TECHNICAL SPECIFICATIONS BASES

FOR

TROJAN

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)

March 1999

B 2.0 ~~FUNCTIONAL AND OPERATING LIMITS~~ *APPROVED CONTENTS*

B2.1.1 *and B2.1.2* Fuel Stored at the ISFSI *and Fuel Storage Configuration Limits*

BASES

BACKGROUND

The design of the *MPC* ~~PWR BASKET, BASKET OVERPACK,~~ and CONCRETE CASK is based on specifications of spent fuel and fuel related material that will be stored. These specifications include type and quantity of fuel and fuel inserts, condition of spent fuel, maximum initial enrichment, maximum burnup, and minimum cooling time prior to storage.

These specifications for spent fuel and fuel related material are included in the thermal, structural, radiological, ~~and~~ criticality*, and materials* evaluations performed for the ~~TranStor™ Storage System~~ *TROJAN STORAGE SYSTEM* components (e.g., the CONCRETE CASK~~, PWR BASKET~~ and *MPC* ~~BASKET OVERPACK~~) described in the ISFSI SAR (Ref. 1).

The design of the ~~TranStor™ Storage System~~ *TROJAN STORAGE SYSTEM* is such that the ~~PWR BASKET~~ *MPC* is placed in the TRANSFER CASK and loaded with INTACT FUEL ASSEMBLIES and FAILED FUEL CANS in the spent fuel pool (i.e., Cask Loading Pit). Administrative controls are used to ensure each ~~PWR BASKET~~ *MPC* is loaded with material meeting the specifications provided in Tables 2-1 and 2-2. Prior to removing the TRANSFER CASK and ~~PWR BASKET~~ *MPC* from the spent fuel pool (i.e., Cask Loading Pit), the loading of the ~~PWR BASKET~~ *MPC* is verified.

After loading and placing the ~~shield~~ *MPC* lid on *the MPC* ~~it~~, the TRANSFER CASK containing the ~~PWR BASKET~~ *MPC* is removed from the spent fuel pool (i.e., Cask Loading Pit) and transferred to the cask preparation area where closure, ~~vacuum~~ *MPC cavity* drying and helium backfilling are performed on the ~~PWR BASKET~~ *MPC*. After installation of the ~~structural lid~~ *MPC vent and drain port cover plates and closure ring*, the ~~PWR BASKET~~ *MPC* is transferred from the TRANSFER CASK to a CONCRETE CASK in preparation for transportation to and storage at the ISFSI.

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After loading, contact surface dose rates and surface contamination of the CONCRETE CASK are measured in accordance with the Radiation Protection Program of Administrative Controls Section 5.5.4. ~~The surface contamination levels must meet the limits of LCO 3.3.1.~~ In addition, the CONCRETE CASK is monitored in accordance with the Thermal Monitoring Program of Administrative Controls Section 5.5.3 to verify that the thermal performance is in accordance with design limits.

APPLICABLE SAFETY ANALYSES

Misloading ~~of~~ a*n* ~~PWR BASKET~~ *MPC* ~~that~~ could *not* result in exceeding the design limits *for thermal, structural, radiological, criticality, and material parameters* ~~is not considered a credible event~~ *assumed in the SAR (Ref. 1)*. As a result of the permanent cessation of power operations, the *characteristics and* inventory available for loading into the ~~PWR BASKETS~~ *MPCs* ~~is~~ *are* known and limited to those fuel assemblies and fuel related material within the Trojan Nuclear Plant (TNP) spent fuel pool. The design of the ~~PWR BASKET~~ *MPC* and CONCRETE CASK ~~without use of the BASKET OVERPACK~~ is based on a maximum heat load of ~~26 KW~~ *17.4 kW, a maximum burnup of 42,000 MWd/MTU, and a minimum cooling time of 9 years*. ~~The design of the PWR BASKET with a BASKET OVERPACK is based on a maximum heat load of 24 KW.~~ Table 2-1 imposes *these* ~~a~~ limit*s* ~~of 24 KW per PWR BASKET which includes the Fuel Assembly Inserts with or without use of the BASKET OVERPACK~~. The ~~PWR BASKET~~ *MPC-24E/24EF* design will not accommodate more than 24 INTACT FUEL ASSEMBLIES *and/or combination of 20-24 INTACT FUEL ASSEMBLIES and up to four FAILED FUEL CANS.*

Based on the actual inventory of spent fuel available for storage, no combination of INTACT FUEL ASSEMBLIES could result in a decay heat load exceeding a limit of ~~24 KW~~ *17.4 kW*. ~~In addition, the actual inventory of FUEL DEBRIS is less than 7.5 kg of fissile material and less than 20 curies of plutonium. Therefore, misloading of a PWR BASKET is not considered a credible event.~~ *Similarly, no combination of INTACT*

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FUEL ASSEMBLIES and FAILED FUEL CANS would result in exceeding the design limits for radiation dose, criticality, materials, and *structural analysis. The specifications limit the spent fuel to a minimum of nine years of cooling prior to loading in a CONCRETE CASK to ensure radiation levels are less than assumed design criteria. No loading combination would result in a critical configuration given the maximum enrichment of the fuel. The geometry of and use of solid neutron absorbing material (Boral) in the MPC is such that a critical mass cannot be achieved. The materials used in the MPC, CONCRETE CASK, and TRANSFER CASK have been evaluated and determined to not have an adverse effect on the safe transfer and storage of spent fuel. The structural analysis of a loaded CONCRETE CASK was performed assuming a weight of 300,000 lbs, which is greater than the maximum weight of a loaded CONCRETE CASK with 24 RCCAs.*

~~FUNCTIONAL AND OPERATING LIMITS~~ *APPROVED CONTENTS* VIOLATIONS

The following ~~Functional and Operating Limits~~ *Approved Contents* violation responses are applicable.

Misloading of a*n* ~~PWR BASKET~~ *MPC* ~~that exceeds~~ *will not result in exceeding any of* the design limits specified in Table 2-1 ~~is not considered credible~~. The actions specified in Section 2.2.1 reflect the reporting requirements of 10 CFR 72.75.

References

1. SAR Section 4.2
2. ~~Letter from PGE to NRC dated 12/23/96 (Question 4-1)~~.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

BASES

LCOs

LCO 3.0.1, 3.0.2, 3.0.4 and 3.0.5 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

LCO 3.0.1

LCO 3.0.1 establishes the Applicability statement within each individual Specification as the requirement for when the LCO is required to be met (i.e., when the facility is in the specified conditions of the Applicability statement of each Specification).

LCO 3.0.2

LCO 3.0.2 establishes that upon discovery of a failure to meet an LCO, the associated ACTIONS shall be met. The Completion Time of each Required Action for an ACTIONS Condition is applicable from the point in time that an ACTIONS Condition is entered. The Required Actions establish those remedial measures that must be taken within specified Completion Times when the requirements of an LCO are not met. This Specification establishes that:

a. Completion of the Required Actions within the specified Completion Times constitutes compliance with a Specification; and

b. Completion of the Required Actions is not required when an LCO is met within the specified Completion Time, unless otherwise specified.

There are two basic types of Required Actions. The first type of Required Action specifies a time limit in which the LCO must be met. This time limit is the Completion Time to restore an inoperable system or component to operable status or to restore variables to within specified limits. If this type of Required Action is not completed within the specified Completion Time, a cessation of operations may be required to place the system or component in a condition in which the Specification is not applicable. (Whether stated as a Required Action or not, correction of the entered Condition is an action that may always be considered upon entering ACTIONS.) The second type of Required Action specifies the remedial measures that permit continued operation that is not further restricted by the Completion Time. In this case, compliance with the Required Actions

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

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provides an acceptable level of safety for continued operation.

Completing the Required Actions is not required when an LCO is met or is no longer applicable, unless otherwise stated in the individual Specifications.

The Completion Time of the Required Actions are also applicable when a system or component is removed from service intentionally. The reason for intentionally relying on the ACTIONS include, but are not limited to, performance of Surveillances, preventive maintenance, corrective maintenance, or investigation of operational problems. Entering ACTIONS for these reasons must be done in a manner that does not compromise safety. Intentional entry into ACTIONS should not be made for operational convenience.

LCO 3.03 This specification is not applicable to an ISFSI. The placeholder is retained for consistency with the power reactor technical specifications.

LCO 3.0.4 LCO 3.0.4 establishes limitations on changes in specified conditions in the Applicability when an LCO is not met. It precludes placing the unit in a specified condition stated in that Applicability (e.g., Applicability desired to be entered) when the following exist:

a. Facility conditions are such that the requirements of the LCO would not be met in the Applicability desired to be entered; and

b. Continued noncompliance with the LCO requirements, if the Applicability were entered, would result in the facility being required to exit the Applicability desired to be entered to comply with the Required Actions.

Compliance with the Required Actions that permit continued operation of the facility for an unlimited period of time in a specified condition provides an acceptable level of safety for continued operation. This is without regard to the status of the facility. Therefore, in such cases, entry into a specified condition in the Applicability may be made in accordance with the provisions of the Required

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

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Actions. The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

The provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to establishing and maintaining the spent fuel in an inert atmosphere.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. These exceptions allow entry into specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

LCO 3.0.5 LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or determined to not meet the LCO to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The equipment being returned to service meets the LCO: or

b. Other equipment meets the applicable LCOs.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed surveillance. This Specification does not provide time to perform any other preventive or corrective maintenance.

LCO 3.0.6 This specification is not applicable to an ISFSI. The placeholder is retained for consistency with the power reactor technical specifications.

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY

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LCO 3.0.7 This specification is not applicable to an ISFSI. The placeholder is retained for consistency with the power reactor technical specifications.

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

BASES

SRs

SR 3.0.1 through SR 3.0.4 establish the general requirements applicable to all Specifications and apply at all times, unless otherwise stated.

SR 3.0.1

SR 3.0.1 establishes the requirement that SRs must be met during the specified conditions in the Applicability for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify systems, components, and variables are within specified limits. Failure to meet a SR within the specified Frequency, in accordance with SR 3.0.2, constitutes a failure to meet an LCO.

Systems and components are assumed to meet the LCO when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components meet the associated LCO when:

a. The systems or components are known to not meet the LCO, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the facility is in a specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a specified condition.

Surveillances, including Surveillances invoked by Required Actions, do not have to be performed on equipment that has been determined to not meet the LCO because the ACTIONS define the remedial measures that apply.
Surveillances have to be met and performed in accordance with SR 3.0.2, prior to

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

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returning equipment to service.

Upon completion of maintenance, appropriate post maintenance testing is required. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 3.0.2. Post maintenance testing may not be possible in the current specified conditions in the Applicability due to the necessary facility parameters not having been established. In these situations, the equipment may be considered to meet the LCO provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a specified condition where other necessary post maintenance tests can be completed.

SR 3.0.2

SR 3.0.2 establishes the requirements for meeting the specified Frequency for Surveillances and any Required Action with a Completion Time that requires the periodic performance of the Required Action on a "once per..." interval.

SR 3.0.2 permits a 25% extension of the interval specified in the Frequency. This extension facilitates Surveillance scheduling and considers facility conditions that may not be suitable for conducting the Surveillance (e.g., transient conditions or other ongoing Surveillance or maintenance activities).

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications as a Note in the Frequency stating, "SR 3.0.2 is not applicable."

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

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or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as a convenience to extend Surveillance intervals or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment as not meeting the LCO or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified Frequency, whichever is less, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with SR 3.0.2, and not at the time that the specified Frequency was not met.

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with Required Actions or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of facility conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements. When a Surveillance with a Frequency based not on time intervals, but upon specified facility conditions or operational situations, is discovered not to have been performed when specified, SR 3.0.3 allows the full delay period of 24 hours to perform the Surveillance.

SR 3.0.3 also provides a time limit for completion of Surveillances that become

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applicable as a consequence of changes in the specified conditions in the Applicability imposed by Required Actions.

Failure to comply with specified Frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 3.0.3 is a flexibility which is not intended to be used as a convenience to extend Surveillance intervals.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered to not meet the LCO or the variable is considered outside the specified limits and the Completion Time of the Required Actions for the applicable LCO Conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment does not meet the LCO, or the variable is outside the specified limits and the Completion Time of the Required Actions for the applicable LCO Conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Completion Time of the ACTIONS, restores compliance with SR 3.0.1.

SR 3.0.4

SR 3.0.4 establishes the requirement that all applicable SRs must be met before entry into a specified condition in the Applicability.

This Specification ensures that system and component requirements and variable limits are met before entry into specified conditions in the Applicability for which these systems and components ensure safe operation of the facility.

The provisions of this Specification should not be interpreted as endorsing the failure to exercise the good practice of restoring systems or components before entering an associated specified condition in the Applicability.

However, in certain circumstances, failing to meet an SR will not result in SR 3.0.4 restricting a change in specified condition. When a system, subsystem,

B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

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component, device, or variable is outside its specified limits, the associated SR(s) are not required to be performed, per SR 3.0.1, which states that surveillances do not have to be performed on such equipment. When equipment does not meet the LCO, SR 3.0.4 does not apply to the associated SR(s) since the requirement for the SR(s) to be performed is removed. Therefore, failing to perform the Surveillances(s) within the specified Frequency does not result in an SR 3.0.4 restriction to changing specified conditions of the Applicability. However, since the LCO is not met in this instance, LCO 3.0.4 will govern any restrictions that may (or may not) apply to specified condition changes.

The provisions of SR 3.0.4 shall not prevent changes in specified conditions in the Applicability that are required to comply with ACTIONS. In addition, the provisions of LCO 3.0.4 shall not prevent changes in specified conditions in the Applicability that are related to the establishment and maintenance of an inert atmosphere in the ~~PWR BASKET~~ *MPC*.

The precise requirements for performance of SRs are specified such that exceptions to SR 3.0.4 are not necessary. The specific time frames and conditions necessary for meeting the SRs are specified in the Frequency, in the Surveillance, or both. This allows performance of Surveillances when the prerequisite condition(s) specified in a Surveillance procedure require entry into the specified condition in the Applicability of the associated LCO prior to the performance or completion of a Surveillance. A Surveillance that could not be performed until after entering the LCO Applicability, would have its Frequency specified such that it is not "due" until the specific conditions needed are met. Alternately, the Surveillance may be stated in the form of a Note as not required (to be met or performed) until a particular event, condition, or time has been reached. Further discussion of the specific formats of SRs annotation is found in Section 1.4, Frequency.

B 3.1 ~~PWR BASKET~~ *MPC* INTEGRITY

B 3.1.1 ~~PWR BASKET Shield~~ *MPC* Lid Weld Helium Leak Rate

BASES

BACKGROUND

The TRANSFER CASK containing a~~n PWR BASKET~~ *MPC* is placed in the spent fuel pool (i.e., Cask Loading Pit) and loaded with spent fuel (i.e., INTACT ~~SPENT~~ FUEL ASSEMBLIES, *DAMAGED FUEL ASSEMBLIES,* and *FUEL DEBRIS* ~~FAILED FUEL CANS~~) meeting the requirements specified in Table 2-1 and Fuel Assembly Inserts meeting the requirements specified in Table 2-2. When ~~PWR BASKET~~ *MPC* loading is complete, the ~~shield~~ *MPC* lid is placed onto the ~~PWR BASKET~~ *MPC* prior to lifting the TRANSFER CASK from the spent fuel pool (i.e., Cask Loading Pit). The TRANSFER CASK containing a loaded ~~PWR BASKET~~ *MPC* is lifted from the spent fuel pool (i.e., Cask Loading Pit), and moved to the cask preparation area for decontamination and assembly to prepare the ~~PWR BASKET~~ *MPC* for STORAGE OPERATIONS. The ~~PWR BASKET~~ *MPC* ~~shield~~ lid is seal welded. To ensure the confinement integrity of the ~~shield~~ *MPC* lid weld, the ~~PWR BASKET~~ *MPC* is *hydrotested and helium leak rate tested* ~~pressurized with helium to a pressure of ≥13 psig~~. A helium leak detector is used to ensure a leak rate of ≤ $5x10^{-6}$ *atm-cc/sec* ~~10^{-4} cc/sec~~ is not exceeded.

APPLICABLE SAFETY ANALYSES

The confinement of radioactivity during the storage of spent nuclear fuel is ensured by the use of multiple confinement barriers and systems. The barriers relied upon are the uranium dioxide fuel pellet matrix, the metallic fuel clad~~ding~~ tubes in which the fuel pellets are contained, and the ~~PWR BASKET~~ *MPC* in which the INTACT FUEL ASSEMBLIES are stored. Long term integrity of the fuel clad~~ding~~ depends on storage in an inert atmosphere. This is accomplished by removing water from the ~~PWR BASKET~~ *MPC* and backfilling it with an inert gas. The failure of all confinement barriers is considered *an incredible event* in the accident analysis (Ref. 1). In addition, the thermal analysis of the ~~PWR BASKET~~ *MPC* and CONCRETE CASK assume*s* that the ~~PWR BASKET~~ *MPC* is filled with dry helium.

DAMAGED FUEL *ASSEMBLIES have* ~~has~~ already released the fission

B 3.1 ~~PWR BASKET~~ *MPC* INTEGRITY

B 3.1.1 ~~PWR BASKET Shield~~ *MPC* Lid Weld Helium Leak Rate

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product gases so ~~it~~ *they* do~~es~~ not have the same confinement requirements of INTACT FUEL ASSEMBLIES. DAMAGED FUEL *ASSEMBLIES* and FUEL DEBRIS have been placed in FAILED FUEL CANS in designated ~~PWR BASKETS~~ *MPCs*.

Short term fuel clad temperatures less than *the* 1058°F *limit* do not result in degradation of the clad~~ding~~. The maximum ~~steady state~~ fuel clad temperature would occur during the vacuum drying process and not during the ~~Shield~~ *MPC* Lid Weld Helium Leak Rate check. ~~It~~ *The maximum fuel clad temperature* has been calculated and found to not exceed ~~888°F~~ *the short term fuel clad temperature limit* for the ~~26~~ *17.4* ~~K~~*k*W loaded basket (Ref. 2).

During STORAGE OPERATIONS, the helium atmosphere provides improved heat transfer characteristics which have been credited in the analysis of off-normal and accident conditions (Ref. 1).

LCO

Verifying that the ~~PWR BASKET~~ *MPC* is sealed by measuring the ~~shield~~ lid weld helium leak rate will ensure that assumptions made in the accident analysis and radiological evaluations are maintained and that the inert gas cover is maintained for the duration of long term storage. The criterion of ≤ $5x10^{-6}$ *atm-cc/sec* ~~10^{-4} cc/sec~~ *is consistent with the leak rate assumed in the confinement analysis. As a practical matter, the MPC is leaktight and will retain all of the helium injected into the MPC cavity during backfill operations* ~~at ≥ 13 psig results in less than 2% loss of helium during the 20 year license~~. The LCO does not have to be met until 72 hours after removal of the loaded TRANSFER CASK from the spent fuel pool (i.e., Cask Loading Pit).

APPLICABILITY

The maximum ~~steady state~~ fuel clad temperature during LOADING OPERATIONS is ~~bounded by the steady state temperature limit of 888~~ *659*°F during vacuum drying operations (Ref. 2). Since this maximum ~~steady-state~~ temperature is ~~170~~ *399*°F below the short term fuel clad~~ding~~

B 3.1 ~~PWR BASKET~~ *MPC* INTEGRITY

B 3.1.1 ~~PWR BASKET Shield~~ *MPC* Lid Weld Helium Leak Rate

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temperature limit of 1058°F, no degradation of fuel clad~~ding~~ is anticipated.

ACTIONS

A Note has been added to the ACTIONS which states that, for this LCO, a separate Condition entry is allowed for each ~~PWR BASKET~~ *MPC*. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each ~~PWR BASKET~~ *MPC* not meeting the LCO. Subsequent ~~PWR BASKETS~~ *MPCs* that do not meet the LCO are governed by subsequent Condition entry and application of the associated Required Actions.

A.1

If the helium leak rate limit is not met, actions must be taken to meet the LCO. A possible corrective action is weld repair. The Completion Time of 48 hours provides adequate time to correct conditions that would prevent satisfying the ~~shield~~ *MPC* lid weld helium leak rate requirements. Specifying a maximum time for completion ensures that completing the ~~shield~~ *MPC* lid leak rate testing is expedited.

B.1

In the event the ~~shield~~ *MPC* lid helium leak rate cannot be satisfied within the allowed Completion Time of ACTION A.1, a cooling flow path will be established to the ~~PWR BASKET~~ *MPC* (Ref. 3). The ~~shield~~ *MPC* lid helium leak rate test will be performed with the ~~PWR BASKET~~ *MPC* filled with water except for approximately ~~75~~ *50 to 120* gallons ~~which~~ *that* has been drained. Therefore, it is relatively easy to establish cooling by ~~means of the Vacuum Drying System (VDS)~~ *recirculating helium or borated water through the MPC cavity*. Establishment of ~~PWR BASKET~~ *MPC* cooling will maintain the fuel clad temperature less than the ~~705~~ *647*°F long term storage limit *(Ref. 4)*, thereby allowing repair of the ~~PWR BASKET~~ *MPC* ~~shield~~ lid weld or helium leak detector as appropriate.

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B 3.1.1 ~~PWR BASKET Shield~~ *MPC* Lid Weld Helium Leak Rate

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The cooling flow path shall remain available, however, flow may be interrupted to complete ~~PWR BASKET~~ *MPC* ~~shield~~ lid weld repairs *and perform the helium leak rate test* ~~provided the bulk water temperature of the PWR BASKET does not exceed 212°F~~.

B.2

In the event the ~~shield~~ *MPC* lid helium leak rate cannot be satisfied within the allowed Completion Time of ACTION A.1, steps will also be taken to return the ~~PWR BASKET~~ *MPC* to the spent fuel pool (i.e., Cask Loading Pit) for unloading within 30 days. Since cooling of the ~~PWR BASKET~~ *MPC* was reestablished by ACTION B.1, there is no thermal limit or safety parameter that could be exceeded by not meeting the helium leak rate LCO. Thirty days was determined to be reasonable to make welding repairs or obtain replacement parts for the helium leak detector.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

A primary design consideration of the ~~PWR BASKET~~ *MPC* is that it is leak tight. The surveillance requirement to verify that the Confinement Boundary helium leak rate is ≤ *$5x10^{-6}$atm-cc/sec ensures that the assumptions in the confinement analysis are preserved. The measured leak rate will be a function of the test pressure and the sensitivity of the instrumentation. The results at test conditions are corrected for comparison with the $5x10^{-6}$atm-cc/sec LCO limit.* ~~$1 X 10^{-4}$ standard cubic centimeters per second at ≥ 13 psig will result in a loss of helium over a 20 year period of less than 2% of the PWR BASKET volume. Since the helium leak rate is proportional to pressure, and 13 psig is a test pressure value conservatively chosen to bound normal operating pressures (the PWR BASKET is stored at essentially 0 psig), the surveillance demonstrates that the Confinement Boundary is sealed~~.

B 3.1 ~~PWR BASKET~~ *MPC* INTEGRITY

B 3.1.1 ~~PWR BASKET Shield~~ *MPC* Lid Weld Helium Leak Rate

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Measurement of the ~~shield~~ *MPC* lid weld helium leak rate must be performed successfully on each ~~PWR BASKET~~ *MPC* prior to performance of ~~vacuum~~ *MPC cavity* drying. Subsequent to the completion of ~~the shield lid leak rate test~~ *MPC cavity drying, the vent and drain port cover plates are welded in place and helium leak rate tested. Finally, after the port covers are helium leak rate tested, a* redundant confinement boundary is provided by the ~~structural lid~~ *closure ring* seal welds.

A Note has been added to the SR to ~~delay its performance~~ *indicate it is not required to be met* until 72 hours after the TRANSFER CASK loaded with the ~~PWR BASKET~~ *MPC* has been removed from the spent fuel pool (i.e., Cask Loading Pit). This allows sufficient time to complete the closure weld, perform the hydrostatic test, and prepare to perform the helium leak rate test.

REFERENCES

1. SAR Sections 8.1.2, *8.2.1,* 8.2.2, 8.2.6, and 8.2.7
2. SAR Table 4.2-12
3. SAR Section 5.1.1.2
4. *SAR Table 3.1-3*

B 3.1 ~~PWR BASKET~~ *MPC* INTEGRITY

B 3.1.2 ~~PWR BASKET Vacuum Drying Pressure~~ *MPC Cavity Dryness*

BASES

BACKGROUND

The TRANSFER CASK containing a*n* ~~PWR BASKET~~ *MPC* is placed in the spent fuel pool (i.e., the Cask Loading Pit). The ~~PWR BASKET~~ *MPC* is loaded with spent fuel (i.e., INTACT FUEL ASSEMBLIES, *DAMAGED FUEL ASSEMBLIES,* and *FUEL DEBRIS* ~~FAILED FUEL CANS~~) meeting the requirements specified in Table 2-1 and Fuel Assembly Inserts meeting the requirements specified in Table 2-2. When ~~PWR BASKET~~ *MPC* loading is complete, the ~~shield~~ lid is placed onto the ~~PWR BASKET~~ *MPC* prior to lifting the TRANSFER CASK from the spent fuel pool (i.e., Cask Loading Pit). The TRANSFER CASK containing a loaded ~~PWR BASKET~~ *MPC* is lifted from the spent fuel pool (i.e., Cask Loading Pit), and moved to the cask preparation area for decontamination and assembly to prepare the ~~PWR BASKET~~ *MPC* for STORAGE OPERATIONS. Prior to commencement of ~~vacuum~~ *MPC cavity* drying, the ~~PWR BASKET~~ *top of the MPC* is decontaminated, ~~and~~ the ~~shield and structural~~ lid ~~are~~ *is* seal welded in place, and the helium leak test is performed on the closure weld of the ~~shield~~ lid. Once the water is removed from the ~~PWR BASKET~~ *MPC* cavity, ~~vacuum~~ drying operations can begin.

There are two options for achieving the required MPC cavity dryness. One method is vacuum drying to a specific pressure. The pressure is chosen to ensure that any remaining water vapor in the cavity is of such a small quantity that it is of no concern for corrosion. The second method involves recirculating helium through a closed loop to condense out any entrained moisture. Recirculation is continued until the amount of remaining water vapor in the gas is below the level at which corrosion is a concern.

Vacuum drying is aided by a temperature increase in the ~~PWR BASKET~~ *MPC* due to heating from the fuel which removes residual moisture. In light of the fact that the helium leak rate test has been performed and ~~both lids are~~ *the MPC lid is* welded in place, the integrity of the ~~PWR BASKET~~ *MPC* has been demonstrated.

B 3.1 ~~PWR BASKET~~ *MPC* INTEGRITY

B 3.1.2 ~~PWR BASKET Vacuum Drying Pressure~~ *MPC Cavity Dryness*

BASES

The purpose of the *MPC cavity drying process* ~~Vacuum Drying Process~~ is to remove ~~all~~ moisture from the ~~PWR BASKET~~ *MPC* prior to backfilling it with helium for long-term storage. *An inert environment in the MPC cavity ensures there will be no significant corrosion of the fuel clad during STORAGE OPERATIONS at the ISFSI.*

APPLICABLE SAFETY ANALYSES

The confinement of radioactivity during the storage of spent nuclear fuel is ensured by the use of multiple confinement barriers and systems. The barriers relied upon are the uranium dioxide fuel pellet matrix, the metallic fuel clad~~ding~~ tubes in which the fuel pellets are contained, and the ~~PWR BASKET~~ *MPC* in which the ~~fuel assemblies~~ *INTACT FUEL ASSEMBLIES* are stored. Long term integrity of the fuel clad~~ding~~ depends on storage in an inert atmosphere. This is accomplished by removing water from the ~~PWR BASKET~~ *MPC* and backfilling it with an inert gas. The failure of all confinement barriers is considered an incredible event and is discussed in the accident analysis (Ref. 1). In addition, the thermal analysis of the ~~PWR BASKET~~ *MPC* and CONCRETE CASK assume*s* that the ~~PWR BASKET~~ *MPC* is filled with dry helium.

DAMAGED FUEL *ASSEMBLIES* has*ve* already released the fission product gases so ~~it~~ *they* do~~es~~ not have the same confinement requirements of INTACT FUEL ASSEMBLIES. DAMAGED FUEL *ASSEMBLIES* and FUEL DEBRIS have been placed in FAILED FUEL CANS in designated ~~PWR BASKETS~~ *MPCs*.

Short term fuel clad temperatures less than 1058°F do not result in gross degradation of the clad~~ding~~. Calculations demonstrate that the maximum steady state fuel clad temperature that would occur during the vacuum drying process is ~~888~~ *659*°F for the ~~26~~ *17.4* ~~K~~*k*W loaded basket. This is ~~170~~ *399*°F less than the *short term fuel clad temperature* limit. Although some clad~~ding~~ creep could occur at this temperature, it will not result in gross failures. The ~~888~~ *659*°F is an equilibrium value and there is no time

B 3.1 ~~PWR BASKET~~ *MPC* INTEGRITY

B 3.1.2 ~~PWR BASKET Vacuum Drying Pressure~~ *MPC Cavity Dryness*

BASES

limit or restriction associated with it; i.e., the fuel clad could be maintained at this temperature for long periods of time without failure. Testing described in PNL-4835 indicated no clad~~ding~~ failure at temperatures of 1058°F for periods in excess of 30 days.

LCO

A vacuum drying pressure of ≤ 3 ~~mm Hg~~ *torr* held for ≥ 30 minutes *or a helium gas demoisturizer exit temperature ≤ 21°F for ≥ 30 minutes* assures that water within the ~~PWR BASKET~~ *MPC* cavity has been evaporated and removed. *The helium gas demoisturizer exit temperature of 21 °F is equivalent to the vacuum drying pressure of 3 torr since the moisture content at each thermodynamic condition is equivalent. The attainment of either limit removes the moisture from the MPC.* Removal of water before filling the ~~PWR BASKET~~ *MPC* with an inert gas ensures optimum long term storage conditions by inhibiting the potential for fuel clad~~ding~~ and ~~C~~confinement ~~B~~*b*oundary degradation.

Fuel ~~C~~clad temperatures are not a concern while the ~~PWR BASKET~~ *MPC* is submerged within the spent fuel pool (i.e., Cask Loading Pit). However, following removal from the spent fuel pool (i.e., Cask Loading Pit), heat transfer is limited with worst case conditions occurring during vacuum drying when conductive heat transfer is limited. Even under these conditions, steady state (i.e., equilibrium) clad~~ding~~ temperatures for a ~~26~~ *17.4* ~~K~~*k*W loaded ~~PWR BASKET~~ *MPC* will not exceed ~~888~~ *659*°F which is ~~170~~ *399*°F below the 1058°F short term clad~~ding~~ temperature limits identified in NUREG-1536. Although no fuel clad damage is anticipated to occur at temperatures less than 1058°F, a 96 hour administrative limit to complete the *MPC cavity* ~~vacuum~~ drying ~~(if that method is used)~~ has been established to minimize the time period in which the ~~PWR BASKET~~ *MPC* cavity atmosphere is not inert. The 96 hour administrative limit allows adequate time to complete the ~~PWR BASKET~~ *MPC* preparations and the ~~vacuum~~ *MPC cavity* drying.

B 3.1 ~~PWR BASKET~~ *MPC* INTEGRITY

B 3.1.2 ~~PWR BASKET Vacuum Drying Pressure~~ *MPC Cavity Dryness*

BASES

APPLICABILITY

This LCO is APPLICABLE during LOADING OPERATIONS but only after the helium leak rate test has been completed. LOADING OPERATIONS begin with the loading of the ~~PWR BASKET~~ *MPC* within the spent fuel pool (i.e., Cask Wash Pit) and end when the ~~PWR BASKET~~ *MPC* is loaded into a CONCRETE CASK.

ACTIONS

A Note has been added to the ACTIONS which states that, for this LCO, a separate Condition entry is allowed for each ~~PWR BASKET~~ *MPC*. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each ~~PWR BASKET~~ *MPC* not meeting the LCO. Subsequent ~~PWR BASKETS~~ *MPCs* that do not meet the LCO are governed by subsequent Condition entry and application of the associated Required Actions.

A.1

If the ~~PWR BASKET~~ *MPC cavity dryness* ~~vacuum drying pressure~~ limit cannot be met, actions must be taken to meet the LCO. Since the ~~vacuum~~ *MPC cavity* drying process is initiated after the ~~PWR BASKET~~ *MPC* has been hydrostatically and helium leak rate tested, and the closure weld dye penetrant tested, there is a high level of assurance that the confinement boundary is intact and properly sealed. The most likely mechanism preventing meeting the LCO is a failure of the Vacuum Drying System, *helium recirculation system,* and/or associated hardware.

Since the maximum steady-state temperature which the spent nuclear fuel would experience during the duration of fuel loading and vacuum drying operations is ~~888~~ *659*°F (Ref. ~~1~~*2*) which is ~~170~~ *399*°F below the short term fuel clad~~ding~~ temperature limit of 1058°F, no degradation of fuel clad~~ding~~ is anticipated. A 48 hour Completion Time is considered appropriate for making repairs and correcting failures of the *affected components* ~~Vacuum Drying System~~.

B 3.1 ~~PWR BASKET~~ *MPC* INTEGRITY

B 3.1.2 ~~PWR BASKET Vacuum Drying Pressure~~ *MPC Cavity Dryness*

BASES

B.1.1 and B.1.2

If the Required Action of A.1 cannot be satisfied, either a method of cooling the spent fuel must be established or an inert atmosphere needs to be established for the spent fuel. If the water is not completely drained from the ~~PWR BASKET~~ *MPC* ~~or the shield lid closure weld is incomplete~~, use of cooling water is desirable to maintain spent fuel temperatures at lower values. However, once the water has been removed, the most efficient means to protect the spent fuel is by creating an inert atmosphere around the spent fuel. The inert atmosphere will help to remove heat. Cooling the spent fuel can be accomplished by establishing cooling water circulation to the ~~PWR BASKET~~ *MPC* as described in the SAR (Ref. ~~2~~*3*). An inert atmosphere can be established by creating a helium atmosphere in the ~~PWR BASKET~~ *MPC* as described in the SAR (Ref. ~~2~~*3*). Either of these actions will reduce fuel clad temperature and minimize the potential for fuel clad degradation until repairs *are completed as* necessary to *perform* ~~successfully complete vacuum~~ *MPC cavity* drying *until the LCO is satisfied* ~~are complete~~.

B.2

In the event the ~~vacuum drying pressure~~ *MPC cavity dryness* limit cannot be satisfied within the allowed Completion Time of ACTION A.1, steps will also be taken to return the ~~PWR BASKET~~ *MPC* to the spent fuel pool (i.e., Cask Loading Pit) for unloading within 30 days. Since cooling of the ~~PWR BASKET~~ *MPC* was reestablished by ACTION B.1, there is no thermal limit or safety parameter that could be exceeded by not meeting the ~~vacuum drying pressure~~ *MPC cavity dryness* LCO. Thirty days was determined to be reasonable to make ~~welding~~ repairs or obtain replacement parts for the *affected components* ~~Vacuum Drying System~~.

B 3.1 ~~PWR BASKET~~ *MPC* INTEGRITY

B 3.1.2 ~~PWR BASKET Vacuum Drying Pressure~~ *MPC Cavity Dryness*

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. Cavity dryness is demonstrated by evacuating the cavity to a very low pressure and verifying that the pressure is held over the specified period of time *or lowering the recirculating gas temperature to or below the specified limit.*

This dryness test must be performed on each ~~PWR BASKET~~ *MPC* within 96 hours after verifying the helium leak rate is within limit. A Note has been added to not require the SR be APPLICABLE until 96 hours after the helium leak rate is acceptable. This allows sufficient time to complete the ~~PWR BASKET~~ *MPC* preparations and perform the ~~vacuum~~ drying *operations. If the vacuum drying method is used, the 96 hours is a reasonable amount of time to allow the MPC to be* ~~while minimizing the time the fuel in the PWR Basket is~~ without an inert atmosphere.

REFERENCES

1. SAR 8.2.1
~~1~~*2.* SAR Table 4.2-12
~~2~~*3.* SAR 5.1.1.2

B 3.2 TRANSFER CASK INTEGRITY

B 3.2.1 TRANSFER CASK Ambient Air Temperature Limits

BASES

BACKGROUND

In addition to its functions in the Fuel Building in transporting a*n* ~~PWR BASKET~~ *MPC* loaded with spent fuel from the spent fuel pool (i.e., Cask Loading Pit) to a CONCRETE CASK, the TRANSFER CASK is used as a support and shielding container in the TRANSFER STATION during the process of transferring a*n* ~~degraded PWR BASKET~~ *MPC* into a *Transport Cask* ~~BASKET OVERPACK or a shipping container~~. The activities involving the TRANSFER CASK in the Fuel Building will be in a controlled environment, ensuring that ambient air temperatures are controlled and that solar heating does not become a factor. However, the activities at the TRANSFER STATION will not be in a controlled environment, and ambient air temperatures cannot be controlled. The analysis of the TRANSFER CASK design assumed *an* ambient air temperatures equal to or greater than *0°F* ~~40°F when used as a lifting device but equal to or less than 100°F when used as a lifting and support device. The 100°F limit of the TRANSFER CASK is based upon the tested properties of the neutron shielding material, RX244, which were verified to 350°F, which corresponds to slightly more than 100°F ambient air temperature~~. Since the TRANSFER STATION ambient air temperatures cannot be controlled, use of the TRANSFER CASK must be restricted to occur only when *the* ambient air temperatures ~~are~~ *is above* ~~within these~~ *this* limits.

APPLICABLE SAFETY ANALYSIS

The design *characteristics* of the TRANSFER CASK ~~is an assumption~~ *are considered* in the Safety Analysis for handling spent fuel. Inherent in the design is the assumption that use of the TRANSFER CASK ~~as a lifting device will occur at or above an ambient air temperature of 40°F or as a support and shielding device~~ will occur at ~~or less than an~~ ambient air temperatures *greater than* ~~of 100~~ *0°*F. Establishing *an* ambient air temperature limits for use of the TRANSFER CASK ensures that its integrity is maintained and the Safety Analysis is valid. ~~The calculation evaluating use of the TRANSFER CASK as a storage device assumed the PWR BASKET was filled with helium and sealed. Use of the~~

B 3.2 TRANSFER CASK INTEGRITY

B 3.2.1 TRANSFER CASK Ambient Air Temperature Limits

BASES

~~TRANSFER CASK as a lifting device only assumed the PWR BASKET was loaded.~~

LCO

Limiting use of the TRANSFER CASK to periods when the ambient air temperature is > ~~equal to or greater than 40~~°F ensures ~~the structural integrity of~~ the TRANSFER CASK ~~is maintained~~ *will not fail by brittle fracture.*

~~Limiting use of the TRANSFER CASK to periods when the ambient air temperature is equal to or less than 100°F ensures the neutron shielding properties of the RX244 material are maintained.~~

APPLICABILITY

The APPLICABILITY for this LCO is LOADING, UNLOADING, and STORAGE OPERATIONS. In the event a loaded ~~PWR BASKET~~ *MPC* is *to be transferrred into a Transport Cask* ~~degraded~~, the TRANSFER CASK will be used to support ~~a PWR BASKET~~ *the MPC* after it has been raised out of a CONCRETE CASK and until it is lowered back into a ~~BASKET OVERPACK in a CONCRETE CASK~~ *Transport Cask* at the TRANSFER STATION. The APPLICABILITY ensures that the LCO applies to all activities involving the use of the TRANSFER CASK.

ACTIONS

A.1 and A.2

~~If TRANSFER STATION or Fuel Building ambient air temperature increases to 100°F or more during use of the TRANSFER CASK, immediate steps will be taken to stop or complete all activities until ambient air temperatures have returned to less than 100°F. In some cases, the prudent action or safe operations might be to complete the activity depending upon the time required to place the TRANSFER CASK in a safe configuration (e.g., lower the PWR BASKET from the TRANSFER CASK in the TRANSFER STATION into a CONCRETE CASK.~~

B 3.2 TRANSFER CASK INTEGRITY

B 3.2.1 TRANSFER CASK Ambient Air Temperature Limits

BASES

~~B.1 and B.2~~

If ~~Fuel Building~~ ambient air temperature decreases to 40°F or less during use of the TRANSFER CASK *to lift, support, or transport a loaded MPC*, immediate steps will be taken to place the TRANSFER CASK in a safe condition and suspend all further operations involving use of the TRANSFER CASK *with a loaded MPC* ~~as a lifting device~~ until ambient air temperatures have returned to greater than 40°F. *A safe condition is one in which the TRANSFER CASK is not being used to lift, support, or transport a loaded MPC. This may involve unloading the TRANSFER CASK.*

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

This SR ensures that the TRANSFER CASK is not used *to lift, support, or transport a loaded MPC* whenever the ambient air temperatures ~~are~~ *is* outside ~~their~~ *its* limits at the locations where the TRANSFER CASK is being used prior to use.

SR 3.2.1.2

~~This SR ensures that when the TRANSFER CASK is being used with an ambient air temperature greater than 90°F, the ambient air temperature is verified equal to or less than 100°F every four hours at the locations where it is being used.~~

~~SR 3.2.1.3~~

This SR ensures that when the TRANSFER CASK is being used *to lift, support, or transport a loaded MPC* with an ambient air temperature less than 45°F, the ambient air temperature is verified greater than 40°F every four hours at the locations where it is being used.

B 3.2 TRANSFER CASK INTEGRITY

B 3.2.1 TRANSFER CASK Ambient Air Temperature Limits

BASES

REFERENCES 1. SAR, Section 4.7.3.1

B 3.3 AIR PADS

B 3.3.1 AIR PAD Limits

BASES

BACKGROUND

The AIR PADS are used to move CONCRETE CASKS (References 1 and 2). The AIR PADS are inserted into the air inlets at the bottom of the CONCRETE CASK. Air compressors are used to inflate and maintain pressure in the AIR PADS. The inflated AIR PADS lift the CONCRETE CASK above the ground and allow it to float on a cushion of air. A transport vehicle is connected to the CONCRETE CASK to move it.

When installed, the AIR PADS partially block the CONCRETE CASK air inlets and reduce the cooling air flow. However, when the AIR PADS are inflated, for analysis purposes, it is assumed that all air flow is blocked even though there is some natural circulation through an unblocked opening and forced air flow from the AIR PAD itself. In either case (i.e., AIR PADS installed or inflated), although the air flow from natural draft air flow or forced air flow, respectively, will provide cooling, the extent of that cooling has not been determined.

APPLICABLE SAFETY ANALYSIS

The CONCRETE CASK bulk concrete temperature is the limiting thermal design parameter (References 3 and 6). Because of the temperature gradient across the concrete, the bulk concrete temperature is difficult to determine and use in analyses. Therefore, the inner concrete temperature is used in lieu of the bulk concrete temperature and is limited to 225°F for long-term normal operational storage (Reference 3). This is conservative since the bulk concrete temperature will not exceed the inner concrete temperature.

In the Full Blockage of Air Flow ~~case~~ *accident evaluated* in the SAR (Reference 4), ~~which assumes~~ *100°F ambient air and* complete air flow blockage of all inlets ~~and all outlets~~ *is assumed.* ~~f~~*F*or a CONCRETE CASK loaded with a*n* ~~PWR BASKET~~ *MPC* with a ~~26~~ *17.4* kW heat load, the inner concrete temperature will increase, reaching the long-term normal operation storage limit of 225°F in approximately ~~9.3~~ *20* hours, the short-term off-normal limit of 300°F in approximately ~~21~~ *39.5* hours, and the short-term accident limit of 350°F in approximately ~~31.5~~ *57.1*

B 3.3 AIR PADS

B 3.3.1 AIR PAD Limits

BASES

hours (Reference 3). In order to prevent the inner concrete temperature from reaching the long-term normal storage limit, installation of the AIR PADS will be restricted to no more than ~~9~~ *20 consecutive* hours ~~in any 24-hour period~~. *Twenty hours is more than sufficient time to complete the movement of the CONCRETE CASK using the AIR PADS. However, the analysis shows that the AIR PADS could be installed for 57 hours without exceeding the short-term accident limit for concrete. Should the 20 consecutive hours limit be approached, it is expected the AIR PADS would be removed until the recirculation is restored and the difficulty in moving the CONCRETE CASK would be resolved.*

Since ~~T~~the Full Blockage of Air *Inlets* ~~Flow case~~ *accident evaluated* in the SAR assumes an initial ambient air temperature of ~~75~~ *100*°F~~; therefore~~, a ~~75~~ *100*°F temperature limit is placed upon installation and use of the AIR PADS on a loaded CONCRETE CASK.

In the Blockage of One-Half of the Air Inlets case in the SAR (Reference 5), which assumes blockage of one-half of the air inlets, for a CONCRETE CASK loaded with a*n* ~~PWR BASKET~~ *MPC* with a ~~26~~ *17.4* kW heat load, the inner concrete temperature will not reach the long-term storage limit of 225 °F. This analysis ~~also~~ assumes an initial ambient air temperature of 75 °F.

LCO

Restricting installation of the AIR PADS to no more than ~~9~~ *20 consecutive* hours ~~in a 24-hour period~~ ensures that the long-term inner concrete storage temperature is not exceeded and that the CONCRETE CASK is not adversely affected.

Restricting the installation of the AIR PADS on a loaded CONCRETE CASK to periods when the ambient air temperature is less than or equal to ~~75~~ *100* °F ensures that the long-term inner concrete storage temperature is not exceeded ~~regardless of the PWR BASKET heat load(the actual heat load is less than the design heat load of 26 kW)~~.

B 3.3 AIR PADS

B 3.3.1 AIR PAD Limits

BASES

APPLICABILITY

The loaded CONCRETE CASKS will only be moved in LOADING, TRANSPORT, or STORAGE OPERATIONS. Therefore, this LCO and SR are only applicable during these conditions.

ACTIONS

A.1

If the AIR PADS are installed for more than ~~9~~ *20 consecutive* hours ~~in a 24-hour period~~, the AIR PADS must be immediately removed. This will reestablish air flow to cool the concrete.

B.1

If ambient air temperature exceeds ~~75~~ *100*°F, the AIR PADS must be immediately removed. This will reestablish air flow to cool the concrete and stay within analyzed limits. ~~Although all of the AIR PADS will be removed, the safety analysis has shown that blockage of one-half of the air inlets, which is more conservative than (but similar to) the case of two of the AIR PADS installed, does not lead to adverse concrete temperatures (Reference 5).~~

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1

Since the long term integrity of the concrete is ensured by maintaining its temperature below the specified limits, the length of time during which the AIR PADS can be installed is monitored *every 10* hour~~ly~~*s* while they are installed.

SR 3.3.1.2

~~Similarly, t~~*T*he ambient air temperature is monitored hourly *whenever it is greater than 90 °F and the AIR PADS are installed. The long term integrity of the concrete is ensured by maintaining its temperature below*

B 3.3 AIR PADS

B 3.3.1 AIR PAD Limits

BASES

the specified limits. To ensure the temperature of the concrete does not exceed its limits, the ambient air temperature during the period of time in which the AIR PADS are installed is monitored hourly when ambient air temperature is > 90 °F.

REFERENCES

1. SAR Section 5.1.1.3
2. SAR Section 5.2.1.1.7
3. SAR Table 4.2-12
4. SAR Section 8.2.7
5. SAR Section 8.1.2.2
6. SAR Table 4.2-2a

ENCLOSURE 1

ATTACHMENT C

to VPN-044-2001

Trojan ISFSI

LCA 72-02

Licensing Basis for Change to the Technical Specification for the Mobile Crane at the Trojan ISFSI

SAIC-01/1057

Licensing Basis for Change to the Technical Specification for the Mobile Crane at the Trojan Nuclear Plant Independent Spent Fuel Storage Installation

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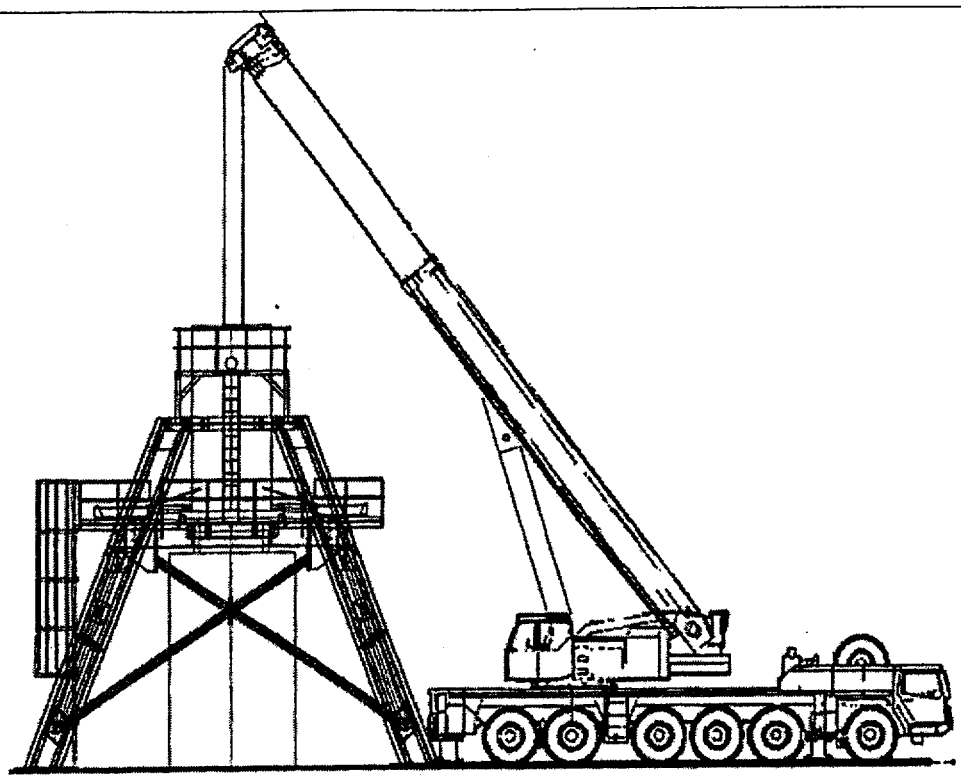
Table of Contents

1. INTRODUCTION

This licensing document was generated in response to a need to change the Trojan Independent Spent Fuel Storage Installation (ISFSI) Technical Specification 4.2.3 (Ref. 1). Technical Specification 4.2.3 states that operations at the Transfer Station which involve lifts of a PWR Basket must be performed using a mobile crane with a safety factor of at least 2 in accordance with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," dated 1980, (Ref. 2). One possible interpretation of the requirements of this Technical Specification is that the mobile crane is incorrectly classified as a "special lifting device" as defined in ANSI N14.6-1978, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More" (Ref. 3).

The ANSI N14.6 standard applies to special lifting devices with single and dual load path hoisting systems, however the special lifting device is defined as a device, that transmits the load from the lifting attachments, which are structural parts of the container, to the hook(s) of an overhead hoisting system. Thus it applies to such devices as lifting trunnions or lifting cleats on containers, to lifting yokes or other flexible or rigid members connecting the container to the hook(s) of the hoisting system, and not to the mobile crane.

The mobile crane (below) will be used at the Trojan ISFSI to perform two types of lifts at the Transfer Station. The mobile crane will perform a non-critical lift of an empty Transfer Cask to the top of the Transfer Station. It will also perform a critical straight vertical lift of an MPC into the Transfer Cask and a critical straight lowering of an MPC into a Shipping Cask for off-site shipment or into a Concrete Cask for further on-site storage. It will not be used to swing a loaded MPC, nor will it be used to lift or swing a Transfer Cask with a loaded MPC.



225 Ton Hydraulic Crane

The proposed changes to Technical Specification 4.2.3 state that operations at the Transfer Station, which involve lifts of an MPC, must be performed with a mobile crane, which shall meet the guidance of Section 5.1.1 of NUREG-0612, except that to assure defense-in-depth the following shall be required. The mobile crane shall have a safety factor of two over the allowable load for the crane in accordance with the guidance of NUREG-0612, and the crane must be able to safely stop and hold an MPC in the event of a safe shutdown earthquake. The mobile crane shall meet the requirements of ANSI B30.5, "Mobile and Locomotive Cranes," (Ref. 4) or the equivalent. The lift height of the MPC shall be restricted to 249 inches. Special lifting devices shall have two times the design safety factors of Section 4.2.1.1 of ANSI N14.6-1993. Slings and wire rope to be used in the vertical lift of the MPC at the Transfer Station shall meet the requirements of ANSI B30.9, "Slings," (Ref. 5) except that the load to be used in selecting the slings and rope is to be twice that specified in NUREG-0612 Section 5.1.1(5).

The licensing bases for the proposed changes are found in NUREG-1576, "Standard Review Plan for Spent Fuel Dry Storage Facilities," (Ref. 6), which in turn references the guidance in NUREG-0612. Section 5 of NUREG-0612 recommends specific guidelines for the control of heavy loads. The guidance is based on a defense-in-depth approach. The two primary objectives of the approach include prevention of accidental dropping of the load, and mitigation of the consequences of the postulated drop.

This document presents the 10 CFR Part 72 licensing requirements for lifting heavy loads at an ISFSI, a description of the operational aspects of using a mobile crane at Trojan, and a clear licensing basis for changing Technical Specification 4.2.3, while maintaining factors of safety for the mobile crane as well as the special lifting devices.

2. BACKGROUND

The Trojan nuclear facility currently has a limited 10 CFR Part 50 license. The handling and loading of spent fuel into the MPC will be conducted inside the spent fuel pool building under the provisions of the Part 50 license. Subsequent to removal of all the spent fuel from the spent fuel pool, and safe emplacement in the MPCs at the ISFSI storage pad, PGE will continue the decommissioning process. The decommissioning process will lead to the termination of the Part 50 license.

In addition to the limited 10 CFR Part 50 license, the Trojan nuclear facility also has a 10 CFR Part 72 license, although authority to load and store the fuel is suspended at the time of writing this licensing document. The suspension of authority to load and store spent fuel is due to coating problems with the TranStor Storage System and is unrelated to the need to change the Technical Specification. Pending NRC approval of Revision 1 to the Trojan ISFSI SAR and Technical Specifications, Trojan will have the authority to proceed with the spent fuel loading campaign, under Parts 50 and 72. Once the Part 50 license is terminated, the Trojan ISFSI will become a stand-alone dry spent fuel storage facility. All operations associated with the storage and handling of the spent fuel at the ISFSI storage site will be licensed under Part 72.

During a Quality Assurance surveillance of the ISFSI, it was not evident how the Trojan ISFSI Technical Specification (ITS) 4.2.3 would be implemented. As a result of the QA surveillance, a Corrective Action Request, CAR C01-0006 (Ref. 7) was written. The concern in CAR C01-0006 is to clarify the intent of PGE to use a mobile crane with a safety factor of two times the rated payload of the crane. Additionally the licensing basis for the mobile crane needs to be changed.

3. 10 CFR PART 72 LICENSING REQUIREMENTS FOR LIFTING HEAVY LOADS

Specific guidance for 10 CFR Part 72 operations is given in Chapter 5 of NUREG-1567, "Standard Review Plan (SRP) for Spent Fuel Dry Storage Facilities". In the SRP, it is stated that the handling of heavy loads such as spent fuel storage casks or spent fuel shipping casks shall conform to the guidance in NUREG-0612.

NUREG-1567, discusses structures, systems and components (SSCs) associated with the transfer of confinement and transfer casks on site, including cranes. However, the NUREG does not identify mobile cranes specifically. As stated in Section 5.5.4.1 of NUREG-1567, "Other SSCs important to safety... may include: SSCs associated with on-site inter-modal transfer of nuclear material containers, such as cranes used at truck, rail and barge/ship docks [note: The cranes may not be important to safety depending on acceptable safety analysis.]" Because the failure of a mobile crane to lift an MPC at the Transfer Station would not jeopardize the basic safety functions of the confinement system or other basic safety criteria, the mobile crane is not considered important to safety. Here a distinction is made between the crane, which supports the hoisting system, and the special lifting devices, which connect the nuclear material container to the hook of the hoisting system and are used to make the lift. Special lifting devices are important to safety and are designed, fabricated, and tested in accordance with guidance in ANSI N14.6.

NUREG-0612 was written for 10 CFR Part 50 licensees, which differ in significant ways from Part 72 licensees. A prime consideration of a Part 50 crane is to preclude the drop of a heavy load by means of various mechanical and electrical safety features in order to prevent failure of safety-related equipment in the plant. In the case of a Part 72 licensee, the heavy load is the confinement boundary and any collateral damage to the Transfer Station, the shipping cask or the Concrete Cask would not result in a loss of

confinement. In other words, the prime consideration for lifting heavy loads by a Part 72 licensee is to preclude loss of confinement. Shielding protection is another major difference between Part 72 and Part 50 facilities. Operations by a Part 72 licensee must ensure continuous shielding protection. Thus all the components designed with a shielding function including the Transfer Cask, the Concrete Cask and/or a shipping cask need to be in place during transfer operations to ensure conformance with ALARA commitments.

4. TROJAN ISFSI MOBILE CRANE AND OPERATING CONSIDERATIONS

One of the operational requirements of the Part 72 license is fuel retrievability, which is discussed in 10 CFR 72.122(l) and further clarified in an NRC Interim Staff Guidance document, ISG-2, Fuel Retrievability (Ref. 8). PGE is in compliance with 10 CFR 72.122(l) by providing a method to repackage the MPC into a 10 CFR Part 71 transportation cask (e.g. the NRC–licensed HI-STAR 100) for shipment offsite for further processing, or disposal.

The method PGE has selected to meet fuel retrievability requirements involves the use of a mobile crane at the Transfer Station. The type of mobile crane that will be used is classified in ANSI B30.5 as a "wheel-mounted crane (multiple control stations)." The crane consists of a rotating superstructure, operating machinery, and operator's station and boom. The crane equipment is mounted on a crane carrier equipped with axles and rubber-tired wheels for travel, and on-board power source (the diesel engine). The crane has separate stations for driving the vehicle and for controlling crane-lifting operations. In function the mobile crane can lift and lower a load using the hoist mechanism, and it can swing a load at various radii using the rotating capability of the superstructure.

The boom on a hydraulic crane is extended and raised by separate hydraulic cylinders and is not supported by cable or rod. Because the boom is cantilevered over the lift cylinder, the boom sections are heavy structural cross sections. Failure of the boom hydraulics would only allow the boom to settle back in (telescoping action) and down (lowering action). However failure is not likely because standard industry practice for cranes, including the hydraulics, involve regular inspection and testing.

Load ratings and load rating charts are developed according to the requirements of ANSI B30.5 or equivalent. There are two basic criteria, which the manufacturer must satisfy to arrive at the load rating charts: stability, and structural competence.

For load rating where stability governs lifting performance the following applies. The margin of stability for determining load ratings with booms of stipulated lengths, at stipulated radii for the wheel mounted cranes with outriggers fully extended and locked and tires off the supporting surface, is 85% of the load that will produce an unstable condition with the boom in the least stable direction relative to the mounting.

For load ratings where structural competence governs lifting performance one of two considerations is involved. The structural competence or structural integrity of the individual mechanical and/or hydraulic components limits the load rather than overall stability of the machine. A non-symmetrical mounting may require a higher loading to produce a condition of tipping in a direction other than the least stable. For this second case, the structural integrity of the individual mechanical and/or hydraulic components limits the load rather than the overall stability of the machine.

The Transfer Cask consists of a cylinder with moveable shield doors at the lower end and a top cover. The moveable shield doors at the lower end allow lowering of the MPC into the Concrete Cask or the shipping cask. The doors slide in steel guides along each side of the Transfer Cask. The top cover of the Transfer Cask extends over the MPC to prevent it from being inadvertently lifted out of the top of the Transfer Cask.

A mobile crane will be used to perform a non-critical (failure will not result in radiological consequences or affect equipment important to safety) lift of an empty Transfer Cask on top of the Transfer Station. The Transfer Station provides a secure platform to locate the Transfer Cask during the MPC transfer operation. A collar fixes the Transfer Cask securely to the Transfer Station. The mobile crane will then be used to perform a straight vertical lift of an MPC from a Concrete Cask into the Transfer Cask. It will also be used to lower an MPC from the Transfer Cask down into a Shipping Cask for shipment off-site or into a Concrete Cask. An impact limiter is embedded in the Transfer Station foundation mat supporting the Concrete Cask or the Shipping Cask, to reduce the decelerations of the MPC in the unlikely event of a vertical drop of the MPC into the Shipping Cask or Concrete Cask. A step-by-step description of the operation follows.

Using the lifting hoist of the mobile crane, the crane operator will lift the Transfer Cask to an elevation sufficient to clear the platform within the Transfer Station. Using the hydraulic boom of the mobile crane, the crane operator will position the Transfer Cask over the support platform of the Transfer Station, and then lower the Transfer Cask to the platform. Lateral frames are then installed at the approximate center of gravity of the loaded Transfer Cask and hydraulic cylinders and hardware for opening the bottom doors on the Transfer Cask are installed. The Transfer Cask doors are then opened.

Prior to performing a vertical lift of the MPC, the mobile crane will be positioned on the transfer pad adjacent to the Transfer Station and the outriggers fixed in place. The hydraulic boom will be positioned such that the lifting hook is directly over the top cover of the Transfer Cask. Once the mobile crane is in position, the movement of the hook will be limited to vertical movement only except for minor horizontal adjustments as the MPC is raised or lowered.

A Concrete Cask loaded with an MPC is prepared for placement in the Transfer Station. The cask lid and shield ring are removed and MPC rigging hardware and slings are attached to the lifting cleats on the MPC lid. The Concrete Cask is then moved under the Transfer Cask using air pads. Shielding is placed in any gaps or potential streaming paths utilizing both installed shield support frames and temporary hangers or wires.

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

The MPC is raised into the Transfer Cask sufficiently to clear the doors. A load cell, load computer, or equivalent, installed on the mobile crane will be used in conjunction with the Transfer Cask lid to prevent the MPC from being raised higher than the top of the Transfer Cask. The doors are closed and the MPC lowered to rest on the doors.

In order to move the Concrete Cask away from the Transfer Station, sufficient shielding is removed from the Transfer Station. Using air pads, operators will relocate the Concrete Cask to an appropriate location on the Storage Pad. The Shipping Cask, which for off-site shipping purposes will be a HI-STAR 100 Shipping Cask, is then moved into position below the Transfer Cask using air pads. Shielding is placed to

prevent streaming. The MPC is then raised sufficiently to just clear the doors, the doors opened, and the MPC lowered into the shipping cask.

Canister rigging is disengaged from the mobile crane hook and lowered to the MPC lid. The Shipping Cask is then moved from the Transfer Station, the MPC rigging is removed, and Shipping Cask lid(s) or other closures are placed.

5. BASIS FOR TROJAN ISFSI MOBILE CRANE COMPLIANCE WITH NRC LICENSING REQUIREMENTS

The use of mobile cranes at an ISFSI is not specifically identified in NUREG-1576, the Standard Review Plan for spent fuel dry storage facilities such as Trojan. However NUREG-1567 states that the use of systems, structures and components, (SSCs), including cranes, for on-site inter-modal transfer of nuclear containers, and the transfer of confinement and transfer casks shall conform to the guidance of NUREG-0612. Although this guidance was developed to address necessary licensing changes to assure the safe handling of heavy loads at nuclear power plants, the recommendations in the NUREG are appropriate to the safe handling of heavy loads at an ISFSI.

Section 5 of NUREG-0612 recommends specific guidelines for the control of heavy loads. Guidance is based on a defense-in-depth approach, which includes the two primary objectives of preventing an accidental load drop, and mitigating the consequences of the postulated drop. These two objectives are discussed separately below.

5.1. Prevention of Accidental Drop: Compliance with NUREG-0612

Trojan complies with the guidance of Section 5.1.1 in NUREG-0612 for the mobile crane by establishing high levels of reliability of certain active components of the handling system through increased factors of safety as outlined in Section 5.1.6. Specifically:

- The Trojan mobile crane shall conform to the guidelines of Section 5.1.1 of NUREG-0612 with the exception that mobile cranes shall meet the requirements of ANSI B30.5, "Mobile and Locomotive Cranes," or equivalent, instead of ANSI B30.2, "Overhead and Gantry Cranes," (Ref. 9). Although not a special lifting device, the mobile crane to be used *shall have a minimum safety factor of two over the allowable load from the table for the crane* and shall be capable of stopping and holding the load during the DBE.

  "Special lifting devices," as defined in ANSI N14.6-1993 *shall have two times the design safety factors* in accordance with the guidance of Section 5.1.6 (1)(a) of NUREG-0612, and Section 7.2 of ANSI N14.6-1993. The lifting cleats attached to the top of the Trojan MPC are designed with safety factors of 6 on yield strength and 10 on ultimate strength, including dynamic effects.

  Slings and the wire rope to be used in the vertical lift at Trojan are designed with safety factors of 10 on ultimate strength, including dynamic effects, in accordance with the guidelines for slings in ANSI/ASME B30.9-1984, "Slings".

- Section 5.1.6 (2) of NUREG-0612 specifies that new cranes should be designed to meet the guidance of NUREG-0554 (Ref. 10). For mobile cranes, this guidance is not applicable because the mobile crane does not supplant the overhead or gantry crane for 10 CFR Part 50 operations.

- In accordance with the guidance in Section 5.1 (1) of NUREG-0612, PGE will assure that mobile crane operators will be properly trained in mobile crane operations. The training will include knowledge of the handling system design, load handling instructions, and equipment inspection to assure reliable operation of the handling system at the Transfer Station.

  - Mobile crane operators are trained and qualified in accordance with the standard for mobile cranes, ANSI B30.5, "Mobile and Locomotive Cranes." ANSI B30.5 contains equivalent requirements to those contained in ANSI B30.2, "Overhead and Gantry Cranes." Appendix 1 contains an item by item comparison of the requirements of these standards. This comparison shows that all the requirements for mobile cranes in ANSI B30.5 are equivalent and/or identical with the relevant requirements in ANSI B30.2.

  - Mobile crane operators will be knowledgeable of the safe load paths and restricted load handling areas in accordance with plant procedures. Transfer Station operations, which include lifting the empty Transfer Cask onto the Transfer Station and lifting or lowering the MPC, will be limited to personnel who are trained and certified in accordance with the Certified ISFSI Specialist Training Program, or these activities will be under the direct visual supervision of a person who is trained and certified in accordance with the Certified ISFSI Specialist Training Program (PGE-1072) pursuant to Technical Specification 5.3.1.

  - The mobile crane will comply with ANSI B30.5 for inspection, testing and maintenance of mobile cranes. Both Federal and State safety regulations impose ANSI B30.5 for the use of mobile cranes.

- In accordance with the guidance in Section 5.1 (2) of NUREG-0612, PGE will define safe load travel paths through procedures and operator training so that heavy loads are not carried over or near irradiated fuel or safe shutdown equipment.

  - Heavy loads will not be handled over fuel storage casks or baskets containing irradiated fuel at the Trojan ISFSI. The only anticipated heavy lifts associated with the operation of the Trojan ISFSI are the placement/removal of the empty Transfer Cask into position on the Transfer Station, vertical lifts of the MPC within the Transfer Station, and the handling of Shipping Casks. Movement of Storage Casks between the storage locations and the Transfer Station is accomplished using an air pad system as described in the ISFSI SAR Section 5.1.1.3, "Transfer to Storage Area Operation," and does not involve crane lifts.

  - Procedures will preclude the movement of the empty Transfer Cask over loaded fuel storage casks during the placement or removal of the empty Transfer Cask from the Transfer Station.

  - Changes to ISFSI procedures are controlled as described in Section 9.4.1 of the SAR. Changes to ISFSI procedures are reviewed to ensure that they do not involve any safety issues that require prior NRC approval as defined in 10 CFR 72.48 and required evaluations are subsequently reviewed by the PGE Trojan ISFSI Safety Review Committee.

  - ISFSI procedures will preclude handling of heavy loads over the loaded Concrete Casks. The use of the mobile crane and lifting the MPC at the Transfer Station are included in the scope of Trojan ISFSI Procedure TIP-10, "Transfer Cask and Concrete Cask Handling and Storage Program," which includes requirements for load handling procedures. Implementing procedures

include appropriate requirements such as the identification of required equipment, inspection and acceptance criteria required before movement of load, steps and proper sequence to be followed in handling the load, and defining the safe load path.

- In accordance with the guidance in Section 5.1 (3) of NUREG-0612, PGE will assure that mechanical stops or electrical interlocks prevent the movement of heavy loads over irradiated fuel. Only minimal horizontal movement of the MPC is possible during MPC lifts at the Transfer Station due to the geometry of the Transfer Cask, Concrete Cask, and Shipping Cask.

  - Although not required for compliance with the guidelines of NUREG-0612, PGE performed an evaluation of a representative mobile crane to quantify the available factor of safety during the lift of a PWR basket. PGE acknowledges the fact that future use of a mobile crane will be to prepare an MPC for off-site shipment and not to handle a PWR Basket. The MPC is 3,400 pounds heavier than the PWR basket, and so the crane load rating must be modified accordingly. Also the type of mobile crane that PGE used for the evaluation, a Liebherr LTM 1225, may not be identical to the type selected in the future. However, if the LTM 1225 is taken as representative, with the hydraulic telescoping boom extended to the 67 feet position, and the crane slewing platform (i.e., the rotational superstructure) located approximately 32 feet south of the Transfer Station north-south centerline, the rated capacity of the crane would be 159,000 pounds.

  - For an MPC lift, a mobile crane configuration (boom length, boom angle, counterweights, etc.) would be selected such as to provide a rated load capacity of at least twice the combined weight of the MPC, load block, rigging, and MPC lifting yoke.

  - In the case of the Liebherr LTM 1225 crane, an overload safety device is designed to prevent lifting more than the rated capacity for the actual crane configuration. The safety device consists of an electronic module (power pack and microprocessor unit) and monitor with controls, pressure sensors, and incline angle and length sensors. The safety device compares actual load with the allowable load as calculated from the measured values from boom length, angle and pressure sensors.

    The safety device provides a warning when the load reaches 90% of the maximum load carrying capacity and shuts down the crane when the maximum permissible load rating is reached or sensor problems are detected that could lead to an unsafe condition. If the crane shuts down due to an overload condition, procedures must be carried out to reduce the load, e.g. setting the load down.

    The overload safety device would also function to preclude any inadvertent lifting of the combined Transfer Cask and MPC from the Transfer Station. The rated capacity of the representative crane in the particular configuration during the MPC lift (159,000 pounds) is less than the combined weight of the Transfer Cask and the MPC (combined weight of 197,000 pounds). Therefore the overload safety device would shut the crane down prior to any inadvertent lifting of both components.

  - Additional safety features include a hoist limit switch, limit switches in the hoist winches, emergency off button, control release switch, slewing platform lock, and hydraulic safety valves. The hoist limit switch monitors the distance from the hook block to the boom pulley head and prevents the hook block from contacting the pulley head. Winch limit switches prevent too few

or too many cable loops on the winch drum. The crane slewing lock prevents any unintentional platform slewing movements. Acoustic signals are provided to alert the operator of approaching limits on wind velocity, crane incline, and outrigger support force.

5.2. *Mitigation of Consequences of a Postulated MPC Drop*

Radiological consequences resulting from an accidental drop of the MPC are mitigated because the MPC has been shown not to breach and therefore maintains the confinement boundary (Ref. 11). Reference 11 demonstrates that a generic MPC weighing 90,000 pounds does not breach if it is dropped 25 feet vertically onto an essentially rigid concrete pad. The following conservative assumptions are made:

- the 90,000-pound weight envelops both the Trojan MPC-24E and MPC-24EF canisters;
- the analytical model assumes an MPC drop directly onto a concrete target pad, whereas kinetic energy would actually be partitioned into the base of the Concrete Cask, and into the impact limiter which is embedded in the Transfer Station pad at the Trojan site;
- the analytical model assumes that the contents of the MPC form a rigid mass, whereas the fuel basket and stored spent fuel would also absorb some of the kinetic energy;
- the 25-foot drop height of the analysis is greater than the 20.75-foot (249-inch) distance between the bottom of the MPC-24E and the bottom of the Concrete Cask.

The computer code used in the analysis of Reference 11 is LS-DYNA (Ref. 12). The results of the finite element analysis for the generic 90,000-pound MPC drop accident show that the maximum calculated plastic strain of the confinement boundary is 15.5%. This strain is less than the maximum plastic strain at failure of 38%. Because the maximum calculated plastic strain of 15.5% is less than the maximum plastic failure strain of 38%, it is concluded that the postulated drop accident of a loaded MPC will not result in a breach of confinement.

6. SUMMARY AND CONCLUSIONS

As shown in the evaluation above, the use of a mobile crane for lifting the MPC within the ISFSI Transfer Station is in compliance with the guidelines of the relevant sections of NUREG-1567 and NUREG-0612. NUREG-1567 states that the use of systems, structures and components, (SSCs), including cranes, for on-site inter-modal transfer of nuclear containers, and the transfer of confinement and transfer casks shall conform to the guidance of NUREG-0612. PGE has demonstrated that the failure of a mobile crane to lift an MPC at the Transfer Station would not jeopardize the basic safety functions of the confinement system or other basic safety criteria, and therefore the mobile crane is not important to safety.

The mobile crane is in compliance with the guidance in Section 5.1.1 in NUREG-0612 because the two main objectives of preventing an accidental drop of a load and mitigating the consequences of the postulated drop are achieved. PGE has improved the reliability of certain active components of the handling system through increased factors of safety as outlined in Section 5.1.6 of NUREG-0612, specifically:

6.1. *Prevention of an Accidental MPC Drop*

The Trojan mobile crane shall conform to the guidelines of Section 5.1.1 of NUREG-0612 with the exception that mobile cranes shall meet the requirements of ANSI B30.5, "Mobile and Locomotive Cranes," or equivalent, instead of ANSI B30.2, "Overhead and Gantry Cranes". A mobile crane to be used in the MPC lift configuration shall have a load rating of at least twice the combined weight of the MPC, load blocking rigging, and MPC lifting yoke, and shall be capable of stopping and holding the load during the DBE.

"Special lifting devices," as defined in ANSI N14.6-1993 shall have two times the design safety factors of Section 4.2.1.1 of ANSI N14.6-1993in accordance with the guidance of Section 5.1.6 (1)(a) of NUREG-0612, and Section 7.2 of ANSI N14.6-1993. The lifting cleats attached to the top of the Trojan MPC are designed with safety factors of 6 on yield strength and 10 on ultimate strength, including dynamic effects.

Slings and the wire rope to be used in the vertical lift at Trojan are designed with safety factors of 10 on ultimate strength, including dynamic effects, in accordance with the guidelines for slings in ANSI/ASME B30.9-1984, "Slings".

In accordance with the guidance in Section 5.1 (1) of NUREG-0612, PGE will assure that mobile crane operators will be properly trained in mobile crane operations. The training will include knowledge of the handling system design, load handling instructions and equipment inspection to assure reliable operation of the handling system at the Transfer Station.

In accordance with the guidance in Section 5.1 (2) of NUREG-0612, PGE will define safe load travel paths through procedures and operator training so that heavy loads are not carried over or near irradiated fuel or safe shutdown equipment.

In accordance with the guidance in Section 5.1 (3) of NUREG-0612, PGE will assure that mechanical stops or electrical interlocks prevent the movement of heavy loads over irradiated fuel. Only minimal horizontal movement of the MPC is possible during MPC lifts at the Transfer Station due to the geometry of the Transfer Cask, Concrete Cask, and Shipping Cask.

6.2. Mitigation of Consequences of a Postulated MPC Drop

In accordance with the guidance in Section 5.1 of NUREG-0612, radiological consequences resulting from a postulated drop of the MPC are mitigated because the MPC has been shown not to breach and therefore maintains the confinement boundary. The analysis demonstrates that a generic MPC weighing 90,000 pounds does not breach if it is dropped 25 feet vertically onto a 22-foot thick concrete pad. These parameters envelope the Trojan ISFSI because, the MPC-24E and MPC-24EF weigh less than 90,000 pounds; the maximum drop height that an MPC could fall is 20.75 feet; and the target is softer than the modeled concrete pad because there is an impact limiter embedded in the Transfer Station pad.

The computer code used in the analysis is LS-DYNA. The results of the finite element analysis for the generic 90,000-pound MPC drop accident show that the maximum calculated plastic strain is 15.5% This strain is less than the maximum plastic strain at failure of 38%. Because the maximum calculated plastic strain of 15.5% is less than the maximum plastic failure strain of 38%, it is concluded that the postulated drop accident of a loaded MPC will not result in a breach of confinement.

7. REFERENCES

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4. American National Standards Institute, American Society of Mechanical Engineers, ANSI/ASME B30.5, "Mobile and Locomotive Cranes," 1994
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11. Holtec International, "Evaluation of the Confinement Integrity of a Loaded Holtec MPC under a Postulated Drop Event", HI-2002572, Rev. 0, November, 2000.
12. Livermore Software Technology Corporation, LS-DYNA3D, Version 950, 1999.

8. APPENDIX 1 Comparison of Requirements of ANSI B30.2 with ANSI B30.5

B30.2 -- Overhead and Gantry Cranes	B30.5 -- Mobile and Locomotive Cranes	Comments
Introduction	**Introduction**	
General	General	Identical -- The Introductory sections of both standards describe the overall structure of the B30 series of standards, the process for submitting suggested changes, and the purpose and scope of the B30 series.
Section I -- Scope	Section I -- Scope	
Section II -- Purpose	Section II -- Purpose	
Section III -- Interpretation	Section III -- Interpretation	
Section IV -- New and Existing Installations	Section IV -- New and Existing Installations	
Section V -- Mandatory and Advisory Rules	Section V -- Mandatory and Advisory Rules	
Section VI -- Metric Conversions	Section VI -- Metric Conversions	
Chapter 2-0 -- Scope, Definitions, and References	**Chapter 5-0 -- Scope, Definitions, and References**	
Section 2-0.1 -- Scope of B30.2	Section 5-0.1 -- Scope of B30.5	Equivalent -- These sections define the scope of the respective standards.
Section 2-0.2 -- Definitions	Section 5-0.2 --Definitions	Equivalent -- These sections provide definitions of specific terms used in the respective standards.
Section 2-0.3 -- References	Section 5-0.3 -- References	Equivalent -- These sections provide appropriate reference documents for each standard.

B30.2 -- Overhead and Gantry Cranes	B30.5 -- Mobile and Locomotive Cranes	Comments
Chapter 2-1 -- General Construction and Installation	**Chapter 5-1 -- Construction and Characteristics**	
Section 2-1.1 -- Markings	Section 5-1.1 -- Load Ratings Section 5-1.2 -- Stability (Backward and Forward) Section 5-1.10 -- Structural Performance	B30.2 specifies that the maximum load rating of the crane be marked on each side of the crane. B30.5 contains substantially more detail related to the load ratings and stability of mobile cranes. Since the load rating of a mobile crane varies with boom angle and direction, the standard specifies that loads not exceed 75-85%, depending on the crane configuration, of the load that could cause tipping of the crane. B30.5 addresses the use of load rating charts to determine the crane capacity in each configuration of boom angle, extension, and direction.
Section 2-1.2 -- Clearances	No corresponding section.	B30.2 addresses required design clearances between an overhead or gantry crane and adjacent buildings or parallel running cranes. This in not applicable to mobile cranes since they are not permanently installed in a fixed location.
Section 2-1.3 -- General Construction - Runways and Supporting Structure	No corresponding section.	B30.2 addresses the crane rail system and anchorages, this is not applicable to mobile cranes.
Section 2-1.4 -- Crane Construction	Section 5-1.9.5 -- General Requirements -- Welded Construction	Equivalent -- B30.2 specifies requirements for steel girders and welded construction. B30.5 specified equivalent requirements for welded construction. Requirements related to steel girders are not applicable to mobile cranes.
Section 2-1.5 -- Cabs - Normal or Skeleton (If Provided)	Section 5-1.8 -- Cabs	Equivalent -- Both standards specify appropriate requirements related to the design and construction of the crane cab.
Section 2-1.6 -- Lubrication	Section 5.1.9.11 -- General Requirements -- Lubricating Points	Equivalent -- Both standards address requirements related to accessibility of lubrication points.

B30.2 -- Overhead and Gantry Cranes	B30.5 -- Mobile and Locomotive Cranes	Comments
Section 2-1.7 -- Service Platforms (Footwalks)	No corresponding section.	B30.2 addresses requirements for service platforms that may be provided to allow access to equipment mounted on the overhead or gantry crane. Such platforms are not typically used on mobile cranes.
Section 2-1.8 -- Stops and Bumpers	No corresponding section.	B30.2 addresses the use of trolley stops and bridge bumpers, these are not applicable to mobile cranes.
Section 2-1.9 -- Rail Sweeps	No corresponding section.	Not applicable to mobile cranes
Section 2-1.10 -- Guards for Moving Parts	Section 5-1.9.6 -- General Requirements -- Guards for Moving Parts	Equivalent -- Both standards address requirements and specifications for guards over moving parts.
Section 2-1.11 -- Wheel and Truck Frames	No corresponding section.	B30.2 requires a means to limit the drop of trolley and bridge truck frames in the case of wheel or axle breakage. This requirement is not applicable to mobile cranes since they do not use trolley or bridge trucks.
Section 2-1.12 -- Brakes	Section 5-1.3 -- Boom Hoist, Load Hoist, and Telescoping Boom Mechanisms Section 5-1.4 -- Swing Mechanism Section 5-1.5 -- Crane Travel Section 5-1.6 -- Controls	B30.2 specifies requirements for various brakes in Section 2-1.12. Equivalent requirements are contained in B30.5 in the various sections covering specific components. The referenced sections of B30.5 also contain a number of additional requirements unique to mobile cranes.
Section 2-1.13 -- Electrical Equipment	Section 5-1.6 -- Controls	This section of B30.2 specifies requirements for electrical systems of overhead and gantry cranes. Mobile cranes are not

B30.2 -- Overhead and Gantry Cranes	B30.5 -- Mobile and Locomotive Cranes	Comments
		electrically powered and these requirements are not applicable to mobile cranes. Section 2-1.13 of B30.2 also specifies suggested control layouts for these cranes. Corresponding specification for control layouts are found in Section 5-1.6 of B30.5.
Section 2-1.14 -- Hoisting Equipment	Section 5-1.7 -- Ropes and Reeving Accessories	Equivalent -- Both standards address requirements related to the design and construction of ropes, sheaves, reeving accessories, and load hooks. B30.2 also addresses the use of lifting magnets which are not typically used with mobile cranes.
Section 2-1.15 -- Warning Devices or Means for a Crane With a Power Traveling Mechanism	Section 5-1.9.12(c) -- General Requirements -- Miscellaneous Equipment	Equivalent -- Both standards require the crane be equipped with an audible signal device.
No corresponding section.	Section 5-1.9 -- General Requirements	This section of B30.5 contains a number of requirements unique to mobile cranes (e.g., booms, outriggers, exhaust gas, etc.). Other subsections of 5-1.9 that address items common to the two standards are referenced above.
Chapter 2-2 -- Inspection, Testing, and Maintenance	**Chapter 5-2 -- Inspection, Testing, and Maintenance**	
Section 2-2.1 -- Inspection	Section 5-2.1 -- Inspection	Equivalent -- Both standards require an initial inspection to verify compliance with the standard. Both standards specify items to be covered in "frequent inspections", (i.e., daily to monthly intervals) and "periodic inspections", (i.e., monthly to yearly intervals). Both standards also address inspections for cranes that are not in regular use and requirements for inspection record keeping.
Section 2-2.2 -- Testing	Section 5-2.2 -- Testing	Equivalent -- Both standards require an operational test of

B30.2 -- Overhead and Gantry Cranes	B30.5 -- Mobile and Locomotive Cranes	Comments
		new cranes and specify a "rated load" test for new, repaired or altered cranes. The test load is limited to 125% of the crane rating for overhead and gantry cranes and 110% of the crane rating for mobile cranes.
Section 2-2.3 -- Maintenance	Section 5-2.3 -- Maintenance	Equivalent -- Both standards require a preventative maintenance program, specify important precautions for maintenance procedures, and address adjustments and repairs. Both standards also address proper lubrication of moving parts.
Section 2-2.4 -- Rope Inspection, Replacement, and Maintenance	Section 5-2.4 -- Rope Inspection, Replacement, and Maintenance	Equivalent -- Both standards address "frequent inspections" of running ropes. B30.5 provides additional guidance on inspections of boom hoist ropes and other features unique to mobile cranes. Both standards also address "periodic inspections" of the entire length of the running ropes, rope maintenance, and criteria for rope replacement.
Chapter 2-3 -- Operation	**Chapter 5-3 -- Operation**	
Section 2-3.1 -- Qualifications for and Conduct of Operations	Section 5-3.1 -- Qualifications for and Conduct of Operators and Operating Practices	Equivalent -- Both standards address the personnel who may operate cranes, the qualifications of crane operators (including equivalent requirements for experience and physical qualifications). Both standards also address appropriate operating practices to ensure safe crane operation.
Section 2-3.2 -- Handling the Load	Section 5-3.2 -- Operating Practices	Equivalent -- Both standards address maximum loads, attaching loads, and movement of loads. B30.2 allows a limited number of overrated lifts ("engineered lifts"), while such lifts are not allowed by B30.5 for mobile cranes. B30.5 also includes additional requirements and restrictions unique to mobile cranes addressing such issues as boom movements,

B30.2 -- Overhead and Gantry Cranes	B30.5 -- Mobile and Locomotive Cranes	Comments
		the use of outriggers, and personnel lifting.
Section 2-3.3 -- Signals	Section 5-3.3 -- Signals	Equivalent -- Both standards specify standard hand signals for communication with the crane operator.
Section 2-3.4 -- Miscellaneous	Section 5-3.4 -- Miscellaneous	Equivalent -- Both standards address such issues as use of ladders, fire extinguishers, and items to be stored in the crane cabs. B30.5 addresses additional items unique to mobile cranes such as ballast or counterweights, refueling, etc.

9. APPENDIX 2 Suggested Modification to Technical Specification 4.2.3

Storage Pad, Service Pad and TRANSFER STATION

- CONCRETE CASKS must have a nominal center-to-center 15 feet spacing with a tolerance of ± 4 inches.
- Operations at the TRANSFER STATION which involve lifts of an MPC must be performed using a mobile crane, which shall meet the guidance of Section 5.1.1 of NUREG-0612, except that to assure defense-in-depth:
  1. The mobile crane shall have a minimum safety factor of two over the allowable load for the crane in accordance with the guidance of NUREG-0612.
  2. In accordance with the guidance of NUREG-0612, the mobile crane must have the ability to safely stop and hold the MPC in the event of a safe shutdown earthquake applicable to the Trojan ISFSI.
  3. The mobile crane shall meet the requirements of ANSI B30.5, "Mobile and Locomotive Cranes," or equivalent, in lieu of the requirements of ANSI B30.2, "Overhead and Gantry Cranes".
  4. The MPC will be restricted to a lift height of 249 inches (bottom of raised MPC in the HI-TRAC to bottom of CONCRETE CASK or shipping cask), when being lifted by the mobile crane in the TRANSFER STATION by the physical limitation of the top of the HI-TRAC. A load cell on the mobile crane will indicate contact with the top of the HI-TRAC to limit the lift height of the MPC.
  5. Special Lifting Devices as defined in ANSI N14.6-1993 shall have two times the design safety factors of Section 4.2.1.1 in accordance with Section 7.2. These special lifting devices include the lifting cleats.
  6. Lifting Devices that are not specifically designed and that are used for handling heavy loads shall meet the requirements of ANSI B30.9, "Slings," except that the load to be used in selecting the slings is to be twice that specified in NUREG-0612 Section 5.1.1(5).
- Movements of the CONCRETE CASK are performed using an AIR PAD System, which restricts the lifting height of the CONCRETE CASK to four inches or less.

TROJAN ISFSI

Proposed Revision
to the
Safety Analysis Report

LCA 72-02
Enclosure 2
Attachment B

ENCLOSURE 2

ATTACHMENT B

to VPN-044-2001

Trojan ISFSI

LCA 72-02

Annotated Pages of the Trojan ISFSI

Safety Analysis Report (SAR)

Chapter 1

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

1.1 INTRODUCTION

The Trojan Nuclear Plant (TNP) was operated for approximately 17 years and was shut down for the last time on November 9, 1992. The plant is jointly owned by Portland General Electric Company (PGE), the City of Eugene through the Eugene Water and Electric Board (EWEB), and Pacific Power and Light/PacifiCorp. PGE is the principal owner and has responsibility for maintaining TNP.

PGE's plans to decommission TNP include prompt decontamination and dismantlement of contaminated structures, systems and components. In order to facilitate decontamination and dismantlement, the contents of the Spent Fuel Pool will be relocated to an Independent Spent Fuel Storage Installation (ISFSI). Use of an ISFSI was determined to be the most economical method for the temporary storage of the TNP spent fuel until a Department of Energy (DOE) or other offsite facility is available. Relocation of the spent fuel and associated radioactive material stored in the pool to the ISFSI would allow decontamination and dismantlement of structures, systems, and components throughout TNP to proceed without impacting the safe storage of the spent fuel.

Activities involving loading of the spent nuclear fuel assemblies into the ~~PWR Basket~~*Holtec International Multi-Purpose Canister (MPC-24E or MPC-24EF; hereafter referred to simply as "MPC" when specification of the particular canister type is unnecessary)*, closure of the ~~PWR Basket~~*MPC,* and placement of the ~~PWR Basket~~*MPC* into a Concrete Cask are considered licensed activities per PGE's 10 CFR Part 50 license and are not described in detail in this Safety Analysis Report *(SAR)*. This ~~Safety Analysis Report~~*SAR* is primarily focused on ISFSI operations, including ultimate transfer to a ~~Shipping~~ *Transport* Cask. However, certain restrictions related to the ~~PWR Basket~~*MPC* loading and sealing operations are included in the ISFSI Technical Specifications.

TNP is located in Columbia County, Oregon, along the west bank of the Columbia River, approximately 42 miles north of Portland, Oregon. Figure 1.1-1 shows the location of TNP. The ISFSI will be located in the northeast portion of the TNP site, as shown in Figure 1.1-2.

1.2 GENERAL DESCRIPTION OF THE INSTALLATION

PGE ~~has selected~~*will use* BNFL Fuel Solutions TranStor™ ~~Storage System~~ *Concrete Casks loaded with Holtec International MPC-24Es or MPC-24EFs at*~~for~~ the Trojan ISFSI. *The MPC-24E is designed to accommodate up to 24 pressurized water reactor (PWR) fuel assemblies. Up to four of the fuel assemblies in any one MPC-24E may be classified as damaged fuel and the balance must be intact fuel. The MPC-24EF is also designed to accommodate up to 24 PWR fuel assemblies. Up to four of the fuel assemblies in any one MPC-24EF may be classified as damaged fuel or fuel debris, and the balance must be intact fuel.*

~~The TranStor™~~*This* ~~S~~storage ~~S~~system is a vertical dry storage system ~~which~~ *that* utilizes a ventilated Concrete Cask and a seal-welded *stainless* steel ~~PWR Basket~~*MPC* to safely store spent nuclear fuel assemblies and fuel debris. *Hereafter in this SAR, this storage system will be referred to as the Trojan Storage System.*

The ISFSI consists of a reinforced concrete Storage Pad, supporting ~~a maximum of 36 BNFL Fuel Solutions TranStor™~~ *34 Trojan* Storage Systems designed to safely store intact spent fuel assemblies, *fuel assembly inserts and metal fragments (e.g., portions of fuel rods and grid assemblies, bottom nozzles, etc.),* assemblies containing damaged fuel, or fuel debris. The storage system is passive and requires minimal surveillances. Significant radioactive waste generation is not anticipated as a result of ISFSI operation.

The system is designed to permit transfer of the ~~PWR Basket~~*MPC* to a ~~Shipping~~ *Holtec HI-STAR 100 System Transport* Cask *(CoC 71-9261)* once a repository or other facility is available. ~~The TranStor™ Shipping Cask is being licensed separately.~~ The system is also designed to accommodate recovery from postulated off-normal events without reliance on the Spent Fuel Pool.

The principal design criteria are discussed in Chapter 3. Chapter 4 discusses the design of the ISFSI.

1.3 GENERAL SYSTEMS DESCRIPTION

The storage system consists of the ~~PWR Baskets~~*MPCs*, Concrete Casks, Storage and Service Pads, Transfer Station, and associated transfer equipment necessary for safe placement of spent nuclear fuel assemblies, *fuel assembly inserts,* and fuel debris into dry storage. The following sections provide an overview of the primary components used during storage. Figure 1.3-1 provides an overview of the ~~PWR Basket~~*MPC* and Concrete Cask.

1.3.1 STORAGE SYSTEM ~~BASKETS~~*CANISTERS*

~~PWR Baskets~~*MPCs* are metal *fuel storage* containers that are seal-welded closed and serve as the confinement boundary for the materials stored within the ~~PWR Baskets~~*MPCs*. The ~~PWR Basket~~*MPC* is ~~a fuel storage canister~~ designed to provide safe storage of intact spent fuel, ~~failed~~ *damaged* fuel, *fuel assembly inserts,* and fuel debris. The ~~PWR Basket~~*MPC* consists of ~~an internal sleeve assembly~~*a honeycomb fuel basket, a baseplate*, an outer shell ~~assembly~~, a ~~shield lid and a structural~~ lid, *vent and drain port cover plates, and a closure ring*. The ~~internal sleeve assembly~~*fuel basket* is fabricated from ~~high strength~~*stainless* steel plates formed into an array of 24 square *fuel* storage ~~sleeves~~*locations (cells)*, each holding one PWR spent fuel assembly. The cells are sized to accommodate storage of control components *and other fuel assembly inserts* within the fuel assembly. The ~~PWR Basket~~*MPC* relies ~~only~~ on geometry *(including flux traps)* for subcriticality during storage. *In addition, credit was taken for the boral (neutron poison) plates in the dry storage criticality analysis, consistent with the analysis for the flooded condition. However, the effect of these plates on reactivity under dry conditions is small. Section 1.2.1.1 of the HI-STORM FSAR provides a complete description of the generally certified Holtec MPC design. Unique features of the Trojan-specific MPC-24E/EF, implemented to customize the generic design for site-specific use, are described in Chapter 4.*

~~The f~~*F*our *specially designated* ~~peripheral~~ cells in each ~~PWR Basket~~*MPC* can accommodate Failed Fuel Cans as well as spent fuel assemblies. Failed Fuel Cans may be loaded with:

- Assemblies containing damaged fuel,
- Process Can Capsules containing fuel debris *(MPC-24EF only)*,
- ~~f~~*F*uel assembly *metal fragments* ~~hardware (non-fuel bearing components~~*(e.g., portions of fuel rods and grid assemblies, bottom nozzles, etc.)*,
- Process Cans containing fuel debris (whole and partial pellets) and fuel assembly hardware *(MPC-24EF only)*, or
- ~~a~~ *A* fuel rod storage container *(MPC-24EF only)*.

Fuel debris mixed with organic filter material ~~is~~ *was* placed in Process Cans, processed to remove the organic material, and sealed in Process Can Capsules, *which will be* ~~prior to~~ place*d*~~ment~~ in the Failed Fuel Can in the ~~PWR Basket~~*MPC-24EF*.
~~A Basket Overpack is provided to be used for continued storage inside a Concrete Cask in the unlikely event of a leaking PWR Basket.~~

1.3.2 STORAGE SYSTEM CONCRETE CASK

The Concrete Cask provides structural support, shielding, and natural circulation cooling for the ~~PWR Basket~~*MPC*. The ~~PWR Basket~~*MPC* is stored in the central steel lined cavity of the Concrete Cask. The Concrete Cask is ventilated by internal air flow paths which allow the decay heat to be removed by natural circulation around the metal ~~PWR Basket~~*MPC* wall. Air flow paths are formed by the skid channels at the bottom (air entrance), the air inlet ducts, the gap between the ~~PWR Basket~~*MPC* exterior and the Concrete Cask interior, and the air outlet ducts. Air outlet temperature is monitored to confirm proper decay heat removal.

The air inlet and outlet vents are steel lined penetrations that take non-planar paths to minimize radiation streaming. Side surface radiation dose rates are limited by the thick steel and concrete walls of the Concrete Cask.

1.3.3 TRANSFER EQUIPMENT

1.3.3.1 Transfer Cask

The Transfer Cask is used to move the loaded ~~PWR Basket~~*MPC* from the Spent Fuel Pool to the Concrete Cask. It is also utilized with the Transfer Station to transfer a*n* ~~PWR Basket~~*MPC* to a ~~Shipping~~ *HI-STAR 100 Transport* Cask *for transportation off site*~~, or to a Basket Overpack, in the unlikely event of a leaking PWR Basket~~. In preparation for use at the ISFSI, the Transfer Cask is first positioned in the Transfer Station. A Concrete Cask is positioned below the Transfer Cask, the retractable doors at the bottom of the Transfer Cask are opened, and a loaded ~~PWR Basket~~*MPC* is hoisted into the Transfer Cask. The Transfer Cask *bottom* doors are then closed, ~~door locking bolts~~*mechanical stops are* installed, and the empty Concrete Cask is moved out of the Transfer Station. The destination ~~Concrete or Shipping~~*HI-STAR 100 Transport* Cask is moved into the Transfer Station and positioned beneath the Transfer Cask. The ~~PWR Basket~~*MPC* is lowered in a process reverse to that described above.

1.3.3.2 Transfer Station

The Transfer Station is utilized for ~~PWR Basket~~*MPC* transfer operations at the ISFSI site. The Transfer Station provides lateral and vertical support ~~which~~ *that* prevents the loaded Transfer Cask from falling or overturning during transfer operations. During transfer to a ~~Shipping~~ *Transport* Cask~~ or Basket Overpack~~, the Transfer Cask is locked into the Transfer Station, while

the destination casks are moved under the Transfer Cask. All ~~PWR Basket~~*MPC* transfers are accomplished by ~~single~~ vertical lifts, with the Transfer Cask secured and stationary within the Transfer Station.

1.3.3.3 ~~Auxiliary~~ *Air Pad* Systems

To move a loaded Concrete Cask from one location to another, an air pad system is used. Air pads are inserted under the *Concrete* ~~c~~Cask and energized with ~~a~~ *one or more* standard air compressor*(s)*. A forklift or other small truck can then be used to move the Concrete Cask.

1.3.4 AUXILIARY EQUIPMENT

1.3.4.1 ~~Vacuum Drying~~*Moisture Removal and Helium Backfill* Systems

A ~~skid-mounted~~ vacuum drying system *or an optional helium recirculation system* is used to ~~remove the water from the PWR Basket (following fuel loading),~~ dry the ~~PWR Basket~~*MPC cavity following fuel loading.* ~~, and~~ *A helium backfill system is used to* backfill the ~~PWR Basket~~*MPC cavity* with helium. The vacuum drying system is designed to evacuate the ~~PWR Basket~~*MPC cavity* ~~in a stepwise fashion~~. During evacuation, the decay heat from the fuel further helps remove residual moisture from the ~~PWR Basket~~*MPC cavity. The helium recirculation unit moves inert gas through the MPC, where it absorbs water, and returns it to a condensation unit for dehumidification. This method of moisture removal allows the stored fuel to be cooled while the drying process takes place. The helium recirculation system may also be used to cool the fuel in the unlikely event that an MPC needs to be unloaded. A detailed description of the helium recirculation system design and operation is provided in the Holtec HI-STORM 100 Final Safety Analysis Report (Docket 72-1014), Appendix 2.B.*

1.3.4.2 Gap Flushing System

A ~~skid-mounted, mixed-resin bed~~ flushing system is used to pump Spent Fuel Pool water through the gap between the Transfer Cask and ~~PWR Basket~~*MPC*. The system is designed to remove Spent Fuel Pool contaminants by filtering water through a ~~mixed~~ *demineralizer using* resin ~~bed~~ to ~~facilitate surface decontamination~~*minimize contamination of the exterior* of the ~~PWR Baskets~~*MPCs* ~~prior to vacuum drying operations~~*while they are in the Cask Loading Pit.*

1.3.4.3 ~~Semi-Automatic~~*Automated* Welding System

The ~~PWR Baskets~~*MPCs* are seal-welded using a*n* ~~semi-automatic~~*automated* welding system as the preferred method.

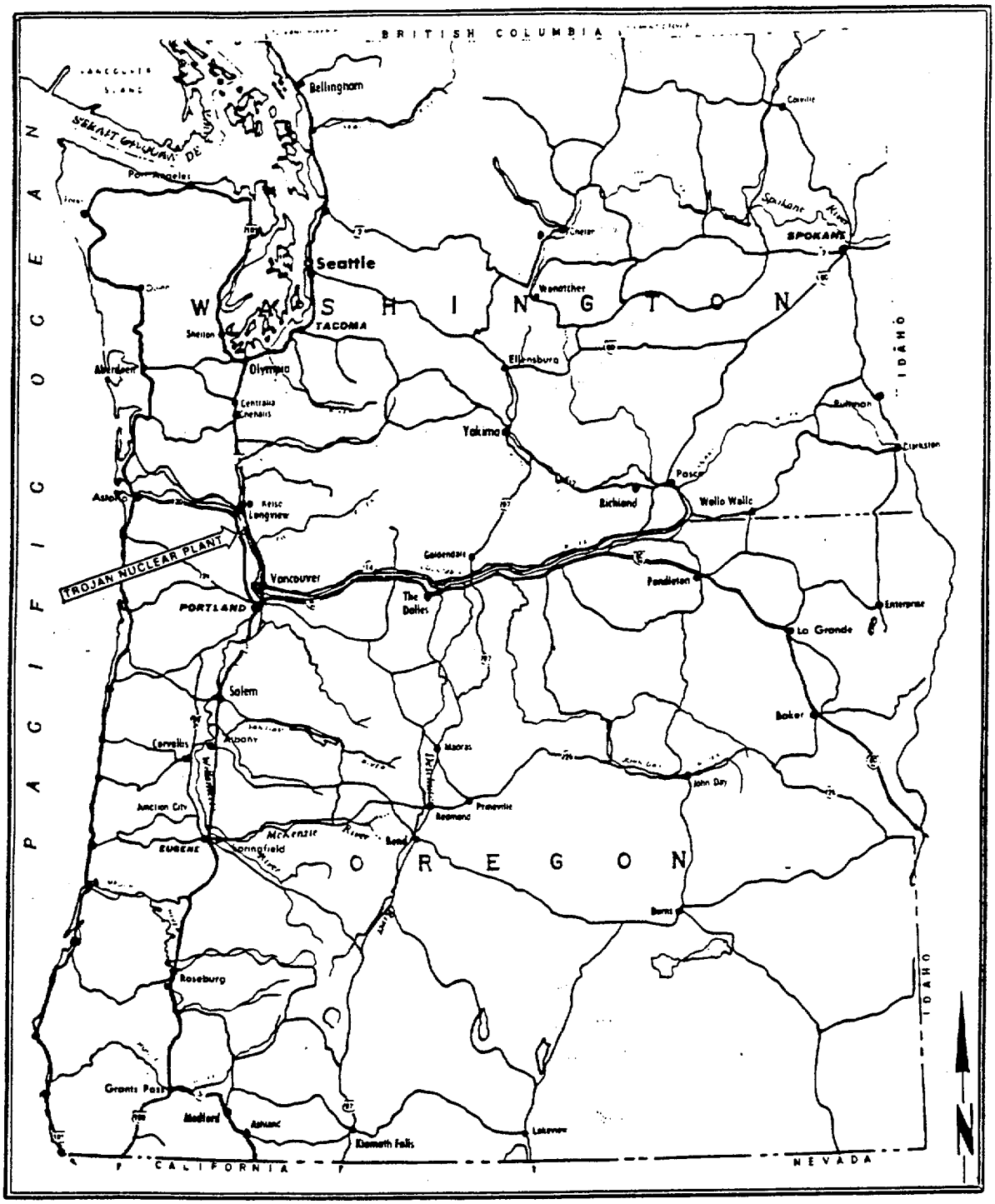
1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

PGE is the principal owner of the Trojan ISFSI and is responsible for fabrication, construction, operation, maintenance and surveillance of the ISFSI. PGE is also responsible for overall project management.

BNFL Fuel Solutions is responsible for *the* design *and fabrication of the Concrete Casks and Failed Fuel Cans,* and *for* some auxiliary equipment fabrication activities including *related* quality assurance services. *Holtec International is responsible for design and fabrication of the MPC and Transfer Cask, and for some of the related auxiliary equipment design and fabrication, including associated quality assurance services.*

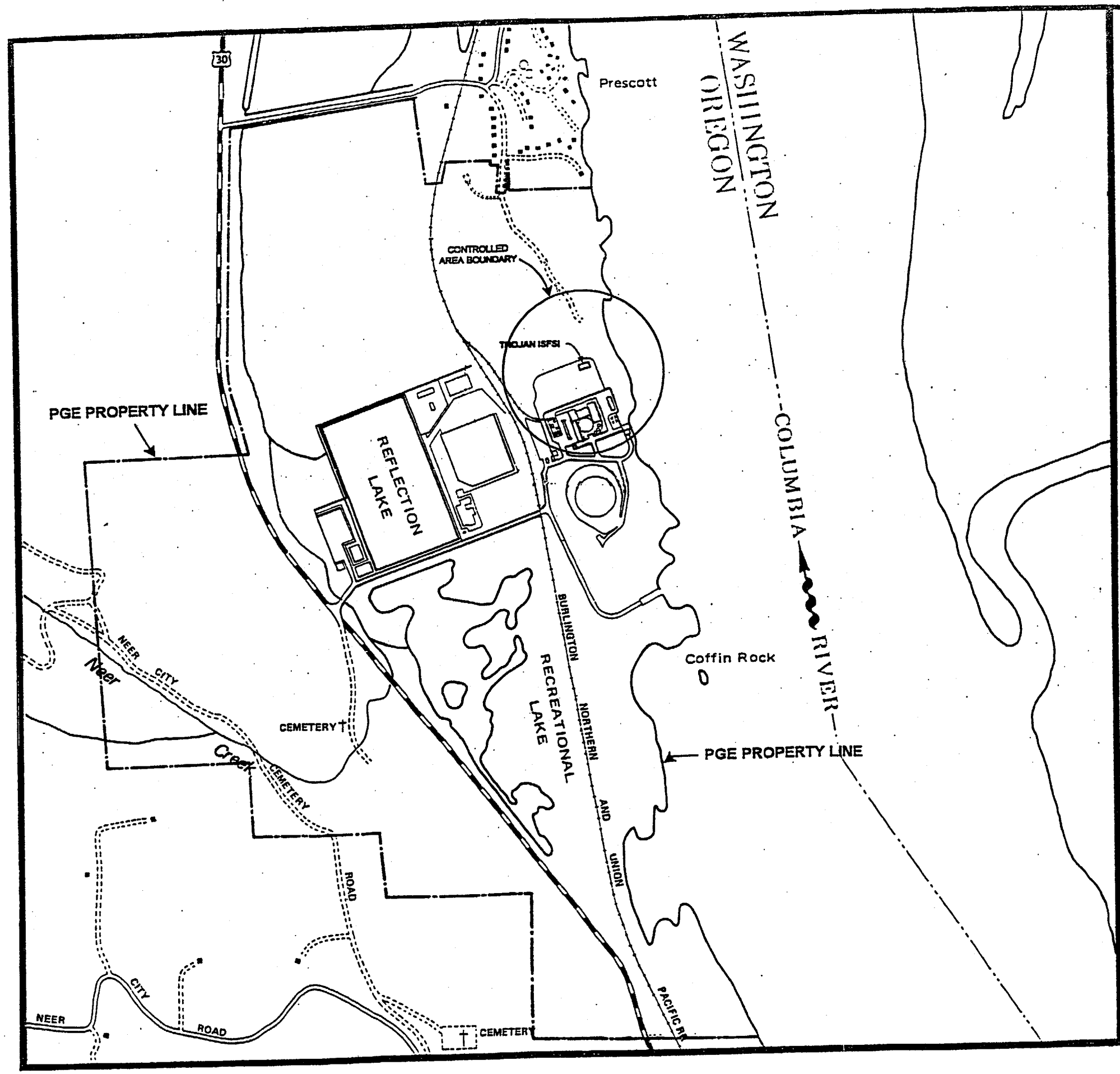
1.5 MATERIAL INCORPORATED BY REFERENCE

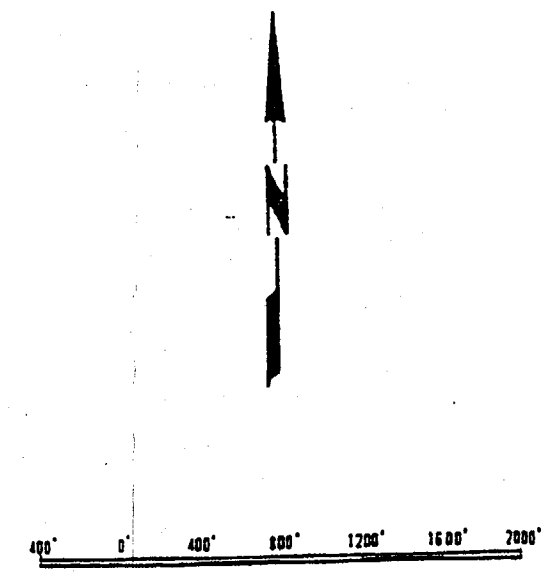
The Trojan ISFSI ~~PWR Basket~~*MPC* and Transfer Cask are based on the same design as components described in the *HI-STORM 100 Final Safety Analysis Report (Docket 72-1014) and in the HI-STAR 100 System Transportation Safety Analysis Report (Docket 71-9261)*~~BNFL Fuel Solutions (BFS) Safety Analysis Report for the TranStor™ Shipping Cask System, Docket No. 71-9268, initially submitted to the NRC on December 20, 1995, under the company name of Sierra Nuclear Corporation (SNC)~~. ~~Most of the calculations and licensing documents prepared by SNC and referenced in the SAR have been changed to reflect the change in ownership of the company to BFS. However, where some previously licensed documents are referenced, i.e., VSC licensing documents or calculations, the SNC company name is still applicable and used. All TranStor™ references should be BFS.~~ PGE intends to register as a user of the ~~TranStor™ Shipping~~ *HI-STAR 100 Transport* Cask ~~once a certification is issued~~*prior to first use of the cask for transporting Trojan spent nuclear fuel off site.*



TROJAN ISFSI
SAFETY ANALYSIS REPORT

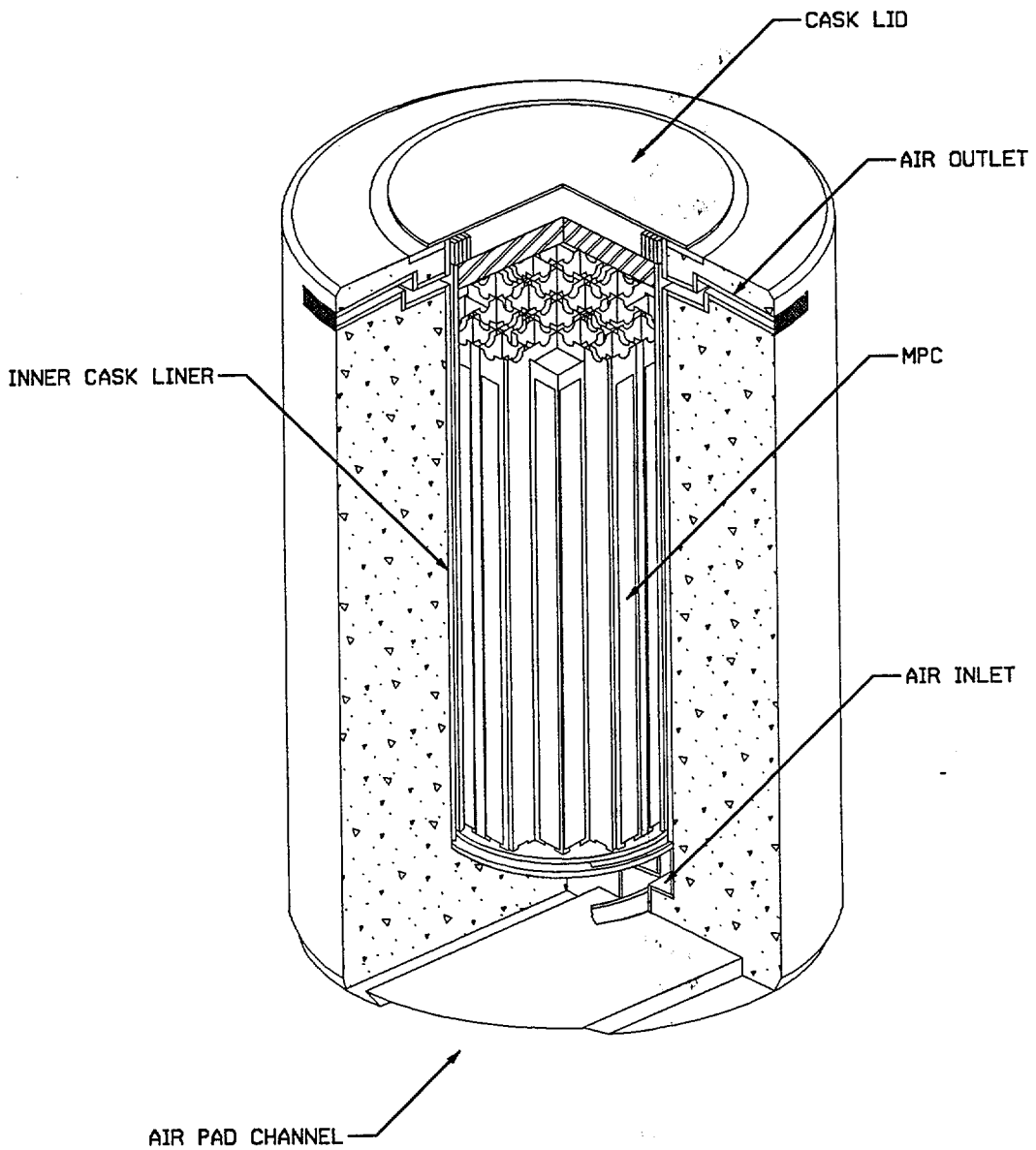
FIGURE 1.1-1
LOCATION OF TROJAN NUCLEAR PLANT





TROJAN ISFSI
SAFETY ANALYSIS REPORT

FIGURE 1.1-2
LOCATION OF ISFSI AT TNP SITE
Revision 2



TROJAN ISFSI
SAFETY ANALYSIS REPORT

FIGURE 1.3-1

STORAGE SYSTEM
REVISION 2

g:/newtroj/common/e66497/isfsi/sarfig1 3-1.dgn

Chapter 2

Figures 1.1-1 and 2.1-1 are maps that show the Trojan Nuclear Plant and ISFSI site location. Figure 2.1-2 is a map that shows the PGE property, the ISFSI site, surrounding topography, and the Controlled Area as defined in 10 CFR 72.106. Figure 2.1-3 shows the ISFSI site layout.

2.1.2 SITE DESCRIPTION

The ISFSI is located on an approximately 634-acre tract of land owned in fee by Portland General Electric Company (PGE) in Sec. 35 and 36, T. 7 N., R. 2 W., W.M., and in Sec. 1 and 2, T. 6 N., R. 2 W., W.M., Columbia County, Oregon. The tract is all-inclusive of individual and separate parcels as described in the following deed records on file in Columbia County: BK 168, Pages 13 and 14, 22, 23 to 26 inclusive, 81 to 83 inclusive, 117 to 121 inclusive; BK 171, Pages 935 and 936; and BK 174, Page 436.

The eastern boundary of the PGE property is the Columbia River. The eastern boundary (owned in fee) extends to mean low water in the southern part of the property and to mean high water in the northern part of the property. By written agreement with the State of Oregon, who is owner of the submerged and remainder of submersible lands in the river, PGE has control of the uses of such areas out to a line at approximately -20 feet MSL. Beyond this line the U. S. Coast Guard has jurisdiction over river operations.

The western boundary of the PGE property is U.S. Highway 30 in the northern one-third of the property. The western boundary in the southern two-thirds of the property extends past (to the west of) U.S. Highway 30 and includes several parcels of land to the west of U.S. Highway 30.

The ISFSI reinforced concrete Storage Pad, which is approximately 105 feet by 170 feet and is designed to accommodate up to 36 Concrete Casks, is located inside the ISFSI Protected Area fence near the eastern edge of the PGE property. The ISFSI Protected Area fence is within the boundary of the *controlled* access ~~Controlled A~~*a*rea fence which defines the ISFSI "site" within which 10 CFR 72 activities will be licensed and will occur.

Transportation routes ~~which~~ *that* are in the immediate vicinity of the ISFSI site include the Columbia River, U.S. Highway 30, and Portland & Western Railroad, Inc. The nearest edge of the ISFSI reinforced concrete Storage Pad is about 160 feet from the Oregon bank (mean low water) of the Columbia River, about 1/2 mile from the U.S. Highway 30 right of way, and about 700 feet from the railway right-of-way.

Approximately six oceangoing commercial vessels pass the ISFSI site on the Columbia River in a typical day (Reference 1). U.S. Highway 30 is a two-lane roadway that carries moderate passenger and freight traffic between communities along the Columbia River. Portland & Western Railroad, Inc. operates an average of two freight trains per day along their railway right-of-way which traverses the PGE property (Reference 2). Railroad property within PGE property boundaries is "operating" property, i.e., not available for lease or other use.

Four 230kV overhead transmission lines terminate in a switchyard approximately 1000 feet from the ISFSI. The switchyard supplies power to the ISFSI site.

The Controlled Area, as defined in 10 CFR 72.106, immediately surrounds the ISFSI and extends out to ~~325~~ *225* meters from the edge of the Storage Pad (Figure 2.1-2). The Controlled Area lies entirely on PGE property with the exception of a portion of the Controlled Area that extends over the Columbia River and the Portland & Western Railroad, Inc. right-of-way. U.S. Highway 30 is not within the Controlled Area. PGE has formal agreements with Portland & Western Railroad, Inc. to restrict traffic over their right-of-way, with the U.S. Coast Guard to restrict traffic on the Columbia River, and with the state of Oregon to evacuate persons from publicly owned lands (i.e., tidelands) in the event of an emergency at the ISFSI.

The doses that could be anticipated at the Controlled Area boundary from an off-normal event or accident are discussed in Chapter 8 and are below the limits of 10 CFR 72.106 and Oregon Administrative Rule (OAR) 345-26-390.

2.1.2.1 Other Activities Within the ISFSI Site Boundary

No activities unrelated to ISFSI operation are performed within the ISFSI *controlled* access ~~Controlled A~~*a*rea boundary.

Several major physical facilities, which were used during Trojan Nuclear Plant operation, are grouped to the south and west of the ISFSI site. These facilities are outside the ISFSI *controlled* access ~~Controlled A~~*a*rea and are intended to be made available for commercial activities upon their release for unrestricted use. Leases issued to commercial users of these facilities will limit activities to ensure that postulated events and accident analyses remain bounding. Access to these facilities will not afford access to the ISFSI.

2.1.2.2 Boundaries for Establishing Effluent Release Limits

The only potential effluent release points are the Concrete Casks themselves located at the ISFSI. *The analyses presented in the HI-STORM FSAR (Reference 14) demonstrate that the MPC remains intact during all postulated off-normal and accident conditions. In summary, there is no mechanistic failure that results in a breach of the confinement boundary. However,* ~~No effluents are anticipated for normal and anticipated occurrences at the ISFSI. In addition, no credible accidents or off-normal events result in effluents. T~~*t*he dose resulting from *an effluent release due to a non-mechanistic ground-level breach of the confinement boundary during* normal operation and anticipated occurrences (i.e., direct radiation) has been estimated at the Controlled Area boundary and is within the limits specified in 10 CFR 72.104 and OAR 345-26-390.

The Restricted Area, as defined in 10 CFR 20, has the same boundaries as the *controlled* access ~~Controlled A~~*a*rea that surrounds the ISFSI Protected Area. Physical access to the ~~r~~*R*estricted ~~a~~*A*rea is restricted by the *controlled* access ~~Controlled A~~*a*rea fence. Access into the Protected Area is controlled as described in the ISFSI Security Plan. Radiation protection procedures specify when dosimetry is required in the Restricted Area.

The minimum distance from any effluent release point (Concrete Casks) to the Restricted Area boundary is approximately ~~40~~ *50* feet. If members of the public have access to the Controlled Area immediately outside the Restricted Area, then the dose to a member of the general public in this area will be shown to comply with the limits of 10 CFR 20.1301.

Recreational uses within the PGE property boundaries include hiking, picnicking, swimming, fishing, and nature observation. In the event of an emergency that could result in a hazard to the general public, members of the general public making recreational or other casual use of the nonrestricted portions of the PGE property or making commercial use of the buildings on the PGE property can be evacuated.

2.1.3 POPULATION DISTRIBUTION AND TRENDS

The 1990 population distribution within 10 miles of the site, shown in Figure 2.1-4, was derived using 1990 census values (Reference 3). Specific place populations were located within the appropriate sectors. Rural population groups were distributed on the basis of the density of roads within each sector.

The population projections for 2000 and 2010, shown in Figures 2.1-5 and 2.1-6, were made using county growth projections based upon three census data points: 1970, 1980, and 1990 (Reference 3). Individual growth projections were developed for Cowlitz and Columbia Counties. Based upon these historical factors, population growth within 10 miles of the site is about 5 percent per decade.

In addition to the resident population, a limited influx of people into the area of the site occurs when river conditions are conducive to fishing and recreation. This influx is primarily on the Columbia and Kalama Rivers and consists of pleasure boaters, boat fishermen and bank fishermen (Reference 5). The Oregon Department of Fish and Wildlife performed aerial surveys of the river from February to October 1995 and estimated that there were 15,335 angler trips on the river from Longview to Prescott (about 6 miles) and 17,236 angler trips on the river from Prescott to Martin Slough (about 9 miles). During the busiest month, September, there were 4,556 angler trips on the river from Longview to Prescott and 6,279 angler trips on the river from Prescott to Martin Slough. Using these estimates, there would be about 80 anglers per day within 5 miles of the ISFSI from February to October. During September, the month of highest utilization, there would be about 241 anglers per day within 5 miles of the site. Because there are

no state or federal parks or campgrounds within 10 miles of the site, any increase in the number of people in the area during the summer months is relatively small.

Public facilities and institutions near the site are listed on Table 2.1-1.

2.1.4 USES OF ADJACENT LANDS AND WATERS

The ISFSI site lies in a heavily timbered area, characterized by rough terrain and suited primarily to logging and other forestry operations. One major population center lies within a 50 mile radius, as do several smaller cities; most heavy industry in the smaller cities is related to forest product processing or agriculture (Reference 6). Well over half of the land is suitable for commercial forestry or grazing, with about 20 percent suitable for farming. Less than 10 percent of the land area is unsuitable for any agricultural pursuit, and a fraction of 1 percent is devoted to urban or incorporated areas.

Lands adjacent to the site lie within Columbia County, Oregon, in which the site is located, and Cowlitz County, Washington, across the Columbia River. The area within a radius of 10 miles of the site lies within these two counties. Both have agriculturally based economies, with land use in the vicinity of the site primarily agricultural. Logs, hay and other feed are the predominant crops. Salient agricultural data for these two counties is indicative of small, generalized family farming, with heavy emphasis on grazing and farm animals. Only 41 of 934, less than 5 percent, of the farms in the counties are larger than 100 acres, while more than one-third have fewer than 20 acres. There are no major milk-producing centers on lands adjacent to the ISFSI site, the major milksheds being 50 or more miles distant (Reference 7).

A land use census completed in 1994 indicated that there were milk cows within five miles of the ISFSI (Reference 8). There were several milk producing goats located within five miles of the site during 1994. Milk from these goats was sampled as part of a Trojan radiological environmental surveillance plan associated with the Trojan Nuclear Plant. The 1994 land use census also surveyed the locations of beef cattle and other meat producing animals as well as vegetable gardens within 5 miles of the ISFSI site. The results of the 1994 land use census are shown in Table 2.1-2.

Other than agriculture, the industrial base of the area around the ISFSI site is centered in forest products and primary metals, and most of the industrial activity is on the other side of the Columbia River. Of the eleven Longview-Kelso industrial facilities listed in Table 2.2-1, four are forest products processors producing lumber, plywood, pulp, paper and paper products, and related wood and wood pulp products. The other large manufacturing firm is an aluminum smelter with an annual capacity of 220,000 metric tons. A small steel furnace smelter and pleasure boat manufacturer are the only other major manufacturers in this area. Near Kalama, upstream of the site, are grain elevators, chemical plants, a steel mill, and a few small mills.

There is relatively little recreational land use within the immediate area of the site. There are no State or Federal parks nearby, nor are there any natural or man-made attractions such as mountains or reservoirs (References 9 and 10).

The 26-acre reflecting lake and 28-acre recreational lake located on PGE property are accessible to the general public from the property entrance road. Fishing activity peaks in the spring (about 30 fisherman per day) when the state of Oregon stocks the recreational lake, and then the fishing activity tapers off to a couple of fisherman per day.

A 3-mile portion of U.S. Highway 30 has been designated as a scenic area by the Oregon State Scenic Area Board (Reference 11). Pleasure boat launchings are located in Rainier, Goble, and Prescott, (Reference 12). Recreational vehicle overnight parking is available in Goble, and Prescott Beach is used for camping and fishing. River access is also available on the Washington shore, directly opposite the site.

The lower Columbia River is well suited to recreational fishing and boating, most of which occurs in the 7 months from March to September. As described in Section 2.1.3, the Oregon Department of Fish and Wildlife performed aerial surveys of the river from February to October 1995. From their estimates, there will be about 8 anglers per day per river mile near the ISFSI from February to October. The heaviest concentration of anglers on the river near the ISFSI will be about 24 anglers per day per river mile in September.

Commercial fishing in the Columbia is regulated by both Oregon and Washington. About 270 miles of the Columbia River and tributaries are open to commercial fishing, with Bonneville Dam being the approximate midpoint. The fishery upriver of Bonneville is reserved for Indians only, while downstream is open to commercial fishing license holders as well.

The Columbia River is a major navigable channel. Approximately 2300 seagoing cargo vessels pass the site annually, carrying wheat and logs outbound and manufactured iron goods, ores and petroleum inbound (Reference 1). Major port facilities are at Portland, Oregon and Longview, Washington.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION AND MILITARY FACILITIES

Potential accidents as a result of external activities in the vicinity of the ISFSI site have been studied to determine their effect on the safety of the ISFSI. This section outlines the activities of the nearby industrial facilities, transportation arterials, and military installations and their potential effects on ISFSI safety. The risk to the operation of the ISFSI resulting from these activities is shown to be minimal.

2.2.1 LOCATIONS AND ROUTES

Most of the local commerce is related to forest products and is centered in Longview, Washington; Rainier, Oregon; and Kalama, Washington.

Due to the emphasis on forest products, industrial development in the area is heavily oriented to river transportation. An aluminum plant, small smelter, and boat manufacturer in Longview, a steel mill, chemical plants, and grain elevators in Kalama, and a fertilizer plant in Columbia City are the only large industries not related to the timber or paper industry. There are also several small quarry sites and gravel pits in the area, the closest being in Goble.

Transportation routes consist of two major highways, two railroads, the Columbia River and an airport and airways. U.S. Highway 30 runs north-south adjacent to the PGE property boundary approximately 1/2 mile from the ISFSI, and is a light-duty, two-lane highway connecting Portland on the south to Astoria, at the mouth of the Columbia River. Interstate 5 (I-5) is part of the West Coast north-south interstate system extending from Mexico to Canada. I-5 in this area is across the Columbia River in Washington approximately 1-3/10 miles east of the ISFSI at its nearest point. The Portland & Western Railroad, Inc. right-of-way passes through the PGE property, approximately 700 feet from the ISFSI. The main line railroad track between Portland and Seattle is located across the Columbia River in Washington, approximately 1-1/10 miles from the ISFSI.

The Columbia River serves as the deep-sea access channel to the important ports of Portland, Oregon and Vancouver, Washington. A 40-foot channel is maintained for deep-draft ocean vessels as far upriver as Portland. The center line of the 600-foot wide ship channel is approximately 3/10 mile from the ISFSI. Upstream from Portland and Vancouver, a 17-foot channel is maintained for barge traffic, extending to Pasco, Washington and a distance into the Snake River (Reference 3). Locks are provided at each of the dams on the river coincident with the 17-foot channel (Reference 4).

About 2300 oceangoing ships a year pass by the ISFSI site on the Columbia River. The major portion of the cargo exported is wheat and logs. Inbound cargo consists of miscellaneous goods such as petroleum, iron and steel products, automobiles, and ores (Reference 1). Portland is one

of the largest ports in terms of tonnage on the Pacific Coast and thus it maintains a large number of supporting facilities.

Longview, Washington, downstream of the site also has facilities for oceangoing ships. The Port of Longview maintains facilities for unloading and storage of ship cargo. Significant facilities are a bulk loader with storage for 14,000 metric tons of talc; storage tanks with capacity for 40,000 tons of calcinated coke; a grain elevator, currently not in use, with a capacity of 7.8 million bushels; and log storage yards. Among the commodities routinely stored at the port are pencil pitch (or coaltar pitch), ammonia sulfate, and potash. Additionally, at the port Wilson Oil (doing business as Wilcox & Flegel) operates a petroleum bulk plant which has 14 storage tanks with a total capacity of 26,190 barrels of storage (Reference 2).

The Kelso-Longview Airport is 5.3-miles north of the site and has a 4,391-foot paved runway oriented northwest-southeast. The airport is not a scheduled airline stop, but is the base for approximately 60 single and twin-engine, private and corporate aircraft. The airport handles about 100 takeoffs and landings per day. The largest planes using the field are a DeHavilland 8 corporate plane, a Siddely Hawker, a Cessna Citation, and a Falcon Jet (Reference 5). The Portland International Airport is located 33 statute miles south of the site, and is the only major airport within a 60-mile radius of the site. Portland inbound and outbound air traffic is controlled for a distance of 30 miles from the airport by Portland Air Traffic Control. Area-wide in-flight traffic control is regulated by Seattle Air Traffic Control (Reference 6).

There are no major military bases in the vicinity of the ISFSI site. The nearest military facilities are Reserve Headquarters for the various branches in Portland and Vancouver (30-40 miles south of the site), and Coast Guard and Naval facilities in Portland, Longview and at the mouth of the Columbia River (Reference 7).

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U.S. Highway 30 provides highway access to the ISFSI site and serves as the traffic arterial between Portland and the communities on the Oregon bank of the Columbia River, carrying an average of 5300 vehicles per day (Reference 9). The highway runs through the communities of Scappoose, Warren, St. Helens, Columbia City, Deer Island and Goble, south of the site; and Rainier, Clatskanie, Westport and Astoria north and west of the site. A bridge at Rainier connects U.S. Highway 30 with Longview, Washington, and a bridge at Astoria, the western terminus of the highway, connects to Megler, Washington.

U.S. Highway 26 provides a shorter Portland-to-Astoria route; thus it carries the bulk of traffic between the two, leaving U.S. Highway 30 to carry local passenger traffic, log trucks, tourists, farm vehicles and truck deliveries to the river communities. There is some shipment of petroleum products via U.S. Highway 30. Gasoline, diesel and heating oils in tank trucks are regularly delivered to towns beyond the site from suppliers in Portland and St. Helens.

Interstate 5 is the primary north-south traffic route between Portland and the Puget Sound area (Seattle, Tacoma, Olympia) carrying an average of approximately 46,000 vehicles per day. Of this total, approximately 20 percent is made up of truck combinations and the remaining 80 percent is passenger traffic (Reference 10). It is estimated that about one-tenth of the truck traffic could be carrying flammable or hazardous material, of which petroleum products would make up the majority.

An average of two freight trains per day pass through the PGE property on the Portland & Western Railroad, Inc. right-of-way, carrying general commodities, with an annual gross tonnage of 6 million tons (Reference 11). Lumber and forest products make up the bulk of the shipping most of the year. During the peak fishing season, some canned and frozen seafood is carried by rail from the Astoria canneries. An average of about 200 shipments per year with 2-3 cars per shipment of chlorine and caustics are shipped to the ~~James River~~*Georgia-Pacific* Corporation in Wauna, Oregon, on the lower river via the Portland & Western line. Other chemicals shipped include preservatives, fertilizer, resins and paints and a small amount of petroleum and propane.

Three railroads use the tracks on the Washington side of the river: Burlington Northern, AMTRAK, and Union Pacific railroads. Thirty-five to forty freight trains and six passenger trains pass the ISFSI site per day on these tracks (Reference 12). The freight carried varies widely with large quantities of wood products, aluminum, paper products, grains, agricultural products and foodstuffs making up the bulk. Chemicals shipped include large quantities of fertilizers, phenols, caustics, propane and various resins, acids, paints and lumber treatments.

Sharply rising ground to the west and similar high ground across the river to the east provide natural barriers for the site. The ISFSI itself is afforded additional protection on the north and east by earthen berms approximately 50 feet high and on the south and west by the buildings ranging from approximately 30 to 100 feet high.

The highest manmade structure at the site is the cooling tower, rising 492 feet above ground level. The cooling tower is marked with lights in accordance with Federal Aviation Administration regulations.

2.2.2 DESCRIPTION OF PRODUCTS AND MATERIALS

Products and byproducts of the timber industry in the area range from unfinished timber to finished construction lumber, cabinetry, plywood and veneer. Some hardwood products are

made in Longview on a small-scale operation, while paper and wood fiber products make up a large percentage of the production of the area. Some chemical use and storage is associated with these industries. Chemicals include resins used in plywood, veneer and chipboard production, acids used in paper and pulp production, and lumber pressure treatments and finish coatings (stains and varnishes). Chemicals are stored either in tank cars on sidings, or in storage tanks connected to the industry involved (Reference 13).

The aluminum plant in Longview is an aluminum reduction facility operated by ~~Reynolds Metal Company~~*Longview Aluminum* which produces raw metal in the form of ingots, billet bars, etc. The use of chemicals at this plant corresponds to that of any aluminum plant; namely coke, pitch, chlorine and liquefied nitrogen. Chemical storage facilities at the plant consist of stockpiles, tanks and rail tankers and transportation is by rail tank cars (Reference 13).

Kalama Chemical, Inc., produces phenols with some secondary production of benzoates. The facility receives its raw material, toluene from tankers and stores it in an 80,000-barrel tank. The finished product is shipped by rail tank car (Reference 13).

Hoechst Celanese Corporation, Inc., is located approximately 3-miles southeast of the ISFSI in Kalama, Washington and produces a bleaching agent used in the pulp and paper industry. The facility receives sulfur dioxide by rail tank car and has a storage capacity for this chemical of 300,000 pounds.

All Pure Chemical Company is located approximately 2-miles southeast of the ISFSI in Kalama, Washington. The company produces a number of products including sodium hypochlorite, household ammonia, and water treatment chemicals. It is involved in the repackaging and distribution of chlorine gas. The chlorine gas is received in 90-ton rail tank cars and is repackaged into 1-ton cylinders. The 90-ton rail tank car is the maximum storage capacity for the chlorine gas at the facility.

A listing of nearby industrial facilities, supplementing the summarization above, is provided as Table 2.2-1. The geographic locations of the nearby industrial facilities are shown on Figures 2.2-1 and 2.2-2.

2.2.3 EVALUATION OF POTENTIAL ACCIDENTS

This section provides an evaluation of the capability of the ISFSI to safely withstand the effects of an accident at, or as a result of the presence of, industrial, transportation and military installations or operations within 5 miles of the site. Potential accidents considered include explosions of chemicals, flammable (including natural) gases or munitions; industrial and forest fires; accidental releases of toxic gases; and collapse of the cooling tower.

2.2.3.1 Explosions

Shipments of commercial cargo past the site create the possibility of nearby explosions. For the most part, the rugged construction of the Concrete Casks would protect the spent nuclear fuel from such explosions. In addition, the ISFSI would be shielded from the direct force of these explosions by the earthen berms on the north and east and by the manmade structures and buildings to the south and west.

Explosions unrelated to transportation are not considered significant. The quarry operations south of the site are located in the hills west of the Columbia River. Presently, there is no storage of explosives at the operating quarry, which is 2 miles from the site. The quarry is not a large operation and only a limited amount of explosives are used. Because of the distance from the site and the protection afforded by the hillside and ridge between the quarry and the site, the quarry operation does not present a hazard to the safety of the ISFSI. The natural gas main runs along the hillside west of the site, approximately 1-1/2 miles from the site. The operation of this line will not present a hazard to the ISFSI from explosion because of the relatively low explosive capacity of the gas and the distance from the ISFSI.

Explosions related to transportation were extensively analyzed for siting of the Trojan Nuclear Plant (same location as the ISFSI). The explosion analysis, which addressed rail, ship, and highway transportation, was described in detail in the Trojan Final Safety Analysis Report (FSAR).

The FSAR analysis used an overpressure limit of 2.2 psi. This overpressure is the maximum overpressure that can be generated by the atmospheric shock from an explosion. As described in Section 8.2.8.2, the Concrete Cask is able to withstand tornado wind pressure up to 2.3 psi and wind pressure as high as 5.87 psi without sliding or overturning. *The MPC is designed for a 60 psig external pressure (Reference 14, Table 2.2.1).*

The FSAR analysis assumed transportation accident rates and numbers of shipments past the site based on estimates from transportation agencies and companies. These agencies and companies were contacted to confirm that the original estimates were still valid for the ISFSI.

The minimum weight of explosives that could cause a 2.2 psi overpressure was calculated by the FSAR analysis as 70,000 (pounds of TNT equivalent). This weight was originally confirmed for Trojan Nuclear Plant operation to exceed any known or planned shipments and has been reconfirmed for ISFSI operation.

In addition, the FSAR analysis calculated that the probabilities of a rail or barge shipment explosion that would cause a 2.2 psi overpressure are less than 10^{-6} per year each. These probabilities would be similar for the ISFSI because the transportation estimates have not changed appreciably.

Therefore, transportation related explosions would not affect the safe storage of spent nuclear fuel.

2.2.3.2 Toxic Chemicals

The effects of toxic chemicals on human habitability were extensively analyzed for operation of the Trojan Nuclear Plant and addressed in detail in the FSAR. These analyses were predicated on maintaining control room habitability during a toxic gas event. Continuous manning of the ISFSI for operational reasons is not required as in the case for an operating nuclear plant. There are no off-normal events or credible accidents for the ISFSI that require operator action within a prescribed amount of time.

Therefore, a toxic gas event would not affect the safe storage of spent nuclear fuel.

2.2.3.3 Fires

The ISFSI does not require automatic suppression and detection systems because the site specific fire hazards will not exceed the design temperature limits of the Concrete Casks. The fire main, which was installed for 10 CFR 50 fire protection requirements, may be operable for general property insurance requirements of the surrounding buildings, but the fire main is not required or credited for ISFSI fire protection.

Industries and oil storage facilities in the vicinity of the ISFSI are separated from the ISFSI, either by considerable distance or by the Columbia River. Therefore, fires at these facilities would not pose a hazard to the ISFSI.

Fires resulting from transportation accidents on I-5, the railway near I-5, or the Columbia River would be separated from the ISFSI by considerable distance and the Columbia River. Fires from transportation accidents on Highway 30 would be separated from the ISFSI by the recreation lake and reflecting lake. Fires from transportation accidents on the Portland & Western railway would be sufficiently far from the ISFSI to not have an effect on the ISFSI. Therefore, fires from transportation related accidents do not pose a hazard to the ISFSI.

The ISFSI is protected from brush or forest fires on two sides by water, the Columbia River to the east and the recreation lake, reflecting lake and Whistling Swan area to the west. The ISFSI is also afforded localized fire protection by the open area immediately surrounding it.

A fire caused by a rupture of the natural gas main west of the ISFSI would be separated from the ISFSI by a considerable distance and by the intervening lake areas. There is the possibility in the future that a natural gas line will be placed in the vicinity of ISFSI to supply gas turbines used to

produce electrical power. The potential hazards posed by placing a natural gas line and gas turbine in the vicinity of the ISFSI are addressed in Chapter 8.

In addition to the natural barriers, the ~~Rainier~~ *St. Helens* Rural Fire Protection District provides fire protection services for the site.

A fire caused by a diesel fuel oil spill from a mobile crane or other diesel fuel oil tank at the ISFSI or in the immediate vicinity of the ISFSI would not affect the safe storage of spent nuclear fuel. This type of fire, which is the only credible fire because of the limited number of fire hazards located at the ISFSI itself, would burn for only a few (6-7) minutes. This short burn time would not be sufficient for much heat transfer to the Concrete Cask or ~~PWR Basket~~*MPC* and the temperatures of the Concrete Cask and ~~PWR Basket~~*MPC* would not be appreciably raised.

The consequences of a forklift fuel (propane) tank explosion and fire are bounded by the diesel fuel oil spill scenario.

Therefore, fires would not affect the safe storage of spent nuclear fuel.

2.2.3.4 Aircraft Impacts

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2.2.3.5 Cooling Tower Collapse

The cooling tower is designed to withstand winds of up to 190 mph and earthquake loads of 0.15g. In the unlikely event of collapse, the hyperbolic design of the structure, coupled with its thin-shell configuration, provide an inherently safe failure characteristic. The structure would tend to collapse inwardly. In addition, the structure is located sufficiently far (over 800 feet) from the ISFSI to prevent damage.

2.2.3.6 Air Pollutants

Air pollutants are not anticipated at the ISFSI site.

2.3 METEOROLOGY

2.3.1 REGIONAL CLIMATOLOGY

2.3.1.1 General Climate

The marine climate of the region around the ISFSI site is typical of the Pacific coast which is characterized by wet winters and dry summers with mild temperatures year long (References 1 and 2).

This region receives substantial annual rainfall, but the rain showers are of light or moderate intensity and continuous rather than heavy for brief periods. Severe storms and thunderstorms are infrequent. On the average, this region receives 2 inches of snow per year.

Regional temperatures are for the most part, mild throughout the year. The average temperature for the summer season is 65°F and for the winter season 40°F. Surface winds seldom exceed gale force. There have been no major hail storms within a 60-mile radius of the site. Tornados rarely occur (References 1 and 2).

2.3.1.2 Severe Weather

The extreme temperatures for Portland, Oregon have been 107 °F on July 30, 1965, and August 8, 1981, and -3 °F on February 2, 1950. The maximum amount of precipitation recorded for a 24-hour period was 7.66 inches in Portland in December 1882. The greatest amount of snowfall ever measured for a 24-hour period was 16.0 inches during January 1937, in Portland. These extremes are based on National Weather Service records for 1880 through 1970 and National Oceanographic and Atmospheric Administration (NOAA) data for 1940 through 1994.

According to Huss (Reference 3), the extreme wind gust expected once in 100 years is 130 mph. However, National Weather Service data for 1928 through 1971 and NOAA data for 1940 through 1994, show that the fastest mile wind speed (1 minute average) at Portland was 88 mph on October 12, 1962. The highest windspeed (1 minute average) in Portland from the windstorm on December 12, 1995, was 52 mph, well below the fastest windspeed.

Tornados have occurred occasionally in the site region, usually associated with the passage of fronts from Pacific storms. From 1916 through 1972, 11 tornados were reported within a 60-mile radius of the site (References 4 and 5). Of these, only four occurred within 30 miles of the site. One occurred near Longview, Washington while the other three occurred in the Portland-Vancouver metropolitan area. Tornados that occur in the northwest region of the United States are usually smaller than tornados typical to the midwestern area.

The series of Mount St. Helens eruptions in 1980 resulted in tephra accumulations at the Trojan site of no more than 1/8 inch. If Mount St. Helens were to have another tephra eruption similar to the May 18, 1980 eruption, only directed towards the ISFSI with winds blowing towards the ISFSI, then the expected ash fall accumulation would be about 1.8 inches.

The greatest air pollution potential in the site region exists during the fall and winter seasons when the tendency is greatest for a quasi stationary anticyclone to develop, associated with wind speeds less than or equal to 5 mph and a shallow mixing depth (References 6 - 9).

2.3.2 LOCAL METEOROLOGY

2.3.2.1 Normal and Extreme Values of Meteorological Parameters

Normals and extremes of available temperature, precipitation, relative humidity and fog for Portland, Oregon and Longview-Kelso, Washington can be found in the "Climatography of the United States No. 20-45, Decennial Census of the United States - Summary of Hourly Observations" (Reference 10).

Meteorological data at the site during the period September 1, 1971, through August 31, 1974, are reported in this section (References 11 - 13). These data compare favorably with National Weather Service data for Portland, Oregon.

The distribution of wind direction and speed is an important factor when considering transport conditions relevant to site diffusion climatology. The topographic features of the site region are a major factor in influencing the wind direction distribution at the ISFSI site. During the 1971-1974 period, the prevailing wind for the 30-foot level was from the south, and south-southeast for the 200-foot level, and the average wind speed at the 30-foot level onsite was 8.2 mph, and was 9.3 mph at Portland.

Wind persistence is extremely important when considering potential effects from a contaminant release. Wind persistence is defined as a continuous flow from a given direction or range of directions. There is only a 5-percent probability of continuous wind direction persistence periods greater than 11.5 hours (References 12 and 13).

Temperatures in the region are generally mild considering its high latitude. During the 1971-1974 period, the annual average temperature onsite was 50°F, the daily annual mean minimum onsite was 44°F, and the annual mean maximum onsite was 58°F.

During the 1971-1974 period, the mean relative humidity onsite was 78 percent, and the annual average precipitation onsite was 62.04 inches.

2.3.2.2 Potential Influence of the ISFSI on Local Meteorology

Operation of the ISFSI is not expected to affect the climate of the region. The natural draft cooling tower is anticipated to remain at the site, but will not be used for ISFSI operation. The physical structure of the cooling tower is expected to locally increase atmospheric turbulence. There is also a potential for somewhat decreased low-level wind speeds in the vicinity of the tower. This effect diminishes rapidly with increasing distance downwind from the cooling tower and is relatively insignificant offsite.

2.3.2.3 Topographic Description

General topography in the vicinity of the ISFSI site is shown in Figure 2.1-2. Topographical cross sections out to 10 miles are provided in Figures 2.3-1 through 2.3-6.

The ISFSI site is located in the Columbia River Valley, which at this location is in a general north-south orientation. North of the site the Columbia River bends to the northwest, and south of the site the river bends to the southeast. Within the immediate vicinity of the site, there is a bluff one-half mile to the west rising sharply to 400-500 feet with a highest peak of 1187 feet MSL. North of the ISFSI, there is a wooded hill which rises to 100 feet. The remaining area in the immediate vicinity of the site is flat and low. The Columbia River Valley is approximately 2 miles wide at the site and widens to 3 miles north of the site at Longview-Kelso. The valley walls at the site rise to an elevation of 1000 feet MSL within approximately 1.8 miles to the west and not quite so high to the east.

The effect of the topographic features on airflow trajectory regimes and dilution is quite significant at the site. Analyses of annual wind roses reveal that the predominant wind flow is in a north-south direction. Winds within the Columbia River Valley will be effectively channeled and therefore will follow the changing orientations of this Valley. Computations of average χ/Q values based on the straight line model for a ground-level release indicate that the greatest potential concentrations would be north and south of the site, corresponding to the predominant wind directions. In addition, a nonbuoyant plume will generally not rise out of the valley for a ground-level release during stable temperature lapse rate conditions. Estimates of dispersion during stable conditions, based on the Gaussian diffusion model, indicate that a plume oriented in a general north-south direction would most likely not intersect with the valley walls. Therefore, the valley walls have only a limited effect as a potential barrier to prevent dispersion of the plume since the width of the valley increases both to the north and south of the site and the plume width is relatively narrow during stable conditions. Turbulence created by the mountainous terrain would increase the dilution of airborne effluents.

2.3.3 ONSITE METEOROLOGICAL MEASUREMENTS PROGRAM

The onsite meteorological program at the site began in October 1969 with wind and temperature instrumentation at four elevations: one 500-foot tower plus a 30-foot satellite tower on the bank of the Columbia River. To more accurately define low wind speed conditions, a Climet system was installed on a 33-foot tower located along the site access road. In addition, one 11-inch rain gauge was installed west of the Turbine Building.

Meteorological data ~~was~~ *were* collected during nuclear plant operation and for a time during defueled operation, but data will not be collected during ISFSI operation. The source terms for ISFSI operation are much lower than the source terms for nuclear plant operation. Accidents and off-normal events do not result in releases that would exceed 10 CFR 72.106 limits and OAR 345-26-390. As a result, meteorological monitoring for the calculation of off-site doses from normal operation and accident conditions is not necessary. These doses can be effectively calculated by using conservative values for atmospheric dispersion (χ/Q) from the onsite historical data.

As stated above, meteorological data for the site during the period September 1, 1971 through August 31, 1974 compared favorably with National Weather Service data for Portland, Oregon. Hence, if real time meteorological data is desired, then data from the National Weather Service for Portland could be used.

2.3.4 DIFFUSION ESTIMATES

Diffusion estimates were made for 3-30 day average conditions only. Worst case postulated long-term (routine) releases associated with operation of the ISFSI were modeled using a 50% probability diffusion value. The long-term diffusion estimate is ~~1.6E-5~~*6.0 x 10^{-5}* sec/m^3 *at 150 meters. Determination of the long-term diffusion value at 150 meters is conservative since the Trojan ISFSI Controlled Area Boundary is at 225 meters.*

A hypothetical accident was also postulated to determine the concentrations and doses that could occur following the release. The 30 day accident diffusion estimate is obtained from the Trojan Nuclear Plant Safety Analysis Report. The hypothetical accident χ/Q at ~~the Controlled Area Boundary (325 meters)~~*150 meters* is ~~1.4~~*5.2* x 10^{-4} sec/m^3, *which as stated above is conservative since the Trojan ISFSI Controlled Area Boundary is at 225 meters.*

2.4 HYDROLOGIC ENGINEERING

The site location and design of the Concrete Casks assures that the systems and structures that are important to safety withstand the additional forces that might be imposed by the hydrology of the area without loss of the capability to protect the public.

Hydrologically-related design bases, performance requirements, and the design for important to safety structures, systems and components reflect thorough consideration of the following phenomena:

1. Runoff-type floods up to and including the probable maximum flood (PMF).
2. Surges and wave actions.
3. Tsunamis.
4. Artificial floods due to dam failures or landslides.
5. Ice jam flooding.
6. Dilution and dispersion characteristics of normal and accidental release to the hydrosphere relating to existing and potential future users of surface and groundwater resources.

The following sections discuss the hydrological characteristics of the ISFSI site and their influence on ISFSI design and operation.

2.4.1 HYDROLOGIC DESCRIPTION

2.4.1.1 Site and Facilities

The ISFSI site is on a rock outcropping located on the Oregon bank of the Columbia River at River Mile 72.5 in the northwest section of the 634-acre tract of PGE property. The concrete reinforced Storage Pad is about 160 feet from the Oregon bank (mean low water) with an intervening hill. Equipment important to safety is located at or above ground elevation 45 feet MSL, which is above postulated flood levels. There is no potential for flood induced erosion because the ISFSI reinforced concrete Storage Pad is founded on impervious rock.

The Columbia River is about 1/2 mile wide adjacent to the ISFSI. In the vicinity of the ISFSI there are holes in the river deeper than -120 feet MSL. Directly across from the ISFSI the deepest profile is about -70 feet MSL. Topography nearby and on the site area is shown on Figures 2.1-2 and 2.4-1.

The ISFSI site has excellent drainage. The east side of the rocky ridge drains directly into the Columbia River, while runoff on the west side flows into the old river channel and thence by Carr Slough northward until it joins the Columbia River. Neer Creek, a small stream, flows off the steep hillside west of the site and old river channel. Its flow varies from over 30 cfs at times during the winter to essentially zero during dry summer periods. Neer Creek provides flow through the recreational lake with the outflow passing into Carr Slough as it did prior to construction of the Trojan Nuclear Plant.

The northern, unpaved area of the PGE property drains to the perimeter drainage ditch, which is approximately 3 feet lower than the ground elevation of the PGE property. This ditch drains to the reflecting lake, and the south drainage ditch empties into the recreational lake. The roads around the PGE property are sloped so that they drain either to the drainage ditch or toward the river except for those portions of roads shown by cross-hatching in Figure 2.4-2, which drain to the indicated catch basins.

2.4.1.2 Hydrosphere

The Columbia River is the major hydrographic feature in the area. It represents one-third of the potential hydropower of the United States, and has an annual discharge of approximately 180,000,000 acre-ft (59 trillion gallons), and drains an area of 260,000 square miles (Reference 1). The Columbia River has an average flow rate of 230,000 cfs at the site with a corresponding average current velocity of 1.8 fps.

A most important factor in considering flows in the Columbia River is the large amount of storage available for flood control and power use. With the dams constructed in the United States and Canada by 1973, more than 30 million acre-ft of storage (Reference 2) is usable in controlling floods on the lower Columbia River.

Tidal effects on the Columbia River can be seen as far upstream as Bonneville Dam, at River Mile 140. The tides at Astoria are typical of the Pacific Northwest tidal pattern. The tides are of a semidiurnal nature with an average period of approximately 12.4 hours.

The effect of tides at the site is dependent to a large part on the flow of the river at the time. Flow reversal occurs at the site on about one-quarter of the tides during a normal year. The extreme tidal range at the site is less than 5 feet, and a maximum upstream flow of 129,000 cfs with an average current velocity of 1.3 fps. The Columbia River has five significant tributaries near the site. None is large enough to have serious effect upon the hydrology at the site.

2.4.1.2.1 Surface Water Use

To determine the extent to which surface water is used within 5 miles and 20 miles of the ISFSI, a survey was made of registered surface water rights. The survey used a 12-mile by 12-mile area and a 36-mile by 36-mile area surrounding the ISFSI in lieu of a 5-mile and 20-mile circle because the Township/Range coordinate system used to identify the location of water rights in the states of Washington and Oregon uses 6-mile by 6-mile boxes. The survey showed that 824 rights are registered within the 36-mile by 36-mile area with an allowed use of 12,480 cfs. Of these rights, 117 are located within the 12-mile by 12-mile area with an allowed use of 401 cfs. The users of surface water will not be adversely affected by the ISFSI because there are no routine effluent releases associated with normal operation and no credible off normal events or accidents that result in liquid effluents.

2.4.2 FLOODS

2.4.2.1 Flood History

Columbia River floods are generally divided into two categories:

1. Spring floods caused by the melting snowpack usually in the upper reaches of the Columbia Basin east of the Cascades.

2. Winter floods caused by intense rain occasionally augmented by melting snowpack in the Willamette and other basins west of the Cascades.

The maximum natural or unregulated flood on record of the Columbia River is that of 1894, which resulted from a combination of hydro-meteorologic conditions including heavy snowpacks and rapid melt plus rainfall. The peak discharge for the Columbia River was 1,240,000 cfs as measured at The Dalles, Oregon. The large floods which have occurred were spring floods resulting from the melt of a large snowpack combined with the spring rain.

On February 8 and 9, 1996, the Columbia River had an estimated flood flow of 850,000 to 900,000 cfs near the Trojan site. This flow resulted in a peak water surface level at the Trojan site of about 22.5 feet. The flood was caused by warm rainstorms from the mid-Pacific falling on snow in the lower Columbia River basin.

2.4.2.2 Flood Design Considerations

The ISFSI ground elevation of 45 feet MSL is sufficient to be considered safe from projected floods. Equipment important to the safety is situated at or above this level. There is no potential for flood induced erosion because the ISFSI reinforced concrete Storage Pad is founded on impervious rock.

Water surface elevations were determined for several cases including: the standard project flood (1000-year); the 10,000-year flood; the probable maximum flood; potential dam failures; probable maximum surge flooding; and tsunami.

In addition, studies were made superimposing more than one case and adding surges from wind activity, wave action and tidal effects. Thus, use of highly conservative methods provide a large degree of assurance that the safety of the ISFSI is guaranteed from any potential flooding. The most critical flood level for the site is the combination of an unlikely failure of Grand Coulee Dam and the resultant surge combined with a 25-year flood.

2.4.2.3 Effects of Local Intense Precipitation

The following historical data shows the probable maximum precipitation (PMP) based on the U. S. Weather Bureau Hydro meteorological Report No. 43:

Hour	Cumulative Precipitation (in)	Hour	Cumulative Precipitation (in.)
½	0.45	6	3.37
1	0.83	12	5.62
1-½	1.19	18	7.28
2	1.52	24	8.63
2-½	1.82	36	10.69
3	2.09	48	12.25
4	2.60	60	13.46
5	3.02	72	14.44

The PMP does not create substantial loads because the tops of the Concrete Casks will not collect standing water. Even if a few inches of water accumulated, the loads from this water would be bounded by the analysis for the worst case snow load.

The site drainage, as described in Section 2.4.1.1, is adequate for the duration and amounts of rainfall listed above.

2.4.3 PROBABLE MAXIMUM FLOOD (PMF) OF STREAMS AND RIVERS

The standard project flood (SPF) and the PMF for the Columbia River were established by the U.S. Army Corps of Engineers. The Corps of Engineers issued their findings in a report dated September 1969 (Reference 1). Because the Columbia River is the major river in the area and is

adjacent to the site, the PMF for the Columbia River is the controlling event in studies of natural river and stream flooding for the site.

The 1000-year flood (SPF) with an 850,000 cfs discharge will result in a maximum water surface elevation in the Columbia River at the ISFSI site of 21 feet MSL (Reference 2). The unregulated flow for the 1000-year flood is 1,550,000 cfs at The Dalles or 1.25 times the maximum historical flood of 1894.

The 10,000-year flood with a discharge of 1,050,000 cfs will cause a river water elevation of 24 feet MSL at the site. The PMF, as computed by the North Pacific Division of the Corps of Engineers, will have a discharge of 2,200,000 cfs associated with a water elevation of 36 feet MSL at the site and it can be safely passed by existing dams, and those under construction on the main stem of the Columbia River and its major tributaries, except Bonneville Dam. The effect of a failure of Bonneville Dam, which is the nearest upstream dam, would be, according to the Corps of Engineers, negligible in terms of additional flooding.

The PMF was considered to be caused primarily by snowmelt over an extended period of 2 or 3 months with significant runoff contributions from storm rainfall during the snowmelt period. The combination of conditions for the PMF derivation was the most severe considered "reasonably possible" in the Columbia River Basin.

The wave run up during the PMF, based on an overland wind speed of 40 mph, would be approximately 3.2 feet. For the maximum discharge of 2,200,000 cfs, the highest water surface level at the site, as a result of wave run up, would be 39.2 feet MSL, which is 5.8 feet below site grade.

Further detailed discussions of the model used to determine the flood levels are contained in the Trojan Nuclear Plant FSAR. The Corps of Engineer report of 1969 has not been revised, indicating that the description in the FSAR is applicable for the ISFSI.

2.4.4 POTENTIAL DAM FAILURES

Studies have been made of potential dam failures that could affect the ISFSI site by the Corps of Engineers (References 3 and 4) and by Portland General Electric Company (PGE). Two types of failures were considered: a seismically induced failure, and a volcanically induced failure. The potential for the most severe flood caused by an earthquake concerns the Columbia River, while the worst possible volcanically induced dam failure concerns the Lewis River in Washington.

2.4.4.1 Seismically Induced Dam Failure

The maximum artificial flood that can occur at the ISFSI site is a catastrophic, massive and sudden failure of Grand Coulee Dam at Columbia River Mile 597. *The seismically induced*

failure of the Grand Coulee Dam, an event of almost inconceivable proportions, has been correlated to Corps of Engineers reports to provide a limiting case conservative assumption. Based on these reports, the resulting flood, depending on the exact conditions of the breach, natural flow conditions, and routing, would cause a discharge ranging from 3,600,000 to 4,400,000 cfs at the site. Steady state discharge versus height curves for the river at Trojan predict maximum river elevations ranging from 39 to 46 feet MSL. It would reach the site approximately 2 days after the failure of the Grand Coulee Dam.

Because the artificial flood peak travels as a surge of much shorter duration than a natural flood, additional calculations were performed assuming an unsteady flow. The results of these calculations show that the water surface elevation for the maximum artificial flood (4,400,000 cfs) would be approximately 41 feet MSL and for the smaller flood (3,600,000 cfs) would be approximately 40 feet MSL.~~This event is almost inconceivable. The resulting flood, depending on the exact conditions of breach and natural flow conditions, would cause a discharge ranging from 3,600,000 to 4,400,000 cfs at the site with maximum river elevations ranging from 39 to 46 feet MSL. It would reach the site approximately 2 days after the failure of Grand Coulee. Considering dam failure permutations and routing of the artificial flood, the water surface elevation for the maximum artificial flood (4,400,000 cfs) was calculated to be 41 feet MSL and for the smaller flood (3,600,000 cfs) was calculated to be 40 feet MSL.~~ Wave run up based on an overland wind speed of 20 mph would be 1.75 feet, resulting in a peak water level at the site of 42.75, which is 2.25 feet below grade level. Further details and assumption of this analysis are contained in the Trojan Nuclear Plant FSAR.

2.4.4.2 Volcanically Induced Dam Failure

The Lewis River enters the Columbia River approximately 14 miles upstream of the ISFSI site. An artificial flood caused by the eruption of Mt. St. Helens and the domino-type failure of Swift, Merwin, and Yale dams on the Lewis River would result in a maximum discharge of 3,300,000 cfs at the site. Considering dam failure permutations and routing of the artificial flood, the peak water at the site corresponding to the peak flow (3,300,000 cfs) will be from 39 to 41 feet MSL. Wave run up based on an overland windspeed of 20 mph would be 1.75 feet, resulting in a peak water level at the site of 42.75 feet, which is 2.25 feet below site grade. Further details and assumptions of this analysis are contained in the Trojan Nuclear Plant FSAR.

Subsequent to the May 18, 1980 eruption of Mt. St. Helens, the Northwest Forecast Center of the National Weather Service performed a failure analysis of the Swift Dam using their dam break and wave models. Their results indicate that the generated flood wave would reach Woodland, Washington, in about 1 hour and inundate the areas to a height of 35 feet MSL. This wave would be expected to reach Rainier, Oregon, in approximately 3 hours with a peak elevation of less than 30 feet MSL. This is well below the site elevation of the ISFSI (Reference 5).

2.4.4.3 Spirit Lake Blockage Failure

Spirit Lake is located on the North Fork of the Toutle River within a few miles of Mount St. Helens. When Mount St. Helens erupted on May 18, 1980, mud and debris caused a blockage at Spirit Lake and raised the lake's surface elevation. The North Fork of the Toutle River flows into the Cowlitz River, which in turn flows into the Columbia River about 4 1/2 miles downstream of the Trojan site. Therefore, if the blockage were to suddenly fail, the released water and sediment presented a potential flooding hazard to the Trojan site.

The blockage at Spirit Lake is not a potential flooding hazard to the ISFSI site. Spirit Lake has been drained to a level where the blockage no longer confines lake water, thus eliminating the potential for flooding should the blockage be seismically dislodged.

2.4.5 PROBABLE MAXIMUM SURGE FLOODING

The ISFSI is located a considerable distance from the outlet of the Columbia River to the Pacific Ocean and is well protected by the terrain. Furthermore, although storms with winds of up to about 70 mph occur off the coast, hurricane-force winds are rare in the Pacific Northwest (Reference 6). Storm wave heights, as reported by the Columbia River Lightship off the mouth of the Columbia River, rarely exceed 30 feet. The Columbia River estuary acts as a dampener to wave action, such that ocean-bred wind waves are indistinguishable from normal river wind-wave action a few miles upstream from the mouth. Surge flooding of the site (ocean-bred), therefore, is considered unlikely.

2.4.6 PROBABLE MAXIMUM TSUNAMI FLOODING

Historically, the evidence demonstrates that the mouth of the Columbia River is relatively insensitive to tsunamis when compared to Crescent City, California, 310 miles south of the Columbia River entrance.

The tsunami effects at the mouth of the Columbia River are further dissipated inside the river due to the characteristics of the estuary, as was demonstrated during the tsunami generated by the Alaskan earthquake of March 28, 1964.

The grade elevation of the ISFSI site is 45 feet MSL. Because of the large margin between the ISFSI grade elevation and the river surface and because of the insensitivity of the river to tsunami effects, tsunamis are not considered in the design criteria for the ISFSI.

2.4.7 ICE EFFECTS

The general climate in the lower Columbia River Basin is not conducive to ice formation. In addition, the flow of the river during periods of freezing temperatures is sufficiently large

(200,000 to 400,000 cfs) that ice formation is impossible in the main streamflow. During extended periods of freezing temperatures, some icing is experienced along the banks of sloughs and inlets where the water is slow moving or stagnant. The lowest recorded river temperature at the site was 34.1°F on February 6, 1971 (period of record 1967 to 1972) (Reference 7). Any surface ice formation would not affect the ISFSI because the ISFSI does not require water for operation.

2.4.8 FLOODING PROTECTION REQUIREMENTS

Facilities/equipment that are important to safety are located at or above elevation 45 feet MSL and none of the postulated floods exceeds that level. Therefore, there are no requirements for flood protection.

2.4.9 ENVIRONMENTAL ACCEPTANCE OF EFFLUENTS

The spent nuclear fuel at the ISFSI is maintained in dry Concrete Casks. There are no routine effluent releases and no credible off normal events or accidents that result in liquid effluents. Therefore, the ISFSI will have no effect on surface or ground waters.

2.5 SUBSURFACE HYDROLOGY

2.5.1 REGIONAL AND SITE CHARACTERISTICS

The ISFSI site is located on an extremely impervious rocky ridge that is bounded on one side and end by the Columbia River and on the other side and end by an old river channel that has been filled with alluvial sediments.

The old channel is bow-shaped, about 2,000 feet wide and 2 miles long. It is carved in the Eocene Goble Volcanics, which borders the old channel on the west side, and forms a rock knob on the east side that separates the old channel from the Columbia River. The ground elevation in the old channel is between 14 feet and 17 feet. The Goble Volcanics have poor aquifer characteristics.

The slough is filled with ~~fine~~ *fine*-grained alluvium of Quaternary age. Previous seismic surveys done in the area indicate the bedrock to be up to 340-foot depth. None of the borings drilled in the deep section of the slough penetrated the full thickness of the alluvium. One of the deep borings - DH-7 - terminated about 9 feet in a gravel bed at the 269-foot depth, which is overlain by 154 feet of fine sand, beneath a 114-foot-thick layer of organic silt with a trace of clay. A layer of volcanic ash occurs near 70-foot depth within the silt layer.

The entire alluvial section appears to be hydraulically connected to the river on both ends of the slough. Water levels in the two domestic water wells located near the south end of the alluvial channel respond to tidal fluctuations. A gravel bed below the fine sand is the aquifer for the two wells ~~which~~ *that* supply the Trojan site. These wells are 8 inches in diameter and were drilled through the 158-foot-thick impermeable gray silt bed and 117 feet of fine sand into about 28 feet of gravel at the bottom of the alluvial channel. The gravel bed appears to be bimodal or strongly gap graded. It is composed of medium-to-coarse gravel with a very fine sand matrix. This fine sand controls the aquifer characteristics. These wells are each capable of producing approximately 250 gpm of high-quality domestic water, but the demand is anticipated to be much less than this capacity.

A survey of existing wells and natural springs was made in the area between Goble and the south edge of Rainier to determine the extent to which groundwater is utilized, and to determine the elevation of the permanent water table in the area of the site. The survey showed that bedrock supplies most of the groundwater to existing wells in the area. Approximate water levels in the existing wells were determined, and piezometers were installed in six of the drill holes at the Trojan site to indicate water levels in the alluvium and in the bedrock.

Static water levels in wells, and the elevation of springs emanating from the ridge west of U.S. Highway 30, show that water levels in the ridge are considerably higher than are the water levels in the alluvium, even during periods of very high flow in the river. Consequently, it is apparent

that the water table in the alluvium does not feed the water table in the ridge. Thus, the precise local direction of movement of the water is not as important as is the fact that the water in the rock and in the alluvium moves toward the Columbia River and not toward existing offsite wells or springs. The hydraulic gradient of the water table precludes contamination of the portion of the bedrock ~~which~~ *that* now supplies groundwater to offsite wells or springs. It is therefore concluded that there is virtually no possibility of contamination of existing or future offsite groundwater supplies by accidental release of radioactive materials onto the alluvium or rock at the site.

2.5.2 CONTAMINANT TRANSPORT ANALYSIS

Four permeability tests of the alluvial material in Drill Holes 9 and 10 showed permeability ranging from 10 feet to 20,000 feet per year. If accidental discharge of contaminated water onto the alluvium should occur, the water would move through the upper portion of the alluvium and toward the Columbia River at a rate of approximately 15 feet per year. If accidental discharge of contaminated water onto the foundation rock would occur the water would also move toward the Columbia River. If it moved through the fractures and pores in the rock, it would move at a much slower rate than the rate in the alluvium.

2.6 GEOLOGY, SEISMOLOGY AND GEOTECHNICAL ENGINEERING

This section describes and evaluates the geologic and seismic conditions for the region around the ISFSI site. Foundation conditions are evaluated. The seismic history of the region is examined, and the earthquake design criteria are developed and described. These discussions have been summarized from the Trojan Nuclear Plant Final Safety Analysis Report (FSAR) with slight modifications to address ISFSI specifics. Further details and figures, such as geologic profiles, may be found in the FSAR.

2.6.1 BASIC GEOLOGIC AND SEISMIC INFORMATION

The ISFSI site is 31 miles north of the city limits of Portland, Oregon on the Oregon bank of the Columbia River. A portion of the PGE property is underlain by a north-south trending steep-sided ridge of volcanic rock that borders the river and rises to a maximum elevation of 134 feet above MSL. The remainder of the PGE property is underlain by a flat alluvian plain with elevation ranging between 5 and 18 feet. Approximately ½ mile west of the site, a north-south trending range of hills rises steeply above the alluvial plain to elevation in excess of 1000 feet MSL. The Columbia River flows in a northerly direction at the site, but turns to the west several miles downstream.

The ISFSI is located on the east side of the PGE property in a flat, yard area at an elevation of 45 feet MSL. The reinforced concrete slab on which Concrete Casks sit is located on competent rock.

Upon original siting for nuclear plant operation, investigations were performed that may be used in determining the suitability of the PGE property for the storage of spent nuclear fuel. The investigations were conducted to determine the characteristics of the foundation material, especially in regard to their suitability for supporting the structures, to determine the depth and configuration of the groundwater table, to determine the characteristics of the soil and rock materials with respect to their effect on the migration of radioactive solutions if such solutions come in contact with them, and to evaluate the seismicity of the area so that appropriate parameters for seismic design could be selected. Consultants in geology and seismology were retained to evaluate independently the results of the field investigations.

A river bottom survey ("Boomer" survey) was made by EG&G of Goleta, California, using continuous seismic profiling, to define the shape of the river bottom adjacent to the PGE property. A geophysical survey was performed by Geo-Recon Inc., of Seattle, Washington, across the alluvial valley to the west of the PGE property. A geophysical survey was made at the reactor site, by P.C. Exploration of Carmichael, California, to measure P-wave and S-wave velocities and to calculate the dynamic modulus of elasticity of the foundation rock. Drilling and sampling was done by Lynch Bros. of Seattle, Washington, under the direction of Bechtel Corporation. Soil tests were designated by Bechtel and done by Shannon and Wilson, Inc.,

Seattle, Washington, in their Portland laboratory. Selected rock core samples from the drill holes were tested by Bechtel in their laboratory in San Francisco.

A comprehensive geophysical survey was made to investigate geologic conditions in the Columbia River channel adjacent to the PGE property. Studies included seismic refraction, resistivity, aeromagnetic, and gravity surveys. The results of these investigations were evaluated by a special advisory board and the results are presented in a geophysical survey report dated August 1, 1972 (Reference 1).

2.6.1.1 Regional Geology

The ISFSI site is located in the Oregon Coast Range section of the Pacific border physiographic province. The Coast Range is farther divided, and the site is on the southeastern margin of the Willapa Hills subsection. The Coast Range section is bordered on the north by the Olympic Range and on the south by the Klamath Mountains. In the area near the site and along the northern two-thirds of the Coast Range, the Puget Trough forms the eastern boundary. The southern third is bounded on the east by the Sierra-Cascade Province.

The Cascade Range east of the site is marked by a chain of volcanic cones whose activity spans most of Tertiary time. Lava flows and pyroclastic deposits range from Eocene to Recent in age. Due to the proximity of the site to these Tertiary features, a detailed study was made to determine the possible effect on the site of lava flows, ash release, or mud flows related to volcanic activity in the region. Special emphasis was placed on studies of Mt. St. Helens, the volcanic cone closest to the site, and one that was active during Recent time.

The rocks exposed in the area are Cenozoic in age. They include marine and terrestrial sediments, and volcanic rocks. The volcanic rocks predominate in quantity. The oldest rocks are a thick sequence (over 5000 feet) of Upper Eocene basaltic flows, pyroclastics, and associated sediments called the Goble series. The foundation rock on the PGE property is part of the Goble series. The unit is widespread in parts of northwestern Oregon and southwestern Washington. Marine tuffaceous sandstones and other sediments, which were derived in part from the erosion of the Goble series, were later deposited in an advancing Oligocene sea. Accompanying or following the retreat of the sea, the rocks were folded and then eroded to form an area of moderately low relief. During Miocene time, intermittent flows of basaltic lavas poured over this eroded Oligocene surface and buried it to depths of as much as 700 feet. After a period of weathering and erosion followed by some folding, the Troutdale sediments were deposited during the Pliocene period by the ancestral Columbia River. Later tectonic activity folded both the Columbia River basalt and the Troutdale sediments. Changes in sea level during and after Pleistocene contributed to considerable erosion which in places has removed the younger geologic units and exposes the older Goble series, as it has on the PGE property. During Pleistocene, the sea level was 300 to 500 feet lower than its present level and the Columbia River

channel near the site was eroded to depths of at least 340 feet below the present sea level. Alluvium has partially filled in the channel since that time.

Folding generally conforms to the northwest Coast Range structural trend. The Eocene formations north of the Columbia River have been folded into a syncline which dips as high as 45 degrees but generally about 10 to 20 degrees. The Goble series underlying the PGE property dips gently to the south or southwest, usually at less than 10 degrees. Pleistocene and recent deposits are apparently flat-lying. No rift-type faults or extensive, continuous faults are in existence in southwest Washington and northwest Oregon. Earthquake activity during the period 1858 to 1965 does not indicate any major active fault near the site. Berg and Baker state in the Bulletin of the Seismological Society of America, January 1963, that the grouping of earthquake epicenters in the Portland area and in other parts of Oregon is associated with the local faulting in those areas. They also state that the probable extension of the San Andreas Fault is clearly exhibited by the alignment of offshore epicenters trending northwest off the coast of Oregon. These offshore epicenters are over 200 miles west of the site.

The presence of some ancient faulting in the area is suggested by topography, but the faulting is apparently minor since mapped faulting is generally of small displacement. No evidence of post-Pleistocene faulting has been found. It was therefore concluded that the evidence indicates there are no active faults in the area.

2.6.1.2 Site Geology

Geologic mapping was performed to locate the various geologic units and their contacts and to determine the geologic structure and the characteristics of the geologic units at and near the site. Seismic lines totaling 4350 feet in length were run across the alluvial-filled valley to the west of the rock ridge to obtain a profile of the subsurface materials. River bottom soundings near the site were obtained by continuous seismic profiling to define the shape of the river bottom. A geophysical survey was made to measure the dynamic modulus of elasticity of the foundation rock for the reactor. A drilling program was conducted, consisting of 59 diamond drill and soil sampling holes totaling about 5200 feet, and three piezometer holes totaling 153 feet. Samples were not taken in the piezometer holes. Piezometers were installed in three of the holes. Existing water wells and natural springs near the site were located, and information obtained on the occurrence, present utilization, and movement of the groundwater in the area.

Diamond drill holes are located in the area in order to define the characteristics and configuration of the bedrock which provides the foundation for the structures. Soil borings were located in the alluvium to check the foundation conditions for the main access road and the railroad track.

Groundwater hydrology and seismology are discussed in Sections 2.5.1 and 2.6.2, respectively.

The PGE property is underlain by bedrock, which is a part of the Goble series of Upper Eocene age, and by recent alluvium. Outcrops of bedrock have not been distinguished or separated from areas where bedrock is obscured by a relatively thin cover of residual soil. The bedrock is exposed on the ground surface along a narrow, elongated ridge bordering the left bank of the Columbia River. The ISFSI reinforced concrete Storage Pad is founded on the rock which forms the ridge. This ridge was formerly an island in the river, but alluvium has since filled in the old river channel west of the ridge to elevations of 5 to 18 feet.

Drill hole DH-4 penetrates the rock in the ridge to an elevation of -240 feet or approximately 285 feet below the general foundation grade. Forty-two other holes also penetrate the rock which forms the ridge. Three holes are located at the intake structure to define foundation conditions. Approximately five holes are situated along the access road and railroad alignment.

The bedrock is volcanic in origin and consists principally of tuffs with lesser amounts of flow breccias, tuff breccias, agglomerates, and basalt flows. Basalt and agglomerate often are exposed on the ground surface since they are more resistant to erosion than is the tuff; however, tuffs and flow breccias are the predominant rock type in the ridge. There are numerous interbedded thin basalt flows, and based on DH-4 and other borings, basalt may be the predominant rock type below river level. Since the basalt often flowed onto eroded or uneven surfaces, it varies greatly in thickness, but in the ridge the basalt flows are generally thin. Some of the individual flows are not continuous across the ridge, probably because at the time they occurred they only filled in depressions in the ground surface.

The tuffs, tuff breccias, and flow breccias are soft to moderately hard, gray with some white spots and veinlets, commonly quite porous, and bedding is commonly not distinguishable. They are usually lightly to moderately fractured but occasionally highly fractured. The fractures are commonly rehealed and tight.

The agglomerate varies in color from reddish blue to blue grey. It is moderately hard, moderately well cemented, and occasionally vesicular. It commonly is more fractured than the other rocks but not as fractured in place as a superficial inspection of the core recovered from drilling might indicate. Nearly all fractures are at least partially rehealed by the deposition of secondary minerals, and the rock is commonly essentially impervious.

The predominant mineral that has rehealed fractures in the rock is calcite, but chlorites and zeolites also occur. The presence of these minerals increases the overall strength of the rock by increasing the cohesion and consequently the shear strength along fractures in which they occur. The presence of these secondary minerals is not, however, required in order for the rock to have sufficient strength to provide an adequate foundation. Materials with no cohesion, such as clean sands and gravels, have high bearing capacities provided they are dense. Fractures in the rock are irregular, and the rock in the foundation is confined, thus the fractures have considerable shear strength.

The basalt varies from vesicular to dense but is generally vesicular. It is blue to blue-black, usually fine grained but occasionally slightly porphyritic. When unweathered it is very hard. Both core and outcrops of basalt commonly exhibit a high degree of fracturing, due to cooling stresses rather than tectonic forces.

The rock in the ridge is often broken by closely spaced fractures and often contains weathered zones, some of which are at considerable depths. Soil cover on the ridge is usually thin. Rock frequently crops out on the ground surface. There are, however, some depressions in the top of the bedrock, and several old stream channels have been eroded across the ridge. An old channel containing potholes filled with sand occurred under the southeast edge of the cooling tower.

Thick alluvial deposits occur in the valley to the west of the ISFSI location. The geophysical survey indicated that the alluvium has a maximum thickness of approximately 340 feet in the area between the ridge and the hills to the west of the site. The accuracy of the geophysical survey was partially verified by DH-7, which was drilled at a point where the survey indicated the alluvium to be approximately 280 feet thick. The hole penetrated alluvium to a depth of 278 feet but it had to be abandoned at that depth. However, the total thickness of the alluvium at the hole is considered to be close to 280 feet since the hole terminated in gravels and boulders which probably occur near the top of bedrock.

The upper approximately 80 to 100 feet of the alluvium usually consists of soft to very soft clayey silt to silty clay with varying amounts of intermixed fine sand and layers of silty fine sand. It also contains considerable amounts of decomposed wood fragments and vegetation, particularly in the 50-foot depth range. In DH-5, -9, and -10, and P-1, -2, and -3, the upper 25 to 35 feet of the alluvium is predominantly silty fine sand, but contains significant quantities of silty clay and clayey silt. DH-7 contains less sand in the upper 35 feet than do the above holes. Holes in the alluvium encountered principally soft clayey silt between approximately 30 to 90 feet in depth.

The seismic survey distinguished a denser material, as indicated by a higher velocity, below a depth of approximately 80 feet, and DH-7 confirmed the existence of the denser material, which is a thick layer of fine and very fine-grained sand. In DH-7, progressively more sand was encountered below a depth of about 100 feet, and between depths of 115 feet and 270 feet the material was essentially fine sand.

2.6.2 VIBRATORY GROUND MOTION

2.6.2.1 Seismicity

The ISFSI is located in an area that experiences moderate seismic activity. Most of the seismic activity has been concentrated in three areas - one about 40 miles east of the site, another

approximately 25 miles south of the site, near Portland, Oregon, and a third approximately 65 to 120 miles north of the site, along a belt between Olympia and Seattle, Washington. It is important to note that there is no alignment of epicenters to suggest the existence of any active fault near the site (References 2 - 5).

2.6.2.2 Geologic Structures and Tectonic Activity

Bedrock at the site consists chiefly of basaltic flows and associated pyroclastics included in the Goble series. This unit contains a thick section of widespread volcanics that crop out along both sides of the Columbia River in this area. The unit is Upper Eocene in age. Generally, the rock ranges from dense to vesicular basalt with interbedded agglomerates. As is common with the volcanics of this area, the rocks show closely spaced fractures and locally contain weathered zones.

The data obtained from the geologic study, core drilling and geophysical surveys indicate that the foundation is composed of moderately hard, competent rock that is suitable for an ISFSI. Rock types include tuffs, tuff breccias, flow breccias, basalts, and agglomerates. Of these, tuffs are the most prevalent. Unconfined compressive strengths of the 41 samples of tuffs, tuff breccias, and flow breccias (200 tons/sq. foot) with an average of 1225 psi (88 tons/sq. foot). The specific gravity of the tuffs range from about 1.84 to 2.33 with an average of about 2.10. Absorption ranges from 5 to 17 percent with an average of around 10 percent. The ISFSI reinforced concrete Storage Pad is founded on rock. Therefore, studies for liquefaction, thixotropy, or differential consolidation of soils were not required.

Geophysical surveys showed compression wave velocities to be 8,200 to 10,600 fps, which indicates adequate foundation conditions. Shear wave velocities of foundation rock ranged from 4,500 to 5,000 fps. A value of 1.9×10^6 psi was used for the dynamic modulus of elasticity, based on the geophysical measurements. Values for static modulus of elasticity were obtained by numerous laboratory tests on representative core samples. A conservative value of 0.8×10^6 was used for the design. No uphole velocity measurements were made.

2.6.2.3 Maximum Earthquake Potential

The largest historically recorded shock center within 50 miles of the site occurred on November 5, 1962. It had an epicentral intensity of VII about 35 miles south of the site near Vancouver, Washington. At Longview, Washington, and at Rainier, Oregon, the intensity was reported as VI; however, the damage was confined to cracked plaster.

On October 12, 1877, an intensity VII earthquake was felt in Portland, Marshfield (now called Clackamas), and Cascades (now called Cascade Locks), Oregon. The location of the epicenter of this shock is uncertain. It is plotted 45 miles from the site in the southern part of Portland, near Clackamas. The epicenter may have been farther east toward Cascade Locks, Oregon. The

original reference to this earthquake is by Rockwood (Reference 6). Unfortunately, he does not state exactly what his source is. His account follows:

> "October 12, 1877. Quite severe shocks were felt in Oregon occurring in Portland at 1:53 p.m. (Two shocks being noticed; at Marshfield, Clackamas Co., at 1:45 p.m.; and at Cascades at 9:00 a.m.) The vibrations were in each case from north to south and were sufficiently violent to overthrow chimneys."

A second reference by Holden (Reference 7) assigned an intensity VIII on the Rossi-Forel (R-F) scale to this earthquake. Townley and Allen (Reference 8) also showed intensity VIII (R-F scale). Rasmussen (Reference 9) and Berg (Reference 2) both used intensity VIII but changed to the Modified Mercalli (MM) scale. The U.S. Coast and Geodetic Survey (Reference 4) shows intensity VII. It is generally accepted that VII (MM) is the equivalent of VIII (R-F).

In correspondence with Bechtel, Mr. Don Tocher, Director of the Earthquake Mechanism Laboratory of the U.S. Coast and Geodetic Survey in San Francisco, stated that he believes an intensity VII (MM) to be correct for the October 12, 1877, earthquake, and states that the intensity VIII (MM) referred to by Berg (Reference 2) and Rasmussen (Reference 5) is probably a result of carelessness in designated scales when changing from the Rossi-Forel scale to the Modified Mercalli scale.

Seven earthquakes of maximum intensity VI were centered within 50 miles of the site. On February 3, 1892, a shock of intensity VI occurred at Portland, and strong vibrations were felt at Astoria, Salem, and Lake Harney, 235 miles southeast of Portland. On December 29, 1941, a shock of intensity VI was felt near Portland, Oregon, about 38 miles south of the site. Another shock, centered near Portland on December 15, 1953, was felt with intensity IV at Kalama, Washington. On September 15, 1961, an earthquake was centered about 33 miles east of the site. The maximum intensity of VI was reported at Swift Dam on the Lewis River, which was designed for 0.10 g, but no damage was done to the dam. At Rainier, the shock on September 15, 1961, was also of intensity IV, and at Carrolls, the intensity was I to III. An after shock on September 17, 1961, was also intensity VI, but was not felt at Rainier or Carrolls. On November 6, 1961, a shock was centered about 30 miles south of the site. At Rainier, the intensity was reported as V. On December 26, 1963, an earthquake of intensity VI occurred about 30 miles southwest of the site. The intensity was reported as V in Longview, and III at Goble. It was not felt in Rainier.

The largest earthquakes within 150 miles of the site were two shocks of epicentral intensity VIII which occurred on April 13, 1949, and April 29, 1965. The epicenters were in the Puget Sound area, approximately 70 miles and 95 miles, respectively, northeast of the site. Heavy damage, deaths, and injuries were reported in the epicentral areas. The accelerograms indicate a maximum resultant horizontal acceleration of about 0.10 g at Seattle, where the intensity was VIII. At Rainier, which is about the same distance from the epicenter as is Seattle, the intensity

was also given as VIII, which is the greatest intensity reported historically at Rainier from any earthquake. However, the intensity at Rainier was based on damage to only one building, and that building was founded on marshy ground. A study of the damage reported at Rainier indicates that a lower intensity might reasonably be assigned. At Goble, the intensity due to the 1949 earthquake was only VI, apparently because Goble is founded on rock. The ISFSI is founded on rock belonging to the same unit as the rock that underlies Goble.

The April 29, 1965, earthquake caused lower intensities in the site area than the 1949 earthquake. The intensity at Kelso and Longview was VI; at Rainier it was V; and at Goble only IV.

2.6.2.4 Seismic Margin Earthquake

The maximum intensity that has been reported at Rainier is VIII. Since this intensity occurred on overburden, it is probable that on rock at the site the intensity for this same shock was not over VII. Intensity VII correlates with a horizontal acceleration of 0.12 g according to Hershberger (Reference 10). This historical data formed the bases for assigning the Safe Shutdown Earthquake (SSE).

The SSE was determined such that any probable earthquake experienced at the site would not exceed the intensity selected. An intensity of VIII was selected since it was probable that an earthquake of that magnitude had never been experienced at the site. An intensity VIII is equivalent to an acceleration of 0.25 g.

There have been significant changes in the perception of earthquake hazards in the Pacific Northwest since the time of the initial design and licensing of the Trojan Plant. It is now commonly believed among the geoscience community that large subduction zone earthquakes likely occurred along the Oregon-Washington-Vancouver Island coast (known as the Cascadia margin, or Cascadia Subduction Zone) within the recent past (Holocene), and that the potential for such events to occur in the future should be considered in any evaluation of safety and reliability of critical facilities during earthquake loading.

In 1987, in response to the emerging issue of potential subduction zone earthquakes, PGE initiated a program of close monitoring of earthquake hazard research conducted along the Cascadia margin. The results of these studies, together with studies initiated by PGE, have been use to characterize the maximum events that could be expected to occur in the region and the resulting free-field ground motions that may occur at the site. This maximum potential earthquake that could affect the site is called the Seismic Margin Earthquake (SME).

These studies determined a value for the SME peak horizontal ground acceleration of 0.38 g (Reference 11). A 1994 earthquake in Northridge, California slightly changed the conclusions of these studies in that the controlling earthquake varies from the intraslab source for peak ground acceleration, to the crustal earthquake for periods between 0.1 and 0.6 seconds, to the interface

source for longer periods, whereas in the original study, only the intraslab and interface sources were controlling (Reference 12). Nonetheless, the response spectra are bounded by the Regulatory Guide 1.60 spectrum shape when anchored at the 0.38 g peak acceleration. Therefore, input from recent earthquakes shows that the SME is the appropriate design basis event for the ISFSI, as required by OAR 345-26-390, and the ISFSI design considers the SME peak horizontal acceleration of 0.38g.

2.6.3 SURFACE FAULTING

The site is in the Willapa Hills geomorphic province, a part of the Coast Range. Most of the region is below 2000 feet in elevation. The descent from the hills to the Columbia River is rather precipitous, but elsewhere the hills merge gradually into the surrounding lowlands.

The bedrock in the area is comprised of a series of moderately folded tertiary formations of both sedimentary and volcanic origin. The folding in the area conforms generally to the northwest Coast Range structural trend. The Eocene formations north of the Columbia River are folded as part of a syncline. Pliocene sediments in the vicinity of the site are only slightly warped and the Pleistocene and Recent deposits appear to be flat lying and undisturbed. A detailed discussion of the regional geology is presented in Section 2.6.1.1.

Faulting is minor in the structural development of the area, and is generally of small displacement. Many of the mapped faults in the area are based on topographic lineations in the pre-Pleistocene strata. No evidence of post-Pleistocene surface displacement has been found in the area.

An extensive investigation to locate faults ~~which~~ *that* might be significant to the site was made as part of the geologic evaluation for siting the Trojan Nuclear Plant. A special effort was made to detect any lineations or indications of offset in the alluvium, terrace deposits, or the Pleistocene alluvial deposits. The details of the investigation are contained in the Trojan Nuclear Plant FSAR. As a result of the investigation, the following conclusions were reached:

1. The "Kelso Fault," as indicated in Bulletin 54 of the Washington Division of Mines and Geology (Reference 13), does not exist.

2. The available geologic evidence indicates that there is not a fault in the old stream channel west of and adjacent to the site.

3. The field evidence indicates that the Clatskanie fault does not extend farther east than indicated on the Oregon State Geologic Map.

4. The available geologic evidence does not indicate that a fault exists along the Columbia River adjacent to the site.

5. The fault zone exposed in the road cut southeast of Kelso and 4.7 miles from the site is apparently not extensive and probably has not experienced movement since deposition of the Troutdale formation during lower Pliocene time. This fault zone is not significant to the site.

6. There is no fault within 5 miles of the site which has experienced movement since Pleistocene time.

The size of faults within a 200-mile radius of the site, together with the known historical activity and distance from the site, suggest that ground accelerations reaching the site from these mapped faults would fall well below those for which the ISFSI was designed.

2.6.4 STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

The existing gravel fill ~~will be~~*was* removed during excavation for the ISFSI reinforced concrete Storage Pad and a nominal 2 ft. layer of graded and compacted gravel fill ~~will be~~*was* placed under the concrete slab. No soluble or cavernous rocks underlie the site area, and no poorly consolidated or mineralogically unstable rocks occur at the site. No oil, gas, other mineral extraction, or subsurface mining occurs or has occurred in the vicinity of the site. It is therefore concluded that future subsurface subsidence is not a problem at the site.

No evidence of recent regional warping was encountered, and USGS Bulletin 1119, 1963, by Donald E. Trimble (Reference 14), states:

> "...deformations of the region apparently ended in early Pliocene time as the Troutdale formation is the youngest one involved in the warping. The undeformed boring lava overlies an erosional surface of considerable relief cut on the Troutdale. Post-Troutdale crustal movements, if any, have consisted only of vertical movement of regional extent."

The geologic mapping near the site, and the study of aerial photographs, did not disclose any indications of recent regional warping.

Because the ISFSI reinforced concrete Storage Pad is founded on the crest of a rock ridge which shows no evidence of deformation since Pliocene time, no unrelieved residual stresses should be expected to exist in the foundation rock. No evidence of unrelieved residual stress was observed during previous excavations for the nuclear plant foundations.

2.6.4.1 Geological Foundation Evaluation

Fifty-five samples of rock core from the drill holes were tested in the Bechtel Geology Laboratory in San Francisco for the original siting of Trojan Nuclear Plant. Standard testing procedures were used in the determination of the physical properties of the rocks.

Rock types include tuffs, flow breccias, basalts, and agglomerates. Since contacts between rock types are not always horizontal, and the rock units are often lenticular, the ISFSI reinforced concrete Storage Pad may rest on one rock type in one portion of the Storage Pad and a different rock type in another part of the Storage Pad. This does not complicate the design, however, since the Storage Pad is designed for the strength of the weakest rock types, which tests have consistently shown to be the tuffs.

Since the tuff is assumed to form the foundation for the reinforced concrete Storage Pad, its strength will generally determine the allowable bearing capacity of the foundation rocks. The lowest and highest unconfined compressive strength of the 41 samples of tuff which were tested are 360 psi (26 tons/ft^2) and 2790 psi (200 tons/ft^2). The average is 1225 psi or 88 tons/ft^2.

Testing showed the specific gravity of the tuff to be in the range of about 1.84 to 2.33 with the average of about 2.10. Porosity varies from 10.3 to 32.4 percent with 22 percent ~~begin~~ *being* about the average. Absorption ranges from 5.0 to 17 percent with an average of around 10 percent.

Soil test data such as grain size, Atterberg limits, water content, soil density, and shear strength were not required because the ISFSI reinforced concrete Storage Pad is founded on rock. Tests were made in the flat alluvial area where access roads are found.

The geophysical survey showed the compression wave velocities in the bedrock to be 8,200 to 10,600 fps, which indicates adequate rock for good foundation conditions. Shear-wave velocities of the foundation rock ranged from 4,500 to 5,000 fps. The flat alluvial area west of the rock ridge had compression wave velocities of 2,000 to 2,500 fps for the overburden and velocities of 4,700 to 5,100 fps for the older compacted overburden. No equipment important to safety is founded on this alluvial material.

In determining the bearing capacity of these rocks, it was noted that the rock which will form the foundation has been preloaded by the weight of overlying material, much of which has been removed by erosion. This, as well as the jointing and weathering, were considered in determining the allowable bearing capacity of the rock.

Values for the static modulus of elasticity were obtained by numerous laboratory tests of representative core samples. A conservative value of 0.8×10^6 psi for the static modulus of elasticity was used for design. For dynamic modulus of elasticity, a value of 1.9×10^6 psi was

used in design. The value was determined by geophysical measurements on in-site foundation rock. Values for Poisson's ratio (dynamic) range from 0.28 to 0.36. From these values, the bulk modulus was computed to be 1.8×10^6, and a value of 0.7×10^6 psi was calculated for the shear modulus. For design purposes, soil structure and rock foundation interaction was assumed to be negligible, thus no damping was used for the rock.

Observations of groundwater levels at the site indicated that the ridge supports a local groundwater mound, probably maintained by rainfall trapped in depressions on the top of the ridge. Previous excavation for the nuclear plant showed the water to be trapped in joints and fractures in the rock and the water drained off rapidly. No artesian pressures were encountered during the previous excavation.

2.6.5 STABILITY OF SLOPES

Permanent excavated slopes to the north and east of the ISFSI are through rock and no problem with long-term stability of such slopes should be anticipated. Sloughing of small amounts of loose weathered surface material would likely not reach the ISFSI, and would not represent a hazard to the ISFSI in any event.

2.6.6 VOLCANOLOGY

Due to the proximity of existing inactive volcanoes in the Cascade Range east of the site, the significance of renewed activity of these volcanoes was considered with regard to the possible effects on the site. The historical seismicity of the Cascade volcanoes was considered as well as the type of volcanic activity that might conceivably occur in the future. The volcanoes that were considered and their distances and direction from the site are:

1. Mt. St. Helens, Washington - 34 miles - ENE
2. Mt. Adams, Washington - 67 miles - E
3. Mt. Hood, Oregon - 74 miles - SE
4. Mt. Rainier, Washington - 77 miles - NE

The details of volcanology are described in the Trojan Nuclear Plant FSAR. In summary, the conclusions of the FSAR were that while predictions related to future volcanic activity are impossible to make, the possibility of volcanic activity significant to the site is considered very remote. In addition, even if activity did occur, it is extremely unlikely that it would occur without warning.

The series of Mount St. Helens eruptions in 1980 resulted in tephra accumulations at the Trojan site of no more than 1/8 inch. If Mount St. Helens were to have another tephra eruption similar to the May 18, 1980 eruption, only directed towards the ISFSI with winds blowing towards the ISFSI, then the expected ash fall accumulation would be about 1.8 inches.

For these reasons, the risk to the ISFSI from volcanic cones in the Cascade Range is considered minimal.

2.7 SUMMARY OF SITE CONDITIONS AFFECTING CONSTRUCTION AND OPERATING REQUIREMENTS

The site-specific phenomena and characteristics described in this chapter have been used to define appropriate design criteria, as described in Chapter 3. Table 2.7-1 is a summary of site-specific information for the ISFSI.

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Table 2.1-1

Public Facilities and Institutions

	Enrollment	Distance (mi)	Direction
WASHINGTON SCHOOLS:			
Longview			
Broadway	114	7	NNW
Carrolls	131	2-1/4	NNE
Cascade Middle School	910	8-1/2	NNW
Columbia Heights	329	8-1/2	NNW
Columbia Valley Gardens	496	8-1/2	NNW
Kalama High School	413	3	SE
Kalama Grade School	410	3	SE
Kessler	535	7	NNW
Lower Columbia Jr. College	3869	7-1/2	NNW
Mark Morris High School	1189	7-3/4	NNW
Mint Valley	688	8-3/4	NNW
Monticello Middle School	866	7-2/3	NNW
Natural High School	24	8-3/4	NNW
Olympic	562	7-2/3	NNW
R. A. Long High School	952	7-2/3	NNW
Robert Gray	582	10-1/2	NNW
Rose Valley	173	5	NNE
St. Helens	421	7	NNW
Kelso			
Barnes	552	8-1/4	N
Beacon Hill	447	9-1/4	N
Butler Acres	528	8-2/3	N
Catlin	426	7-2/3	NNW
Coweeman Middle School	662	7	N
Huntington Middle School	583	8	N
Kelso High School	1222	7	N
Wallace	467	6-2/3	N

Table 2.1-1

Public Facilities and Institutions

	Enrollment	Distance (mi)	Direction
Private			
Columbia Heights	175	8-3/4	NNW
Kelso Christian Academy	200	7-1/4	NNW
Longview Christian School and Day Care	700	8	NNW
St. Rose	237	7	NNW
Seventh Day Adventist	31	7	NNW
OREGON SCHOOLS:			
Columbia City Elementary	151	10-1/4	SE
Goble Elementary	119	2-1/4	S
Hudson Park	349	6-1/4	WNW
Rainier Elementary	307	4-1/4	NW
Rainier High School	441	6	NW
Rainier Middle School	244	3-3/4	NW
HOSPITALS			
St. Johns Medical Center	340	6-2/3	NNW
NURSING HOMES			
American Convalescent & Retirement Home	74	7	NNW
Campus Towers	110	7-2/3	NNW
Canterbury Retirement Inn	104	7	NNW
Cowlitz Convalescent Center	46	7-1/4	NNW
Delaware Plaza	87	6-1/2	NNW
Frontier Extended Care Facility	136	6-3/4	NNW
Manor Nursing Home	51	7	NNW
Northwest Continuum Care Center	66	9-1/2	NNW
Park Royal	49	7	NNW
Woodland Care Center	58	10	SSE

Table 2.1-1

Public Facilities and Institutions

	Enrollment	Distance (mi)	Direction
PARKS			
Bailey Park		8-3/4	NNW
Clearview Park		6-1/2	NNW
Cloney Park		7	NNW
Gerhart Gardens Park		5	N
Highlands Park		6	NNW
Hudson Parcher Park		4	NW
John Null Park		8-1/3	NNW
Kellogg Park		8	NNW
Kelso Rotary Park		8	N
Lake Sacajawea Park		7	NNW
Prescott Beach Park		1	NNW
R. A. Long Square		7-1/4	NNW
Riverside Park		10	N
Roy Morse Park		10-3/4	NNW
Scott Hollow Park		7-1/4	N
Seventh Avenue Park		6-1/2	NNW
Tam O'Shanter Park		6-3/4	N
Vandercook Park		7-1/4	NNW
Windermere Park		10-3/4	NNW
COUNTY FAIR GROUNDS			
Cowlitz County	56,000 during fair - 54,000 total during rest of year	7	NNW

Table 2.1-2

1994 LAND USE CENSUS

NEAREST LOCATION TO TROJAN WITHIN A FIVE-MILE RADIUS

	Radial Mileage for Nearest Location				
Directional Sector	Residence	Garden	Milk Cow	Milk Goat	Meat Animal
N	0.70	0.80	None	None	None
NNE	2.00	2.80	4.00	2.00	None
NE	1.60	2.00	None	None	2.00
ENE	2.30	None	None	None	4.00
E	1.30	1.30	None	None	1.30
ESE	0.80	2.40	None	None	2.40
SE	2.30	2.80	None	None	2.70
SSE	1.40	3.00	None	None	3.00
S	1.20	1.40	None	None	2.00
SSW	0.90	2.60	None	2.60	1.00
SW	1.50	3.00	None	3.00	2.10
WSW	1.40	1.60	None	None	3.20
W	1.70	2.10	None	None	2.20
WNW	1.70	1.70	None	None	1.70
NW	1.20	1.20	None	None	2.00
NNW	0.60	0.60	None	None	None

Table 2.2-1

Nearby Industrial Facilities

Security-Related Information Table Withheld Under 10 CFR 2.390.

Table 2.2-1

Nearby Industrial Facilities

Security-Related Information Table Withheld Under 10 CFR 2.390.

Table 2.2-1

Nearby Industrial Facilities

Security-Related Information Table Withheld Under 10 CFR 2.390.

Table 2.2-1

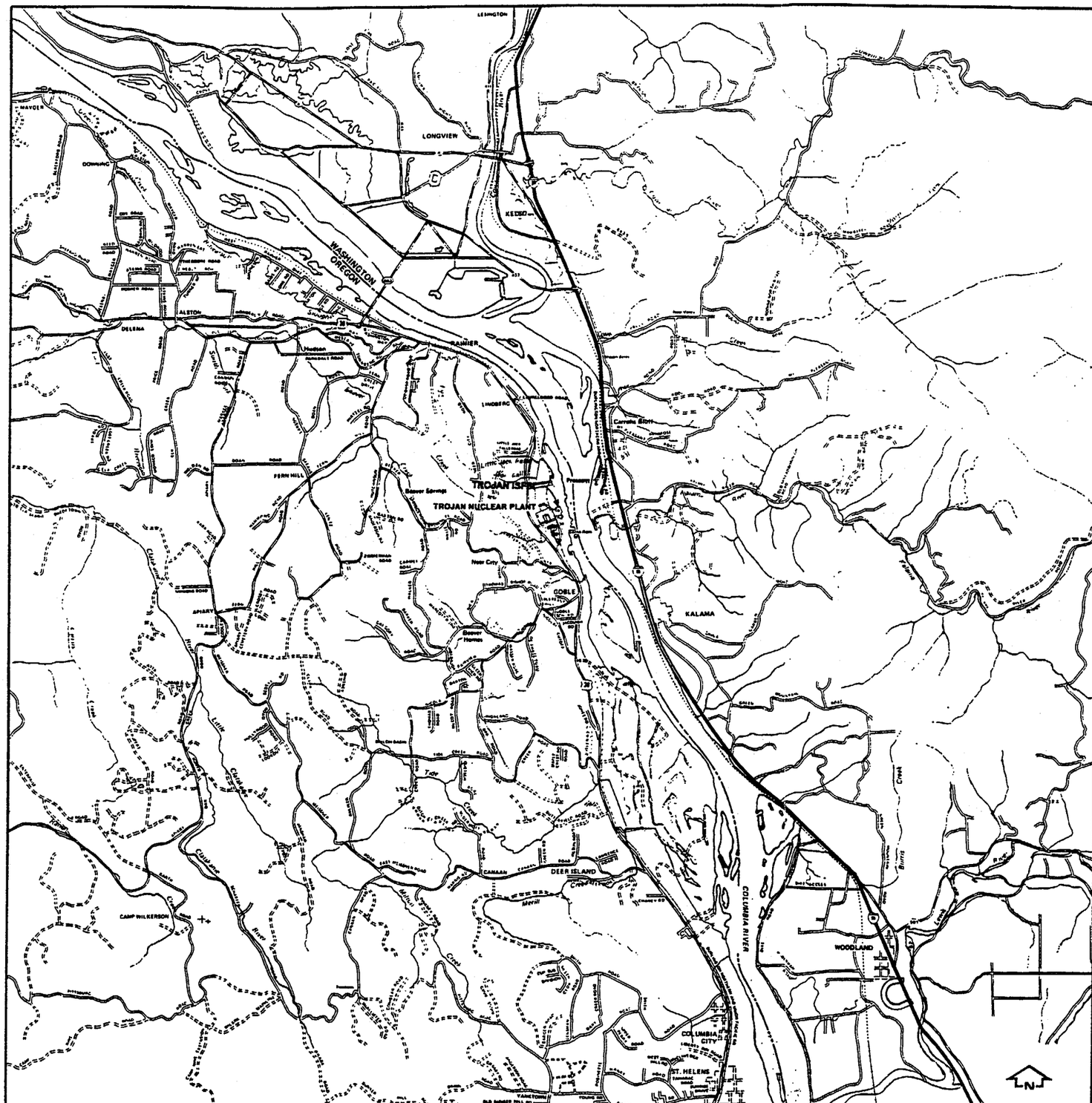
Nearby Industrial Facilities

Security-Related Information Table Withheld Under 10 CFR 2.390.

Table 2.7-1

Summary of Site Conditions Affecting Construction and Operating Requirements

Parameter	Extreme Value Measured or Estimated	Design Value
Ambient Temperature	-20°F minimum, 105°F maximum (Longview/Kelso, Washington) -3°F minimum, 107°F maximum (Portland, Oregon)	-40°F minimum 100°F maximum 125°F short term extreme
Tornado Winds/Pressure Drop	240 mph, maximum 190 mph, rotational 50 mph, translational 1.5 psi, pressure drop 0.6 psi/sec, rate of pressure drop (Regulatory Guide 1.76, Region III)	360 mph, maximum 290 mph, rotational 70 mph, translational 3.0 psi, pressure drop 2.0 psi/sec, rate of pressure drop
Maximum Flood Level	42.75 ft MSL (estimated - seismically induced dam failure with coincident wave runup)	None required - nominal ISFSI grade elevation is 45 ft MSL
Snow and Ice Loading	10 inches of snow in 24 hours (Longview/Kelso) 16 inches of snow in 24 hours (Portland)	100 psf ground load (12 inches of snow exerts from 1 to 6 psf)
~~Atmospheric Dilution Factor~~	~~$7.10 \cdot 10^{-5}$ sec/m^3~~ ~~(measured at Trojan 1976-1993)~~	~~$4.08 \cdot 10^{-3}$ sec/m^3~~
Seismic Margin Earthquake	0.38g, horizontal acceleration (estimated) 0.25g, vertical acceleration (estimated)	0.38g, horizontal acceleration 0.25g, vertical acceleration



LEGEND

HIGHWAY, HIGH TYPE

HIGHWAY, LOW TYPE

ALL-WEATHER ROAD

SECONDARY ROAD

N

0 1 2 3 4 5 miles

TROJAN ISFSI
SAFETY ANALYSIS REPORT

FIGURE 2.1-1
SITE LOCATION

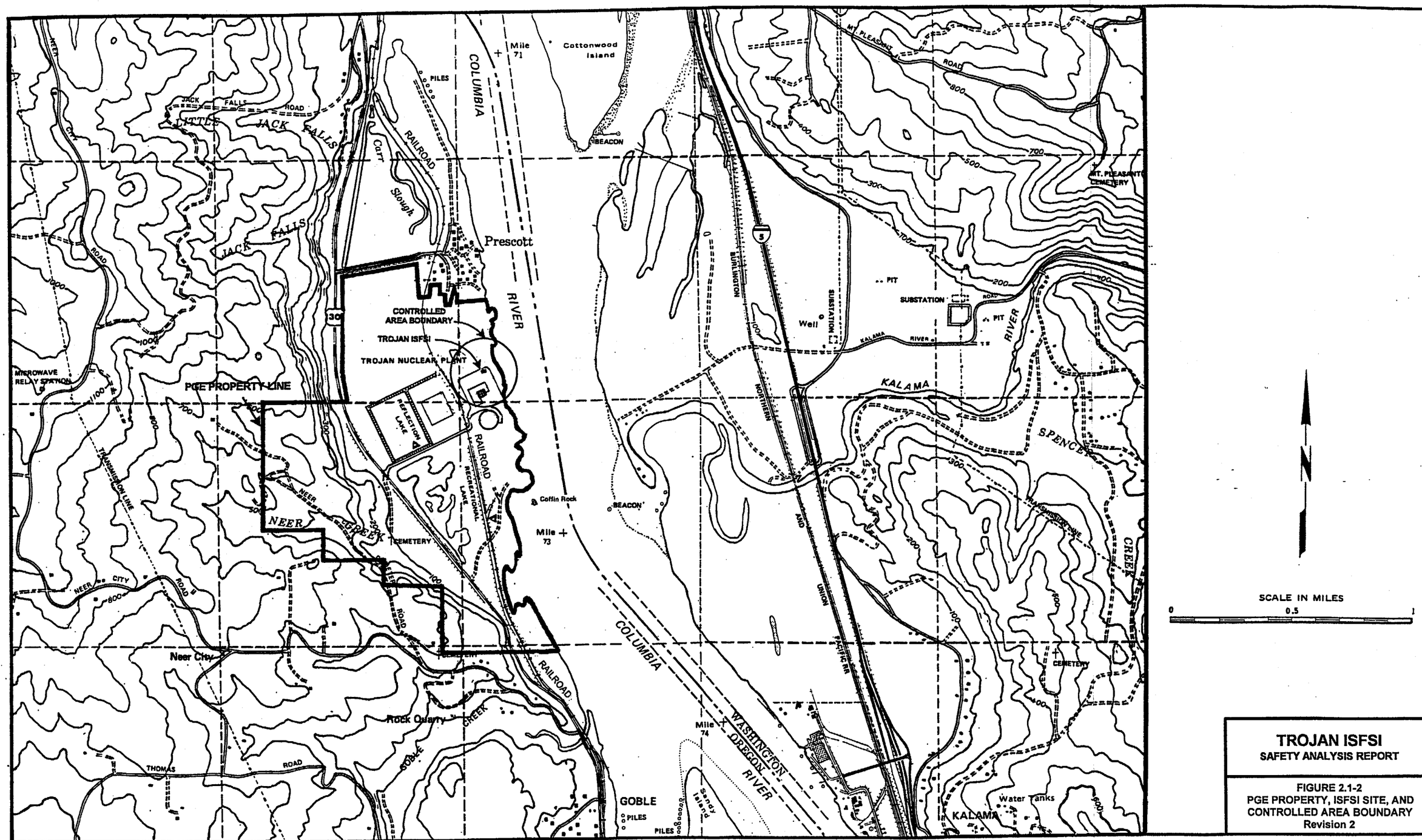


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FIGURE 2.1-2
PGE PROPERTY, ISFSI SITE, AND CONTROLLED AREA BOUNDARY
Revision 2

Security-Related Information Figure Withheld Under 10 CFR 2.390.

TROJAN ISFSI
SAFETY ANALYSIS REPORT

FIGURE 2.1-3
ISFSI LAYOUT

POPULATION IN ANNULAR RINGS	
0 — 1	179
1 — 2	432
2 — 3	1981
3 — 4	2467
4 — 5	3507
5 — 10	58502

POPULATION WITHIN RADIAL DISTANCE of SITE	
1	179
2	611
3	2592
4	5059
5	8566
10	67068

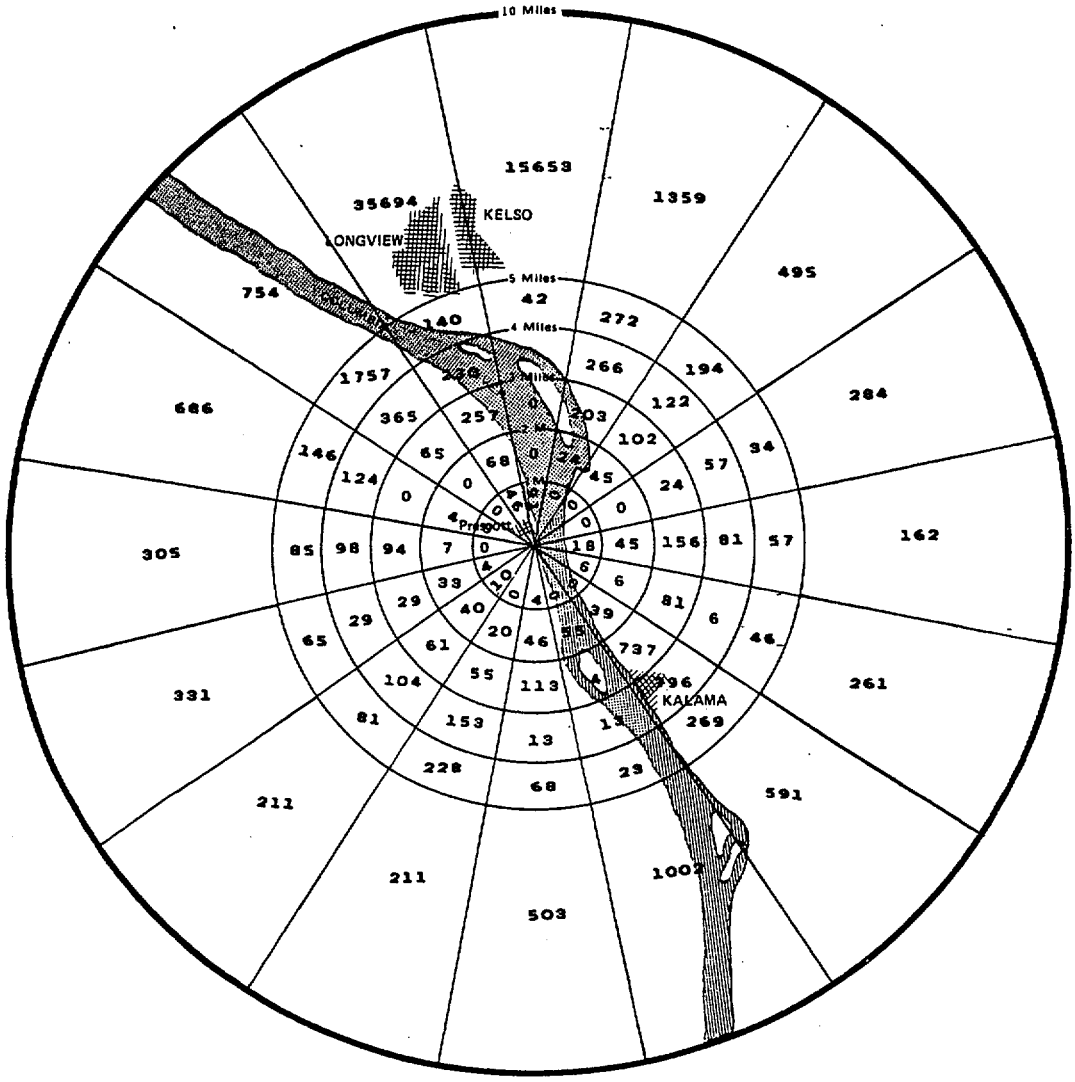

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FIGURE 2.1-4
1990 POPULATION DISTRIBUTION WITHIN 10 MILES

POPULATION IN ANNULAR RINGS	
0 — 1	188
1 — 2	449
2 — 3	2059
3 — 4	2572
4 — 5	3671
5 — 10	60458

POPULATION WITHIN RADIAL DISTANCE of SITE	
1	188
2	637
3	2696
4	5268
5	8939
10	69397

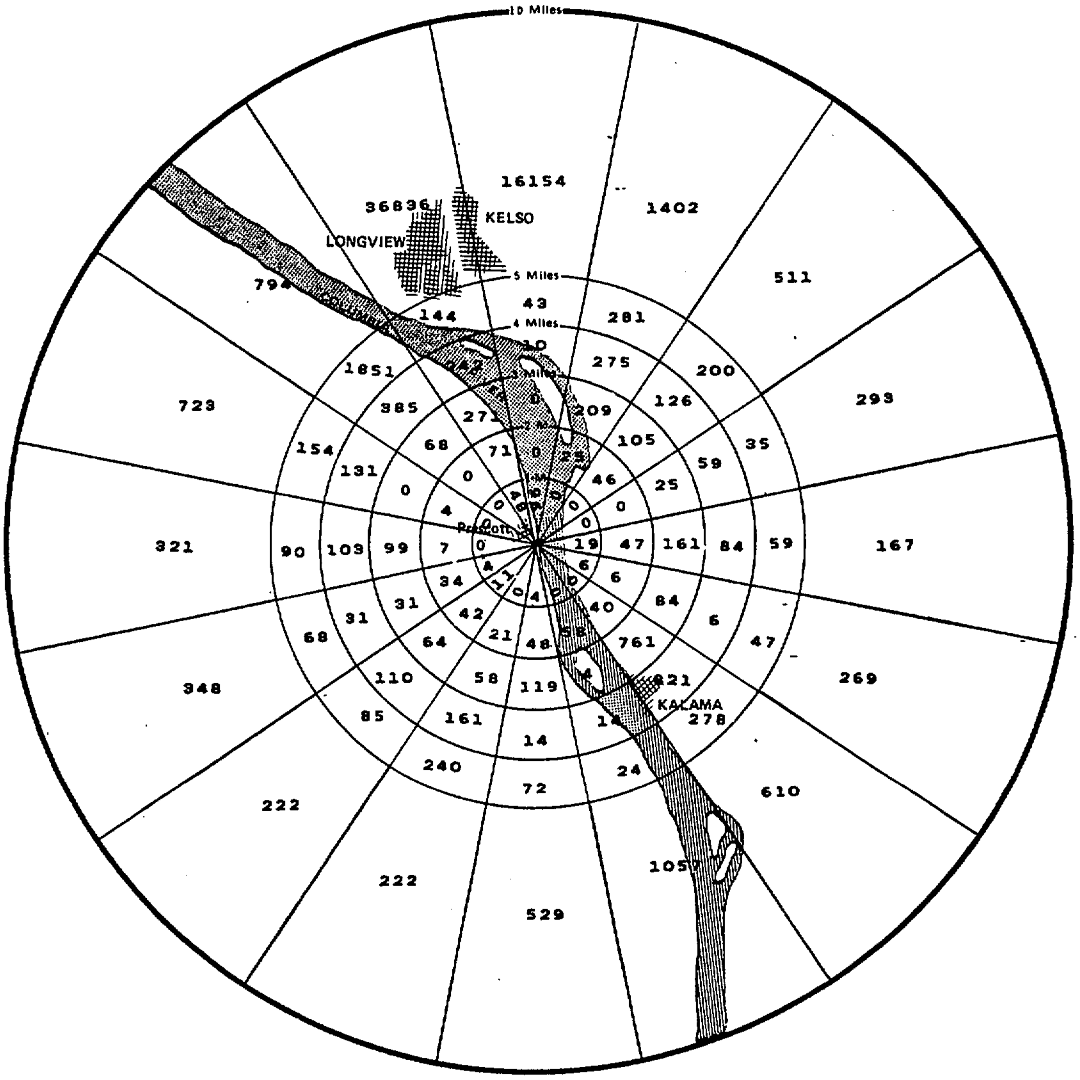

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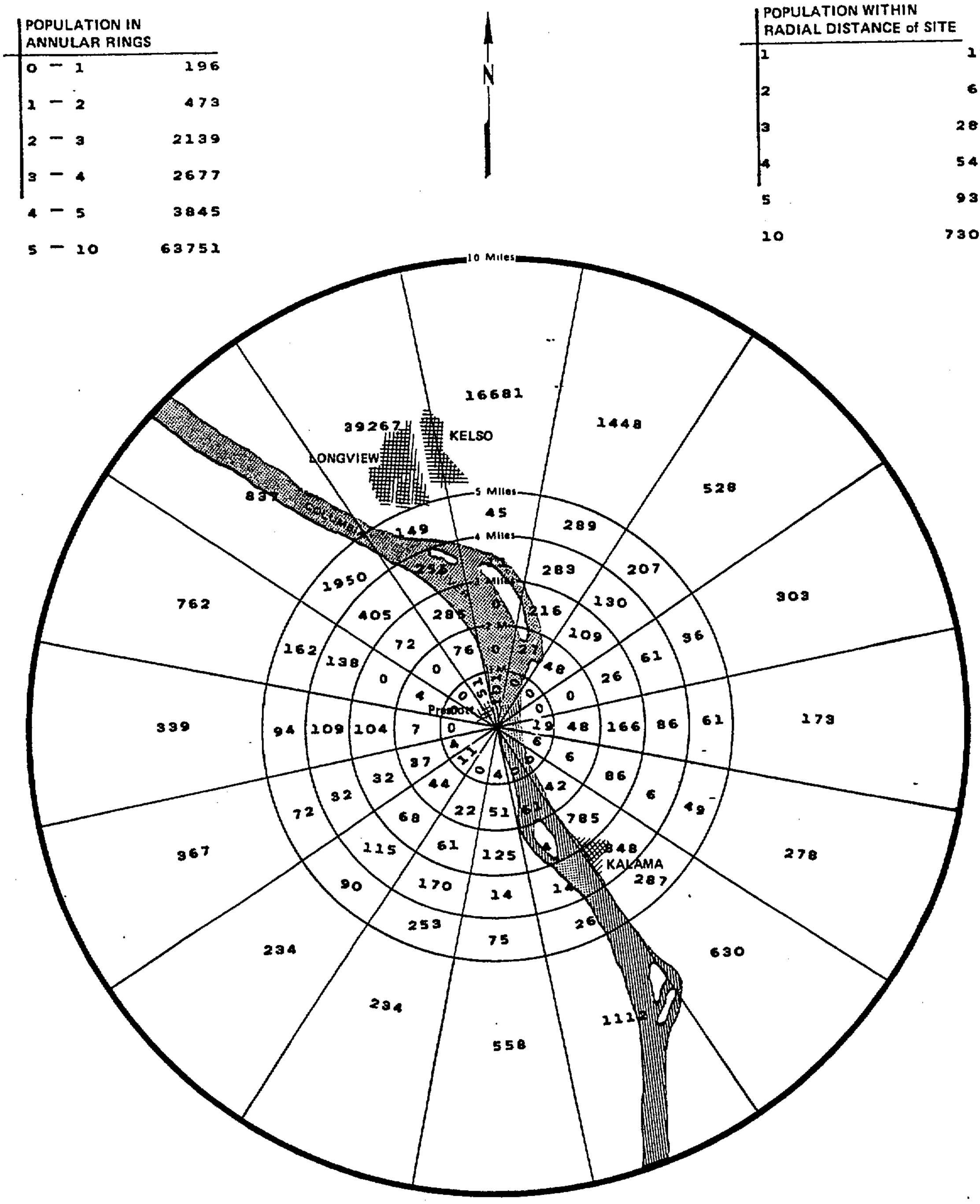
FIGURE 2.1-5
2000 PROJECTED POPULATION DISTRIBUTION WITHIN 10 MILES

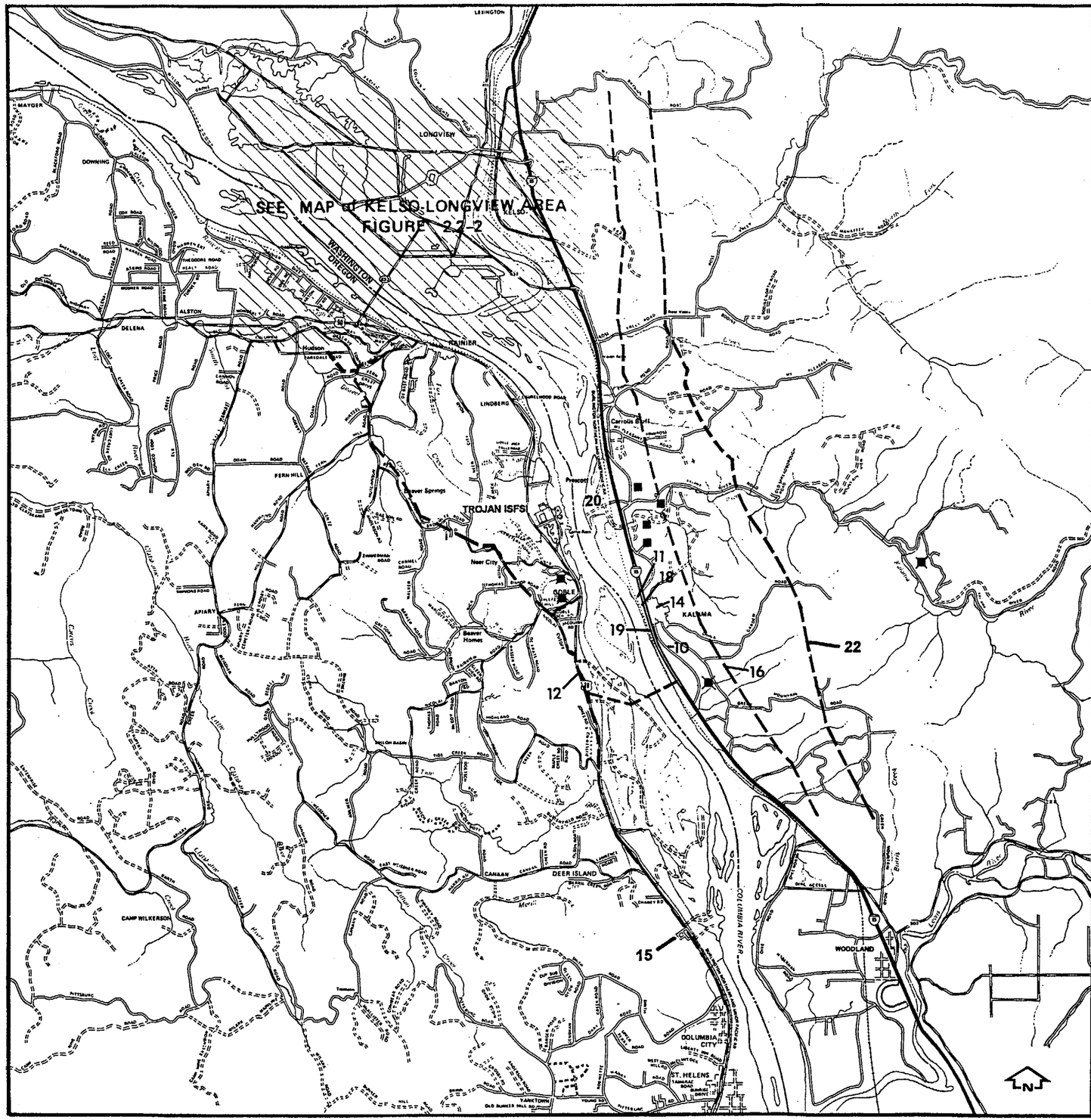
POPULATION IN ANNULAR RINGS	
0 — 1	196
1 — 2	473
2 — 3	2139
3 — 4	2677
4 — 5	3845
5 — 10	63751

POPULATION WITHIN RADIAL DISTANCE of SITE	
1	196
2	669
3	2808
4	5485
5	9330
10	73081



TROJAN ISFSI
SAFETY ANALYSIS REPORT

FIGURE 2.1-6
2010 PROJECTED POPULATION DISTRIBUTION WITHIN 10 MILES



GAS or OIL TANKS
QUARRY
GRAVEL PIT

Note: Numbers Refer to Table 2.2-1

LEGEND
HIGHWAY, HIGH TYPE
HIGHWAY, LOW TYPE
ALL-WEATHER ROAD
SECONDARY ROAD

N

0 1 2 3 4 5 miles

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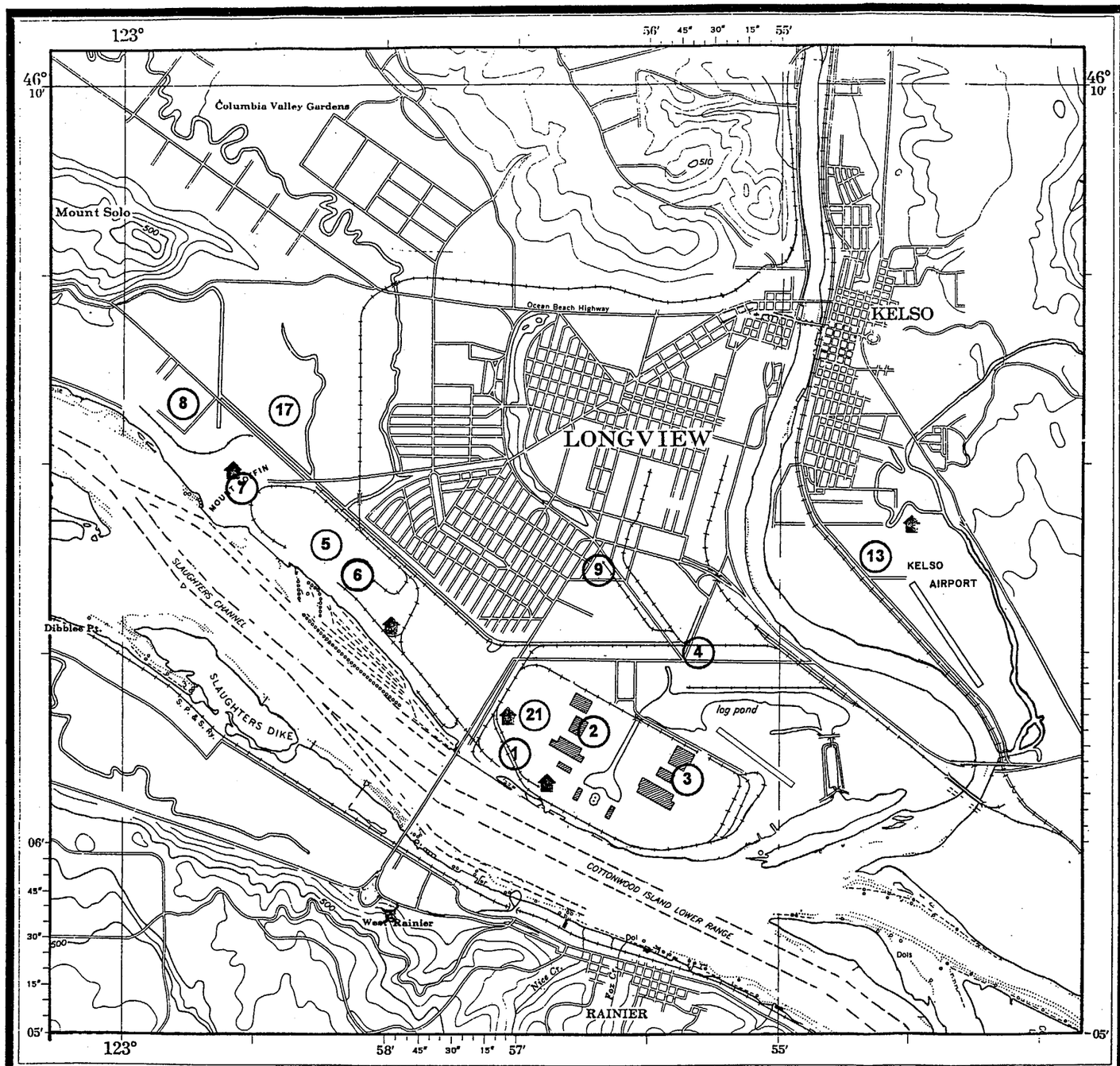
SAFETY ANALYSIS REPORT

FIGURE 2.2-1
NEARBY INDUSTRIAL ACTIVITY



Source: U S Coast and Geodetic Survey 68

GAS or OIL TANKS
QUARRY

Note: Numbers Refer to Table 2.2-1

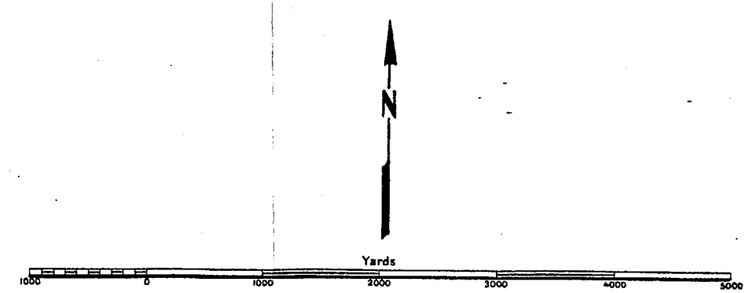


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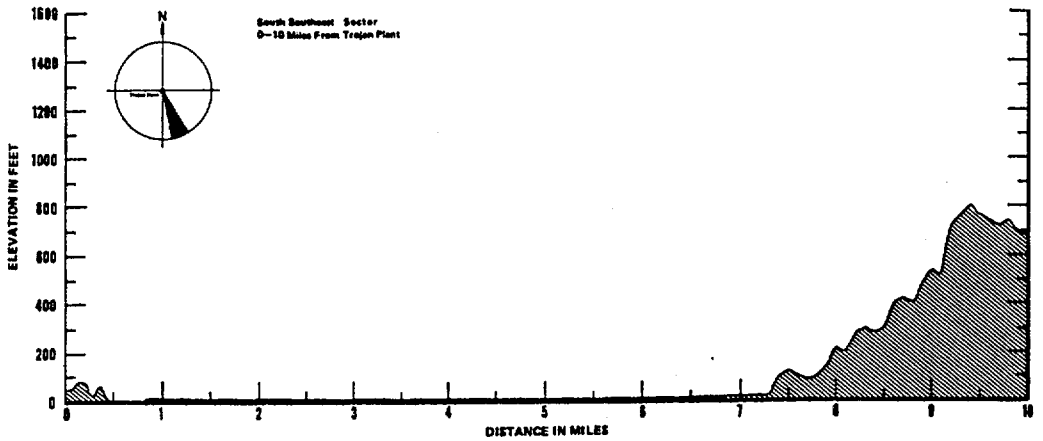
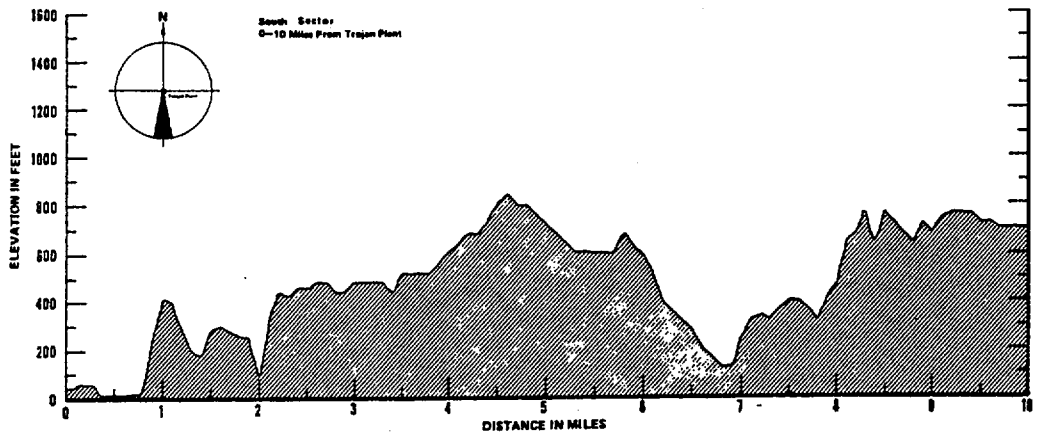
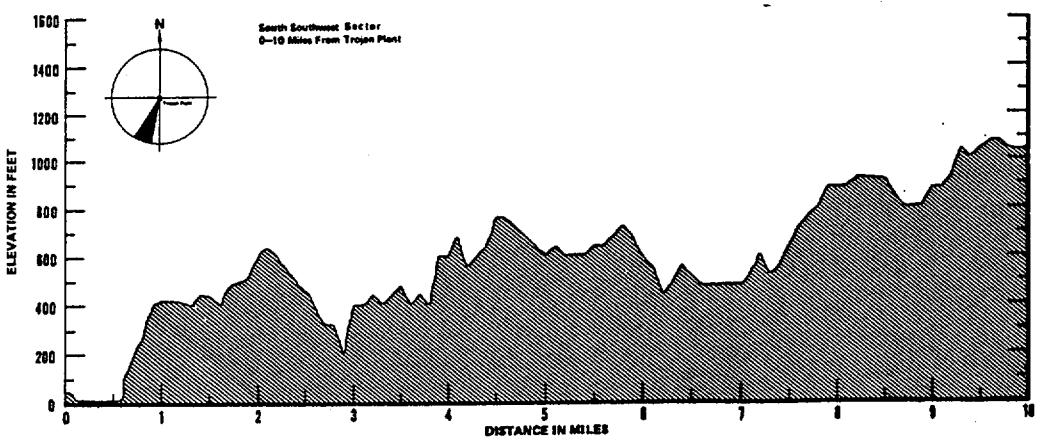
FIGURE 2.2-2
LONGVIEW AREA
INDUSTRIAL ACTIVITY







TROJAN ISFSI
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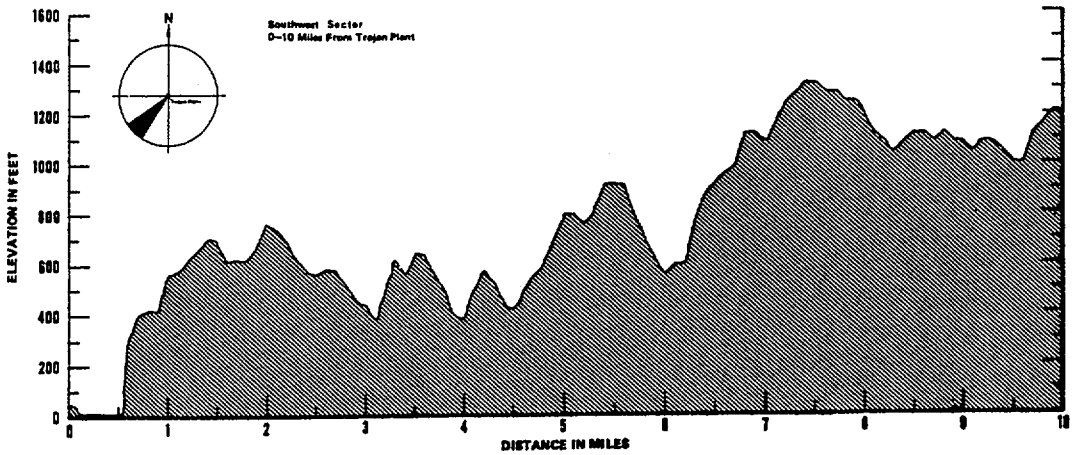
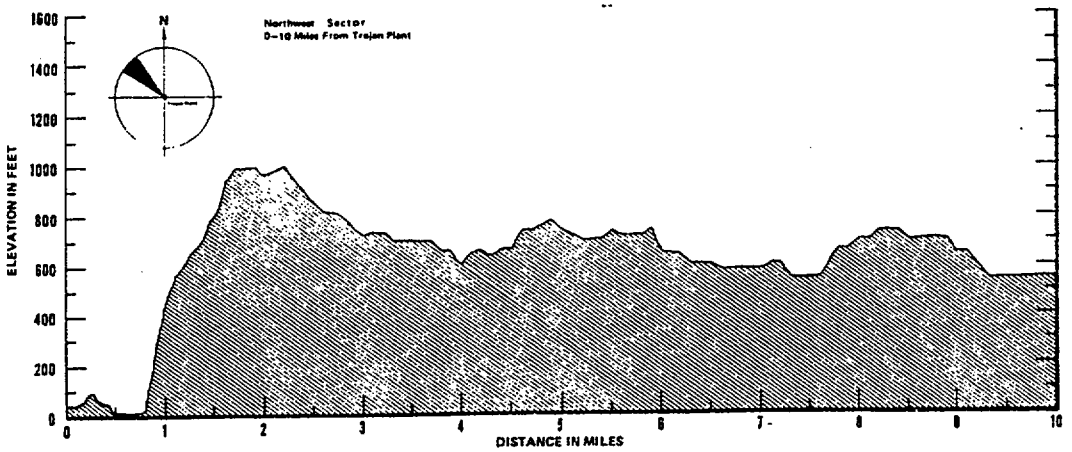
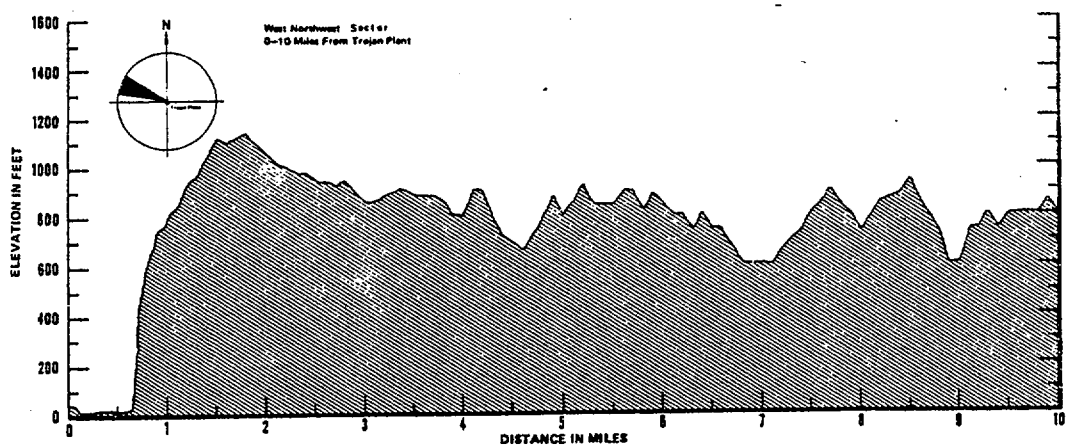
FIGURE 2.3-1
TOPOGRAPHICAL CROSS SECTIONS
(MAXIMUM ELEVATIONS)
SSE, S, SSW

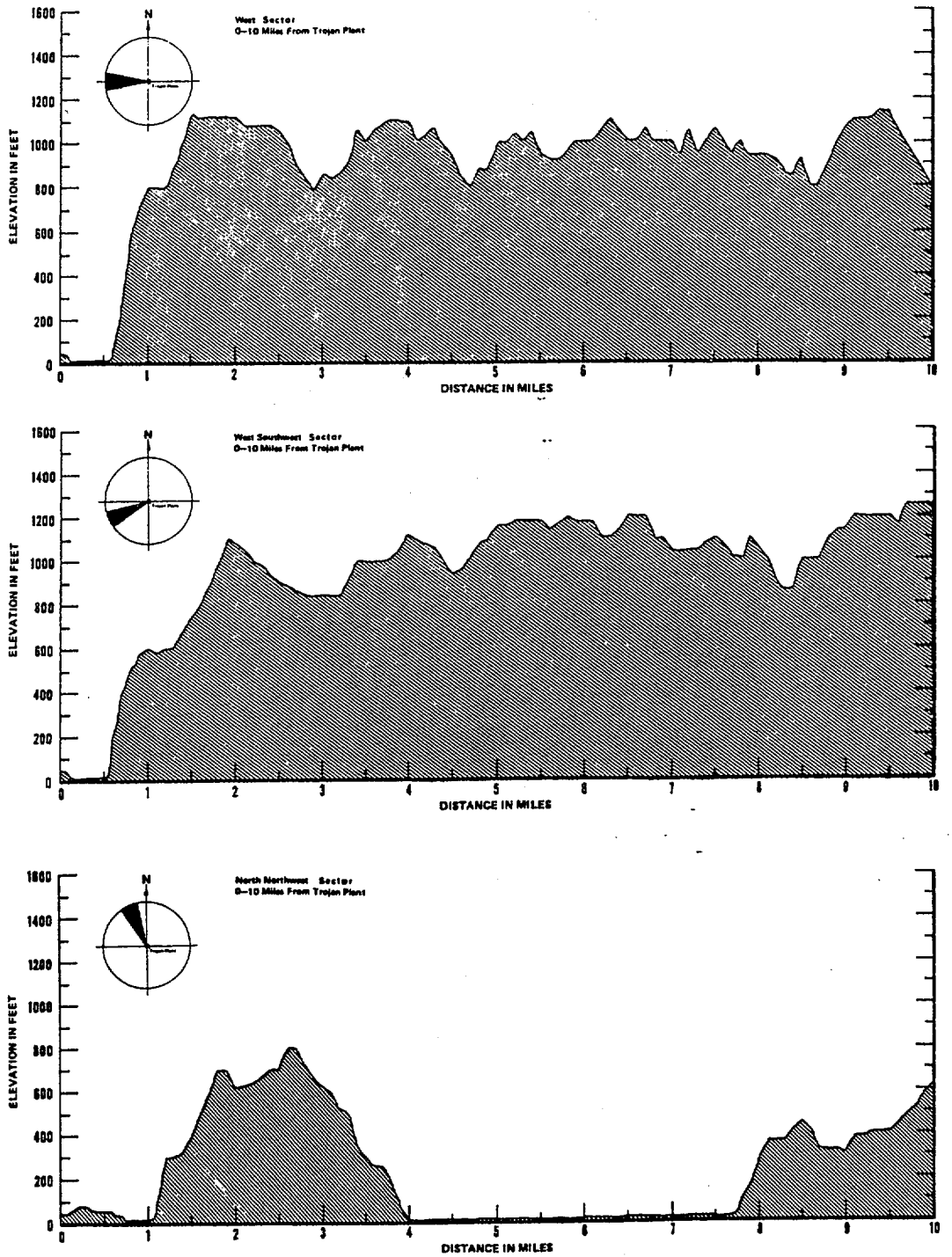






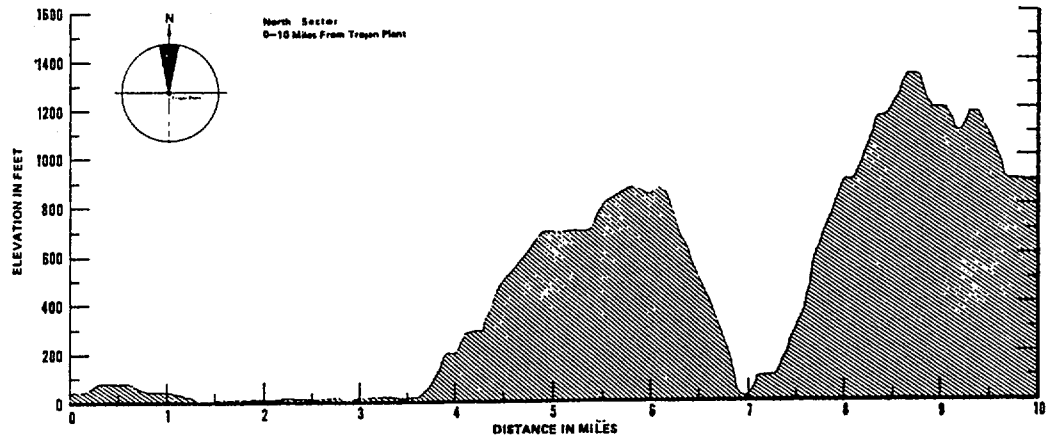
TROJAN ISFSI
SAFETY ANALYSIS REPORT

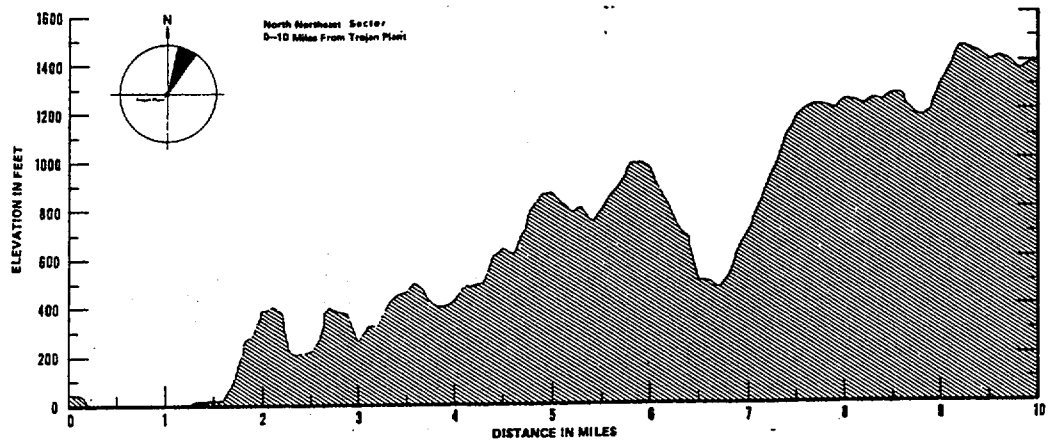
FIGURE 2.3-2
TOPOGRAPHICAL CROSS SECTIONS (MAXIMUM ELEVATIONS) SW, NW, WNW



TROJAN ISFSI
SAFETY ANALYSIS REPORT

FIGURE 2.3-3
TOPOGRAPHICAL CROSS SECTIONS (MAXIMUM ELEVATIONS) W, WSW, NNW





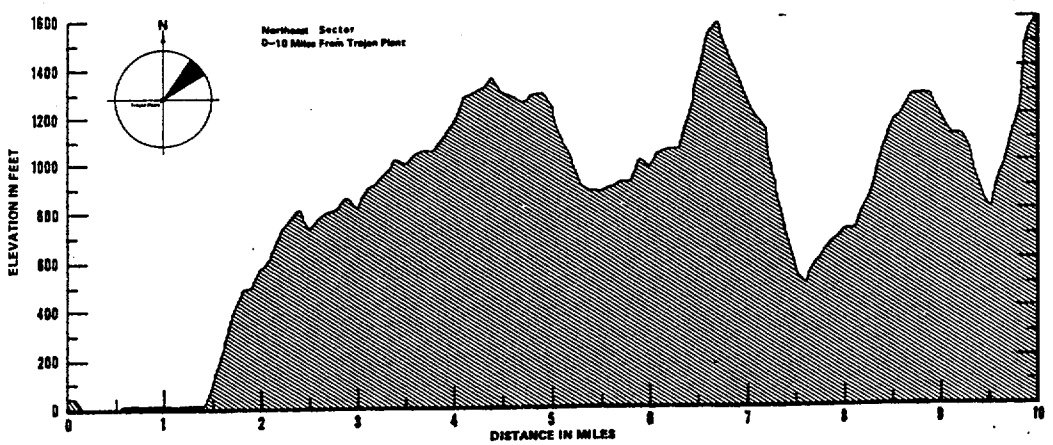


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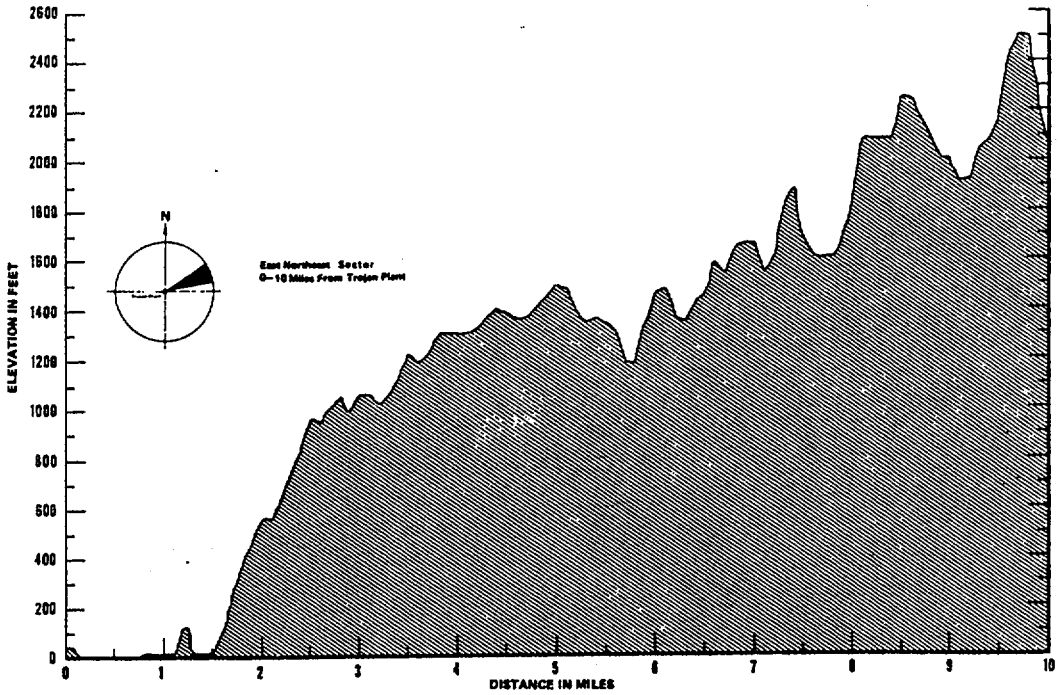
FIGURE 2.3-4
TOPOGRAPHICAL CROSS SECTIONS (MAXIMUM ELEVATIONS) N, NNE, NE



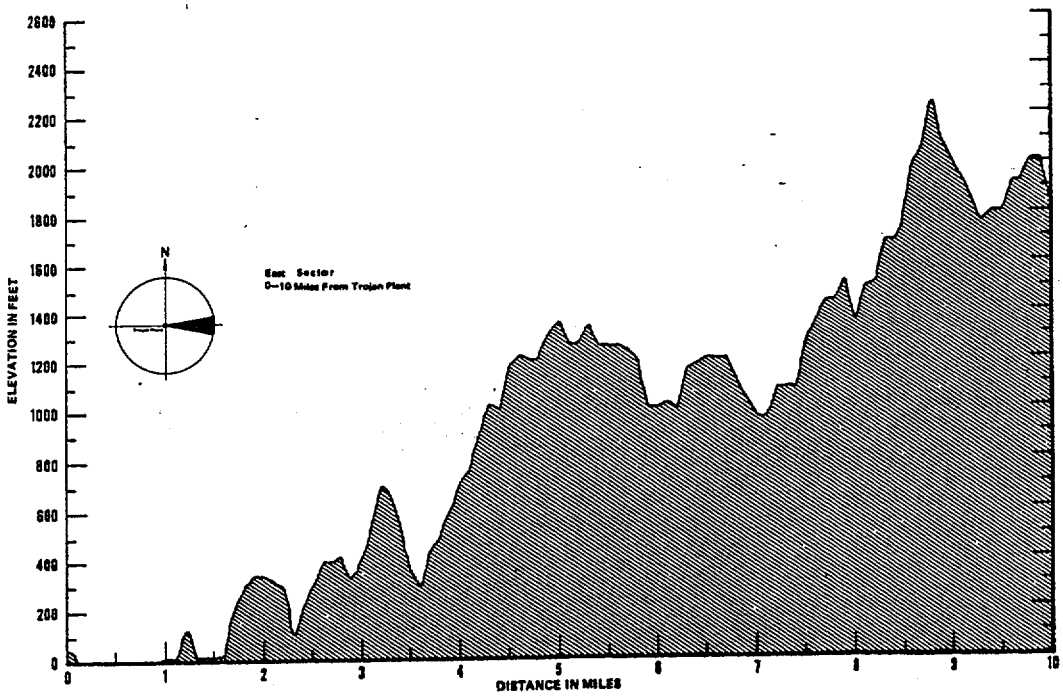


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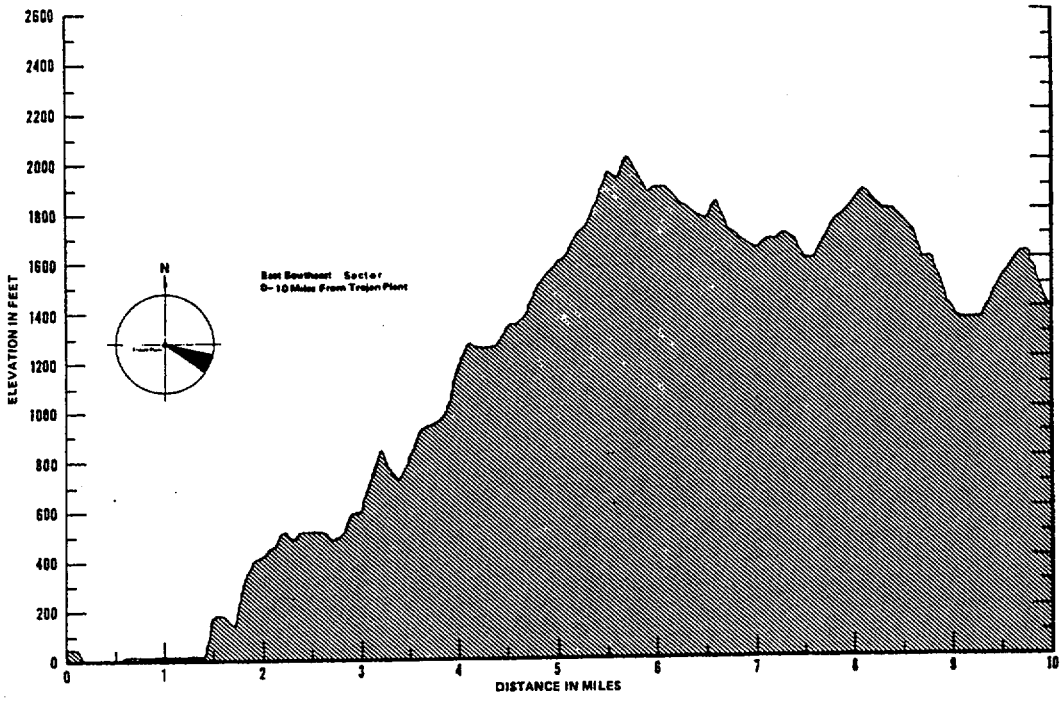
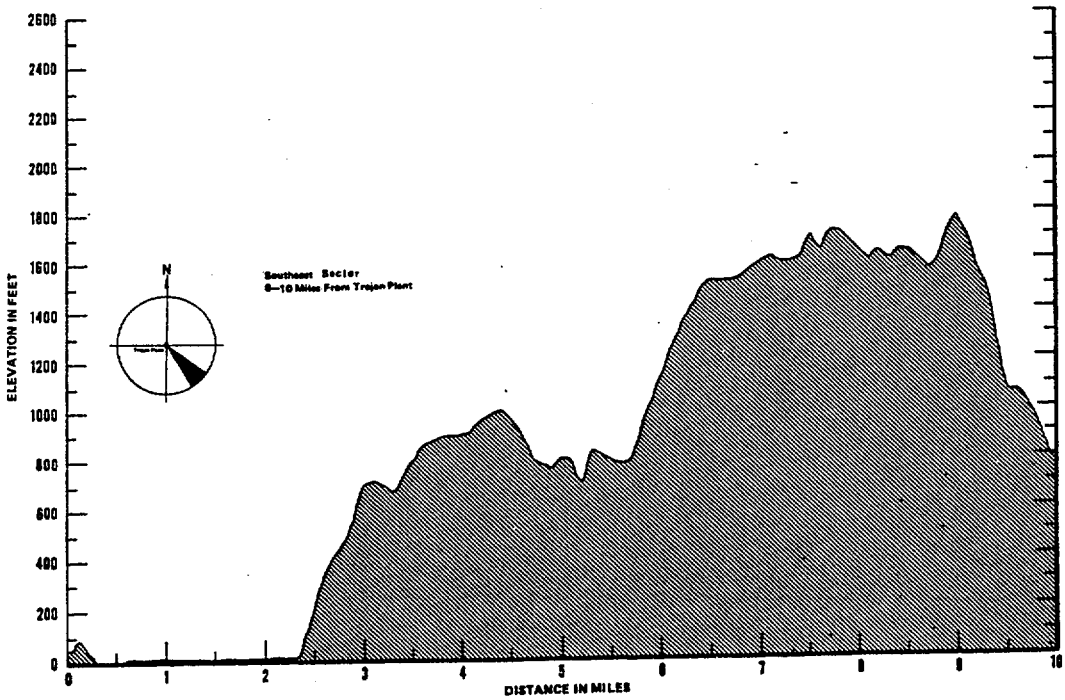
FIGURE 2.3-5
TOPOGRAPHICAL CROSS SECTIONS (MAXIMUM ELEVATIONS) ENE, E





TROJAN ISFSI
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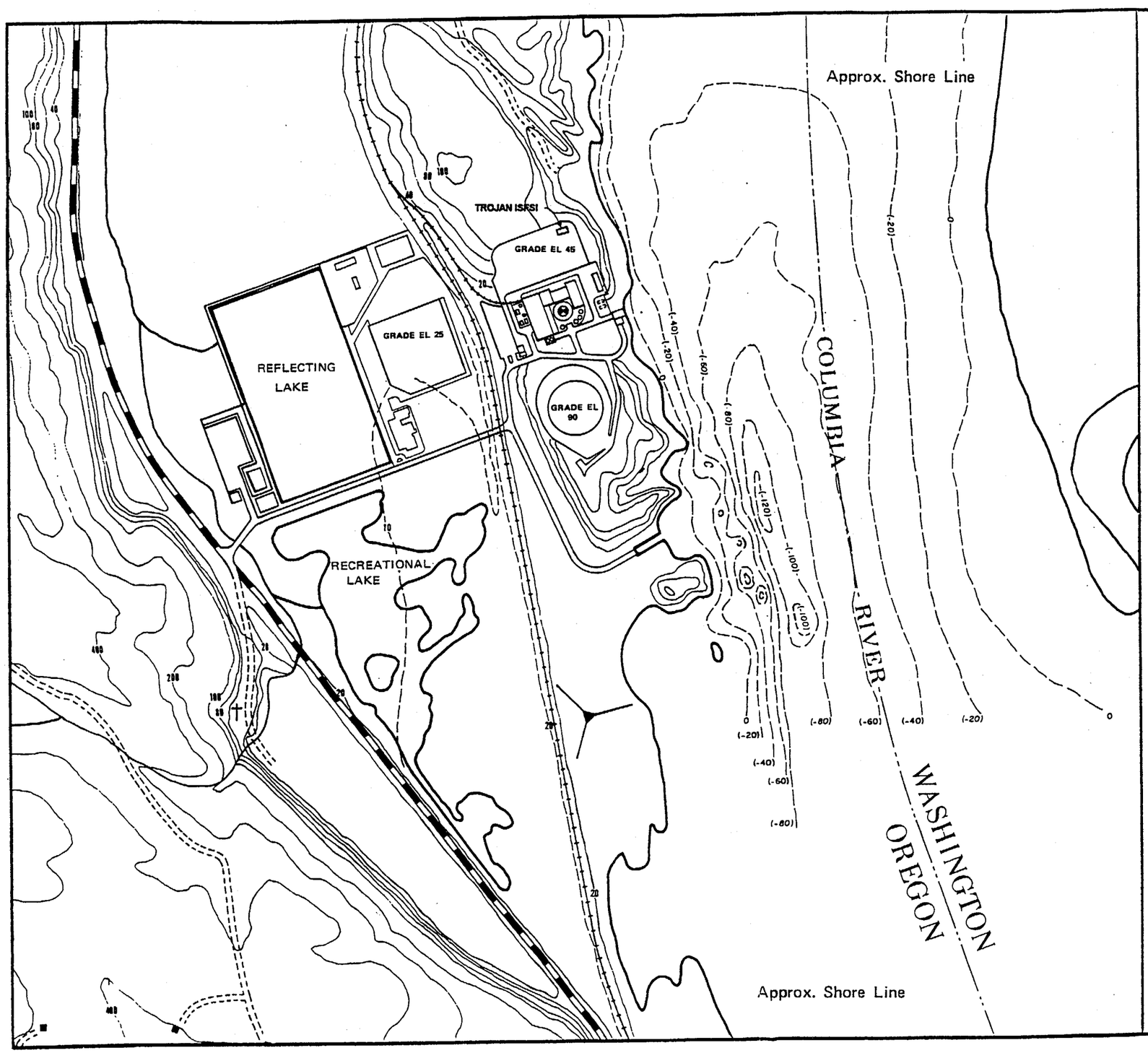
FIGURE 2.3-6
TOPOGRAPHICAL CROSS SECTIONS (MAXIMUM ELEVATIONS) ESE, SE



CONTOUR INTERVAL 100 FEET EXCEPT AS NOTED
CONTOUR LINES DASHED WHERE APPROXIMATE

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FIGURE 2.4-1
ISFSI TOPOGRAPHY

Security-Related Information Figure Withheld Under 10 CFR 2.390.

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FIGURE 2.4-2
ISFSI SITE DRAINAGE

Chapter 3

3.0 PRINCIPAL DESIGN CRITERIA

The following sections provide a discussion of the principal design criteria for the ISFSI. The design criteria have been derived from Title 10, Code of Federal Regulations, Part 72 (10 CFR 72), and applicable industry codes and standards.

3.1 PURPOSES OF INSTALLATION

The Trojan ~~Independent Spent Fuel Storage Installation (~~ISFSI~~)~~ is designed for dry, above ground storage and ultimate offsite transport of intact and ~~failed~~ *damaged* spent nuclear fuel assemblies and fuel debris. The material will be sealed in ~~PWR Baskets~~*MPCs* and stored within ventilated Concrete Casks arranged on a reinforced concrete Storage Pad. The stand alone ISFSI, when operational, allows deactivation of the Trojan Nuclear Plant Spent Fuel Pool and decommissioning of the reactor site.

3.1.1 MATERIALS TO BE STORED

The physical, thermal, and radiological characteristics of the material to be stored in the ISFSI are provided below.

3.1.1.1 Intact Fuel Assemblies

A total of 780 intact fuel assemblies will be loaded. Of these, 732 are Westinghouse 17x17 fuel assemblies and 48 are B&W Fuel Company 17x17 fuel assemblies. Fuel assemblies may contain inserts, which consist of rod cluster control assemblies (RCCA*s*) or burnable poison rod assemblies (BPRA*s*), thimble plugs, and sources. Not all fuel assemblies contain inserts.

Physical design parameters of the fuel assemblies are shown in Table 3.1-1. Limiting radiological and thermal characteristics are shown in Tables 3.1-2 and 3.1-3, *respectively*.

For 17x17 fuel assemblies, 264 grid locations contain fuel pins, 24 grid locations contain thimble guides to allow for fuel assembly inserts, and the center grid location contains an instrument sheath. RCCAs consist of 24 absorber rods which can be inserted into the thimble guides. A total of 61 RCCAs will be stored in the ISFSI. BPRAs are similar to RCCAs but consist of fewer absorber rods (9 to 20). Thimble plugs were used to "plug" thimble guides which did not contain absorber rods or sources during reactor operation. Sources are similar in shape to absorber rods but a portion of the length contains a secondary neutron source. The primary design concern associated with these components is weight. Table 3.1-4 summarizes the physical characteristics of the inserts.

The main physical parameters of concern are the fuel assembly dimensions, weight and envelope (cross-sectional dimension). These parameters establish the mechanical and structural design

aspects of the Concrete Cask and ~~PWR Basket~~*MPC*. The thermal and radiological characteristics establish the thermal and shielding aspects of the design.

3.1.1.2 ~~Failed~~ *Damaged* and Partial Fuel Assemblies

Ten (10) partial fuel assemblies and one (1) fuel rod storage container, which contain intact, suspect, or ~~failed~~ *damaged* fuel rods, will require storage.

3.1.1.3 Fuel Debris

Fuel debris consists of loose fuel pellets, fuel pellet fragments, ~~and~~ fuel assembly *metal* fragments (portions of fuel rods, portions of grid assemblies, *bottom nozzles,* etc.), *and those fuel assemblies classified as fuel debris.* ~~A criticality analysis was performed and determined that the quantity of fuel debris contained as fissile material will not exceed 7.5 kg per PWR Basket. This 7.5 kg of fissile material per PWR Basket is also less than the 10 kg limit imposed by the license conditions for the TranStor™ Shipping Cask (Reference 1). An additional limit for fuel debris of no more than 20 curies of plutonium is imposed to meet the offsite transportation requirements of 10 CFR 71.63.~~

~~3.1.1.4 GTCC Waste~~

~~This section has been deleted.~~

3.1.2 GENERAL OPERATING FUNCTIONS

The Trojan ISFSI design accounts for a maximum of 36 Concrete Casks. *A total of 34 Concrete Casks are required to store Trojan spent fuel and fuel-related hardware at the ISFSI.*

The Concrete Casks and associated ~~PWR Baskets~~*MPCs* are oriented vertically and arranged above ground on a reinforced concrete Storage Pad.

~~PWR Basket~~*MPC* loading operations and preparations for storage are discussed in Section 5.1.1. After preparations for storage are complete, the ~~PWR Basket~~*MPC* is transferred into a Concrete Cask within the Fuel Building. The loaded Concrete Cask is then transferred to the ISFSI storage location on an air pad system.

Due to the passive design of the ISFSI, operations primarily consist of inspecting the Concrete Cask air vents for blockage and monitoring outlet vent air temperature *on scheduled intervals.* Handling operations are anticipated to be limited to transferring the ~~PWR Baskets~~*MPCs* to a ~~Shipping~~ *HI-STAR 100 Transport* Cask for off-site transport for disposal or storage. ~~In the unlikely event of a PWR Basket leak, the PWR Basket may be repaired or transferred to a Basket Overpack for continued storage inside a Concrete Cask.~~

3.2 STRUCTURAL AND MECHANICAL SAFETY CRITERIA

The storage system is designed to be stored outdoors without additional weather protection. The system is designed to withstand the daily and seasonal temperature and environmental fluctuations as well as tornado, flood, seismic and handling loads. Various loads are considered per the recommendations of ANSI 57.9 (1984). Trojan ISFSI site specific environmental and geological features are addressed in Chapter 2.

Average summer and winter air temperatures for the region are 65°F and 40°F, respectively. Short term extreme temperatures for the site are not anticipated to be less than -20°F or greater than 107°F based on historical data. Steady state temperature limits for the site were assumed to be bounded by ambient air temperatures between -40°F with no solar loads and 100°F with maximum solar loads. Since short term operation with ambient air temperature in excess of 100°F is credible, an analysis was also performed assuming 125°F steady state air temperature with maximum solar loads for comparison to ISFSI short term operating limits. Thermal evaluation of ISFSI performance is provided in Section 4.2.6.

3.2.1 TORNADO AND WIND LOADINGS

The ISFSI storage structures are designed to withstand loads associated with the most severe meteorological conditions, including extreme wind and tornado, which are postulated to occur at the storage site. Tornado design parameters used to evaluate the suitability of the Concrete Cask include high winds, wind generated pressure differentials and tornado generated missiles. NUREG 0800 (1987), Regulatory Guide (R.G.) 1.76 (1974), ANSI 57.9 (1984), ANSI A58.1 (1982), and National Defense Research Committee (NDRC) methodologies were used to guide the tornado analyses.

As noted in Section 3.2.5.6, the tornado wind forces on the Transfer Station are bounded by the seismic design acceleration load for the Transfer Station.

3.2.1.1 Applicable Design Parameters

The design basis tornado characteristics (consistent with R.G. 1.76 -1974) are presented in Table 3.2-1. The design values shown are for generic storage system design and bound the postulated wind and tornado conditions for the Trojan ISFSI site. The generic values are for Region I, as defined in R.G. 1.76, and are more conservative than those defined for Region III, in which the Trojan site is located.

3.2.1.2 Determination of Forces on Structures

Wind and tornado loadings are applied to the Concrete Cask in storage only. The Concrete Cask is designed to withstand the effects of wind loading, wind generated pressure differentials, and

tornado generated missiles without toppling, or significant impact damage. The methods used to convert wind and tornado loadings into forces on the Concrete Cask are based on NUREG-0800 (1987), Section 3.3.1 (wind loadings), Section 3.3.2 (tornado loadings and combined loadings), and Section 3.5.3 (barrier design procedures).

3.2.1.3 Ability of Structures to Perform Despite Failure of Structures Not Designed for Tornado Loads

Each Concrete Cask is designed to operate independently without reliance on support structures. The Concrete Cask is designed to withstand tornado loadings and maintain its required safety function.

During storage, each ~~PWR Basket~~*MPC* ~~or Basket Overpack~~ is enclosed and protected by the Concrete Cask from tornado missiles. In addition, the ~~PWR Baskets~~*MPCs* can withstand the atmospheric pressure drop (-3 psid) associated with tornadoes.

3.2.1.4 Tornado Missiles

The Concrete Cask is designed to withstand the effects of impacts associated with postulated tornado generated missiles as identified in NUREG-0800 (1987), Section 3.5.1.4.III.4. Spectrum I missiles are used and assumed to impact in a manner that produces the maximum damage to the Concrete Cask. The design basis tornado generated missiles for the Trojan ISFSI are shown in Table 3.2-2. The effect of postulated impacts on the Concrete Cask is discussed in Section 8.2.4.2.2.

As shown in Section 8.2.4.2.8, the annual probability of a tornado-driven missile hit on the Transfer Station with ~~Concrete~~ *Transfer* Cask is on the order of 4.3×10^{-8} to 7.1×10^{-9} for tornado intensities F'1 to F'4. These probabilities are much less than the threshold of significant risk, considered to be in the range of 10^{-6} to 10^{-7} per year. These probabilities are for missile strikes, and not necessarily for unacceptable damage. Considering the short time it takes to complete the transfer of the ~~PWR Baskets~~*MPCs*, the risk associated with tornado-driven missiles is not considered to be significant.

3.2.2 WATER LEVEL (FLOOD) DESIGN

The floods postulated for the Trojan ISFSI do not exceed the site elevation~~,~~; therefore, flooding of the ISFSI is not a credible event. Trojan ISFSI site hydrology is discussed in Section 2.4.

3.2.2.1 Flood Elevations

The nominal ISFSI elevation is 45 ft. MSL. Section 2.4 discusses the potential flood levels for the ISFSI location. A summary of potential floods is as follows:

Standard Project Flood (SPF, 1000 yr.):	21 ft. MSL
Probable Maximum Flood (PMF):	39.2 ft. MSL
Seismically Induced Dam Failure:	42.75 ft. MSL

3.2.2.2 Flood Protection

There are no ISFSI storage system components classified as important to safety located at an elevation below the most limiting credible flood level (42.75 ft. MSL).

3.2.3 SEISMIC DESIGN

3.2.3.1 Input Criteria

ISFSI design criteria were evaluated using the Trojan Nuclear Plant Seismic Margin Earthquake (SME) as the design basis seismic event (Reference 2). The SME represents the maximum potential earthquake ground motions for the ISFSI site.

3.2.3.1.1 Design Response Spectra

The Trojan Seismic Margin Earthquake has the following peak conditions:

Horizontal ground acceleration:	0.38g
Vertical ground acceleration:	0.25g

The design earthquake response spectra are in accordance with Regulatory Guide 1.60 (Rev. 1, December 1973), with a maximum horizontal ground acceleration equal to the Trojan SME value. Damping values of Regulatory Guide 1.61 (Rev. 1, October 1973) were used.

A minimum factor of safety for overturning of 1.1 was used based on guidance provided in NUREG-0800 (1987), Section 3.8.5. This document provides criteria used in determining the acceptability in the design of Seismic Category I foundations at nuclear facilities.

3.2.3.1.2 Design Response Spectra Derivation

The response spectral shapes of Regulatory Guide 1.60 (Rev. 1, December 1973) are used for the design of structures important to safety. Earthquake time functions or other data, therefore, are not required for derivation in accordance with Regulatory Guide 3.48 (Rev. 1), Section 3.2.3.1.2.

3.2.3.1.3 Design Time History

The earthquake is defined by the response spectra, therefore, time histories are not used.

3.2.3.1.4 Use of Equivalent Static Loads

The equivalent static loading is used for the storage system evaluation since the Concrete Cask is a very rigid body with natural frequencies in excess of the zero period acceleration cut-off. No dynamic amplification by the Concrete Cask is expected. Refer to Section 3.2.3.2.1 for a discussion of seismic analysis methods.

3.2.3.1.5 Critical Damping Values

Damping values are developed in accordance with Regulatory Guide 1.61 (Rev 1, October 1973) for a Safe Shutdown Earthquake (SSE).

3.2.3.1.6 Bases for Site-Dependent Analysis

The SME was developed as part of the U.S. Nuclear Regulatory Commission's Individual Plant Examination for External Events (IPEEE) program. The results of this study focused on determining the maximum potential earthquake ground motions that could affect the site. The SME for the Trojan Site conservatively bounds the Design Basis Earthquake (DBE). In addition, the SME is a specified design criterion of Oregon Administrative Rule (OAR) 345-26-0390.

3.2.3.1.7 Soil-Supported Structures

The reinforced concrete Storage Pad on which the Concrete Casks rest and the adjacent concrete Service Pad and Transfer Station are the only at-grade ISFSI structures. The concrete pads are located on approximately 24" *inches* of engineered fill founded on competent rock. The foundation beneath the Transfer Station extends to competent rock.

3.2.3.1.8 Soil-Structure Interaction

The foundation rock contains joints and fractures, as do essentially all rocks exposed to the earth's surface. However, none of these features should be expected to affect the stability of the foundation rock during vibratory motion. The foundation rock is confined by natural, in situ materials, and foundation loads are small in comparison to the foundation rock's ultimate bearing capacity. There will be no loss of strength or stability of the foundation rock during vibratory motions. Since the reinforced concrete pads are located on engineered fill founded on competent rock, soil-structure interaction is negligible.

3.2.3.2 Seismic-System Analyses

3.2.3.2.1 Seismic Analysis Methods

The Concrete Cask is a very stiff structure. Its lowest natural frequencies exceed the zero period acceleration threshold. No dynamic amplification of the ground motion is expected from the Concrete Cask. For the purpose of calculating seismic loads, the Concrete Cask is treated as a rigid body, and equivalent static analysis methods were used to calculate loads, stresses, and overturning moments. Although free-standing, it is conservatively analyzed for stresses as a cantilever, fixed at the base.

The storage system is evaluated statically for overturning by conservatively applying static loads to the Concrete Cask equivalent to the peak horizontal component acting simultaneously with 40~~%~~ *percent* of the peak vertical component applied upward.

3.2.3.2.2 Natural Frequencies and Response Loads

The fundamental natural frequency of vibration for the Concrete Cask was determined as shown below:

$$f_n = [(K_n)/2\pi]\ [(E)(I)(g)/(w)(L^4)]^{0.5}$$

where:

f_n = Frequency of the n-th mode

K_n = 3.52 for first mode of vibration

E = Modulus of Elasticity

I = Moment of Inertia

g = Gravitational acceleration

L = Height of Concrete Cask

w = Uniform weight per unit length of cantilever

3.2.3.2.3 Procedure Used to Lump Masses

The storage system components are very rigid and no dynamic modal analysis needs to be performed as explained above. Thus, no mass lumping procedure was used.

3.2.3.2.4 Rocking and Translational Response Summary

The Concrete Cask restoring moment is always higher than tipover moments from seismic loads. Therefore, no Concrete Cask tipover can occur.

A minor amount of sliding may occur under SME ground motions, but no Concrete Cask impacts will result.

3.2.3.2.5 Methods to Determine Overturning Moments

The storage system has been evaluated conservatively by applying the static loads to the Concrete Cask equivalent to the horizontal component acting simultaneously with the vertical component (40~~%~~ *percent* peak) applied upward. For the SME ground accelerations listed in Section 3.2.3.1.1 the margin of safety against overturning was determined and shows that the Concrete Cask will not overturn during the SME.

3.2.3.2.6 Analysis Procedure for Damping

No damping was assumed in the storage system.

3.2.3.2.7 Seismic Analysis of Overhead Cranes

There are no overhead cranes included in the ISFSI design. The use of mobile cranes is discussed in Section 4.7.3.6.

3.2.3.2.8 Seismic Analysis of Specific Safety Features

The ISFSI must be designed to meet the requirements of 10 CFR 72.122(b)(2) and OAR 345-26-0390. The ISFSI shall be capable of withstanding a SME without overturning the Concrete Cask or compromising the ~~PWR Basket~~*MPC* confinement boundary. A SME will not result in uncontrolled release of radioactive material or increased radiation exposure to workers or members of the general public.

3.2.4 SNOW AND ICE

The criterion for determining design snow loads is based on ANSI A58.1 (1982), Section 7.0. Flat roof snow loads apply and are calculated from the following formula:

$$p_f = 0.7C_eC_tIp_g$$

where:

p_f = Flat roof snow load (psf)

C_e = Exposure factor = 0.8

C_t = Thermal factor = 1.0

I = Importance factor = 1.2

p_g = ground snow load, pounds per square foot = 100 psf

The numerical values of C_e, C_t, and I, are obtained from Tables 18, 19, and 20 respectively, of ANSI A58.1 (1982). A conservative value of 100 pounds per square foot was assumed for snow load (p_g).

The exposure factor accounts for wind effects. The Trojan site is assumed to have siting category A which is defined to be a "windy area with roof exposed on all sides with no shelter afforded by terrain, higher structures, or trees." The thermal factor accounts for the thermal condition of the structure. Due to the presence of substantial heat load in the storage system, it is classified as a heated structure. The storage system Concrete Cask is conservatively classified as Category III which is the highest category in the ANSI standard. Ground snow loads for the contiguous United States are given in Figures 5, 6, and 7 of ANSI A58.1 (1982).

Based on the above, the design criterion for snow and ice loads is:

Flat Roof Snow Load p_f = (0.7) (0.8) (1.0) (1.2) (100)

= 67.2 psf

3.2.5 COMBINED LOAD CRITERIA

The storage system components are subjected to normal, off-normal, and accident loads. These loads are defined as follows:

Normal Loads	-	Dead weight, pressure, handling, thermal, winds, snow, rain
Off-Normal Loads	-	Severe environmental conditions, interference during ~~PWR Basket~~*MPC* lowering from Transfer Cask to Concrete Cask, off-normal handling
Accident Loads	-	Complete blockage of all air inlets~~ and outlets~~, maximum heat load, ~~PWR Basket~~*MPC* drop into ~~Shipping~~ *Transport*

Cask or Concrete Cask, tornado (wind and missiles), flood, seismic, fuel pin rupture (for the ~~PWR Basket~~*MPC*)

Normal loads due to pressure, temperature, and dead weight act in combination with other loads. No two unrelated accident events are postulated to occur simultaneously. However, loads due to one event, such as tornado wind and tornado missile loads, are assumed to act in direct combination.

3.2.5.1 Load Combinations and Design Strength - Concrete Cask

The load combinations specified in ANSI 57.9 (1984) and ACI-349 (1985) for concrete structures were used and are shown in Table 3.2-3. The steel liner, air ducts and bottom plate of the Concrete Cask are stay-in-place forms. They are designed to ACI--349 (1985) requirements for steel forms and reinforcement.

3.2.5.2 Load Combinations and Design Strength - ~~PWR Basket and Basket Overpack~~*MPC*

The ~~PWR Basket and Basket Overpack~~*MPC* ~~are~~*is* designed to the ~~1992~~ *1995* edition of the ASME Boiler and Pressure Vessel Code, Section III, including Addenda through ~~1994~~*1997*. The *MPC* pressure retaining components (shell and lid~~s~~) are designed to Section III, Subsection ~~NC~~*NB*. The ~~PWR Basket internals~~*MPC fuel basket is* ~~are~~ designed to Section III, Subsection NG. Buckling/instability of the internal members are evaluated according to the requirements of NUREG/CR-6322. The load combinations for normal, off-normal, and accident conditions and corresponding Service Levels are shown in Table 3.2-4. The structural design criteria are summarized in Table 3.2-5.

Service Levels A and ~~C~~*B*, as specified by the ASME Code, are used for normal and off-normal conditions, respectively. The analyses methods allowed by the ASME Code are employed. Stress intensities caused by pressure, temperature and mechanical loads are combined before comparison to ASME code allowables.

Service Level D, as specified by the ASME Code, is used for accident conditions. The design strength criteria for the evaluation of Service Level D events are that any damage to the ~~PWR Basket~~ *MPC* will not prevent the ~~PWR Basket~~ *MPC* from performing the following safety functions: heat dissipation, criticality prevention, confinement, and shielding. Limited plastic deformation is allowed. Stresses caused by normal condition loads are combined with the stresses caused by accident or off-normal loads. These stresses are then compared to the stress limits defined in Appendix F of the ASME Code.

The ~~PWR Basket~~*MPC* design weight is based on 24 intact fuel assemblies, each containing a RCCA ~~(149 lbs.)~~. *Physical parameters of RCCAs and other fuel assembly inserts are summarized in Table 3.1-4.*

3.2.5.3 Load Combinations and Design Strength - Transfer Cask

The *structural steel components of the* Transfer Cask, *with the exception of the lifting trunnions, are* ~~is a special lifting device~~ designed and fabricated *in accordance with the applicable requirements of Section III, Subsection NF, of the ASME Code, 1995 Edition with Addenda through 1997. The lifting trunnions and associated attachments are designed in accordance with the guidance*~~to the requirements~~ of ANSI N14.6 ~~(1993)~~ and NUREG--0612 ~~(1980)~~ *for non-redundant special lifting devices.* The *design* criteria for ~~its load-bearing components~~*the Transfer Cask* are summarized in Table 3.6-2. ~~and are as follows:~~

> ~~Maximum principal stress during the lift (with 10% dynamic load factor) will be less than $S_y/3$ or $S_u/5$.~~
>
> ~~Load bearing members of the Transfer Cask shall be subject to drop weight test (ASTM E208) or Charpy impact test (ASTM A370) per ANSI N14.6 (1993) paragraph 4.2.6.~~

3.2.5.4 Load Combinations and Design Strength - Failed Fuel Can

The Failed Fuel Can is designed to be placed into one of the four corner storage locations of the ~~PWR Basket~~*MPC*. The Failed Fuel Can is designed to allow for water draining and ~~vacuum drying~~*moisture removal* during ~~PWR Basket~~*MPC* closure *operations*. Once placed into its storage location, the Failed Fuel Can is not subjected to external loadings applicable to ASME Service Level A (normal). Specific structural design criteria and load combinations are not applicable.

3.2.5.5 Load Combinations and Design Strengths - Fuel Debris Process Can Capsule

The fuel debris Process Can Capsule material and welds are selected based on ASME Section III, Division I, Subsection NG (1992). The fuel debris Process Can Capsule is structurally analyzed for external pressure, internal pressure, dead weight, thermal stresses, and drops. The stresses calculated by classical equations are less than the allowable stresses provided in ASME, Section III, Division I, Subsection NG (1992) for service levels A and D.

3.2.5.6 Load Combinations and Design Strength - Transfer Station

The Transfer Station is designed to the load combinations presented in Table 3.2-6. Soil pressure load (H) and temperature loads (T, Ta) are zero, and the live load (L) is negligible (the loaded Transfer Cask was considered to be a dead load). Concrete Cask handling loads and wind loads are bounded by seismic loads. Transfer Station tornado loadings are addressed in Section 8.2.4.2.8.

The only loads that need to be considered are the dead load (D) and seismic load (E). As a result, only one bounding combination is considered in the Transfer Station analysis. This load combination is as follows:

$$1.6S > D + L + E$$

where 1.6S is used as the bounding (lowest) allowable, and E represents the bounding (highest) load that can exist.

The analysis and design are performed in accordance with the AISC "Manual of Steel Construction," Ninth Edition, and NUREG 0800, "Standard Review Plan," Section 3.7.2, "Seismic System Analysis," and Section 3.8.4, "Other Seismic Category I Structures," U. S. Nuclear Regulatory Commission, Revision 1, July 1981.

A three dimensional ANSYS finite element model was used for the analysis. All member and connection stresses remain within AISC allowables.

3.3 SAFETY PROTECTION SYSTEMS

3.3.1 GENERAL

The ISFSI is designed for safe, long-term storage of the radioactive material described in Section 3.1.1. The ISFSI withstands normal, off-normal and credible accident conditions without release of radioactive material to workers or members of the general public.

The primary functions of the ~~PWR Basket~~*MPC* (including ~~internals~~*fuel basket*) are:

1. To provide ~~containment and~~ confinement for the spent nuclear fuel during normal storage, off-normal events and postulated accidents,

2. To provide criticality control in the absence of a moderator under design conditions and postulated accidents,

3. To provide adequate heat transfer so that the fuel clad temperature does not exceed allowables under design conditions and postulated accident conditions, and

4. To provide adequate shielding (together with a Concrete Cask or Transfer Cask) to meet 10 CFR 72 requirements.

~~The primary functions of the Basket Overpack are:~~

~~1. To provide the same containment and confinement function as a PWR Basket in the highly unlikely case the PWR Basket develops a leak, and~~

~~2. To allow adequate heat transfer so that the fuel clad temperature does not exceed allowables under design conditions.~~

The primary functions of the Concrete Cask are:

1. To protect the ~~PWR Basket~~*MPC* from weather and postulated environmental events such as earthquakes and tornado missiles,

2. To provide adequate heat transfer for the ~~PWR Basket~~*MPC*, and

3. To provide adequate shielding (together with a*n* ~~PWR Basket~~*MPC*) to meet 10 CFR 72 requirements.

The primary functions of the Transfer Cask are:

1. To ~~serve as a special lifting device meeting the requirements of NUREG-0612 (1980)/ANSI 14.6 (1993)~~*provide* for movement of a*n* ~~PWR Basket~~*MPC* within the Fuel Building, and

2. To provide radiation shielding to minimize exposure rates during transfer operations.

The primary function of the Transfer Station is to secure the Transfer Cask in position during ~~PWR Basket~~*MPC* transfer operations.

The primary function of the Failed Fuel Can is to provide a containment boundary for ~~failed~~ *damaged* fuel such that the ~~failed~~ *damaged* fuel will be constrained within its ~~PWR Basket~~*MPC* storage location. The primary function of the fuel debris Process Can Capsule is to provide a ~~containment~~ *confinement* boundary for fuel debris processed during the ~~spent f~~*F*uel ~~d~~*D*ebris *Processing* ~~p~~*P*roject. *The Process Can Capsules and those fuel assemblies classified as fuel debris are also placed in the containment boundary provided by the Failed Fuel Can.* Constraining ~~failed~~ *damaged* fuel and fuel debris to fixed storage locations is required to maintain the assumptions in the criticality analysis and heat transfer modeling.

As discussed in the following sub-sections, the ISFSI design incorporates features addressing each of the above design considerations to assure safe operation during fuel loading, storage system handling, and storage.

3.3.2 PROTECTION BY MULTIPLE CONFINEMENT BARRIERS AND SYSTEMS

3.3.2.1 Confinement Barriers and Systems

Oregon Administrative Rule (OAR) 345-26-0390 prohibits the storage of spent nuclear fuel or radioactive materials other than that generated or used in the operation of the Trojan Nuclear Plant. Spent nuclear fuel and fuel related material will be confined within ~~PWR Baskets~~*MPCs*.

The ~~PWR Basket~~*MPC* is designed to provide a confinement barrier for spent nuclear fuel in accordance with the general design criteria requirements of 10 CFR 72, Subpart F. The ~~PWR Basket~~ *MPC confinement boundary* is *defined as the MPC shell, bottom baseplate, lid (including the vent and drain port cover plates), the closure ring, and associated welds (Reference 1, Section 7.1). The MPC is* a stainless steel seal welded enclosure. The ~~PWR Basket structural~~*MPC* lid closure is accomplished by multi-pass welding *of the lid to the MPC shell.*

~~The smaller weld on the shield lid may be performed in a single-pass or multi-pass depending on the welding process used.~~ *The MPC lid weld maintains confinement integrity under all normal, off-normal, and accident conditions of storage. Additionally, the vent and drain port cover plates are welded to the top of the MPC lid and the closure ring is welded to the MPC lid on the inner diameter and to the MPC shell on the outer diameter. The closure ring provides a second welded boundary beyond the MPC lid-to-shell weld and the vent and drain port cover welds.* The ~~PWR Basket~~*MPC* confinement ~~barrier~~ *boundary* is designed in accordance with ASME, Section III, Subsection ~~NC~~ *NB* (~~1992~~ *1995 Edition* including Addenda through ~~1994~~*1997*).

The ~~PWR Basket internals~~*MPC fuel basket*, which ~~are~~ *is* used to constrain fuel assemblies and Failed Fuel Cans during storage, ~~are~~ *is* designed in accordance with ASME, Section III, Subsection NG (~~1992~~ *1995 Edition* including Addenda through ~~1994~~*1997*). The ~~PWR Basket~~*MPC-24E/EF* ~~internals~~ *fuel basket* provides 24 storage locations. ~~The four (4) corner~~*Fuel storage* locations *3, 6, 19, and 22* are designed slightly larger *than the rest of the fuel storage locations* to accommodate a Failed Fuel Can *(see Figure 4.2-1b)*.

The fuel debris Process Can Capsule provides a ~~containment~~ *confinement* boundary for fuel debris loaded into a Failed Fuel Can within the ~~PWR Basket~~*MPC*. It is designed using the guidance in ASME, Section III, Subsection NG (1992 including Addenda through 1994) (see Section 3.2.5.5).

The Failed Fuel Cans do not provide a confinement boundary and are considered to function as part of the ~~PWR Basket~~*MPC* internals. The Failed Fuel Can is designed in accordance with applicable portions of ASME, Section III, Subsection NG (1992 including Addenda through 1994).

The MPC is designed to be leak tight under all normal, off-normal, and accident conditions of storage. Based on this design and the very low probability that a flawed confinement boundary weld would be performed, pass NDE inspection, and subsequently leak, the leakage of an MPC is not considered a credible event.. ~~In the unlikely event of a PWR Basket confinement boundary failure, the affected PWR Basket may either be repaired or sealed within a Basket Overpack. The design criteria for the Basket Overpack are the same as those specified for the PWR Basket confinement boundary.~~

The ~~PWR Basket~~*MPC* ~~must be~~*is* designed to withstand credible drop accidents without damaging the stored fuel (i.e., the storage cells do not ~~deform such that they bind the fuel~~*exceed ASME Section III, Level D stress limits*). The ~~PWR Basket~~*MPC is* ~~must~~ also ~~be~~ designed to provide confinement in the event of a fuel clad failure.

3.3.2.2 ~~PWR Basket~~*MPC* Closure Welds

The primary function of the ~~PWR Basket~~*MPC* is to act as a confinement barrier for the radioactive material associated with the spent nuclear fuel as it is stored in the ISFSI awaiting transfer to a permanent repository. As such, once the ~~PWR Baskets~~*MPCs* are filled and sealed closed, there must be a high degree of confidence that the ~~PWR Basket~~*MPC* ~~seal~~ welds will remain strong and leak free throughout the life of the ISFSI.

This high degree of confidence is achieved by ensuring that the welds performed are free from defects. Defect free welds are achieved by following the well-established processes in the ASME Boiler and Pressure Vessel Code for design, material specification, welding procedure qualification, and NDE, for this type of service.

Industry experience has shown that weld cracking in austenitic stainless steel welds can be effectively avoided by ensuring that specific metallurgical requirements for the base metal, filler metal, and the "as-welded" joint are met. Austenitic stainless steel that meets rigid ~~ASTM~~ *ASME* material standards is being specified for the boundary material to ensure that the properties required for weldability, strength, and corrosion resistance are maintained. A welding filler metal with a controlled chemical content will also be specified in accordance with the code. Extensive weld procedure qualification testing to meet the requirements of ASME Section IX is being performed to develop welding procedure specifications for both automated and manual welding. This procedure qualification process will ensure that the "as-welded" condition of the weld joint is within the metallurgical parameters that will prevent weld cracking from being an issue. Based on this ability to control the conditions that lead to cracking, cracking in these welds is a very unlikely event.

The ASME Code specified for the closure welds requires a radiograph as the final NDE. A radiograph of these welds cannot be performed due to the inability to access the inside of the weld joint. This is compensated for by meeting the criteria set forth in ~~the~~ Interim Staff Guidance No. 4 *(Reference 3)*, ~~"Cask Closure Weld Inspections."~~*as discussed below.*

1. The ~~PWR Basket~~*MPC* is closed with redundant seal welds.

2. ~~Shield~~ *MPC* lid weld strength and *MPC* leak tightness ~~is~~ *are* verified by performing:

   - a visual exam *of the MPC lid-to-shell, vent port cover plate, drain port cover plate, and closure ring welds* per Section V, Article 9, and acceptance *criteria* per *Section III,* ~~NG-5361~~ *NF-5360* ~~and NG-5362(b)(1)~~,

   - a hydro-test per Section III, ~~NG-6221~~ *NB-6000 (with Table 4.2-1a exception),*

- a helium leak test *of the MPC lid, vent port cover plate, and drain port cover plate welds* using the guidance of ANSI N14.5, ~~and~~

- *a multi-layer dye penetrant examination of the MPC lid-to-shell weld per Section V, Article 6, with acceptance criteria per Section III, NB-5350. The multi-layer dye penetrant examination includes the root and final weld passes and each approximately 3/8-inch layer of weld depth, and*

- a dye penetrant exam*ination of the vent port cover plate, drain port cover plate, and closure ring welds* per Section V, Article 6, ~~and~~ *with acceptance criteria per Section III,* ~~NC-5300~~*NB-5350.*

~~Structural lid seal weld strength and leak tightness is verified by performing:~~

~~a visual exam per Section V, Article 9, and acceptance per NG-5361 and NG-5362(b)(1),~~
~~a multi-layer dye penetrant exam per Section V, Article 6, and NC-5300,~~
~~a design stress-reduction factor of 0.8 has been applied to the weld design.~~

As an alternative to volumetric examination, an analysis has been performed to determine the critical flaw size that could lead to weld failure under the most extreme loads anticipated in the ~~TranStor™ structural~~*MPC* lid weld. The critical flaw size for the ~~TranStor™ structural~~*MPC* lid weld was calculated to be *well in excess of* ~~0.36"~~*0.375 inches* ~~using data for the 30' PWR Basket drop~~*based on fracture mechanics evaluations performed by Holtec International.* ~~To be conservative, the 1.39 safety factor used in ASME Section XI for accident condition loads was applied to the critical flaw size which gives a maximum allowable flaw size of 0.26".~~ Using this size as a target, a multi-layer dye penetrant (PT) exam will be performed on the ~~¾" thick structural lid~~*root and final* weld *passes and each* ~~at~~ approximately ~~¼"~~*3/8-inch layer of weld depth* ~~intervals (after the first ¼" for the root PT, the second ¼" for the mid-layer PT, and after the third ¼" for the cap PT)~~. By performing a PT at this interval within the *MPC lid* weld, any flaw greater than the ~~0.36"~~*0.375 inches* ~~critical flaw size~~ would break the weld surface and be readily detected by the PT.

~~The shield lid weld, which is only ¼" thick, will be dye penetrant examined after welding (i.e., no root PT planned) based on the same criteria. The shield lid weld will be hydro-tested, helium leak checked, and dye penetrant examined in its final condition to ensure that it is leak tight.~~

A permanent record of the PT examinations, including all relevant indications, shall be made using video, photographic, or other means providing a retrievable record of weld integrity. The records shall be taken during the final interpretation period described in ASME Section V, Article 6.

The ~~TranStor™ Basket~~*MPC* has all-welded construction. The ~~shield~~ *MPC* lid, *vent port cover plate, and drain port cover plate are* ~~is~~ helium leak tested after welding to demonstrate that the*se* inner closure weld*s* meet~~s~~ leakage limits. The ~~structural lid~~*closure ring* is then welded in place. The leak tight ~~shield~~ *MPC* lid, *vent port cover plate, and drain port cover plate* prevent~~s~~ helium from reaching the ~~structural lid~~*closure ring*, making a helium leak test of ~~this lid~~*the closure ring* unnecessary.

Ultrasonic Testing (UT) is listed in the Code as an acceptable alternative to radiography. UT of the ~~structural~~ *MPC* lid weld is not being performed because, at this time, a proven method for performing a UT that can consistently detect weld flaws or cracks in these austenitic stainless steel welds has not been identified. Meaningful UT of stainless steel weld joints, within the confines of Code requirements, is recognized to be difficult to achieve due to the inherent nature of stainless steel to attenuate UT signals. This is further complicated by the geometry *and location* of the ~~structural~~ *MPC* lid weld joint, which ~~includes a backing bar, counter bore, and vertical shell wall~~*is located in such proximity to the inner surface of the Transfer Cask that locating a UT transducer on the outside of the MPC shell is virtually impossible.* ~~These features add to the attenuation and scatter of the UT signals and increase the degree of interpretive difficulty for the UT operators.~~ This interpretive nature of UT examinations in austenitic stainless steel, under these conditions, significantly reduces the reliability of this examination.

To summarize, the needed level of assurance that the ~~PWR Baskets~~*MPCs* will remain leak free over their service life is provided by:

- use of redundant ~~seal~~ welds,
- use of controlled materials and *adequately qualified* weld processes,
- *use of adequately qualified welders,*
- the crack resistant nature of stainless materials,
- *visual examinations of the MPC lid-to-shell, vent port cover plate, drain port cover plate, and closure ring welds,*
- *dye penetrant examinations of the vent port cover plate, drain port cover plate, and closure ring welds,*
- *multi-layer* dye penetrant exam*inations of the MPC lid-to-shell weld, to include examination of the root and final* ~~at~~ weld *passes and each approximately 3/8-inch layer of weld depth* ~~build-up thickness less than critical flaw size, and~~

- *MPC hydrostatic test*
- helium leak test of the ~~shield~~ *MPC* lid, *vent port cover plate, and drain port cover plate* weld~~s.~~, *and*
- *very low probability of flawed weld being performed in the shop or the field and remaining undetected by inspection.*

3.3.2.3 Ventilation Offgas

The ISFSI is designed to confine radioactive materials within a sealed enclosure for the life of the facility. There are no expected radioactive releases during normal operations or credible accidents. ~~In the unlikely event a leaky PWR Basket must be placed in a Basket Overpack, evacuation of the Basket Overpack and backfilling with helium would be required. The operation is discussed in Chapter 5. A suitable filtration system such as a high efficiency particulate air (HEPA) filter would be used for the vacuum system vent path during this evolution.~~

3.3.3 PROTECTION BY EQUIPMENT AND INSTRUMENTATION SELECTION

3.3.3.1 Equipment

The equipment/components that have been identified as important to safety for the ISFSI are:

1. Concrete Cask,
2. ~~PWR Basket~~*MPC*,
3. ~~Basket Overpack,~~
4. ~~f~~*F*uel ~~d~~*D*ebris Process Can Capsule,
5. ~~5~~*4*. Failed Fuel Can,
6. ~~6~~*5*. Transfer Cask,
7. ~~7~~*6*. Transfer Station,
8. ~~8~~*7*. Transfer Station Pad and Impact Limiter,

~~9~~8. ~~Basket Hoist Rings~~*MPC Lift Cleats*, and

~~10~~9. Mobile Crane

The design criteria for the ~~PWR Basket and Basket Overpack~~*MPC* are summarized in Table 3.2-5. The design criteria for the Concrete Cask are summarized in Table 3.6-1.

3.3.3.2 Instrumentation

3.3.3.2.1 Temperature Monitoring

A temperature monitoring device is provided for each of the air outlet vents per Concrete Cask (four per Concrete Cask). The temperature monitoring devices are commercial grade. Additional discussion of temperature monitoring is provided in Section 5.1.3.4 and Section 5.4.1.

3.3.3.2.2 Seismic Monitoring

Seismic instrumentation is located at appropriate locations at the ISFSI facility so that the seismic response of equipment/components identified as important to safety can be determined and compared with the design bases.

3.3.4 NUCLEAR CRITICALITY SAFETY

The storage system is designed to maintain subcritical conditions (K_{eff} ≤0.95) under normal handling and storage conditions, off-normal handling and component functioning, and hypothetical accident conditions.

3.3.4.1 Control Methods for Prevention of Criticality

Subcritical conditions are to be maintained by ~~PWR Basket internal~~*MPC fuel basket* geometry *and the use of Boral*. The ~~PWR Basket internals~~*MPC fuel basket* will establish fuel assembly spacing. The design will assume a fuel assembly enrichment equal to or greater than the maximum initial fuel assembly enrichment that will be stored (3.56 wt% U^{235}). No credit will be taken for burnup or fuel assembly control inserts. ~~Although~~ Boral is used as a neutron absorbing material in the ~~PWR Basket internals~~*MPC fuel basket* design ~~to meet the transportation requirements of 10 CFR 71~~, ~~it~~ *and* is ~~not~~ credited in the criticality analysis for dry storage conditions. ~~An exemption from the requirements of 10 CFR 72.124(b) to "...provide for positive means to verify their continued efficacy," has been requested since no credit is taken for the Boral.~~

Boral is a patented product of AAR Advanced Structures that has been licensed for and used as a neutron poison material in many spent nuclear fuel storage applications. Boral is made of two

chemically compatible materials, boron carbide and 1100 alloy aluminum. The boron carbide has a high boron content in a physically stable and chemically inert form. The aluminum is a lightweight metal with high tensile strength that is protected from corrosion by a highly resistant oxide film. The materials are ideal for long-term use in a radiation, thermal, and chemical environment of a nuclear reactor, Spent Fuel Pool, or dry Concrete Cask. There is a substantial amount of experience demonstrating the capability of Boral to function as designed for extended periods of time. ~~However, since no credit is taken for Boral in the criticality analysis, it is not necessary to verify its continued efficacy in~~*As discussed in Sections 3.4.12 and 6.3.2 of the HI-STORM FSAR (Reference 1), the MPC fuel basket is designed such that the fixed Boral neutron absorber will remain effective for the 40-year design life of* the ISFSI. *There are no credible means to lose the Boral in dry storage service. Therefore, there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber as required by 10 CFR 72.124(b).*

Table 3.1-1 lists the fuel characteristics. Fuel debris *(except for fuel assemblies classified as fuel debris)* is placed in fuel debris Process Cans. Fuel debris Process Cans that contained organic filter material ~~are~~ *were* processed to destroy the organic material, sealed in fuel debris Process Can Capsules, and *are* placed in Failed Fuel Cans. Fuel debris Process Cans containing only fuel pellets or fragments do not require processing *and are placed directly in Failed Fuel Cans. Fuel assemblies classified as either damaged fuel or fuel debris are placed directly in Failed Fuel Cans.* ~~Administrative controls limit the amount of fuel debris which can be placed within a PWR Basket.~~

3.3.4.2 Error Contingency Criteria

The values of K_{eff} include error contingencies and calculational and modeling biases. K_{eff} equals the calculated K_{eff}, plus criticality code bias, plus two times code uncertainty. K_{eff} has been evaluated for *an* infinite Concrete Cask array. *Fuel assemblies are assumed to be centered* ~~and optimum assembly position~~ within ~~the~~ *each* cell. *Optimization of fuel assembly position within the cells (i.e., all fuel assemblies assumed to be toward the center of the MPC) has not been included in the criticality analysis. Positioning of all assemblies toward the center of the MPC due to random or accident-induced relocation is not considered credible. The criticality analyses were conducted assuming the MPC fuel basket as-fabricated dimensions to be at their most conservative values with respect to criticality. This is consistent with the approach described in Chapter 6 of the generally certified HI-STORM 100 System Final Safety Analysis Report (Docket 72-1014).*

3.3.4.3 Verification Analyses

The criticality analysis was performed using an NRC accepted code *as described in the FSAR for the HI-STORM 100 System.* The code was validated in accordance with ~~BFS Quality Assurance Program (Reference 3)~~*NUREG-1536 and Holtec's Quality Assurance Program.*

3.3.5 RADIATION PROTECTION

3.3.5.1 Access Control

Personnel exposure and access to radioactive material is minimized by the ISFSI design. Personnel exposure is minimized by incorporating shielding into the design of the ~~PWR Baskets~~*MPCs*, Transfer Cask, and Concrete Casks. Chapter 7 discusses the administrative procedures designed to limit personnel exposure. In addition, physical access to the ISFSI is restricted by a security fence. Access to the facility is limited to those persons who have satisfactorily completed access training or are escorted by a person who has satisfactorily completed access training. Training requirements for facility access are discussed in Section 9.3.

In addition to the security fence which restricts access to the ISFSI, access to stored radioactive material is further prevented by facility design as the radioactive materials are confined within welded steel enclosures (~~PWR Basket~~*MPC*).

3.3.5.2 Shielding

The storage system in conjunction with appropriate administrative control is designed to maintain radiation exposure As Low As Reasonably Achievable (ALARA). The storage system ~~is~~ design~~ed to provide~~ *results in* an average external side surface dose (gamma and neutron) of less than 100 mrem/hr on the sides and ~~250~~ *300* mrem/hr on the top and at the air vents.

Expected dose rates associated with ISFSI operations are contained in Section 7.4.

3.3.5.3 Radiological Alarm Systems

The Concrete Cask system does not produce routine solid, liquid, or gaseous effluents. Section 8.1.3 discusses an inadvertent release of surface contamination from the exterior of the ~~PWR Basket~~*MPC*. The consequences of this event are negligible (~~2.4~~*2.50* mrem at 100 meters). Therefore, an alarm for airborne radioactivity is not required to protect personnel or the environment.

The estimated working dose rate for the Concrete Cask (maximum fuel burnup) is ~~10.2~~*6.5* mrem/hr and the highest dose rate at 100 meters from the edge of the ISFSI Storage Pad is calculated to be ~~0.023~~*0.077* mrem/hr. These dose rates do not warrant a radiation alarm to protect personnel or the environment.

Based on the above, radiological alarms are not required for the Trojan ISFSI.

3.3.6 FIRE AND EXPLOSION PROTECTION

The potential for fires at the ISFSI are minimized by the use of paved open areas and minimum combustible materials within the ISFSI security fence. As discussed in Section 2.2.3.3 the facility is well protected from industrial and forest fires by natural barriers. Sections 8.2.9 and 8.2.14.2.2 provide additional discussion on fires.

Explosion analyses for the ISFSI are presented in Sections 8.2.8, 8.2.14.2.3 and 8.2.14.2.4.

3.3.7 MATERIALS HANDLING AND STORAGE

3.3.7.1 Spent Fuel Handling and Storage

The loading of each ~~PWR Basket~~*MPC* is limited to ~~a~~ *the design basis maximum* decay heat load *limit shown in Table 3.1-3.* ~~of 24 kWt per PWR Basket with no individual fuel assembly heat generation exceeding 1.08 kWt.~~ The ~~ISFSI~~ *Trojan Storage System* is designed to accommodate *the design basis maximum* decay heat load~~s~~ ~~of 26 kWt (24 kWt for Basket Overpacks)~~ and maintain fuel cladding temperature below limits established for inert dry storage (Reference 4). In addition, temperature limits for storage system components ~~must~~ *are* also ~~be~~ maintained below design limits. The Technical Specifications establish surveillances to preclude exceeding material design temperature limits.

The fuel clad temperature limit is a function of fuel burnup, fuel pin fill gas pressure, and fuel age. For the Trojan ISFSI, the fuel clad temperature limit ~~was determined to be 374°C (705°F)~~*is shown in Table 3.1-3.* This limit was determined using Westinghouse 17x17 fuel with *a limiting combination of* ~~41,900 MWd/MTU and 5 years of~~ cooling *time and burnup to produce the highest decay heat emission rate, as shown in Table 3.1-3.* This fuel is limiting because any assembly with *burnup* greater than ~~41,900 MWd/MTU~~*that listed in Table 3.1-3* will have greater than ~~nine years~~*the minimum amount* of cooling *time* prior to loading in the ~~PWR Basket~~*MPC,* and *will have* lower initial fill gas pressures.

Concrete Cask temperature limits are based on guidance provided by ACI-349 (1985). Section 4.2.6 presents the thermal analysis of the Concrete Cask and ~~PWR Basket~~*MPC* for the anticipated range of operating conditions.

The design criteria for maintaining *a* subcritical condition are presented in Section 3.3.4. Contamination control is addressed in Section 3.3.2.

~~Although n~~*N*o credible events result in loss of confinement due to damage to a*n* ~~PWR Basket~~*MPC.* ~~provisions for this unlikely occurrence are provided in the ISFSI design. If the affected PWR Basket cannot be repaired, the affected PWR Basket can be removed from the Concrete Cask and placed in a Basket Overpack. The Basket Overpack is designed and~~

~~fabricated to the same criteria as the PWR Basket. The design of the Basket Overpack must also ensure fuel clad temperature limits and Concrete Cask temperature limits are not exceeded. Operations involving the Basket Overpack are presented in Chapter 5. Design analyses for the Basket Overpack are presented in Chapter 4.~~

The *storage system components and the* ISFSI ~~shall have~~*are designed for* a minimum design life of 40 years in accordance with the requirements of Oregon Administrative Rule 345-26-0390(4)(j).

3.3.7.2 Radioactive Waste Treatment

ISFSI operations do not result in the generation of liquid or gaseous radioactive waste. Although generation of solid radioactive waste is not expected, it is possible that small quantities of low level radioactive waste could be generated as a result of routine radiological surveys (e.g., swipes). This waste would be stored in an appropriate container pending disposal.

3.3.7.3 Waste Storage Facilities

Chapter 6 discusses waste confinement and management for the ISFSI.

3.3.8 INDUSTRIAL AND CHEMICAL SAFETY

There are no required special industrial or chemical design criteria that are important to personnel or plant safety.

3.4 CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

Structures, systems, and components important to safety are those features of the ISFSI whose function is:

1. To maintain the conditions required to store spent fuel safely,

2. To prevent damage to the spent fuel container during handling and storage, or

3. To provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

ISFSI structures, systems and components important to safety are identified in Section 3.3.3.1.

3.5 DECOMMISSIONING CONSIDERATIONS

Decommissioning activities consist primarily of transferring the ~~PWR Baskets~~*MPCs* from the Concrete Casks into ~~a Shipping~~ *Transport* Cask*s*. The ~~PWR Baskets~~*MPCs* are then shipped off-site for disposal or storage.

The storage system has been designed to minimize contamination of the Concrete Cask exterior during loading and unloading operations. Although no contamination of the Concrete Cask is expected, the interior steel liner can be decontaminated and the complete Concrete Cask broken up (or left whole) and shipped to a landfill.

3.6 SUMMARY OF DESIGN CRITERIA

The design criteria for the ~~PWR Basket and Basket Overpack~~*MPC* are presented in Table 3.2-5. A summary of the Concrete Cask design criteria is presented in Table 3.6-1.

3.7 REFERENCES

1. ~~"~~*Final* Safety Analysis Report for the ~~TranStor™ Part 71 Shipping~~*Holtec International HI-STORM 100 Dry* Cask *Storage* System,~~" BNFL Fuel Solutions~~ *Holtec Report No. HI-2002444*, Revision ~~A~~*0*, ~~February~~*July* ~~1999~~*2000, and proposed Revision 1F.*

2. "Seismic Margin Earthquake Study," letter from PGE to NRC dated May 26, 1993, Docket 50-344.

3. ~~BNFL Fuel Solutions Quality Assurance Program for the Ventilated Storage Cask System, Pacific Sierra Nuclear Associates and BNFL Fuel Solutions, October 1991, Docket Number 72-1007~~*U. S. Nuclear Regulatory Commission Interim Staff Guidance (ISG)-4, "Cask Closure Weld Inspections," Revision 1.*

4. E.R. Gilbert et al., "Control of Degradation of Spent LWR Fuel During Dry Storage in an Inert Atmosphere," PNL-6364, Pacific Northwest Laboratory, Richland, WA (1987)

Table 3.1-1

Fuel Characteristics

	Parameters	Westinghouse 17x17	B&W 17X17
Fuel Assemblies:	UO_2 rods per assembly	264	264
	Rod Pitch (in)	0.496	0.496
	Overall dimensions (in)	8.426 x 8.426	8.425 x 8.425
	Fuel Weight, (lb U/ assy)	1154	1129
	Structural Weight (excludes fuel)	313	302
	Total Fuel Assy Weight (dry)	1467	1431
Fuel Rods:	Outside diameter (in.)	0.374	0.374
	Clad thickness (in.)	0.0225	0.024
	Active fuel length (in.)	144	144
	Clad Material	Zircaloy-4	Zircaloy-4
Fuel Pellets:	Material	UO_2 sintered	UO_2 sintered
	Density (% of theoretical)	95	95
	Diameter (in)	0.3225	0.3195
	Maximum Enrichment	3.46 wt% U^{235}	3.56 wt% U^{235}

Table 3.1-2

Design Maximum Radiological Characteristics of Stored Material

Characteristic	Value
Fuel:	
~~Maximum burnup 5 yr. cooled[1]~~	~~40,000 MWd/MTU~~
Maximum burnup ~~6~~ *9*-yr cooled[2]	~~45~~*42*,000 MWd/MTU
Initial Enrichment[3]	~~3.02 wt% U^{235} (for 40,000 MWd/MTU)~~ ~~3.30~~*3.09* wt% U^{235} (for ~~45~~*42*,000 MWd/MTU)
Gamma Source (*0.45 – 3.0 MeV;* 24 fuel assemblies)	~~1.856E+17~~*5.856E+16* γ/sec
Neutron Source (24 fuel assemblies)	~~1.187E+10~~ *7.872E+9* η/sec

~~[1] The 5 year cooled fuel results in the most conservative gamma source term.~~

~~[2] The 6 year cooled fuel results in the most conservative neutron source term. Neutron source includes factors to account for axial burnup profile and sub-critical neutron multiplication.~~

~~[3] Low initial enrichments will yield higher gamma and neutron source terms for a given burnup. The enrichment values provided bound the Trojan ISFSI spent fuel inventory.~~

Table 3.1-3

Design Maximum Thermal Characteristics of Stored Material

Characteristic

Fuel	Value
Decay heat per assembly	~~1,080~~*.725* KWt
Decay Heat per ~~PWR Basket~~*MPC*	~~26~~*17.4* KWt ~~(24 KWt for Basket Overpacks)~~
Burnup (~~5~~*9*-year cooled)[1]	~~40,000~~*39,345* MWD/MTU
Clad Temperature	
Long Term Limit	~~374~~*341.7*°C (~~705~~*647*°F)
Accident Limit	570°C (1058°F)

[1] Assemblies with greater than ~~40,000~~*39,345* MWD/MTU are not limiting because they will have at least ~~9~~*13* years of cooling at time of loading in the ~~PWR Basket~~*MPC*.

Table 3.1-4

Physical Parameters of Fuel Assembly Inserts

RCCA	Neutron Absorber	Ag-In-Cd
	Cladding Material	304 SS
	Number of Absorber Rods per Assembly	24
	Number of Assemblies	61
	Weight of Assembly	149 lbs.
	Overall Length	161 in.
BPRA	Cladding Material	304-SS
	Number of Absorber Rods per Assembly	Varies between 9 and 20
	Weight of Assembly	Weight is bounded by Weight of RCCA
	Overall Length	156 in.
	Number of BPRA	92
Thimble Plugs	Material	304 SS
	Weight	Bounded by Weight of RCCA
Source	Cladding Material	304-SS
	Weight	Bounded by Weight of RCCA

Table 3.2-1

Wind and Tornado Design Specifications

Environmental Condition	Value
Rotational Wind Speed, mph	290
Translational Speed, mph	70
Maximum Wind speed, mph	360
Radius of Max. Wind Speed, ft.	150
Pressure Drop, psi	3.0
Rate of Pressure Drop, psi/sec	2.0

Table 3.2-2

Design Basis Tornado Generated Missiles

Missile Description	Weight (lbs.)	Velocity (mph)
Automobile	3960	126
Armor Piercing Shell (8 in. diameter)	275	126
Steel Sphere (1 in. diameter)	0.22	126

Table 3.2-3

Design Load Combinations for Concrete Cask

Load Comb	Dead	Live	Wind	Norm/ Acc Temp	Seismic	Tipover	Tornado	Soil Pressure
1	1.4D	+1.7L						
2	1.4D	+1.7L						+1.7H
3	0.75(1.4D	+1.7L	+1.7W	$+1.7T_O$				+1.7H)
4	0.75(1.4D	+1.7L		$+1.7T_O$				+1.7H)
5	D	+ L		$+ T_O$	$+E_{SS}$			+ H
6	D	+ L		$+ T_O$		+A		+ H
7	D	+ L		$+ T_a$				+ H
8	D	+ L		$+ T_O$			$+W_t$	+ H

D = Dead Load
L = Live Load
W = Wind Load
T_o = Normal Temperature Load
H = Soil Pressure Load
T_a = Accident Temperature Load
E_{ss} = Earthquake Load
W_t = Tornado Load
A = Cask Tipover Load

Table 3.2-4
Summary of Load Combinations
for ~~PWR Basket and Basket Overpack~~*MPC*

~~LOAD~~		~~Normal~~			~~Off-Normal~~			~~Accident~~							
~~ASME Service Level~~		~~A~~			~~C~~			~~D~~							
~~Load Combination Number[1]~~		~~1~~	~~2~~	~~3[2]~~	~~1~~	~~2[2]~~	~~3~~	~~1~~	~~2~~	~~3~~	~~4~~	~~5[2]~~	~~6~~	~~7~~	~~8~~
~~Dead Weight~~	~~Basket with fuel~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~
~~Thermal~~	~~Inside Concrete Cask: 75°F~~ ~~Inside Transfer Cask: 75°F~~ ~~Inside Concrete Cask: 40°F or 100°F~~ ~~Inside Concrete Cask: Max Heat Load (125°F)~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~
~~Pressure~~	~~Normal~~ ~~Accident~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~	~~X~~
~~Handling Load[3]~~	~~Normal~~ ~~Off-Normal~~		~~X~~	~~X~~	~~X~~	~~X~~	~~X~~					~~X~~	~~X~~		
~~Drop (Vertical or Horizontal)[3]~~								~~X~~							
~~Seismic~~									~~X~~						
~~Flood~~										~~X~~					
~~Tornado~~											~~X~~				

~~[1] Load Combination Number corresponds to load combinations provided on Table 3.2-2.~~

~~[2] Controlling load combination for the Service Level.~~

~~[3] Not applicable for Basket Overpack. Basket Overpack is only a storage component and will not be used for fuel handling.~~

LOAD		Design (ASME Code Pressure Compliance)	Normal		Off-Normal		Accident		
ASME Service Level		A	A		B		D		
Load Combination Number		1	1	2	1	2	1	2	3
Dead Weight	MPC with fuel		X	X	X	X	X	X	X
Thermal	Normal		X	X			X		X
	Off-Normal				X	X			
	Accident							X	
Internal Pressure	Normal (100 psig)	X	X				X		
	Off-Normal (100 psig)				X				
	Accident (125 psig)							X	
External Pressure	Normal (40 psig)	X		X					
	Off-Normal (40 psig)					X			
	Accident (60 psig)								X
Handling Load			X	X	X	X			
Drop (Vertical or Horizontal)							X		

Table 3.2-5
~~PWR Basket and Basket Overpack~~*MPC* Design Criteria

~~Component (Applicable Code or Criteria)~~	~~Criteria~~
~~PWR Basket or Basket Overpack~~ ~~Normal Operation - Service Level A~~ ~~ASME Section III, Subsection NC (shell)~~ ~~ASME Section III, Subsection NG (internals)~~	~~$P_m < 1.0\ S_m$~~ ~~$P_L + P_b < 1.5\ S_m$~~ ~~$P + Q < 3.0\ S_m$~~
~~PWR Basket or Basket Overpack~~ ~~Off-Normal Operation - Service Level C~~ ~~ASME III, Subsection NC (shell)~~ ~~ASME III, Subsection NG (internals)~~	~~$P_m < 1.2S_m$ (shell), $< 1.5S_m$ (cells)~~ ~~$P_L + P_b < 1.8S_m$ (shell), $< 2.25S_m$ (cells)~~
~~PWR Basket or Basket Overpack~~ ~~Accident Condition - Service Level D~~ ~~ASME III, Subsection NC (shell)~~ ~~ASME III, Subsection NG (internals)~~	~~$P_m < 2.4\ S_m$ or $0.7\ S_u$~~ ~~(whichever is less)~~ ~~$P_L + P_b < 3.6\ S_m$ or $1.0\ S_u$~~ ~~(whichever is less)~~

Component (Applicable Code)	Stress Category	Design	Levels A & B	Level D[1]
MPC Confinement Boundary (ASME Section III, Subsection NB)	Primary Membrane, P_m	S_m	N/A[2]	Lesser of $2.4S_m$ and $0.7S_u$
	Local Membrane, P_L	$1.5S_m$	N/A	150% of P_m Limit
	Membrane plus Primary Bending	$1.5S_m$	N/A	150% of P_m Limit
	Primary Membrane plus Primary Bending	$1.5S_m$	N/A	150% of P_m Limit
	Membrane plus Primary Bending plus Secondary	N/A	$3S_m$	N/A
	Average Shear Stress[3]	$0.6S_m$	$0.6S_m$	$0.42S_u$
MPC Fuel Basket (ASME Section III, Subsection NG)	Primary Membrane, P_m	S_m	S_m	Lesser of $2.4S_m$ and $0.7S_u$[3]
	Primary Membrane plus Primary Bending	$1.5S_m$	$1.5S_m$	150% of P_m Limit
	Primary Membrane plus Primary Bending plus Secondary	N/A[2]	$3S_m$	N/A

[1] Governed by Appendix F, Paragraph F-1331 of the ASME Code, Section III.
[2] No specific stress intensity limit applicable.
[3] Governed by NB-3227.2 or F-1331.1(d).

Table 3.2-6
Design Load Combinations for Transfer Station

Load Comb	Allowable	Dead	Live	Wind	Normal/Accident Temp	Seismic	Handling	Soil Pressure
1	S	D	+ L					
2	S	D	+ L					+ H
3	1.33S	D	+ L	+W				+ H
4	1.5S	D	+ L		+ T			+ H
5[1]	1.6S	D	+ L		+ T	+ E		+ H
6	1.7S	D	+ L		+ T		+ A	+ H
7	1.7S	D	+ L		+ Ta			+ H

D = Dead Load
L = Live Load
W = Wind
T = Normal Temperature
H = Soil Pressure
Ta = Accident Temperature
E = Earthquake (seismic margin earthquake)
A = Cask Handling
S = Allowable strength per AISC Manual

[1] Indicates controlling load combination.

Table 3.6-1
Summary of Concrete Cask Design Criteria

Design Load Type	Design Parameters	Applicable Criteria and Codes
Tornado	360 mph, maximum 290 mph, rotational 70 mph, translational 3.0 psi, pressure drop 2.0 psi/sec, rate of pressure drop	NRC Reg. Guide 1.76 (1974) ANSI A58.1 (1982)
Tornado Missile	At 126 mph: Automobile 8 in dia. shell 1 in solid sphere	NUREG ~~0800~~ *0800* (1987) Section 3.5.1.4
Flood	Not Applicable -- ISFSI elevation above credible flood levels	10 CFR 72.122
Seismic	0.38g horizontal acceleration 0.25g vertical acceleration	10 CFR 72, NRC R.G. 1.60 Rev. 1 , NRC R.G. 1.61 Rev. 1
Dead Loads	Dead weight, including ~~PWR Basket~~*MPC* weight (concrete density at 145 lb/ft^3)	ANSI 57.9 (1984)
Air Temperature	-40°F minimum 100°F maximum 125°F short term extreme	10 CFR 72.122
Concrete Temperature	225°F- normal 300°F - off normal 350°F - accident	ACI-349 (1985)[1]
Snow and Ice Loads	67.2 psf included in live loads	ANSI A58.1 (1982) and ANSI 57.9 (1984)

1 ACI-349 (1985) establishes a normal operating temperature limit of 150°F, except for local areas which may not exceed 200°F. Refer to Section 4.2.4.2.4 for discussion of elevated limits.

Table 3.6-2
Summary of Transfer Cask Design Criteria

~~Design Function~~	~~Design Parameters~~	~~Applicable Criteria and Codes~~
~~Basket Lift~~[1]	~~Lifting Stress < S_y/3 or < S_u/5~~	~~ANSI N14.6 (1993) and NUREG 0612 (1980)~~
~~Material Properties~~	~~Charpy Impact Energy Absorption ≥ 25.6 ft. lbs. at 0°F~~	~~Charpy Impact Testing per ASTM A370~~
~~Neutron Shielding Material~~	~~Ambient Air Temperature ≤ 100°F~~	~~Material Qualification Test Report Submitted on 11/25/98 via VPN-076-98~~

Design Load Type	*Design Parameters*	*Applicable Criteria and Codes*
Tornado Missile	*At 126 mph:* *Automobile* *8 in dia. shell* *1 in solid sphere*	*NUREG 0800 (1981)* *Section 3.5.1.4*
Dead Loads	*Dead weight, including MPC weight with water*	*R.G. 3.61 (1989)*
Air Temperature	*0°F minimum* *100°F maximum*[1]	*ANSI/ANS 57.9 (1992)*
Design Temperatures	*Structural materials: 400°F* *Shielding materials:* *Lead: 350°F (max.)* *Solid Neutron Shield: 300°F (max.)*	*ASME Code Section II, Part D and Manufacturer Data*

[1] ~~The Transfer Cask is only used as a lifting device in the Fuel Building. At the Transfer Station, it is used as a stationary shielded structure for support of a PWR Basket.~~ *Value is used as input for the thermal analysis, and is based on a 72-hour average ambient temperature.*