents shall be inspected to identify degradation mechanisms (not identified in the Safety Analysis Report) that may affect system performance.

The Concrete Cask Interior Inspection Program requires the results of the inspections to be documented and a report summarizing the findings to be submitted to the NRC within 30 days of the inspections. The report shall be submitted in accordance with 10 CFR 72.4 with a copy sent to the appropriate NRC regional office.

9.8 ISFSI DECOMMISSIONING PLAN

This section describes the plans for decommissioning the ISFSI. Included are discussions of the method of decommissioning, anticipated costs, design and operational features that facilitate decommissioning, and record keeping utilized during the life of the ISFSI. Details pertaining to the financial assurance for ISFSI decommissioning (10 CFR 72.30) are provided in ~~the Trojan Nuclear Plant~~ *PGE-1061, "TNP* Decommissioning Plan *and License Termination Plan* (PGE-*1078*~~1061~~)."

9.8.1 DECOMMISSIONING PROGRAM

Decommissioning of the ISFSI primarily consists of transferring the spent nuclear fuel contained in the sealed ~~storage PWR Baskets~~*MPCs* to a facility for final disposal or storage. The spent nuclear fuel that will be stored at the ISFSI is not eligible for near surface disposal in accordance with 10 CFR 61. The DOE is responsible for the acceptance of spent nuclear fuel and related nuclear material in accordance with the terms of the 1982 Nuclear Waste Policy Act.

After the spent nuclear fuel is transferred to the DOE for disposal or storage, contamination and radiation surveys will be performed to determine if the ISFSI is contaminated or if ISFSI components are activated. If contamination is detected, then decontamination can be accomplished by routine radiation protection practices. The resultant radioactive waste would be packaged and shipped off site as radioactive waste. If the ISFSI components are activated, then the components would be packaged and shipped off site as radioactive waste.

9.8.2 COST OF DECOMMISSIONING

The cost for decommissioning the ISFSI is estimated at approximately $7.9 million (1997 dollars). A breakdown of cost estimates based on activities is provided in Table 9.8-1. Further details of the ISFSI decommissioning costs are contained in ~~the Trojan Nuclear Plant~~*PGE-1061, "TNP* Decommissioning Plan *and License Termination Plan* (PGE-*1078*~~1061~~)."

9.8.3 DECOMMISSIONING FACILITATION

The ISFSI was designed to minimize the decontamination efforts required for decommissioning. The design of the ~~PWR Basket~~*MPC* and the operational process for handling the ~~PWR Basket~~*MPC* ensure that the radioactive materials are contained within the sealed ~~PWR Basket~~*MPC,* which minimizes the potential for contamination of the ISFSI components and structures. The procedures developed for the transfer of the ~~PWR Baskets~~*MPCs* from the Concrete Casks to the ~~t~~*T*ransport~~ation~~ ~~system~~*Casks* will need to include similar requirements for monitoring and controlling contamination during ISFSI decommissioning.

9.8.4 RECORD KEEPING FOR DECOMMISSIONING

Records of information important to the safe and effective decommissioning of the ISFSI will be maintained for the life of the ISFSI. The types of information that will be maintained as records for decommissioning are listed in 10 CFR 72.30(d).

TABLE 9.1-1

ISFSI STAFFING QUALIFICATIONS
Operation Organization

1. ISFSI Manager:

The ISFSI Manager, at the time of appointment to the position, shall have a minimum of eight years of power plant experience, of which a minimum of three years shall be nuclear power plant experience. A maximum of two years of the remaining five years of power plant experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one basis. The ISFSI Manager will be trained and certified in accordance with the Trojan Certified ISFSI Specialist Training Program (PGE-1072).

In addition to the above specified requirements, the ISFSI Manager will also be required to be qualified as an Independent Safety Reviewer (ISR). The qualifications for an ISR are provided in Section 9.6.1.

2. ISFSI Specialists:

The ISFSI ~~s~~Specialists, at the time of appointment to the position, shall have a High School diploma or successfully completed the General Education Development (GED) test and two years of power plant experience of which a minimum of one year shall be nuclear power plant experience. Consistent with the assigned duties, ISFSI Specialists will be trained and certified in accordance with the Trojan Certified ISFSI Specialist Training Program (PGE-1072) and the Trojan ISFSI Security Plan (PGE-1073) training and qualification requirements.

Table 9.2-1

Pre-Operational, Startup, and Other Tests

Component	Type	Test Purpose/Objective(s)
~~PWR Basket~~*MPC* lifting equipment (attaches to ~~structural~~ *MPC* lid)	Other	1. Check fit up with ~~structural~~ *MPC* lid and lifting cranes. 2. Load test demonstrates ability to safely lift a fully loaded ~~PWR Basket~~*MPC*.
~~PWR Basket~~*MPC* automated welding system and cutting equipment	Other	1. Check fit up of ~~shield lid, structural~~*MPC* lid, ~~quick connect/ball~~ *remote* valve~~s~~ *operating assemblies (RVOAs)*, *closure ring,* and *vent and drain* ~~valve access~~ port cover plates. 2. Demonstrate ability to install the *MPC* lid~~s~~, ~~and valve access~~*vent and drain* port cover plates, *and closure ring.* 3. Demonstrate ability to remove ~~valve access~~*vent and drain* port cover plates, *closure ring, and MPC lid.*
~~PWR Basket shield~~*MPC* lid ~~retainers~~*retention system*	Other	1. Check fit up of ~~retainers~~ *the lid retention system* with ~~shield~~ *the MPC* lid. 2. Demonstrate ability to keep the ~~shield~~ *MPC* lid on the ~~PWR Basket~~*MPC when subjected to forces equivalent to those experienced* during/after mishandling event or *postulated* Transfer Cask tip over.
~~PWR Basket internal assembly~~*MPC basket*	Other	~~1. Check fit up with PWR Basket.~~ ~~2~~*1*. Load dummy fuel assembly into ~~PWR Basket~~*MPC* ~~internal assembly~~*basket.* *2. Load Failed Fuel Can into MPC basket corner cell.*
~~PWR Basket~~*MPC* hydrostatic test, dewatering, ~~vacuum drying~~*moisture removal*, and helium backfill systems	Other	1. Check fit up with ~~PWR Basket quick connects on ball valves~~*MPC RVOAs*. 2. Demonstrate ability to pressurize~~/evacuate~~ ~~PWR Basket~~*MPC* to required test pressure~~/vacuum~~. 3. Demonstrate ability to dewater and ~~evacuate~~ *complete moisture removal from MPC*~~PWR Basket~~ in the time required to prevent boiling. 4. Demonstrate ability to ~~vacuum~~ dry and backfill the ~~PWR Basket~~*MPC* with helium.

Table 9.2-1

Pre-Operational, Startup, and Other Tests

Transfer Cask lifting crane(s)	Other	Load test demonstrates ability to safely lift a fully loaded Transfer Cask.
Transfer Cask ~~and~~ *lifting* trunnions	Other	300% load test *to* demonstrate~~s~~ ability to safely lift a loaded Transfer Cask.
Lifting Yoke	Other	1. Check fit up with Transfer Cask and crane. 2. ~~300~~*150*% load test *to* demonstrate~~d~~ ability to safely lift a loaded Transfer Cask.
Transfer Cask bottom doors	Other	Demonstrate proper operation of bottom doors after supporting the weight equivalent to ~~a fully loaded PWR Basket~~*150% of design load.*
Transfer Cask annulus water injection system	Other	Demonstrate the ability to inject sufficient water into the ~~PWR Basket~~*MPC*/Transfer Cask annulus to minimize contamination of ~~PWR Basket~~*MPC* external surfaces.
Concrete Cask air pads	Other	Demonstrate ability to lift the weight *at least* equivalent ~~of~~ *to* a fully loaded Concrete Cask.
Concrete Cask air outlet temperature monitoring system *components*	Pre-op	Demonstrate proper operation of the temperature monitoring system *components* prior to placing a loaded ~~PWR Basket~~*MPC* into the Concrete Cask.
Concrete Cask shield ring and Concrete Cask lid	Other	Check fit up.
~~Overpack automated welding system and cutting equipment~~	~~Other~~	~~Check fit up of Basket Overpack, structural lid, quick connect, and quick connect cover and demonstrate the ability to install and remove the lid and quick connect cover.~~

Table 9.2-1

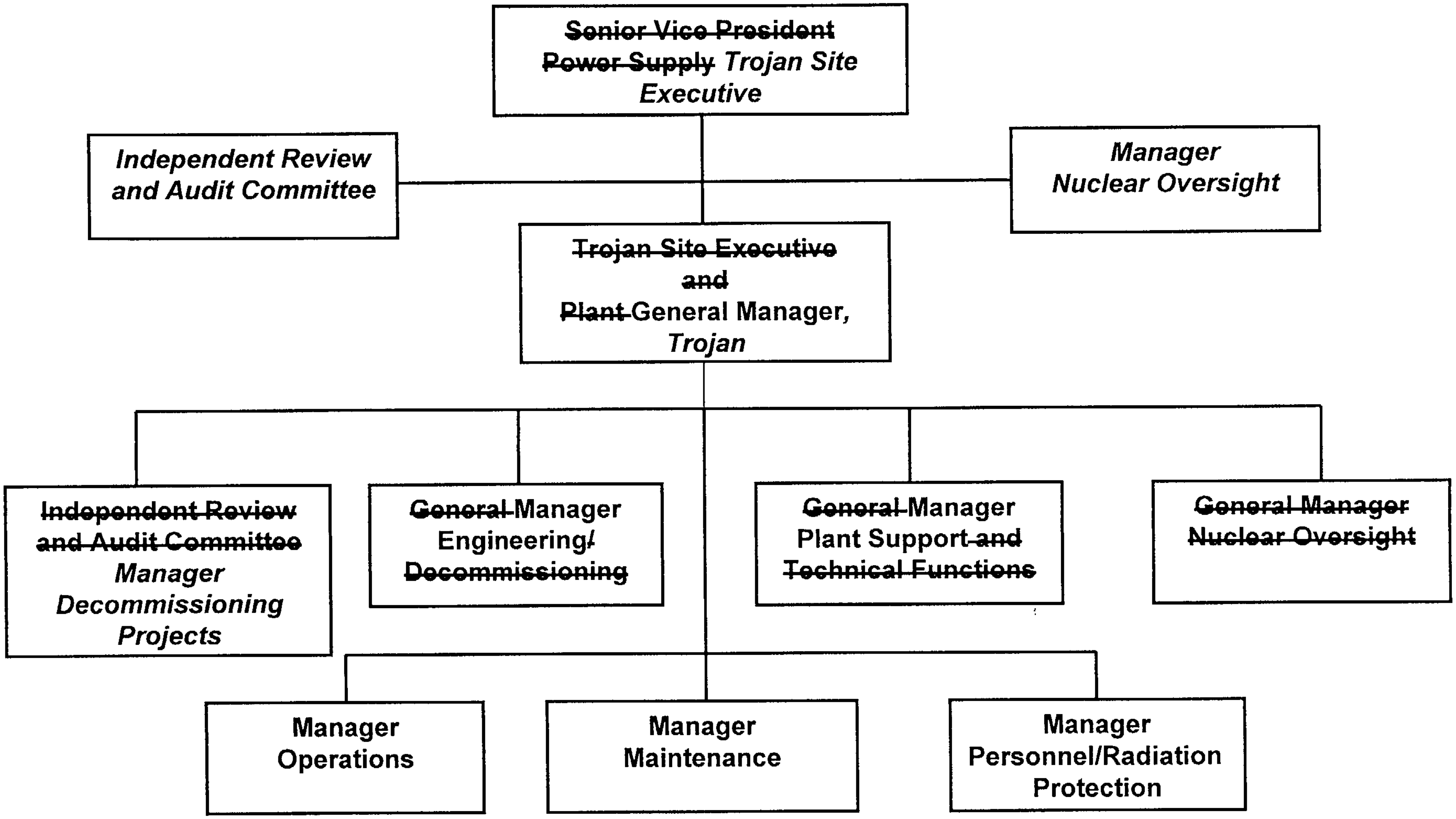
Pre-Operational, Startup, and Other Tests

~~Overpack shield ring~~	~~Other~~	~~Check fit up.~~
Transfer Station and side members	~~Pre-op~~*Other*	1. Check fit up of Transfer Station components. 2. Demonstrate ability to move Transfer Cask through a transfer sequence.
Storage system performance	Startup	~~1.~~ Measure external radiation dose rates *for each Concrete Cask* to confirm *actual dose rates are less than maximum expected dose rates of Table 7.4-1, such that* estimated personnel exposures *are bounded by the safety analysis.* ~~2. Measure fuel decay heat to confirm proper loading of PWR Basket and proper heat removal by storage system.~~
Heat transfer validation testing	*Other*	*For the lowest heat load (expect to be the first cask loaded) and highest heat load Concrete Cask (expected to be the third cask loaded), confirm the analytic methods and predicted thermal behavior described in the storage system thermal evaluation.*
Component compatibility/load travel path	Other	1. Check fit up of components with each other. 2. Check load path from Spent Fuel Pool to pad. - *Transfer Cask from Fuel Building crane bay to Cask Wash Pit.* - ~~Basket~~ *MPC* into Transfer Cask. - Transfer Cask into Cask Loading Pit. - Transfer Cask moved from Cask Loading Pit to ~~Decon and Assembly Station~~*DAS*. - Transfer Cask moved from ~~Decon and Assembly Station~~*DAS* to Fuel *Building crane* ~~Loading B~~*b*ay and placed on top of Concrete Cask. - ~~PWR Basket~~*MPC* lowered from Transfer Cask into Concrete Cask. - Concrete Cask shield ring and lid installed. - Concrete Cask moved from Fuel *Building crane* ~~Loading B~~*b*ay to Storage Pad. ~~- Basket Overpack placed in Concrete Cask.~~ ~~- Basket transferred into a Concrete Cask containing an Basket Overpack.~~

TABLE 9.8-1
ISFSI DECOMMISSIONING COSTS

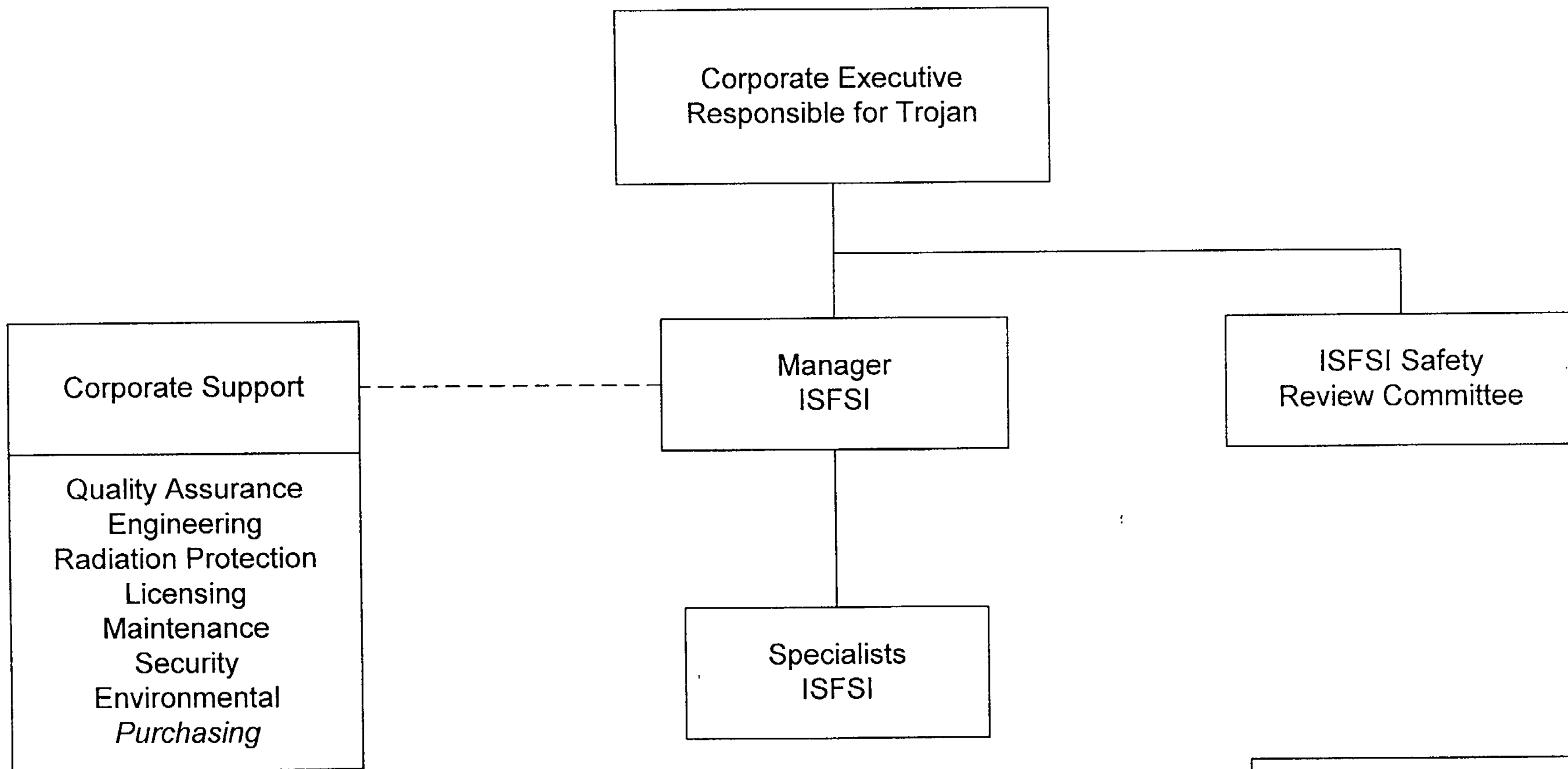
ACTIVITY	ESTIMATED COST (thousands of 1997 dollars)
Demolition of ISFSI	417
Transfer Spent Nuclear Fuel and Miscellaneous Costs	3,013
Professional Services	750
Burial Cost, Low Level Waste[1]	3,673
Total Decommissioning Cost	7,853

1 Separate burial of the Concrete Casks as Low Level Radioactive Waste.



TROJAN ISFSI
SAFETY ANALYSIS REPORT

FIGURE 9.1-1
ISFSI ORGANIZATION
CONSTRUCTION AND FUEL LOADING
Revision 2



TROJAN ISFSI
SAFETY ANALYSIS REPORT

FIGURE 9.1-2
ISFSI ORGANIZATION OPERATION

Revision 2

Chapter 10

10.0 OPERATING CONTROLS AND LIMITS

The ~~Concrete Cask~~*Trojan Storage* ~~s~~*S*ystem is passive during the storage mode and requires few operating controls. Design criteria and functional descriptions of safety features are contained in Chapter 3 (Principal Design Criteria) and Chapter 4 (Installation Design). Required functional and operating limits, limiting conditions for operations, surveillance requirements and administrative controls are contained within the ISFSI Technical Specifications.

As described in Chapter 5 (Operations), fuel loading and ~~PWR Basket~~*MPC* sealing operations will be performed within the Fuel Building of the Trojan Nuclear Plant (TNP). These activities will be performed in accordance with the requirements of the TNP 10 CFR 50 license and TNP procedures developed pursuant to the requirements of that license. However, certain restrictions related to fuel loading and ~~PWR Basket~~*MPC* sealing operations are also included in the ISFSI Technical Specifications. These Technical Specifications include restrictions and requirements to ensure the initial spent fuel packaging conditions are consistent with the ISFSI design requirements for safe long-term storage of spent nuclear fuel.

The Trojan ISFSI Technical Specifications have been developed consistent with the format of improved standard technical specifications contained in NUREG-~~1431~~-*1745*, "Standard *Format and Content for* Technical Specifications *for 10 CFR Part 72 Cask Certificates of Compliance*~~, Westinghouse Plants~~."

Chapter 11

11.0 QUALITY ASSURANCE

Portland General Electric Company implements a Nuclear Quality Assurance (QA) Program which directs quality-related activities at the Trojan Nuclear Plant. This QA Program is described in PGE-8010, "Trojan Nuclear Plant Nuclear Quality Assurance Program."

PGE-8010 complies with Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

In addition to 10CFR50 activities, PGE-8010 applies to activities covered by 10 CFR 71, Subpart H, "Quality Assurance for Packaging and Transportation of Radioactive Material," and 10 CFR 72, Subpart G, "Quality Assurance for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

The BNFL Fuel Solutions Quality Assurance Program, *which is applicable to the Concrete Casks and Failed Fuel Cans,* is discussed in Section 13.2 of the Safety Analysis Report (SAR) for the Ventilated Storage Cask (VSC) System by Pacific Sierra Nuclear Associates and Sierra Nuclear Corporation, dated October 1991. The Sierra Nuclear Corporation SAR is addressed in the Certificate of Compliance issued by the NRC effective May 7, 1993, on Docket Number 72-1007. The program was *originally* approved by the NRC as noted in Section*s* 10.0 and 14.1.3 of the NRC "Safety Evaluation Report for the Pacific Sierra Nuclear Associates Safety Analysis Report for the Ventilated Storage Cask System," dated April 1993. *Subsequent to the original NRC approval, the BNFL Fuel Solutions Quality Assurance Program was revised, when necessary. As of September 2001, the current effective revision of this Quality Assurance program was Revision 8, dated April 28, 2000, and the associated NRC approval was documented in NRC letter dated August 1, 2000, and titled, "Quality Assurance Program Approval for Radioactive Material Packages No. 0804."*

The Holtec International Quality Assurance program, which is applicable to the MPCs and related support systems, is discussed in Section 13.3 of the HI-STORM 100 System Final Safety Analysis Report (Holtec International Report No. HI-2002444). The HI-STORM FSAR is addressed in HI-STORM 100 System Certificate of Compliance Number 72-1014, effective May 31, 2000. This QA program description, as augmented by the Holtec QA program manual, was originally approved by the NRC as meeting the requirements of 10 CFR 72, Subpart G, as described in Section 13.0 of the NRC Safety Evaluation Report for the HI-STORM 100 Cask System, dated May 4, 2000. Revision 11 of the Holtec International QA program was also approved for use under 10 CFR Part 71 as documented in NRC Quality Assurance Program Approval for Radioactive Material Packages No. 0784, Revision 2, dated August 23, 1999. Changes to the Holtec International Quality Assurance program will be made in accordance with approved Holtec procedures and NRC regulations.

Appendix A

Security-Related Information Figure
Withheld Under 10 CFR 2.390.

TITLE		
MULTI-PURPOSE CANISTER ASSEMBLY		
DRAWING NO.	SHEET	REVISION
PGE-001	1/1	2

Security-Related Information Figure Withheld Under 10 CFR 2.390.

TITLE		
CONCRETE CASK ASSEMBLY		
DRAWING NO.	SHEET	REVISION
PGE-002	1/1	2

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Security-Related Information Figure Withheld Under 10 CFR 2.390.

TITLE

TRANSFER CASK
ASSEMBLY

DRAWING NO.	SHEET	REVISION
PGE-004	1/1	2

Security-Related Information Figure Withheld Under 10 CFR 2.390.

TITLE		
TRANSFER CASK LIFTING YOKE		
DRAWING NO.	SHEET	REVISION
PGE-005	1/1	2

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Figure PGE-006 Deleted

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