Oregon Office of Energy
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STAFF EVALUATION
of
PORTLAND GENERAL ELECTRIC INDEPENDENT SPENT FUEL STORAGE INSTALLATION

(ISFSI)

January 27, 1999

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I. EXECUTIVE SUMMARY

The Oregon Office of Energy (OOE) has reviewed the proposed Independent Spent Fuel Storage Installation (ISFSI) plan for Portland General Electric (PGE) Company's Trojan Nuclear Plant (TNP). The purpose of this review is to:

1. Evaluate whether PGE's ISFSI Safety Analysis Report (SAR) complies generally with the requirements of Oregon Administrative Rules, Chapter 345, Division 26 , and with OAR 345-026-0370 and 0390 in particular.
2. Determine whether activities associated with transfer of spent nuclear fuel from the spent fuel pool (SFP) to the ISFSI will be performed safely, and
3. Recommend amendments to OAR 345 Division 26 regarding the proposed ISFSI and transfer of spent nuclear fuel from the SFP to the ISFSI for storage.

The Oregon Energy Facility Siting Council (EFSC or "the Council") approved the PGE Decommissioning Plan for Trojan on March 11, 1996. Decommissioning activities commenced shortly thereafter. The Decommissioning Plan states PGE's intention to remove the spent nuclear fuel from the SFP and transfer it to an ISFSI on the Trojan site. This would enable PGE to remove certain contaminated components that are associated with safe storage in the pool. OAR 345-26-390 currently prohibits storage of spent nuclear fuel outside the SFP and requires rulemaking to approve of any alternate method of spent fuel storage. By submitting its SAR, PGE requested this rulemaking approval.

PGE issued the SAR on March 26, 1996. OOE's review of the SAR included an extensive review of the SAR and supporting documents, including calculations in support of structural and thermal analyses, quality assurance (QA) audit and surveillance reports, and draft procedures; numerous site visits and interviews with PGE representatives; observation of two PGE QA surveillances at the principal vendor's facilities, Sierra Nuclear Corporation (SNC)[1]; a visit to the major fabricator of the PWR baskets; and several meetings with the Council.

In our review of calculations that were important-to-safety, whether by PGE or SNC, OOE confirmed that the assumptions were conservative, that appropriate calculational methods were used, and that the calculations were reviewed and approved according to PGE's QA Program. For calculations involving the use of large computer codes, such as ANSYS, OOE verified that the computer codes were NRC approved. For calculations used in support of statements in the SAR, OOE verified that the SAR accurately reflected the calculation, its assumptions, inputs, and results.

OOE found that the SAR describes the design and safety features of the proposed ISFSI and demonstrates compliance with OAR 345 Division 26 in general and particularly OAR 345-026-0390. The NRC must independently find that the SAR complies with Title 10 Part 72 of the Code of Federal Regulations (10 CFR 72). The SAR includes an accident analysis demonstrating that the risk to the public is acceptably low.

Activities related to transfer of fuel from the SFP to the ISFSI are described in a separate document, License Change Application (LCA)-237. This document describes safety measures for fuel transfer activities and analyzes the consequences of postulated accidents. Fuel transfer activities at the SFP are

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[1]In 1998, SNC became BNFL Fuel Solutions Corporation, also referred to in this report as BNFL/SNC. Depending on the context, both SNC and BNFL/SNC are used in this report.

governed by PGE's existing NRC license under Title 10 Part 50 (10 CFR 50), and, therefore, are not described in the SAR. However, OOE reviewed both the SAR and LCA-237, and recommends amending EFSC rules based on both analyses.

This report includes: (1) an introduction to PGE's proposal to establish an ISFSI along with a background and schedule of the Trojan decommissioning, (2) a description of the ISFSI and its associated SAR, (3) OOE's evaluation of PGE's Safety Analysis, (4) OOE's evaluation of LCA-237, (5) an assessment of PGE's compliance with the OARs, and (6) OOE's conclusions and recommendations.

In summary, OOE concludes that the proposed plan for alternative spent nuclear fuel storage, as described in the SAR, meets the criteria set forth in OAR 345-26-390. OOE further concludes that measures have been taken to assure that fuel transfer activities described in LCA-237 will be performed safely. OOE therefore recommends new rules approving the Trojan ISFSI plan and requiring specific safety measures for fuel transfer and storage.

II. INTRODUCTION AND BACKGROUND

A. TROJAN DECOMMISSIONING AND SCHEDULE

The Trojan Nuclear Plant is located in northwest Oregon on the Columbia River approximately 40 miles north of Portland. Trojan is operated by Portland General Electric (PGE) and co-owned with the Eugene Water & Electric Board/Bonneville Power Administration (EWEB/BPA) and Pacific Power and Light.

Trojan began commercial operation in 1976. In January 1993, PGE decided to permanently close the plant, primarily for financial reasons related to repairing the plant's steam generators. On May 5, 1993, the U.S. Nuclear Regulatory Commission (NRC) amended the plant's federal operating license to a "Possession Only License" (POL). This license prohibits reactor operations but allows PGE to continue storing its spent nuclear fuel in the Trojan spent fuel pool (SFP). That license, issued under Title 10 Part 50 of the Code of Federal Regulations (10 CFR 50), will remain in effect until the site is shown to meet federal and state requirements for removal of residual radioactive contamination.

The process of dismantling the plant and removing all residual radioactive contamination is called "decommissioning." The Code of Federal Regulations defines decommissioning as the process of removing the facility safely from service and reducing residual radioactivity to a level that permits release of the property for unrestricted use and termination of the license. Pursuant to NRC and EFSC regulations, PGE submitted a Decommissioning Plan to the NRC and to EFSC on January 26, 1995. Council rules provided the opportunity for contested case review of the Decommissioning Plan on request; however, there was no request. After approximately a one year review, the Council approved the plan on March 11, 1996. The NRC approved the plan in April 1996.

PGE chose to decommission Trojan using a method called DECON.[2] This is the decommissioning alternative that takes place shortly after cessation of operations in which the equipment, structures, and portions of the facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use. The Trojan Decommissioning Plan calls for the removal of all radioactive and contaminated components and material on the site. A final survey is planned for approximately 2001. This survey is expected to confirm that all residual radioactive material has been removed.

Much of the decommissioning is completed. The Council approved of the removal of the steam generators and pressurizer in a review that was handled separately from the Decommissioning Plan. The was the Large Component Removal Project (LCRP). PGE removed those components in 1995. As of this writing, much of the plant equipment formerly associated with reactor operations has been removed, along with many of the non-safety related components that are no longer needed. However, the Trojan SFP and its associated pumps and cooling systems cannot be removed or decommissioned while spent fuel is stored in the pool. Further, the presence of spent fuel in the pool limits decommissioning activities, because nothing can be done that might endanger the pool, its water supply, or its associated systems.

To allow decommissioning to be completed by 2002, the Decommissioning Plan states that PGE intends to remove the fuel from the SFP and place it in dry storage casks. The casks would be air cooled using

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[2] The other primary method that was evaluated in the Decommissioning Plan and rejected was SAFSTOR. This is a deferred decontamination process that involves partially "mothballing" the facility until a portion of the plant's radioactivity has decayed to nonhazardous levels before proceeding with decontamination to levels that permit release for unrestricted use.

natural air circulation. They would be located on the Trojan site but outside the existing Fuel Building. This would allow the complete decommissioning of all buildings and structures related to plant operations.

This method of storage in dry casks is called an "Independent Spent Fuel Storage Installation" (ISFSI). The term ISFSI is defined under 10 CFR 72, which also contains the NRC standards for the ISFSI's design and operation. PGE's Decommissioning Plan states that PGE intends to transfer its spent fuel to an ISFSI in 1999 in order to complete decommissioning the rest of the plant by 2002. The Decommissioning Plan discusses the ISFSI in general terms but does not include details on its design, construction, or operation. These details are described in the ISFSI Safety Analysis Report (SAR), which PGE submitted on March 26, 1996. The SAR identifies the vendor that would construct the proposed ISFSI, demonstrates compliance with EFSC and NRC regulations, and provides an analysis of projected radiation exposure to the public under normal and accident conditions.

Under EFSC rule OAR 345-26-390(4), PGE cannot transfer fuel to an ISFSI without prior Council rulemaking approving the ISFSI. In submitting the SAR, PGE has requested that rulemaking. This report reviews the SAR, as well as other supporting documents.

B. SPENT FUEL STORAGE ISSUES

1. Current Spent Fuel Storage

There are 780 intact fuel assemblies currently in the SFP. There are also 10 partial spent fuel assemblies and 1 fuel rod storage container, which contain intact, suspect, or failed fuel rods. The fuel itself is in the form of ceramic uranium pellets, about one quarter of an inch in diameter and one half inch long. The pellets are stacked in hollow zircalloy (an alloy similar to steel) rods, approximately one third of an inch in diameter and 12 feet long. The rods ("fuel rods") are arranged in a 17 X 17 square bundle array, held together with grid assemblies and grid straps. This 17 X 17 square bundle of fuel rods is called a fuel assembly, and is approximately 12 feet long and 10 inches on a side. Fuel rods comprise 264 of the 289 available locations in a bundle. The remaining 25 locations contain guides for fuel assembly inserts, such as control rods or burnable poison rods, and instruments. All of the fuel assemblies used at Trojan during its operating period are presently stored in the SFP. In all, the 791 assemblies contain approximately 345 metric tons of uranium.

The SFP is located in the Fuel Building, which is part of the original power plant. The pool is approximately 40 feet deep. PGE must keep at least 23 feet of water over the spent fuel at all times to provide radiation shielding. The 23 feet is enough shielding so that even fuel handling personnel working directly over the pool receive minimal exposure from the fuel. Other personnel receive none. An NRC and EFSC approved security plan precludes unauthorized entry into the spent fuel area.

Spent fuel is still highly radioactive long after removal from a reactor due to daughter products from nuclear fission. The fission products produce both radiation and decay heat. For at least one year after removal from the reactor, spent fuel assemblies produce enough decay heat to require continued storage in water. The regulations at 10 CFR 72 require that spent fuel cannot be stored in an ISFSI unless it has cooled for at least one year. After that period, the decay heat has diminished to the point that air cooling is sufficient. By 1999, all Trojan fuel will have cooled in the SFP for more than six years.

The risks from spent fuel storage are much less than those from an operating reactor. Spent fuel produces very little heat compared to an operating reactor. The hottest assemblies at Trojan produce about 1 kW of heat (about the same as ten 100 watt light bulbs), and most assemblies produce less. This heat production will decrease further over time. The radiation dose from the spent fuel decays at a rate similar to the heat load. After about 10 years, the radiation is approximately half of what it was after being removed from the reactor.

The danger with any stored radioactive material is the dispersion of that material into the environment or biosphere. In order to have dispersion, there must be a driving force, such as heat or pressure. Unlike the high temperature and pressure conditions in a reactor plant, there is insufficient driving force in an ISFSI basket to create a release and dispersion of radioactive materials. The pressure in an ISFSI basket is approximately atmospheric; and after six years of cooling, the heat generation rate of spent fuel is too low to cause significant particulate dispersal in the event of a cask failure. Furthermore, the radioactive material is not readily dispersible. Except for some gaseous fission products produced during reactor operations, most of the radioactive material in spent fuel is in the form of solid ceramic material encapsulated in a zircalloy cladding. The fuel assemblies, in turn, are stored in airtight stainless steel baskets. The structural integrity of these baskets was a key element in OOE's review.

2. GTCC Waste Storage & RVAIR

The SAR originally described the ISFSI as storage for both high level waste and "Greater Than Class C" (GTCC) waste. The term "high level waste" generally refers to spent nuclear fuel and related radioactive material, such as control rods, which are part of the fuel assemblies. GTCC waste is a classification of radioactive waste defined in 10 CFR 61.55. Low level radioactive waste is classified as either Class A, B, C, or GTCC. The classifications are based on the concentrations of long and short-lived radionuclides present. Classes A, B, and C are suitable for disposal in near-surface disposal sites, but GTCC waste must be placed in a geologic repository such as the one proposed at Yucca Mountain, Nevada. OAR 345-26-390(2) defines "GTCC waste" as waste that is not eligible for disposal as low-level radioactive waste at a land disposal facility, such as the one operated by U.S. Ecology on the Hanford reservation in Washington.

The GTCC waste at Trojan consists of the reactor vessel internals, which are comprised of the structural supports for the fuel, guide structures for control rods and instruments, core support plates and baffles, and other internal components that were located close to the fuel during reactor operations. The internals are highly radioactive due to the presence of activation products, which means that the steel components themselves have become radioactive as a result of neutron irradiation during reactor operations. The radioactivity is in the steel itself and is not in a form that can be easily dispersed.

As originally approved, the Decommissioning Plan called for removal of the reactor vessel internals by segmentation and storage of this waste in the ISFSI in the same manner that the spent nuclear fuel would be stored. Two of the original 36 ISFSI casks would have been for this GTCC waste. However, on January 15, 1997, PGE submitted a proposed change to the Decommissioning Plan that would permit the removal and disposal of the reactor vessel and its internals as a single piece. This single unit would be shipped by barge up the Columbia River and disposed of at the U.S. Ecology low level waste site at the Hanford reservation in Washington.[3] PGE called this proposal the Reactor Vessel and Internals Removal (RVAIR) Project.

The OOE staff reviewed the RVAIR Project and recommended its approval in our staff report of September 29, 1998.[4] The Council then met and approved RVAIR on October 15, 1998. PGE also

[3] PGE letter Yundt, P. D. to D. Stewart-Smith, OOE, Supplemental Information to Support the Proposed Change to the Trojan Decommissioning Plan, CPY-004-097, January 30, 1997

[4] OOE Staff Evaluation of PGE RVAIR Project, September 29, 1998

received approval of RVAIR from the NRC, the U.S. Department of Transportation (USDOT,) and the Washington Department of Health (WDOH). In August 1998, PGE revised the SAR to delete references to GTCC waste storage in the ISFSI. This report assumes that GTCC waste will not be stored in the ISFSI.

3. Industry Experience with Dry Cask Storage

Spent nuclear fuel has been stored under dry conditions since the mid-1940s when fuel examinations began in hot cells. Irradiated light water reactor fuel, such as Trojan's, have been handled in dry cells since the early 1960s, and some fuels have been in storage under dry conditions for approximately 20 years.[5] Experience with the storage of spent fuel in dry casks is extensive and growing. At least ten nuclear power plant sites are already using dry cask storage: Virginia Electric and Power's Surry plant; Carolina Power and Light's H. B. Robinson plant; Duke Power's Oconee plant; Public Service of Colorado's Fort St. Vrain plant; Consumers Power's Palisades plant; Baltimore Gas and Electric's Calvert Cliffs plant; Northern States Power's Prairie Island plant; Arkansas' ANO plant; Toledo Edison's Davis-Besse plant; and Wisconsin Electric Power's Point Beach plant. These ten facilities use a variety of dry storage technologies, including vaults, and several types of metal casks, concrete casks, and baskets.

4. History and Status of a Permanent Federal Repository

EFSC rule OAR 345-26-390 states that onsite spent fuel storage in an ISFSI is an interim measure only. Onsite storage is needed because of delays in the development of a federally licensed permanent storage facility. The following is a brief history and current status of the development of a federal repository for high-level radioactive waste at Yucca Mountain, Nevada.

Permanent disposal of high-level radioactive waste is ultimately the responsibility of the Federal Government. The government's program for geologic disposal of nuclear waste was originally established by Congress in the Nuclear Waste Policy Act of 1982, and reaffirmed by the Nuclear Waste Policy Amendments Act of 1987.

At that time, Congress directed the U.S. Department of Energy (DOE) to determine the suitability of Yucca Mountain, Nevada, as the sole candidate for a repository site. The Energy and Water Development Appropriations Act for 1997 further directed DOE to address the remaining technical uncertainties at Yucca Mountain and to complete a viability assessment of the site by the end of 1998. Progress on the project has been slow. A major milestone, construction of the five mile underground exploratory studies facility, was completed in 1998.

The purpose of the Viability Assessment is to provide sufficient financial and technical data to allow the Congress and DOE to decide whether a repository at Yucca Mountain is viable. Following an assessment favorable to continuing the development of a repository, DOE will prepare the draft Environmental Impact Statement in 1999, and, if appropriate, recommend the repository site to the President and Congress in 2001. If the President and Congress approve the site, DOE will submit a license application to the NRC in 2002. Following a successful NRC licensing review, construction and operation will proceed. These are target dates established by DOE. Delays may be expected, plus there will be an extensive public decision-making process, rulemaking process, and Congressional reviews.

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[5] NRC, Part 72 Statements of Consideration, September 29, 1995, p.72-SC-71

5. Plans for an Interim Federal Repository

In addition to the ongoing research to develop a geologic repository at Yucca Mountain, Congress continues to consider legislation that would initiate an interim storage program for both spent fuel and high-level radioactive waste. According to DOE estimates, by 2010, about 55 sites in 30 states will require new spent nuclear fuel storage capacity for over 11,000 metric tons of uranium. (By comparison, the Trojan ISFSI will store approximately 360 metric tons.) DOE is conducting safety design and engineering analyses for a non-site specific, generic interim storage facility in order to reduce the time required for facility licensing once a decision is made. In 1997, Senate Bill 1936 would have created a centralized interim storage facility in Nevada. However, the bill did not pass, and the outcome of future legislation is uncertain.

In 1996, several utilities along with a group of states and public utility commissions sought a ruling from the U.S. Court of Appeals for the District of Columbia that the DOE is obligated under the Nuclear Waste Policy Act to begin disposing of the utilities' spent nuclear fuel by January 31, 1998. In a decision issued in July 1996, the Court agreed with the utilities, holding that the Act creates an obligation in the DOE, reciprocal to the utilities' obligation to pay fees, to start disposing of spent nuclear fuel no later than January 31, 1998. The Court rejected DOE's position that it does not have an obligation in the absence of a repository or other facility constructed under the Act. The Court remanded the case to the DOE for further proceedings, stating that it would be premature for a court to attempt to fashion an appropriate remedy. As of this writing, the DOE is still reviewing the Court's ruling to determine what steps it should take. On January 31, 1998, a DOE spokesperson stated that the "Department is committed to fulfilling its obligation to dispose of the nation's spent nuclear fuel and high-level waste as soon as possible — but not sooner than it takes to fully study and understand the science of a permanent waste disposal site."[6]

Due to the uncertainty in the plans for a federally licensed spent fuel storage site, EFSC requires a 40 year design life for the ISFSI. The federal ISFSI license issued under 10 CFR 72 would expire after 20 years. And, many experts in the industry believe that a federally licensed facility will be open within 20 years. Nevertheless, EFSC believed it was prudent to require the 40 year life.

C. REGULATORY BASIS FOR OOE REVIEW

1. State Regulation

The Council has historically regulated spent nuclear fuel storage at Trojan through rules in OAR 345 Chapter 26. Prior to the plant's closure in 1993, the rule governing spent fuel storage at Trojan was OAR 345-26-145. That rule stated that:

(a) Storage of spent nuclear fuel at a nuclear power plant is an interim measure; otherwise utilities and residents of Oregon would face the financial burden of maintaining, operating, and safeguarding the on-site storage facilities indefinitely;

(b) The purpose of this rule is to cooperate with the federal government in accordance with Oregon's siting policy in ORS 469.310 to ensure the safety of interim on-site storage and to ensure spent nuclear fuel will not be an undue financial burden to utilities or people of Oregon.

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[6] USDOE Press Release, Barrett, L., Acting Director of OCRWM, , January 31, 1998

OAR 345-26-145 limited the number of assemblies that could be stored at Trojan. In 1983, EFSC amended OAR 345-26-145 to permit expansion of the SFP capacity to 1400 assemblies. This would have accommodated spent fuel expected to be produced if Trojan continued operating until the expiration of its operating license.

In 1994, following the plant's closure, the Council amended all of its Trojan related rules. The Council deleted OAR 345-26-145 and replaced it with a new rule, OAR 345-26-390. OAR 345-26-390 limited spent fuel storage to 791 complete and partial assemblies, the number of assemblies at the site at the time of plant shutdown. It retained requirements from the old rule concerning effluent releases and off-site radiation exposures in case of accidents. OAR 345-26-390(3) further states :

"Spent Nuclear Fuel shall be stored only in the Trojan spent fuel pool (SFP). Storage of spent nuclear fuel in any other type of facility shall require the prior adoption of rules by the Council allowing the specific type of proposed facility."

In November 1995, EFSC amended OAR 345-26-390 to allow spent fuel storage in other than the SFP, and setting forth general criteria that any alternate spent fuel storage installation must meet. This evaluation is based on the rules adopted in November 1995. OAR 345-26-390(4) states that:

"The Council may approve by rule a plan by the owner for storage of spent nuclear fuel or other related radioactive materials and wastes in an interim storage facility other than the SFP, and in doing so the Council may impose criteria in addition to those set forth in this rule."

The rule requires compliance with current EFSC criteria and also allows the Council to set additional conditions specific to the particular ISFSI proposed. This provides the regulatory basis for the safety requirements that OOE proposes in this report.

2. Federal Regulation

The U.S. Nuclear Regulatory Commission (NRC) is the primary regulatory agency for all nuclear plants in the U.S. Its regulatory responsibilities for nuclear power plants include: protecting public health and safety, protecting the environment, protecting and safeguarding nuclear materials and nuclear power plants in the interest of national security, and assuring conformity with antitrust laws. Agency functions are performed through standards setting and rulemaking; technical reviews and studies; conduct of public hearings; issuance of authorizations, permits, and licenses; inspection, investigation, and enforcement; evaluation of operating experience; and confirmatory research.[7] These responsibilities include the independent storage of spent nuclear fuel and high level radioactive waste.

The specific rules that govern the construction and operation of an ISFSI are contained in 10 CFR 72 or Part 72. In accordance with these rules, the NRC requires that an application for a license be submitted by the utility and approved by the NRC. This application is required to contain a Safety Analysis Report or SAR, which describes the proposed ISFSI "for the receipt, handling, packaging, and storage of spent fuel or high-level waste, including how the ISFSI ... will be operated."[8] Part 72 also provides the requirements for the content of the application, including the SAR; the issuance and conditions of the license; requirements for records, reports, inspections, and enforcement activities; siting evaluation factors; design criteria; quality assurance practices; security measures; and the training and certification of personnel.

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[7] 10CFR1.11(b)

[8] 10CFR72.24

The NRC approval process begins with the licensee's submittal of an application and SAR. The Commission staff reviews the SAR to determine whether the ISFSI design can be fabricated and used without endangering the health and safety of the public. The SAR is the principal document in which the applicant provides the information needed to understand the basis upon which a determination can be reached; however, the NRC may (and usually does) submit "requests for additional information" (RAIs).

At the completion of the review process, the NRC issues a Safety Evaluation Report (SER). This document describes the review process; which matters were modified by the applicant, require additional information, will be resolved in the future, or remain unresolved; where the design deviates from the standard criteria; and the bases for any deviations or exemptions from the regulations. Once it is determined that the proposed ISFSI complies with the regulations, then the NRC will issue a license under the provisions of 10 CFR 72. After issuing the license, the NRC continues to enforce its regulations through its inspection program.

PGE filed its application for a spent fuel storage license under 10 CFR 72 on March 26, 1996. The NRC issued public notice in the Federal Register on April 25, 1996,[9] including notice of the public comment period. The deadline to intervene in the NRC licensing process was May 28, 1996. No one filed to intervene.

3. Coordination of the NRC and OOE Review Efforts

Oregon statute explicitly provides for OOE oversight of spent fuel storage at Trojan in cooperation with the NRC. The specific requirement is contained in ORS 469.594(3) which states:

"*The Office of Energy and the operators of nuclear-fueled thermal plants shall pursue agreements with the United States Office of Energy and the United States Nuclear Regulatory Commission to fulfill the provisions of this section.*"

OOE and NRC coordinate their Trojan nuclear safety regulatory programs in accordance with a Memorandum of Understanding,[10] which focuses principally on the separate inspection programs at Trojan carried out by the two organizations. Furthermore, Oregon regulations on the Trojan decommissioning direct that OOE "to the extent practicable, coordinate its technical review with that of the U.S. Nuclear Regulatory Commission."[11] The Oregon rule on the storage of spent nuclear fuel also states that part of its purpose is "to cooperate with the federal government in accordance with Oregon's siting policy ... to ensure the safety of interim onsite storage."[12]

OOE and NRC work toward separate conclusions concerning the proposed ISFSI. In some cases, OOE may recommend requirements for the ISFSI that go beyond those imposed the NRC. However, OOE cannot recommend anything that might create a conflict with NRC requirements.

Cooperation between OOE and the NRC is regularly demonstrated through meetings at the Trojan plant, NRC headquarters, and in telephone conferences. There are also frequent interactions at the plant site between OOE inspectors and NRC inspectors. OOE often has a representative at NRC meetings regarding Trojan, and the NRC occasionally sends a representative to OOE and EFSC meetings regarding Trojan. Additionally, information is regularly exchanged between the two organizations. For

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[9]Federal Register, Volume 61, No. 81, p. 18448

[10]<u>Memorandum of Understanding between the NRC and the State of Oregon</u>, January 1980

[11]OAR 345-26-370(1)(c)

[12]OAR 345-26-390(1)(b)

the ISFSI review, the RAIs submitted by both organizations and the licensee's responses to those RAIs have been exchanged.

Cooperation during the ISFSI review was also demonstrated in May 1996 when the NRC invited OOE's site inspector to visit Arkansas's ANO plant with an NRC inspection team to witness a mock loading of a dry spent fuel storage basket that is similar to the ones planned for use at Trojan. During this visit, the inspector examined elements of the licensee's operational procedures, ALARA (As Low As Reasonably Achievable) radiation program, QA practices, and training program.

There is necessarily some overlap between the two review efforts. Both the NRC and OOE have performed an extensive review of the SAR and its supporting documents, therefore examining much of the same material. However, NRC regulations are highly detailed in terms of design requirements and technical content, while the Oregon rules are more broadly written and focus more on concerns specific to this region (for example, seismic concerns).

One difference between the two review efforts is that the NRC has more extensive resources to evaluate complex computer software. For example, the evaluation of the cask system's ability to maintain the spent nuclear fuel subcritical under all credible conditions requires the use of such software. For this aspect of the ISFSI design, the NRC provided the expertise in nuclear criticality computer codes. The OOE staff discussed the criticality calculations with the NRC to verify their intention and method in performing this criticality evaluation. Based on these discussions and on the NRC review, OOE concluded that PGE used approved computer codes and applied them properly.

Conversely, OOE was able to observe PGE's programs and procedures firsthand, both through its resident inspector and through frequent site visits by its consultant. Because of its on site presence, OOE was able to interview PGE's and the engineers of their principal vendor directly.

In summary, the NRC reviewed PGE's SAR largely by running independent calculations using their own computer codes, and making sure their results were consistent with PGE's. OOE reviewed the SAR largely by ensuring that PGE used appropriate and conservative assumptions and calculation methods, by direct interviews with PGE staff and observation of PGE construction and pre-operational preparations. This dual review by OOE and NRC therefore resulted in a review that examined PGE's proposed ISFSI from different perspectives.

D. OOE's REVIEW PROGRAM

1. Criteria and Methodology

To approve the proposed ISFSI, the Council must find it meets the criteria set forth in OAR 345-26-390. These criteria are intentionally broad in scope. They specify that the ISFSI must be approved by the Council, and they identify key parameters related to the design and operation of the ISFSI, including the design's ability to maintain discharges and radiation exposures within limits, and to contain the spent nuclear fuel in the event of an accident. To adequately address these criteria, the OOE staff reviewed the SAR and supporting documents to a depth sufficient to provide the Council with an adequate basis to approve the proposed ISFSI. For example, in order to verify that projected off-site radiation doses did not exceed limits in the event of an accident, we had to go beyond the accident analyses in the SAR to verify that the structures themselves were built to industry standards.

The OOE review included the SAR and dozens of supporting documents. These supporting documents included: industry codes and standards; selected NRC and vendor documents; LCA-237 on fuel transfer activities accident analysis; PGE draft procedures, including those for loading, emergencies, and testing; the ISFSI Technical Specifications; the PGE QA program, including selected QA audit and surveillance reports; and various calculations in support of the SAR.

The calculations reviewed by OOE fell into four major categories: heat removal (thermal analyses), structural analyses, radiation dose projections, and accident analyses. Our review of these calculations, whether by PGE or SNC, confirmed that the assumptions were conservative, that appropriate calculational methods were used, and that they were reviewed and approved according to PGE's QA Program. For calculations involving the use of large computer codes, such as ANSYS, OOE verified that the computer codes were NRC approved. For calculations used in support of statements in the SAR, OOE verified that the SAR accurately reflected the calculation, its assumptions, inputs, and results.

It was not necessary or practical to review every calculation. OOE selected a representative sample of calculations of key parameters and of the potential consequences of accidents or natural events. We based our selection on the calculations' relevance to EFSC Standards and their importance to safety, and on the degree to which commitments or statements in the SAR were based on calculation results. For all calculations, we asked the following types of questions:

(1) Verification

Was the calculation verified by PGE engineering?

Was it appropriately reviewed and approved? If performed by a vendor, did the vendor perform an appropriate review?

Was the calculation verified or audited by PGE QA/QC personnel?

(2) Assumptions

Are the assumptions appropriate and reasonable?

Are the assumptions conservative?

Do the assumptions agree with those used in accepted methodologies?

Are appropriate limits (structural, thermal, or other) used?

(3) Analytical Methods

Have the equations been used previously in accepted methodologies?

Were industry standards used or referenced?

Does a units check give the appropriate results?

Was input data appropriately entered?

Is the calculation mathematically accurate?

Would an alternate method yield similar results?

Do the results seem reasonable?

(4) Computer Codes

Are the computer codes used widely in the nuclear industry?

Are the computer codes accepted by the NRC?

Will the NRC verify this code for this application?

Was the computer code validated or benchmarked appropriately?

Since OOE does not have the resources to validate computer software codes, calculations involving such codes were reviewed primarily to ensure assumptions were conservative, inputs to the codes were reasonable and accurate, and the results were reasonable. These calculations included the criticality analysis (Section IV.D) and portions of the thermal analyses (Section IV.B). We also confirmed that such calculations were evaluated in depth by the NRC and their principal engineering consultant. In addition, OOE verified that PGE QA had audited and found satisfactory BNFL/SNC's validation and benchmarking program for the computer codes used in the ISFSI calculations. (See a detailed discussion of computer software validation in Section IV.B.3 of this report.)

2. Requests for Additional Information (RAIs)

OOE issued five formal RAIs to PGE. We based many of our conclusions on PGE's written responses, and also on less formal responses given in interviews with Trojan personnel. We held several face-to-face meetings and conference telephone calls with NRC representatives to resolve issues and coordinate review efforts. We also reviewed the NRC's RAIs and PGE's written responses to them.

3. Site Reviews and Inspection Program

OOE conducted numerous site visits and interviews with PGE representatives in addition to the routine inspection work conducted by the OOE site inspector. We observed the construction of the ISFSI storage pad, inspected the areas of the Fuel Building where fuel loading operations will occur, witnessed the fit-up testing of the prototype basket and components, witnessed test operations of the Vacuum Drying System, and observed various training sessions of ISFSI personnel.

In addition to our reviews at the Trojan site, the OOE site inspector visited Arkansas' ANO plant at the invitation of the NRC to witness a mock loading of a dry spent fuel storage basket similar to the ones planned for use at Trojan. We also twice visited the SNC headquarters[13] to interview SNC personnel directly, review quality records, and to observe PGE QA surveillances. In January 1999 we will visit the major fabricator of the PWR baskets, ABB Combustion Engineering Nuclear Systems.

The SAR, supporting documents, and responses to RAI questions describe procedures that will direct the loading process, ISFSI operations, and emergency responses in the event of accidents. These licensee documents also commit to various actions in preparation for or during ISFSI operations. PGE uses an administrative procedure to track these commitments to ensure their completion. OOE conclusions regarding safety and compliance with EFSC standards are in part based on PGE's written commitments to implement procedures and/or administrative controls. OOE verification of these commitments are met as part of its normal inspection program.

4. Verification

The term "verify" appears frequently in this report. The American Heritage Dictionary defines "verify" as "to determine or test the truth or accuracy of, as by comparison, investigation, or reference." OOE's use of the term is consistent with this definition.

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[13]In 1998, SNC became BNFL Fuel Solutions Corporation, also referred to in this report as BNFL/SNC. Depending on context, SNC and BNFL/SNC are used interchangeably in this report.

OOE verified statements, calculations, and assumptions made by PGE in a variety of ways. Where practical, we verified PGE calculations through direct review. For large computer calculations, we verified that the computation method was valid by using any of the review techniques described in Section II.D.1 of this report. In certain cases, we verified that inputs and assumptions were appropriate by going over them with the PGE engineer responsible for the calculation. We verified assumptions by comparing them with known data about the plant or the ISFSI design, or by comparison with known data from other plants. In cases where PGE assurances of safety were based on the existence of approved written procedures, we verified these assurances by reviewing the procedures. In cases where calculations were represented as being "NRC approved," we verified this by directly contacting the NRC. In cases where PGE statements and assumptions were based on the recommendations of national laboratories, such as Pacific Northwest Laboratories or Lawrence Livermore National Laboratories, we verified these statements and assumptions by reviewing the documentation from those laboratories and comparing them to the representations made by PGE.

In summary, the term "verify" refers to any or all of the review techniques used to ensure that PGE representations in the SAR were true.

5. Outline and Organization of the Report

The essential elements of this report consist of the following: (1) an introduction to PGE's proposal to establish an ISFSI along with background information on Trojan's decommissioning, spent fuel storage, regulatory bases, and the OOE review; (2) a description of the ISFSI and its associated SAR, (3) OOE's evaluation of PGE's analyses, including thermal, structural, radiation protection, criticality, and accident; (4) an evaluation of the fuel loading operations; (5) an assessment of PGE's compliance with the OARs, and (6) OOE's conclusions and recommendations.

For each of the principal areas that we reviewed, the following sections of this report attempt to respond to four basic questions: (1) What did PGE state in their SAR and supporting documentation? (2) What aspects of the these documents did we review? (3) How did we verify what we reviewed? And, (4) what are OOE's conclusions?

III. DESCRIPTION OF THE PROPOSED ISFSI

A. ISFSI SITE

PGE plans to locate the ISFSI on the site of the Trojan Nuclear Plant, approximately 100 meters north of the existing Fuel Building and 100 meters east of the Trojan Central Building. This is an appropriate location because it limits the likelihood that workers associated with Trojan's decommissioning activities might enter the ISFSI area or receive significant occupational radiation exposure from it. The relatively short distance between the Fuel Building and the ISFSI site minimizes the need for spent fuel transportation. Fuel casks can be loaded in the Fuel Building and moved to the ISFSI site without the need for motorized transportation, other than an air pad system and modified fork lift truck. The Trojan site was originally chosen for its seismic stability. The ISFSI site is located on essentially the same basalt outcropping as the nuclear plant, which makes the site highly stable in the event of the maximum credible earthquake.

The ISFSI pad has been constructed and is located in the northeast corner of what is now the Trojan Industrial Area, as described in the Trojan Decommissioning Plan. The pad measures about 105 feet by 170 feet and is designed to accommodate up to 36 storage casks, the Transfer Station, and a small storage building. (See Figure1in Appendix A of this report.) The ISFSI pad is surrounded on two sides, north and east, by a berm of earth, more than twenty feet high, that was excavated from the original terrain at the time of site construction. The berm provides natural radiological shielding for persons north and east of the ISFSI. The areas to the south and west are part of the Trojan site and are under PGE control.

There are several important boundaries associated with the ISFSI site, which serve a variety of functions. These are shown on Figure1in Appendix A, reprinted from the SAR. They include:

<u>ISFSI Protected Area</u>**.** This area is designed for security purposes. The Protected Area immediately surrounds the ISFSI and is the area in which federally licensed activities under 10 CFR 72 will occur. This boundary includes a Protected Area Fence that is 25 feet outside the ISFSI pad and precludes entry by unauthorized persons in accordance with the security requirements of 10 CFR 73.55.

<u>ISFSI Controlled Access Area</u>**.** This area encompasses the Protected Area. The ISFSI Controlled Access Fence (also known as the Perimeter Fence) is 25 feet outside the Protected Area Fence and is used to control access to the ISFSI. This area will also tentatively coincide with the Radiologically Restricted Area, a radiological term, until after the spent fuel is moved into the ISFSI and actual dose measurements are recorded. The Restricted Area is defined as the area in which the radiation dose rate is equal to or greater than 2 millirem per hour (mrem/hr), such that a person (occupational radiation worker) could receive an annual dose equal to or in excess of 100 mrem. For practical reasons, the dose at this boundary is likely to be less. Persons authorized for entry to this area must be trained in radiation protection and have dose monitoring equipment.

<u>Controlled Area.</u> This area is within 325 meters from the edges of the ISFSI pad. This distance was selected by PGE and is the distance at which federal requirements at 10 CFR 72.106 require that exposure to the public in the event of a design basis accident be no more than 5 Rem to the whole body or any organ. (ISFSI SAR 8.2.1 and 10 CFR 72.106). There will be no physical barrier at this distance, but PGE must have the ability to exclude people from the Controlled Area in an emergency. Calculations show that radiation dose from the worst case credible accident would in fact be much less at this distance.

Industrial Area**.** This is the area outside the ISFSI Controlled Access Area in which activities are still covered by the 10 CFR 50 license and access is controlled. This term is for security purposes. It may be eliminated upon termination of the 10 CFR 50 license.

PGE's property boundary includes the entire 643 acre tract owned by PGE. The boundary of this property extends to the Columbia River on the east, across Highway 30 to the west, includes the forested area to the north of the plant, and the public picnic grounds and recreational lake to the south of the plant. This property is defined as the Controlled Area for emergency purposes in the Trojan Nuclear Plant Decommissioning Plan, and will remain the Controlled Area boundary for decommissioning purposes until the activities described in the Decommissioning Plan are complete and the 10 CFR 50 license is terminated. There is no fence at this boundary, but PGE has the ability to exclude unauthorized persons from this area in an emergency. This boundary is not part of the SAR and applies only to decommissioning activities.

B. THE VENDOR - SIERRA NUCLEAR CORPORATION

1. Vendor Selection

PGE selected a spent fuel storage system designed and built by Sierra Nuclear Corporation (SNC) (now BNFL Fuel Solutions Corporation or BNFL/SNC). BNFL/SNC performed most (but not all) of the calculations regarding structural integrity, heat removal, nuclear criticality, radiation, and accident analysis. BNFL/SNC is the prime fabrication contractor, as well, coordinating many of the manufacturers involved in building the ISFSI facility.

The design PGE selected is called the TranStor™ system. This system is designed for both storage and shipping. The TranStor™ design is essentially a modified version of SNC's VSC-24 design. The two designs are not identical but they are very similar, and many of the design calculations for the TranStor™ are modified from previous VSC-24 calculations. The similarities are important because OOE's conclusions regarding safety are based partly on VSC-24 fuel storage systems that are used elsewhere in the country.

2. SNC Experience in Other States

VSC-24 fuel storage systems are currently in use at the following operating nuclear plants: Consumers Power's Palisades plant in Michigan; Wisconsin Electric Power's Point Beach plant; and Arkansas' ANO plant.

OOE reviewed radiation dose measurements at the Palisades plant in Michigan and the ANO plant in Arkansas as part of our review of radiation dose calculations. OOE also reviewed welding concerns at Arkansas and at the Point Beach plant in Wisconsin to ensure those concerns were properly addressed at Trojan. OOE reviewed safety concerns raised by the State of Michigan, which intervened in the NRC licensing process for the ISFSI at the Palisades plant. That case was resolved in the NRC's favor. However, OOE considered the safety issues raised by the State of Michigan in its review of the PGE plan.

OOE also verified that PGE took steps to learn from the experience of previous SNC cask users. PGE has implemented a "lessons learned" program in which problems at other plants are reviewed by PGE personnel to ensure that specific problems are addressed at Trojan.

3. Recent Concerns Regarding SNC Performance in Other States

During the course of this review, two significant events at other plants caused the NRC to question the adequacy of SNC's QA program. In May 1996, there was a hydrogen explosion during welding operations while sealing an SNC VSC-24 cask at Point Beach. Hydrogen was released by the interaction of boric acid in the SFP water with the zinc coating on the SNC fuel basket. Boric acid is used to ensure that nuclear fission is impossible in a SFP. The zinc coating on the SNC fuel basket is applied to prevent corrosion. These two substances interacted to release hydrogen, which was then ignited during the welding of the shield lid. This event resulted in an extensive investigation by the NRC.

In April 1997 at the Arkansas ANO plant, an 18 inch crack was found in the weld used to seal the lid of an SNC basket. As with the hydrogen explosion at Point Beach, this brought an immediate halt to all fuel loading operations and resulted in an extensive investigation.

Both events could have been prevented by better welding procedures and practices at those plants. OOE has confirmed that PGE's welding engineers have reviewed the events at both plants in order to incorporate the lessons learned into PGE procedures.

These events prompted an expanded NRC investigation into SNC's fabrication controls, which led to them finding further discrepancies in the implementation of their QA program. As a result, the NRC on October 6 ,1997, issued to SNC a Demand for Information (DFI) letter. In this DFI, the NRC demanded that SNC provide reasons why the NRC should not immediately suspend all fabrication activities of SNC VSC-24 casks, as well as all review activities concerning applications for VSC-24 casks.

Although the DFI was directed at SNC, PGE provided an independent response. In this response, PGE pointed out the constant involvement of its own Nuclear Oversight Department in any QA related activity concerning the spent fuel casks. PGE also described its "lessons learned" program, in which PGE staff reviews all events and problems at other SNC client facilities in order to ensure that PGE procedures would prevent those problems from occurring at Trojan. Based on PGE's response, the NRC agreed to treat PGE as separate from Point Beach and ANO.

OOE reviewed the events at Point Beach and ANO, as well as the NRC DFI to SNC and PGE's response. We observed that PGE took several steps to ensure that the QA program was effectively implemented. These steps included the stationing of a full-time PGE QA representative at the vendor's engineering facility, implementing the lessons learned program mentioned above, sending PGE welding engineers to the subcontractor's fabrication facility to ensure that welding procedures met PGE specifications, and having a prototype spent fuel cask built in order to work out any procedure problems out in advance. The prototype basket will not be used for the actual storage of spent fuel.

On August 21, 1998, the NRC issued a Closure of Demand for Information letter with the finding that SNC’s corrective action program had satisfactorily resolved all of the identified problems.

In summary, OOE observed that PGE learned from SNC's previous experience and has been aggressive in its QA oversight of SNC and their subcontractors. For these reasons, we conclude that PGE has taken appropriate preventive actions regarding the events at Point Beach and ANO.

C. STORAGE COMPONENTS

The principal storage components of the ISFSI include the baskets, transfer cask, concrete casks, failed fuel cans, and the fuel debris process can capsules. There are many other smaller components, but these are the major ones. A description of each of these principal components follows.

Basket or PWR Basket. This is not a basket in the normal sense of the word but a cylindrical stainless steel canister, about 15.1 feet tall and 66 inches in diameter. The steel shell is 0.75 inches thick, and the basket weighs approximately 75,300 pounds when loaded with dry spent fuel. There will be 34 baskets for storing spent fuel.

Inside the basket is an internal sleeve assembly with capacity for 24 fuel assemblies. The fuel racks are carbon steel with Boral plates attached. Boral is a material that prevents nuclear fission by absorbing neutrons. The Boral plates provide additional safety margin against criticality. The internal carbon steel storage cells are coated with an inorganic, radiation resistant, high temperature coating to provide corrosion protection and to promote radiant heat dissipation from the spent fuel to the basket shell. After being loaded with fuel, dried, and sealed, the basket will be filled with helium at atmospheric pressure. This helps to prevent corrosion and also provides good heat transfer from the spent fuel to the basket shell.

Each basket will have two lids, both approximately 64 inches in diameter. The inner lid is a shield lid. It is 8 inches thick and made of a combination of carbon and stainless steel. There are two small penetrations in the shield lid for venting and draining during fuel loading operations, and for filling the basket with helium once the fuel is loaded. The lid will be welded to the basket shell and pressure tested to ensure an airtight seal.

The outer lid is the structural lid. It is 3 inches thick, fabricated from stainless steel, and also welded to the basket shell. A hole through the lid provides access to the drain and vent connections. After the fuel is loaded and the basket is filled with helium, this hole is filled with two stainless steel plugs and welded. Eight eye bolts will be threaded into the top of the structural lid to allow PGE to pick up the basket once the lids are in place.

Once loaded, the basket will be placed in a concrete cask made of reinforced concrete. The cask is approximately 11.3 feet in diameter, 17.6 feet high, with walls 29 inches thick. The concrete cask interior has a 2 inch thick carbon steel liner. There is a 4 inch air space between the inner wall of the concrete cask and basket outer wall. Vents at the top and bottom of the cask permit natural air circulation, which is the passive method of decay heat removal. The cask weighs approximately 290,000 pounds when loaded with a sealed basket and spent fuel. Weather protection is provided by a carbon steel lid.

The transfer cask is used to transfer a basket between the SFP and a concrete cask. It provides shielding for plant personnel during fuel loading operations. The transfer cask has hydraulically-operated, retractable doors at the bottom to permit lowering of a loaded basket into a concrete cask. The transfer cask serves as a special lifting device which meets the requirements in NUREG-0612 (1980) and ANSI 14.6 (1993) for movement of a basket.

In the event that a basket is found to be breached or damaged once it is on the ISFSI pad, an overpack will be used. This is a cylindrical shell of carbon steel that fits around the basket and provides a leak tight confinement space. It provides the same containment and confinement functions as a basket in the unlikely event that a basket develops a leak. It also provides adequate heat transfer so that the fuel clad temperature does not exceed design limits.

Fuel debris process cans are sealed containers, approximately 29 inches long, that hold loose fuel debris which is not confined by cladding. Five of these cans fit into one fuel debris process can capsule, which is approximately 13 feet long. PGE has 41 fuel debris process cans currently stored in the SFP. These will go into eight fuel debris process can capsules. Each of these will go into a failed fuel can that will be located in one of the four oversized corner locations of a basket. The remaining fuel debris process can will go into a failed fuel can with eight fuel assembly bottom nozzles remaining from a modification that occurred during reactor operations and a spacer to prevent movement during eventual transport to a federal fuel repository.

Failed fuel cans are approximately 13 feet long and are designed to provide an extra barrier for fuel assemblies with failed cladding and for fuel debris process can capsules. These will be stored in one of the four oversized corner locations in a basket. PGE plans to use 20 failed fuel cans.

The transfer station is located on the ISFSI pad just west of the storage casks and will be used to transfer: (1) a loaded basket form one concrete cask to another in the event that the original concrete cask has been damaged, (2) a loaded basket from a concrete cask to a shipping container for transfer to a federally licensed repository, and (3) a leaking or damaged basket to an overpack. The transfer station will be the only method of transferring fuel from one concrete cask to another once the SFP is decommissioned.

Not used in storage, but important to the loading operation and at the transfer station are several impact limiters. These are rigid polyurethane foam blocks positioned to absorb energy from a falling object. There are several of these used during fuel loading on the 93 foot elevation of the Fuel Building, one on the ground floor of the Fuel Building under the hoistway, and one in the pad at the transfer station.

D. THE LOADING PROCESS

The activities inside the Fuel Building are controlled by PGE's existing NRC license under 10 CFR 50. Therefore, they are not described in the SAR but in a separate PGE document: License Change Application (LCA)-237. OOE reviewed LCA-237 in the same level of detail as used to review the SAR. Our review of LCA-237 is described in Section V of this report. However, the following is a brief summary of the loading process.

An empty basket is placed in the transfer cask and lowered into the Cask Loading Pit, adjacent to the SFP. The Cask Loading Pit is filled with water, the gates opened between the Cask Loading Pit and SFP, and the spent fuel is loaded into the basket. The shield lid is lowered onto the basket. The transfer cask and loaded basket are lifted to the 93 foot level. The transfer cask and basket exterior are cleaned and the shield lid is welded onto the basket. The basket is then hydrostatically tested with 15 psig of water and pressure tested with 15 psig of helium. The structural lid is welded on top of the shield lid. The basket is then drained, vacuum dried, and filled with helium at atmospheric pressure. The transfer cask and sealed basket are lifted and carried to the hoistway where they are lowered to a concrete cask on the Fuel Building ground floor. The doors of the transfer cask are opened, and the basket is lowered into the concrete cask.

Once the loaded basket is in the concrete cask, the lid is placed on the cask, and the loaded cask is moved to the ISFSI pad. This is accomplished with air pads under the loaded concrete cask which lift the cask and enable it to be pushed to the ISFSI pad with a modified fork lift truck. Once on the ISFSI pad, the air pads are removed, the inlet screens are installed, and the process is complete.

A more detailed description of the ISFSI Loading process and the associated accident analysis is contained in Section V.B of this report.

E. RADIATION PROTECTION PROGRAM

PGE relies on various programs and procedures to ensure public health and safety. Some of the programs are carried over from the Trojan Nuclear Plant; others are written specifically for the ISFSI.

The Radiation Protection (RP) Program for the ISFSI is based on PGE's existing program for the Trojan Nuclear Plant. The program must meet the same requirements of 10 CFR 20, which governs radiation exposure to workers and the public, that the operating plant had to meet. In particular, PGE must have a program to maintain personnel exposures As Low As Reasonably Achievable (ALARA), which is defined at 10 CFR 20.1003 and in the glossary, Appendix B of this report. The program uses existing procedures, many with little or no change. The radiation worker training is essentially the same training that was required when the plant was operating.

OOE evaluated PGE's RP program by reviewing procedures, directly observing plant activities to verify continued good radiation protection practices in use by PGE staff, attending the radiation worker training, reviewing exposure records from ongoing decommissioning activities, reviewing NRC inspection reports, and working with NRC inspectors.

When the ISFSI is in operation, worker dose will be limited to the few PGE personnel who perform maintenance and surveillance activities. These activities are described in the SAR, and PGE has taken appropriate steps to maintain worker exposure ALARA, such as minimizing the time workers must spend close to the casks and using remote temperature monitoring instrumentation. The largest uncertainty in personnel exposure will come during fuel loading activities. These activities are allowed under current EFSC rules and are governed by PGE's existing NRC license under 10 CFR 50. OOE observed fuel handling activities during PGE's fuel inspection program in February 1997. At that time, fuel handling personnel followed PGE RP procedures, and PGE RP technicians were constantly present.

PGE has procedures in place to ensure that personnel exposures are ALARA. The procedures require a review of all jobs in radiologically controlled areas. The RP reviewer answers specific questions concerning radiation exposure and contamination potential for every activity. The reviewer specifies a level of protection based on the exposure potential. RP has the authority to specify critical hold points where an RP technician must give his or her approval before a job can proceed. The RP technician has the authority to stop work if necessary. The procedure also provides for a mid-job assessment, so that changes in protective measures can be made.

Some techniques used to reduce personnel exposures include pre-job briefings, mockup training for complicated jobs, portable shielding in high dose areas, and use of mechanical aids (such as robotic welding) where practical. To date, PGE has used all these techniques in decommissioning. The pre-job ALARA briefing is required by procedure, mockup training for fuel loading will be required by the NRC, and the use of portable shielding and other mechanical means to reduce exposure is specified by the RP Department in cooperation with the various working groups.

PGE reports personnel exposure to EFSC quarterly. To date, the reports show actual doses less than projected doses for decommissioning. OOE also reviews NRC inspection reports and maintains a working relationship with the NRC Radiation Protection inspectors who visit the site. NRC inspection reports since the plant shutdown indicate that PGE has complied with 10 CFR 20.

In conclusion, OOE has evaluated the written RP Program at Trojan and has extensively monitored their performance. The written program meets all of the regulatory requirements, and is essentially the same program that was in place when the plant was operating. Implementation of the program on the decommissioning activities thus far have exceeded expectations in reducing worker dose. OOE is confident that the existing RP Program will be effective in maintaining ALARA principles during the loading and storage activities of the ISFSI.

F. MONITORING AND SURVEILLANCE

1. Temperature Monitoring

Section 5 of the SAR describes the ISFSI instrumentation. Since the ISFSI is passive by design and requires no instrumentation to operate, only monitoring instrumentation is required.

As required by NRC regulations at 10 CFR 72.26 and 10 CFR 72.44, PGE has proposed Technical Specifications for ISFSI operation. Section 5.5 of the proposed Technical Specifications includes a program for temperature monitoring. The specification includes measurements of ambient temperature and air outlet temperature, and requires further action if any air outlet temperature reaches 195 ºF. We requested information describing the temperature monitoring devices that would be used to meet this requirement. PGE responded that "the air outlet temperature monitoring system consists of 4 resistance temperature detectors (RTDs), one of which will be located in each of the 4 air outlets. The RTDs are connected to a junction box that will be mounted about 5 feet from the bottom of the concrete cask."[14]

Since there are four installed devices per cask, we asked PGE what would be the allowable difference in readings between the four devices. PGE stated that :

> "Any difference in temperature readings between the four devices will be addressed in the surveillance procedure. When a technical specification operating limit is established, a surveillance limit for the recorded readings will be conservatively set such that there is enough margin between the two limits to include the instrument uncertainty."[15]

OOE recommended that a trending program be established that could indicate either a problem with a cask or a faulty RTD. Experience at the Palisades plant in Michigan shows that measured temperatures can very greatly between the four air outlets due to wind and weather conditions. PGE agreed to an accelerated temperature monitoring program when any concrete cask outlet temperature reaches 180 °F, 15 °F below the Technical Specification limit of 195 °F. Further, the ISFSI Manager will periodically review the temperature data and will compare any unusual readings to baseline or predicted values and take appropriate corrective actions. OOE concurs with these measures. However, OOE has seen no details regarding this review program, nor is a program of data review described in the SAR or other correspondence. Therefore OOE recommends a rule codifying this PGE commitment.

PGE does not consider the temperature monitoring system as important-to-safety and has stated that they will be purchased as commercial grade items. We asked how they will ensure that these commercial grade devices will remain accurate over the 40 year expected life of the ISFSI. PGE responded that the devices will be calibrated in accordance with QA requirements. After further review of PGE's program, we could not verify that there are any QA requirements. However, PGE committed to periodic recalibration of the electronic components of the temperature monitoring devices. The sensing element of an RTD is not electronic and cannot be readily be calibrated in the field. Therefore PGE agreed to trend the outlet temperatures to detect any RTD whose output is consistently less conservative than other RTDs. OOE considers this adequate for three reasons:

- The sensing element of an RTD does not rely on active electronics and is not prone to degradation because it is constructed of highly corrosion resistant materials.

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[14]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-032-96, question 116, June 27, 1996

[15]ibid., question 21

- There is considerable margin between the expected temperature readings and the temperatures assumed in the safety analysis.
- A review of temperature trends by the ISFSI manager would detect an instrument problem long before there was an actual temperature problem.

Because OOE has not seen the details of the trending or calibration programs, we recommend a rule requiring that these programs be available for inspection prior to fuel loading. Also, the SAR states that PGE will monitor temperatures every 12 hours. OOE agrees that this is an appropriate frequency, but recommends a rule requiring prior OOE concurrence if PGE wishes to reduce that frequency.

We asked PGE if remote temperature sensing was planned to reduce radiation exposure for the ISFSI Specialists, the personnel assigned to monitor the ISFSI. The SAR does not describe this capability, and PGE stated that the use of remote monitoring instrumentation was not planned. However, the ISFSI pad includes the capability for the ISFSI personnel to read cask temperatures by a remote readout system, which, if used, could reduce personnel dose, consistent with ALARA goals. OOE, therefore, recommends the ability for remote temperature be required by rule, with hand-held temperature measuring devices used as backup.

2. Visual Inspections

The SAR requires PGE to verify each air inlet and outlet wire mesh screen is intact and free of blockage once per 24 hours. OOE agrees that this is an appropriate frequency. Since this requirement does not appear in the Technical Specifications, OOE proposes a requirement for OOE notification if PGE reduces the scope or frequency of this examination. Due to 17.6 foot height of the concrete casks, we requested additional information to determine if adequate visual inspections of the air outlets were possible. PGE responded that the air outlets are covered with a stainless steel wire mesh, making it unlikely that foreign objects will be drawn into them; however, devices such as portable stairs or mirrors will be available to conduct positive visual inspections.[16]

3. Structural Inspection Program

The SAR describes a Structural Inspection Program that established periodic inspections of the concrete surface of the concrete casks and the ISFSI pad. The intent of this program is to ensure that the structural integrity of the concrete was maintained through the establishment of acceptance criteria, degradation evaluation methods, and repair instructions. This program is important to OOE's finding that the casks are designed to last 40 years, a requirement that exceeds NRC requirements. Therefore OOE recommends that PGE's commitment to implement a Structural Inspection Program procedure be added to the Council's rules.

4. Interior Surface Inspections

The NRC imposed a requirement on the VSC-24 design that licensees remove the concrete lid from each cask every five years to inspect for degradation. The apparent reason for this is that the VSC-24 basket shells are fabricated out of carbon steel which corrodes and could affect heat transfer performance. PGE has taken the position with the NRC that this requirement is unnecessary for the TranStor™ design, because the basket shells are fabricated out of stainless steel, which is far less vulnerable to corrosion than carbon steel. PGE also asserts that degradation of the air flow path would be detected by the proposed surveillance requirements. PGE contends that a requirement for interior surface inspections

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[16] ibid., question 107

would increase "the potential risk of unnecessary exposure to those employees performing the inspection and the possibility for operational events such as basket drops."[17]

OOE concurs with PGE's argument that a reduced inspection requirement is sufficient to ensure structural integrity, and would reduce the radiation exposure to ISFSI workers. The program described in the SAR includes inspecting the interior of one cask every five years, with further actions if any degradation is noted. OOE finds this acceptable, and recommends that Council concurrence be required for changes in the Structural Inspection Program that would decrease the amount of inspection.

G. QUALITY ASSURANCE PROGRAM

1. PGE's Written QA Program

Section 11 of the SAR describes PGE’s QA Program and references PGE's topical report PGE-8010, Trojan Nuclear Plant Nuclear Quality Assurance Program,[18] This is identical to the program that was reviewed in OOE's Review of PGE's Decommissioning Plan for the Trojan Nuclear Plant, issued January 22, 1996. That review stated that:

> “The Quality Assurance and Quality Control programs appear effective in assessing plant performance through a comprehensive program of reviews, audits, surveillances, and involvement in plant staff meetings.”[19]

This conclusion also holds true for the proposed ISFSI. This section provides a brief overview of the QA program at Trojan and a description of OOE's review efforts.

The definition of quality assurance, as it is applied to nuclear plants, is provided in 10 CFR 50, Appendix B, for operating plants and those undergoing decommissioning, and in 10 CFR 72, Subpart G, for an ISFSI. In both references, the definition of quality assurance is identical. Quality assurance (QA) comprises “all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service." QA is further defined in the nuclear industry as "the systematic actions necessary to provide adequate confidence that ... work is performed according to plan.”[20]

At Trojan, the QA program is the primary responsibility of the Nuclear Oversight Department. As part of our review of the Decommissioning Plan, we interviewed members of that department, and we reviewed PGE-8010 and many of the sub-tier QA procedures and directives that implement the program. We concluded from our review that the written program was adequate. For the ISFSI, we compared the elements of PGE's QA program with 10 CFR 72, Subpart G, which contains provisions similar to 10 CFR 50, Appendix B, on QA organization, design control, procurement, procedures, material controls, special processes, tests, nonconformances, corrective actions, records, and audits. We also interviewed

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[17] PGE, letter from S. M. Quennoz to U.S.NRC, Trojan ISFSI (TAC No. L22102) Response to NRC Request for Additional Information, VPN-001-97, question 10-1(c)3, January 9, 1997

[18] Portland General Electric Company, PGE-8010, Trojan Nuclear Plant Nuclear Quality Assurance Program, Rev. 22, August 13, 1998

[19] Oregon Department of Energy, Review of PGE's Decommissioning Plan for the Trojan Nuclear Plant, January 22, 1996

[20] U.S. Nuclear Regulatory Commission, NUREG/CR-5884, PNL-8742, Revised Analyses of Decommissioning for the Reference Pressurized Water Reactor Power Station, October 1993, p.7.14

members of the Nuclear Oversight Department about the application of the QA program to the ISFSI. As a result, the OOE staff found that the written program satisfactorily described all of the required elements. Moreover, NRC approval of PGE's ISFSI project will include consideration by the NRC of PGE's QA program and its conformance to 10 CFR 72.

2. PGE's QA Audits and Surveillances

PGE provided a copy of the Nuclear Oversight Integrated Assessment Plan for the Independent Spent Fuel Storage Installation, Revision 2, December 11, 1996, which includes their 1995-1998 Integrated Audit/Surveillance Schedule. This is PGE's schedule of audits and surveillances for work involving the ISFSI. It includes both internal and external (also known as vendor or supplier) audits and surveillances.[21] The scheduled external audits and surveillances include several on SNC and one or more on the various fabricators of ISFSI components, including the transfer cask, basket, lifting yoke, and concrete cask steel liner. The scheduled internal audits and surveillances include all of the major ISFSI program activities, such as compliance to Technical Specifications and procedures; material audits; license document reviews; adequacy of calculations, work packages, and modifications; and receipt inspections.

To verify the implementation of the written QA program for the proposed ISFSI, OOE reviewed PGE QA internal audits and surveillances listed below, which cover various aspects of the ISFSI project. Although the adequacy of the actual surveillances or audits could not be assessed from the reports alone, these surveillances and audits provided verification that PGE's QA program addressed significant portions of the ISFSI project.

October 1995 surveillance report (95-031-SURV) by PGE QA on an inspection of spent fuel filters to provide information for the eventual disposal of these filters. This surveillance appeared comprehensive in its coverage of the filter inspection program. It included a review of applicable documents, verification of prerequisites identified in various SFP maintenance and engineering analysis activities, participation in radiation protection briefings, and observation of the inspection activities.

March 1996 surveillance report (96-006-SURV) by PGE QA on activities associated with the submittal of the ISFSI license application. The report indicated that documents required as part of the license application, including the ISFSI SAR, addressed all of the requirements.

July 1996 surveillance report (96-019-SURV) by PGE QA on four calculation packages for the ISFSI. These packages were reviewed against several Trojan procedures and supplier documents. The report concluded that "All PGE Calculation Packages contained necessary documentation, reviews appeared to be complete and thorough, and personnel performing the reviews were properly qualified."

November 1996 PGE QA audit report (DLN-077-96) of PGE's Engineering and Decommissioning activities as part of their commitment to audit all safety-related functions every two years. The scope of the audit included compliance to Technical Specifications, organizational structure, personnel training and qualifications, the Decommissioning Plan, calculations, corrective actions, and system modifications. The calculations reviewed included two in support of the ISFSI. Three observations were made by the PGE auditors that did not have significant consequences. The overall finding of the

[21] Audit and surveillance are defined in the glossary, Appendix B. Basically, an audit covers a wide range of programmatic requirements, and a surveillance is a limited scope audit involving verification or adherence to specific requirements.

audit was that the Engineering and Decommissioning activities had satisfactorily implemented Trojan's QA program.

OOE also reviewed PGE QA audit and surveillance records on the primary vendor, Sierra Nuclear Corporation (SNC) (now BNFL Fuel Solutions Corporation or BNFL/SNC). We also directly observed two PGE QA surveillances at the offices of SNC. The results of these reviews and observations are described in the following section.

3. PGE's Surveillance of Sierra Nuclear Corporation (SNC)

In 1998, SNC was bought by British Nuclear Fuels Limited (BNFL) and renamed BNFL Fuel Solutions Corporation, also referred to as BNFL/SNC. Since OOE's review covered both SNC and the new combined organization, both SNC and BNFL/SNC are used in this report depending on the context or when our review was conducted in relation to the merger.

At the beginning of OOE's review, we verified that SNC was on the PGE "Approved Suppliers List" for purchase of the TranStor™ system. We examined PGE's oversight of SNC, including the audits and surveillances described below. We also found that PGE approval was required for SNC's QA Plan for the project, all applicable SNC QA procedures, and all sub-tier suppliers performing quality-related work prior to the use of those suppliers. Additionally, in response to a question about QA/QC involvement with procedures, PGE responded that QA also reviewed sub-tier suppliers' procedures.[22]

To evaluate the adequacy of SNC's QA program, we reviewed the written program, which was included in Section 13 of their Safety Analysis Report for the VSC-24, dated October 1991. This program was reviewed and approved by the NRC in 1993. The OOE staff also concluded that the written program satisfactorily meets the requirements of 10 CFR 72, Subpart G. We reviewed SNC's Quality Assurance Plan for the Portland General Electric - Trojan - Dry Spent Fuel Storage Project, October 15, 1996. This plan provides a description of how the SNC QA program is applied to the Trojan ISFSI project. It references the various implementing SNC QA procedures and describes the interfacing relationships and contractual agreements between SNC and PGE. There were no regulatory requirements or guidance for this document, and OOE had no comments on its adequacy.

To evaluate the implementation of SNC's QA program, we reviewed PGE's QA audit and surveillance reports on SNC activities. Although the adequacy of the actual surveillances or audits could not be assessed from the reports alone, these surveillances and audits provided verification that PGE's QA program was addressing significant issues with their primary vendor.

May 1995 audit report (95-05) by PGE QA that qualified SNC to be on the Approved Suppliers List as a supplier of an ISFSI. This audit used a comprehensive Nuclear Utilities Procurement Issues Committee (NUPIC) checklist, which is the industry standard. The single "finding" of this audit appeared substantive. PGE QA found that corrective actions to previous 1994 and 1995 internal discrepancy reports had not been addressed in a timely manner. The audit observations, which are considered less significant than findings, were largely minor administrative issues, including a lack of documentation, lack of procedural compliance, and aspects of SNC vendor audits not adequately covered.

October 1995 surveillance report (95-18) of the SNC offices in Roswell, Georgia, that focused on verification of the implementation of their quality program, design control, software quality assurance, and SNC's QA Plan for the Trojan project. No findings or observations were reported.

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[22]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-032-96, question 129, June 27, 1996

December 1995 surveillance report (95-22) of SNC that addressed concerns with design input, the bases of certain calculation statements. the overall calculation process, and training. No findings and five observations were identified. The observations on calculations appeared substantive. Some calculations referred to others that were not completed. Some design input references were found to be in error. The bases for statements made in calculations were not always provided. In a letter to PGE dated January 24, 1996, the SNC Vice President of Quality Assurance/Control responded to each of the observations, and PGE QA found this response acceptable.

In addition to reviewing PGE's audit and surveillance reports, two members of OOE observed a surveillance performed by a joint PGE QA and engineering team at the head offices of SNC in Scotts Valley, California, in January 1997. Our primary purpose was to observe oversight activities by PGE auditors; however, we also made direct observations of the implementation of SNC's QA program and interviewed SNC representatives. Specifically, we reviewed SNC's QA policy and implementing procedures, nonconformance and corrective actions programs, internal and external audit programs, and procedures for calculations used in support of the PGE ISFSI, including their procedures for validating and benchmarking computer codes used in calculations.

As a result of this surveillance, we found that the implementing procedures of SNC's QA program meet the NRC requirements of 10 CFR 72, Subpart G. We found that SNC aggressively audits its fabrication subcontractors, but has not always followed up on the findings from those audits. In the case of one subcontractor, we found that SNC had identified numerous concerns, but we could not verify that all those concerns had been satisfactorily corrected. We also found that the trending program for SNC's corrective actions was ineffective. These observations were consistent with those of the PGE auditors. We observed that BNFL/SNC took corrective actions on all PGE QA findings. PGE recommended further that SNC take steps to ensure better compliance with its own procedures, and they planned continued surveillance of SNC throughout the project. To this end, PGE stationed a full-time representative at SNC offices to monitor activities related to the Trojan ISFSI project.

Our observations of PGE's conduct of this surveillance at SNC served also as a measure of the effectiveness of implementation of PGE's QA program. During the surveillance, PGE auditors followed up on findings from previous audits at SNC and reviewed calculations for compliance with SNC procedures. They also reviewed audits of SNC by other utilities and the NRC. PGE auditors showed an excellent working knowledge of SNC's QA program and procedures, and they covered the material in depth. OOE concluded that PGE is applying an appropriate level of oversight with SNC to ensure that the quality requirements of the proposed ISFSI are met.

4. Error in an SNC Calculation

In June 1997, due to OOE questions about inputs to a thermal calculation, SNC discovered an error in a thermal analysis resulting in the calculated maximum equilibrium temperature of fuel cladding in a vacuum being lower than it should have been. This temperature is a key parameter, but the greater concern was that this error had not been discovered by either SNC's or PGE's calculation review process. As a result, PGE issued a stop work on all engineering calculations by SNC and a Corrective Action Report, CAR 97-0013. SNC also issued their own CAR and an internal nonconformance report (NCR). Correction actions by both PGE and SNC were taken immediately to prevent further errors. OOE reviewed the CARs, the NCR, the associated documentation, PGE's investigation of their internal engineering review process, and the subsequent corrective actions taken by both PGE and SNC.

On July 9, 1997, OOE accompanied a PGE audit team to Sierra Nuclear Corporation (SNC) to interview SNC personnel and to review documentation, the calculation review process, various contributing factors, and the proposed corrective actions. OOE concluded that PGE conducted a thorough investigation of SNC's design and calculation review process, and that the proposed corrective actions were appropriate. The Stop Work Order remained in effect until PGE's Nuclear Oversight Department (QA) had verified that SNC had completed the required corrective actions. OOE also verified that the corrective actions had been satisfactorily completed.

Based on our own review of selected thermal calculations, the corrective actions taken in the review process for calculations both at SNC and PGE, and the NRC's independent review of all SNC calculations, OOE considers the thermal calculations, as revised and corrected, to be reliable and sufficiently conservative.

5. BNFL/SNC Vendor Audit of ABB Combustion

As mentioned previously, SNC was bought by British Nuclear Fuels Limited (BNFL) in 1998. In June 1998, at a meeting with PGE representatives, the new BNFL/SNC organization was described. BNFL/SNC currently has four people dedicated to QA plus contractors available for support as necessary. This appears to be sufficient to effectively implement their QA Program.

In November 1998, OOE reviewed BNFL/SNC external audit report (BNFL/SNC 98076), which was performed to qualify one of two fabricators of the TranStor™ baskets, ABB Combustion Engineering Nuclear Systems (ABB). This audit report covered all the elements of a QA program required by 10CFR72, Subpart G, and appeared to adequately evaluate ABB's QA program and their ability to fabricate the baskets. In January 1999, OOE inspected the ABB fabrication facility in New Hampshire to review the PGE and BNFL/SNC's oversight program, ABB's QA program, and various implementing procedures; evaluate ABB's closure of nonconformances; and observe the status of fabrication of the baskets. ABB's program appeared effective in meeting PGE's QA requirements.[23]

6. PGE QA Involvement with Fabricators

In response to a question about QA/QC involvement in the manufacturing process for the ISFSI components, PGE responded that source surveillance would "be performed during the complete fabrication process for the first basket and concrete cask. Additional periodic surveillances are planned during the fabrication of subsequent containers as well as the Transfer and Shipping Casks. The frequency of these surveillances will be dictated by the quality and complexity of the work performed. Total surveillance time is expected to be approximately 50% of the fabrication time."[24] OOE has verified that PGE QA has assigned a member of their staff to witness the fabrication of the baskets and related components. To date, this person has been assigned virtually full time between the two manufacturing facilities: Hi Tech in North Carolina and ABB in New Hampshire.

In addition, throughout our review, OOE periodically monitored PGE's QA Program to ensure that they verified the satisfactory resolution of all outstanding issues involving subcontractors that could affect the quality of the fabrication of the baskets and other components.

7. PGE QA Receipt Inspections

With the delivery of hardware to Trojan in November 1998, OOE observed PGE QA's receipt inspections of the prototype basket, shield lid, structural lid, transfer cask, and transfer cask yoke. Most

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[23] Jim Woessner letter to Adam Bless, OOE "ABB Audit Report", January 24, 1999

[24] ibid., question 131

of the technical reviews had been performed by a PGE QA representative prior to the components leaving the fabrication facilities, and the receipt inspections at the Trojan site were largely to ensure that damage had not occurred during shipment. We then reviewed the surveillance records performed at the fabrication facility (Hi Tech Manufacturing) to document PGE's verification that procurement of materials, fabrication, inspection, and assembly of several critical components were in accordance with purchase specifications.[25] These surveillance records also serve as QA receipt inspection records. The records indicated numerous problems in the manufacturing process; however, we verified that all had been satisfactorily resolved.

8. PGE QA Involvement in Procedures

In response to a question about QA/QC involvement in procedures and activities, PGE stated that:

> "certain procedures and work instructions for ISFSI construction, testing, and fuel loading will contain Circle Q Notification points and QC Inspection Hold Points. Additionally ... Nuclear Oversight plans to review fabrication and construction work packages in order to evaluate the inclusion of inspection hold points by the fabricator/constructor and SNC. Nuclear Oversight may also choose to include their own inspection hold points."

OOE verified these hold points in several of the test procedures to be performed prior to loading fuel, and also in the draft loading procedure.

9. Conclusion

OOE extensively monitored PGE's QA involvement in the ISFSI project. PGE's QA program meets all regulatory and procedural requirements. PGE's program of reviews, audits, surveillances, and other activities appears to go into sufficient depth to assure of satisfactory implementation of the QA program elements. When an error was found in a safety-related calculation, the Nuclear Oversight Department took strong action to resolve the problem and ensure that proper calculation reviews were performed. OOE also accompanied PGE QA inspectors on two vendor surveillances and observed excellent auditing practices. We believe PGE is applying effective oversight to their vendors to ensure that the quality requirements of the ISFSI are met. And, as on each of the other phases of decommissioning that OOE has reviewed — LCR, RVAIR, and other dismantling activities — there has been and continues to be a high level QA involvement in all critical onsite activities for the ISFSI project.

OOE reviewed the QA program of BNFL/SNC, PGE's primary contractor. Their QA program also appears adequate to ensure that the ISFSI is designed and will be constructed in compliance with EFSC rules. Our observations of BNFL/SNC quality activities were consistent with those of the PGE auditors. We observed that BNFL/SNC took corrective actions on all PGE QA findings. Further, PGE stationed a full-time representative at BNFL/SNC offices to monitor activities related to the ISFSI project. OOE also visited the fabrication facilities of ABB/Combustion-Engineering, and the QA oversight of this fabricator by PGE and BNFL/SNC appears thorough.

Overall, the quality assurance efforts for the Trojan ISFSI appear effective in providing adequate oversight of the fabrication, construction, loading, testing, and operational activities. We conclude that these efforts have been and continue to be effective.

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[25] PGE, D. L. Nordstrom to D. Olson (Hi Tech Manufacturing), Surveillance Plan No. 98-26, December 9, 1998, and PGE, D. L. Nordstrom to D. Olson (Hi Tech Manufacturing), Surveillance Plan No. 97-42, December 9, 1998

IV. ANALYSIS OF THE ISFSI DESIGN

A. STRUCTURAL ANALYSIS

1. Review Criteria

The structural analysis for normal operation of Trojan's ISFSI design is described in Sections 3 (Principal Design Criteria) and 4 (Installation Design) of the SAR, and is reviewed here. The structural analysis for loading operations is described in LCA-237 and reviewed in Section V.C of this report. For normal ISFSI operations, OOE evaluated the following areas of concern, some of which are further evaluated in the thermal and accident analyses portions of this report:

Did PGE identify the structures, systems, and components that are important-to-safety?

Did PGE identify principal design criteria and design bases for the important-to-safety structures, systems, and components?

Do the design criteria specify that the ISFSI components be able to satisfactorily handle the combined loads of normal, off-normal, and accident conditions?

Are the analytical tools used to determine stresses and stress combinations identified, and are these tools appropriately applied?

Will the cask system provide adequate confinement of the spent fuel and other radioactive material under normal, off-normal, and accident conditions?

Is the cask design capable of safely storing the spent fuel for 40 years?

2. Structures, Systems, and Components Considered Important-to-Safety

Section 3.4 of the SAR and NRC regulations at 10 CFR 72.3 define "Structures, Systems and Components Important-to-Safety" as those features of the ISFSI whose function is:

To maintain the conditions required to store spent fuel or high-level radioactive waste safely,

To prevent damage to the spent fuel or the high-level radioactive waste container during handling and storage, or

To provide reasonable assurance that spent fuel or high-level radioactive waste can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

The identification of important-to-safety components is significant because NRC regulations at 10 CFR 72.122 state that:

"Structures, systems and components important to safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed."

Section 3.3.3.1 of the SAR identifies the equipment and components that are classified as important-to-safety. These include the concrete cask, basket, overpack, fuel debris process can capsule, failed fuel can, transfer cask, transfer station, hoist rings, mobile crane, transfer station pad and impact limiter.

OOE requested additional information to determine whether the air pad system used to move the concrete casks to the ISFSI pad and the Fuel Building Crane used to move the transfer cask during loading should be considered important-to-safety. Both appeared to meet part of the definition. PGE responded that the air pad does not prevent damage to the spent fuel but is used to transport the casks to the ISFSI pad. A failure of the air pad would not result in a radiological release or damage to the spent fuel. We concur with this conclusion. (See Section IV.A.4.a for further discussion of the air pad system.)

PGE responded that the Fuel Building Crane was not subject to the ISFSI criteria but was subject to the controls of their 10 CFR 50 license and LCA-237. Although the crane is not formally classified as important to safety, we verified that there has been significant Nuclear Oversight Department (QA) involvement in crane maintenance and modification work. Also, visiting NRC inspectors have found PGE's crane maintenance program acceptable. Therefore OOE concludes that the Fuel Building Crane is appropriately maintained, and we have no further concerns regarding its classification.

In summary, OOE reviewed PGE's classification of important-to-safety components and considers it acceptable.

3. Principal Design Criteria

a. ASME Boiler and Pressure Vessel Code

The design basis for the ISFSI components, as described in Section 3 of the SAR, is the ASME[26] Boiler and Pressure Vessel Code. ASME has provided internationally accepted codes and standards for components used in the power industry since 1884. The ASME Code is the accepted standard for materials, design, fabrication, examination, and testing of steel confinement casks in the nuclear industry. It provides authoritative technical guidelines for safety and reliability, and helps to ensure that the components provide adequate strength, pressure integrity, and support. The NRC uses ASME Section III as the basis for fracture toughness and inspection requirements for reactor pressure vessels and coolant systems.[27]

The steel components of the ISFSI, including the basket, basket internals, and overpack, are designed to the specifications of Section II of the ASME Code, *Materials*, and Section III, Division 1, *Rules for Construction of Nuclear Power Plant Components*. Applicable subsections of Section III, Division 1, include subsections NC and NG. Subsection NC covers pressure retaining components in the temperature, pressure, and service range that include ISFSI components such as basket shells and lids. Subsection NG covers core support structures, which provide support for the fuel assemblies. These include the basket internal members, failed fuel cans, and fuel debris process can capsules. Because the ISFSI structures and components must be fabricated to meet a range of pressure, temperatures and other stresses, the ASME Code specifies rules or limits in terms of "service levels." The code defines service levels in terms of the conditions expected during the life of the component. The applicable service levels are as follows:

Service Level A is for normal conditions, which are those associated with the normally expected range of environmental, operational, and storage conditions. These include weight, pressure (both internal and external), and thermal loads caused by operating temperature.

Service Level C is for off-normal conditions, which are defined as other than normal conditions or events that may reasonably be expected during the life of the cask system. These include potential mishandling, negligence of operators, equipment malfunction, loss of power, severe weather temperature, and blockage of one-half the air inlets to a concrete cask.

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[26] ASME: American Society of Mechanical Engineers

[27] 10 CFR 50 App.. G

Service Level D is for accident conditions, which are unusual events that are not expected to occur. These include a basket drop, a cask drop and tipover, explosive overpressure, fire, extreme weather such as tornado and earthquake.

b. ASME Code Stress Analysis

A key parameter in the ASME evaluation of any component is stress intensity, or the measure of the internal resistance experienced in a material in reaction to an applied force, such as weight, internal and external pressures, thermal loads, forces created by weather or handling, and fatigue. The ASME Code requires that calculated stress intensities not exceed specified allowable limits and it prescribes methods for calculating stress intensities.

For the important-to-safety components, OOE verified that the yield strengths (S_y), ultimate strengths (S_u), and allowable stress intensities (S_m) provided in the SAR were appropriately derived from Section II of the Code on materials. We also verified for each Service Level application that the allowable stress intensities (S_m) for the important-to-safety components were based on conservatively determined temperatures.

We verified that load combinations for the basket and overpack were appropriately added before they were compared to the ASME service level limits. For example, for Service Level A conditions, weight, thermal loads, pressure, and handling loads were added to determine the maximum stress before comparing this number to the ASME specified stress limit. For Service Level D conditions, applicable Service Level A loads plus accident pressures and loads, such as seismic or tornado wind pressure, were added to determine the maximum Service Level D stress before comparing this figure to the ASME specified stress limit.

OOE staff verified that the methodology for determining stress intensity limits (P_m, $P_L + P_b$, and P+Q) for dead weight, design pressure, maximum thermal, and normal handling loads was appropriate and that the combined values for calculated stresses fell conservatively within the allowable service level stress intensity limits. We also verified that individual calculations of stresses for various pressure and temperature conditions were reviewed in detail by the NRC.

Fatigue effects on the cask system were also evaluated using criteria in ASME Section III. Since casks are typically subjected to non-cyclic loads, PGE concluded that fatigue was not a significant concern. OOE reviewed the criteria and the analysis performed by PGE and reached a similar conclusion.

c. Design Criteria for Concrete Casks

American Concrete Institute (ACI) guidelines are generally accepted in the civil engineering field where concrete specifications are required. The American National Standards Institute (ANSI) also publishes standards for materials and testing that are generally accepted in the nuclear industry. The design criteria for the concrete storage casks are provided in ACI-349 and ANSI-57.9, except as described below. The casks will be constructed in accordance with ACI-318. The concrete is Type II Portland Cement, 145 pounds per cubic foot, 4000 psi concrete.

Section 4.2.4.2.4 of the SAR states that the concrete cask design deviates from the ACI-349 generic limits for the temperature of concrete. ACI-349 specified a normal operating concrete temperature limit of 150 ^{o}F, except for local areas which may not exceed 200 ^{o}F, and a short-term or accident temperature limit of no more than 350 ^{o}F. The SAR, however, sets temperature limits of 225 ^{o}F for normal conditions, 300 ^{o}F for off-normal conditions, and 350 ^{o}F for accident conditions. Table 4.2-12 shows a projected concrete temperature of 189 ^{o}F for normal operations and 223 ^{o}F for a "steady state severe hot" condition, both in excess of ACI-349 limits.

The SAR referenced a Bechtel study[28] showing that there is no reduction in strength for bulk concrete temperatures up to 250 ^{o}F. A review of that study, however, raised further questions. It stated that "since temperatures may exceed the allowable limits of ACI-349, Appendix 4, the proposed mix should be subjected to short-term thermal degradation testing prior to acceptance." Furthermore, it was not clear from the SAR whether the concrete mix was different from that prescribed in ACI-349. The original revision of the SAR stated in Section 4.2.4.2.4 that, "The concrete mix used to fabricate the Concrete Casks is intended to allow satisfactory long-term concrete temperatures as high as 250 ^{o}F." We requested additional information from PGE concerning the concrete mix and the deviation from ACI-349. PGE responded that:

> “The concrete mix intended to be used for the Trojan concrete cask is not different from that prescribed in ACI-349. The deviation from the 200 oF bulk limit in ACI-349 is acceptable based on the low concrete stress for the applicable load combinations.... Past studies and tests indicate that temperatures considerably higher than 200 oF can be justified without loss in strength. Long-term bulk temperatures up to 500 oF should not cause any degradation in strength while at temperature. Reducing the bulk temperature from 500 oF to room temperature may result in a loss in strength of 25% to 30%, while reducing the bulk temperature from 350 oF to room temperature may cause a strength loss of up to 10%. For bulk temperatures of 250 oF and below, no reduction in strength is expected even after cooling to ambient temperature.”[29]

PGE also referenced other material on the temperature effects on concrete to prove their point, including references to the NRC's past acceptance of deviations to ACI-349 requirements. We reviewed the NRC's acceptance of deviations to ACI-349 requirements on SNC's VSC-24 cask system for use by Consumers Power Corporation at the Palisades Nuclear Generating Station. The NRC stated:

> “The NRC has accepted deviations from the ACI-349 Code, Appendix A.4 for the concrete temperature criteria. However, while accepting the deviation, the NRC has identified a specified maximum thermal expansion coefficient for fine and coarse aggregates in the concrete which allows operation at higher temperatures.... Thus, deviation from the ACI-349 temperature criteria is not a cause for concern and does not compromise safety.”[30]

The NRC's RAI of November 25, 1996, stated in greater detail their policy for deviation from ACI-349 limits for the Trojan ISFSI. The NRC required PGE to provide tests to evaluate the reduction in strength of the concrete or modify the specifications for the fine aggregate composition of the aggregate mix by removing quartz sand and sandstone sands to limit the maximum thermal expansion coefficient. PGE chose the latter option. The August 1998 revision to the SAR states:

An alternative approach (other than testing) is to specify material properties for the concrete ingredients. PGE has opted for this method and thereby provided assurance that there is no reduction in strength as a result of exposure to high temperatures.

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[28]Bechtel Power Corporation letter to PGE, Concrete Cask Temperature Review, T040898, March 4, 1996

[29]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-040-96, July 31 1996, question 40

[30]NRC, Part 72 Statements of Consideration, p.72-SC-26, item 24, September 29, 1995

Based on the data we have reviewed, we concur with the NRC's policy on the use of concrete at greater than 150 ^{o}F and PGE's decision to alter the fine aggregate mix for the fabrication of the concrete casks. Furthermore, we agree with PGE's assertion that the calculated temperature limits are conservative. When OOE requested additional information about the effect of high temperatures on the concrete for up to 40 years, PGE responded that the maximum temperatures given in the SAR (189 ^{o}F for normal operations and 223 ^{o}F for a "steady state severe hot" condition) are based on a heat load of 26 kW per basket.[31] The actual heat load is for the "hottest" cask will be approximately 16.5 kW. Thus, the actual concrete temperatures should be much lower than the calculated maximum values.

d. Criteria for Tornado and Earthquake Safety

The ISFSI concrete casks are designed to withstand loads associated with severe weather conditions and maintain their safety function. OOE verified that the methods used to analyze the force of high winds and tornadoes on the concrete casks were appropriately based on the NRC's "Standard Review Plan," NUREG-0800 (1987). (See Section IV.E.5 for more discussion on the accident analyses associated with tornadoes.)

Oregon Administrative Rule OAR 345-26-390(4)(b) requires that the design for earthquake safety be based on the Trojan Nuclear Plant Seismic Margin Earthquake (SME) as described in PGE's 1993 SME study.[32] We verified that the accelerations used in the SAR agreed with those in the SME study. We verified that the calculation which demonstrates that the cask would not overturn in an SME event used conservative assumptions, including: the cask was treated as a rigid body, fixed at the base, and the peak horizontal and peak vertical accelerations acted simultaneously. We requested additional information concerning the method used to determine the peak kinetic energy input to the loaded concrete cask. PGE did not use the "square root of the sum of the squares" methodology, which is commonly used for nuclear plants. PGE used instead one peak acceleration at 100% and the other two at 40%, referred to as the 100/40/40 method. PGE responded that their methodology conformed with the NRC guidance in NUREG/CR-0098 and EPRI R NP-6041, Rev 1, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin."

In addition, we requested the Oregon Department of Geology and Mineral Industries (DOGAMI) to review the adequacy of the seismic constraint for the ISFSI based on both the 1993 SME study and an updated assessment dated July 18, 1995. Their letter of May 1, 1997, from J. Beaulieu, Deputy State Geologist, reached the following conclusions:[33]

- Testimony at the May 1993 EFSC hearings indicated that the SME study includes conservative safety factors in the design for spent fuel storage.
- The 1993 Site Characterization and the 1995 update properly accommodate more recent research.
- Recent earthquake research does not contradict earlier conclusions regarding site exposure to earthquake hazard.
- There is no need at this time to modify earlier conclusions based on earthquake exposure.

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[31]PGE, letter from C. P. Yundt to D. Stewart-Smith, Supplemental Response to Request for Additional Information, CPY-040-96, question 40, July 31, 1996

[32]PGE, Seismic Margin Earthquake Study for the Trojan Site, May 1993

[33]DOGAMI letter , Dr. J. Beaulieu to A. Bless, May 1, 1997

From this information, we concluded that PGE selected appropriate design criteria in considering the potential effects of earthquake and tornado hazards on the ISFSI. (See Section IV.E.6 for more discussion on the accident analyses associated with earthquakes.)

e. Transfer Cask Trunnions and Safety Factors

The American National Standards Institute (ANSI) publishes nationally recognized standards for crane safety. According to ANSI 14.6-1993, the load-bearing members of a special lifting device shall be capable of lifting three times the combined weight of both the lifting device and the container without exceeding the minimum tensile yield strength (S_y) of the materials of construction. The S_y is the point at which materials begin plastic deformation. The standard says that these load-bearing members shall also be capable of lifting five times that weight without exceeding the ultimate tensile strength (S_u) of the materials. The S_u is the maximum load that a component can sustain immediately prior to failure. These capabilities are referred to as safety factors of 3 and 5, respectively. For "critical lifts" where the lifting design is for a single load path and not a dual load path, the factors of safety on yield and ultimate must be doubled to 6 and 10. These doubled safety factors apply to the transfer cask.

The transfer cask is considered a special lifting device and is designed to the requirements of ANSI 14.6-1993 and NUREG-0612 (1980). The load-bearing components of the transfer cask are the trunnions. For these components, we reviewed the calculations presented in Section 4.7.4.1.1 of the SAR for shear stress, bending stress, maximum principle stress, and the resulting safety factors. The safety factors, presented in Section 4.7.4.1.1 of the SAR, were determined to be 11.8 for yield strength and 25.9 for ultimate strength, which is in excess of the required 6 and 10. We verified the classical structural analysis formulas used, the assumptions, and the constants; and we found the results accurate and acceptable.

The lifting yoke to be used in conjunction with the Fuel Building Crane to lift the transfer cask is also considered a special lifting device as described in NUREG-0612 and ANSI N14.6. However, the lifting yoke does not require doubled safety factors or load paths for single failure proof handling systems. Loads that satisfy the evaluation criteria of NUREG-0612 Section 5.1 need not meet the doubled safety factor requirement. Those evaluation criteria require that heavy loads in the SFP area be limited so that in case of accidents:

i. Radioactive doses caused by potential damage from dropping a heavy load must not exceed 25% of 10 CFR 100 limits, which are 300 rem thyroid and 25 rem whole body.

ii. K_{eff} must remain below 0.95.

iii. No damage may occur that will uncover the spent fuel in the spent fuel pool.

iv. The accident analysis presented in LCA-237 addresses and meets these criteria. Therefore, OOE accepts PGE's conclusion that safety factors of 3 and 5 are sufficient. In addition, the lifting yoke will be tested to 150% of rated load.

Based on the above, OOE considers the safety factors used in design of the transfer cask trunnions and lifting yoke to be adequately conservative.

f. ISFSI Storage Pad

The ISFSI storage pad is a reinforced concrete slab approximately 170 feet by 105 feet and constructed on shallow compacted fill over rock. The pad provides an unsheltered, outdoor place for the storage of the concrete casks. Because it performs no function in heat removal, structural integrity, or radiation shielding, PGE concluded that the pad is not considered important-to-safety. OOE concurs. OOE staff also observed construction of the storage pad, including placement of steel rebar and concrete pouring and testing. All elements of the pad construction were found to be satisfactory. Also, PGE's Nuclear Oversight Department (QA) was directly involved in reviewing the pad's design and construction.

Prior to the pad's construction, the NRC raised concerns that the pad design may not have sufficient impact absorbing capabilities in the unlikely event of a cask tipover. The NRC imposed additional requirements on the pad's design, particularly for the portion of the pad that serves as the foundation for the transfer station. These changes were completed as specified.

4. Structural Evaluation of Other ISFSI Components

a. Air Pad System

An air pad system will be used to move the concrete casks between the Fuel Building and the storage pad. This system consists of four individual air pads that are slipped into each 48.5 inch wide air inlet vent opening at the bottom of a concrete cask and then pressurized. The forced air will lift a nearly 300,000 pound loaded concrete cask for movement with a modified fork lift truck. The system requires a smooth surface to transport the casks, and PGE has constructed such a concrete roadway between the Fuel Building and the storage pad. After positioning the cask in the desired location on the pad, the air pads are depressurized and removed.

OOE requested additional information concerning the procedure to be taken if an air pad fails during transport. PGE responded that the amount of time that the air pad system could be inserted into the air inlet openings was limited to 12 hours to ensure that adequate decay heat is being removed from the basket. They also stated that the potential problems with the air pad system were limited (ruptured air hose or leaking air pad) and could be repaired in situ.[34]

OOE inspectors observed the first use of the air pad system on a prototype concrete cask in November 1998. During this test, two air pads were damaged due to roughness on the concrete roadway. As a result, we were able to observe that the air pads could be quickly removed and repaired with no adverse consequences to the concrete cask or its load.

b. Basket Pressure

We reviewed the PGE calculation on basket pressure, TI-021, TranStor™ PWR Basket Accident and Normal Pressure Analysis, Rev. 0. We verified that the assumptions were appropriate and conservative, the analytical methods were appropriate; and the results were reasonable. For maximum basket pressure, SNC assumed 100% of all fuel rods fail and release 30% of all fission gas to the inside of the basket. The assumption of 100% fuel rod failure is clearly conservative, and the assumption of 30% fission gas release is in accordance with the guidance in NRC Regulatory Guide 1.25. The resulting maximum pressure was less than 52.9 psig, and the resulting stresses were within those allowed by the ASME Code. Based on these calculations, PGE concluded, and OOE concurs, that the basket is conservatively designed to withstand any potential internal pressures.

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[34]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-032-96, question 111, June 27, 1996

c. Helium Leakage

OOE requested information concerning the potential for helium leakage from the basket. PGE responded that the criteria for measuring the leak tightness of the sealed basket is that there be no detectable helium using a helium sniffer with a known accuracy of at least 1×10^{-4} scc/sec.[35]

OOE also requested information about the consequences if helium leaked out of the basket during the lifetime of the ISFSI. The routine monitoring described in the SAR does not appear to include a determination of helium leakage. PGE responded:

> "The consequences of a helium leak have not been formally analyzed because the basket, transfer cask, and storage cask are designed such that the integrity/confinement properties of the basket are maintained for the spectrum of credible off-normal events and accidents. However, the procedures for responding to the off-normal events and accidents discussed in SAR Chapter 8 will include methods to determine if the basket is leaking. These methods include, depending on the event, nondestructive examination of welds and basket lid surface, use of a helium sniffer, sampling for radioactive gases or airborne contamination, and/or monitoring the air outlet temperature of the storage cask for subtle increases or decreases."[36]

In response to a second RAI question about the circumstances that would trigger a determination of helium leakage, PGE stated:

> "It should be noted that after the initial leak testing, the helium pressure in the baskets will be vented to or at near-atmospheric pressure. There will be no significant differential pressure during cask storage that could contribute to subsequent weld failure or helium leakage. There are no credible events analyzed in the SAR that could produce a helium leak while the cask is on the pad. If a design basis earthquake were to occur, PGE would initiate an investigation into possible helium leakage from the storage casks but routine inspections are not required."[37]

OOE reviewed the initial draft of off-normal procedures and could not find actions to detect a problem with helium or to investigate for possible leakage. Additionally, a review of temperature data from Consumers Power's Palisades plant in Michigan indicated wide fluctuations in air outlet temperatures due to air movement and ambient temperatures, so that finding a leak by monitoring outlet temperatures would not be practical. We put these questions to PGE in a subsequent RAI. PGE responded in July 1998 that the off-normal instructions would be revised to direct the ISFSI Manager to evaluate the need to perform a helium check following an seismic margin earthquake. OOE confirmed in November 1998 that these changes had been incorporated into the procedures.

5. Design Lifetime of the ISFSI

a. Minimum Design Life

EFSC requires that the ISFSI have a minimum design life of 40 years, as stated in OAR 345-26-390(4)(j). In response to an RAI question, PGE responded that the design life of the cask is 40 years and that SNC has warranted their design for 50 years. PGE also provided a table listing ISFSI important-to-safety components, materials, functions, and the effects of potential degradation mechanisms. These

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[35]ibid., question 66

[36]ibid., question 113

[37]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-001-97, question 11, February 6, 1997

degradation mechanisms included radiation, thermal, corrosion, fatigue, and weather. A second table provided similar information on the fuel cladding and storage pad. These tables showed that the ISFSI components are designed to perform their required functions for 40 years. OOE concurs with this analysis.

The rule also requires that the plan discuss the options available if the expected lifetime is reached and no Federally licensed permanent disposal or storage facility is available. We could not find this discussed in the SAR. In response to an OOE RAI question, PGE provided this information and described three options: (1) build and license a new ISFSI, (2) re-license the present ISFSI based upon a revised and re-analyzed design life, or (3) ship the spent nuclear fuel to an offsite temporary storage facility.[38] OOE staff considers this response acceptable.

b. Corrosion

OOE staff considers corrosion to be the major impediment to a 40 year design life for ISFSI components. However, the ISFSI design calls for all of the important-to-safety materials that would be subject to corrosion to be either inherently resistant to corrosion or protected from it. The basket shells, bottoms, lids, and overpack will be constructed from 304L stainless steel. The basket internals will have radiation resistant, high temperature coatings in addition to being in an inert helium environment. The basket exterior and the carbon steel liner of the concrete casks will have corrosion resistant coatings and be separated by ceramic tiles to prevent galvanic corrosion. The concrete cask lids will also be coated. (See the following section on coatings.) Neutron embrittlement, which is a major source of concern for reactor vessels, will not be a factor because of the extremely low neutron flux in the casks. For these reasons, PGE concluded, and OOE concurs, that the ISFSI is satisfactorily designed for a 40 year life.

c. Coatings for Corrosion Protection

PGE had planned to coat the basket internals, the external side of each basket shell, and the exposed carbon steel lining of the concrete casks with Carbo Zinc; however, the use of this material came into question with NRC Information Notice 96-34, Hydrogen Gas Ignition During Closure Welding of a VSC-24 Multi-Assembly Sealed Basket. This notice described an explosion that occurred at Wisconsin Electric Power's Point Beach plant due, in part, to the use this coating material. Explosive hydrogen had apparently been produced by the chemical interaction between the Carbo Zinc and the boric acid in the SFP during welding operations.

In response to the NRC Information Notice, PGE began an extensive testing program with other coatings. These evaluations included irradiation, immersion in boric acid, and high temperature testing.[39] Based on the test results, PGE and SNC selected one coating material for the exterior surfaces and one for basket internals. Section 4.2.4.2.1 of the SAR states:

> "The exterior bottom plate and shell are coated with a radiation resistant, high temperature, gloss epoxy coating: (1) to ease decontamination following loading operation, and (2) to promote radiant heat dissipation....

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[38]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-032-96, question 23, June 27, 1996

[39]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-001-97, question 6, February 6, 1997

> The baskets internal carbon steel components are coated with an inorganic, radiation resistant, high temperature coating: (1) to provide corrosion protection during immersion in fuel pool water, and (2) to promote radiant heat dissipation.... "

PGE will also add controls to the welding process to ensure that a hydrogen ignition could not occur at Trojan during welding.

6. Failed Fuel and Failed Fuel Cans

Between 1979 and 1982, Trojan experienced significant fuel cladding failures, in which fuel pellets were released into the primary coolant system. Damaged fuel assemblies and recovered fuel pellets were stored in the SFP. Section 3.1.1.2 of the SAR states that there are 10 damaged fuel assemblies and 1 fuel rod storage container with damaged fuel rods. These failed fuel assemblies will be placed into failed fuel cans which will be located in one of the four oversized corner locations or peripheral cells of a basket.

Section 4.2.4.2.5 of the SAR states that the failed fuel cans will be vented to the basket atmosphere to enable vacuum drying. This means that fuel pellets contained in the failed fuel assemblies will have only a single confinement boundary. Other fuel material (intact fuel assemblies and fuel debris) have two barriers. We requested additional information from PGE to show that the use of a single confinement barrier for failed fuel is acceptable. PGE responded that "The gaseous radioactive content of the failed fuel has dissipated to the reactor coolant system or the Spent Fuel Pool over the years. This reduces the potential releases from failed fuel in the unlikely event of a storage or transportation event."[40] Given the results of the accident analyses on potential releases, PGE has concluded, and OOE concurs, that the use of a single fission product barrier for failed fuel is adequate to prevent potential radiation exposure to the public.

To ensure all failed fuel is identified and placed in cans, PGE inspected each assembly in the pool in February 1997. OOE reviewed the inspection procedure and observed a portion of the inspections. No fuel assemblies had failed, other than those known to be failed through historical records. PGE concluded, and OOE concurs, that all failed fuel will be contained in failed fuel cans, as designed.

B. THERMAL ANALYSIS

1. Review Criteria

The thermal analysis for normal operation of Trojan's ISFSI design is described in Section 4 (Installation Design) of the SAR and reviewed here. The thermal analysis for loading operations is described in LCA-237 and reviewed in Section V.D of this report. For normal operations, OOE evaluated the following general areas of concern, some of which are further evaluated in the accident analyses portion of this report.

Will the cask system have an adequate, passive heat-removal capability?

What are the allowable temperature limits for the important-to-safety components and the spent fuel cladding under normal, off-normal, and accident conditions? What is the basis for these limits?

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[40] PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-001-97, question 3, February 6, 1997

What are the maximum credible temperatures that important-to-safety components and the spent fuel cladding could reach? Are these temperatures below the allowable limits?

Were computer software codes used, and were they appropriately validated?

2. Principal Design Criteria

The design for cooling the spent fuel in the casks is passive and relies on the natural circulation of outside air. Heat generated by spent fuel is transferred to the basket shell. Air in the annulus, the space between the concrete cask liner and the basket shell, is heated by the shell temperature. This heating causes a reduction in the density of the air. It rises through the annulus and exits at the four outlets near the top of the concrete cask. Cooler, ambient air enters the annulus via the four inlets at the bottom of the cask to replace the hot air that has exited, and the cycle is repeated.

For the steel components (basket, basket internals, overpack, fuel debris process can capsule, failed fuel can, and transfer cask), the principal thermal criteria comes from the ASME Code, Section III, which is described in Section IV.A.3.a of this report. These criteria specify the acceptable limits on thermal cycling, which is the accumulated stress on a component from repeated heatups and cooldowns. The SAR shows that the expected thermal cycles on the basket and other steel components are well within the ASME code criteria. Temperature changes experienced by these components are primarily due to fluctuations in weather and outdoor air temperature, and are small and slow compared with the rapid and large temperature changes of an operating reactor. PGE concluded, and OOE concurs, that thermal cycles will have little adverse effect on the basket and other steel components.

The bases for the thermal limits on dry-stored spent fuel are established in the NRC's Standard Review Plan NUREG-1536, and in the Pacific Northwest Laboratory reports PNL-6364, <u>Control of Degradation of Spent LWR Fuel During Dry Storage in an Inert Atmosphere</u>, October 1987, and PNL-4835, <u>Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases</u>, September 1983.

PNL-6364 establishes that the most likely failure mechanism for spent fuel is a phenomena called "clad creep" or "creep rupture," and the principal variable that can be controlled to limit the likelihood of this mechanism is temperature. The report documents laboratory testing in which spent fuel was subjected to adverse temperature conditions at an accelerated rate and tested for failure. The report presents temperature guidelines designed to limit long-term cladding degradation. The guidelines were selected to achieve a less than 0.5% probability for a limiting spent fuel rod of more than 1% cladding degradation over the designed life of the ISFSI, or 40 years. Based on these guidelines, fuel clad temperature is limited to 705 ^{o}F for normal storage. OOE staff has studied the industry literature and we concur with PGE's basis for the long-term temperature limit.

PNL 6364 also indicates that it may be permissible to exceed the long term limit for short periods of time without significantly increasing the risk of clad degradation. The study recommends a limit of 0.1% clad strain during vacuum drying. This translates to 816 ºF for no more than 24 hours or 842 ºF for no more than 8 hours. However, based on studies in PNL 4835, NUREG 1536 permits short term temperatures up to 1058 ºF. Such temperatures are not credible during long term fuel storage, but they could occur during vacuum drying, which is one of the steps in the fuel loading operation. Fuel loading operations are discussed in detail in Section V.D of this report.

The thermal limits for the concrete cask were established by guidelines of the ACI-349, which were described in Section IV.A.3.c of this report. PGE has selected limits of 225 ^{o}F for normal conditions, 300 ^{o}F for off-normal conditions, and 350 ^{o}F for accident conditions, and OOE accepts the inherent conservatism in these limits.

In summary, OOE has found that PGE chose appropriate design criteria for the important-to-safety ISFSI components. We also verified the statement in Section 4.2.6.1 of the SAR that "Based upon evaluation of these limits it was determined that the fuel cladding and concrete temperature limits were the limiting conditions."

3. Validation of Computer Codes

a. PGE and SNC Validation of Computer Codes

OOE reviewed numerous calculations to support the SAR structural, thermal, radiation, and accident analyses, using the process described in Section II.D.1 of this report. Where large computer codes were used, OOE checked to ensure that PGE or SNC followed the requirements in PGE's QA program to ensure that the computer codes were accurate and reliable. PGE procedures refer to this program as "software validation."

The SAR refers to the following computer software codes used in various analyses in the ISFSI design. OOE confirmed with the NRC that these computer codes have been widely used in the nuclear industry and have been accepted by the NRC in connection with previous applications.

Criticality analysis: SCALE-PC Modular Code, Version 4.1

Shielding and dose calculations: QAD-S, QAD-CGGP, and MCNP

Spent fuel gamma and neutron source strengths: OCRWM computer database

Thermal stress analysis: ANSYS and COBRA-SFS

OOE requested from PGE additional information about the verification/validation program for the use of computer software programs in ISFSI calculations. PGE responded that they "verified the adequacy of the SNC software validation process and determined it to be acceptable by approval of the SNC QA Program. SNC compliance with their validation process is verified by periodic audits performed by PGE."[41] In response to an RAI question regarding the concrete cask thermal analyses, which used the ANSYS program, PGE stated that they had verified the appropriate use and inputs for these calculations. They added that "the analytical method used (ANSYS) was deemed appropriate, mathematically accurate, and the outputs seemed reasonable. The design inputs were clear and input data was appropriate, the assumptions were properly referenced and correct."[42]

This review seemed adequate for reviewing manual calculations. However, for computer calculations, we took additional steps to ensure that PGE had not simply accepted SNC's validation process by virtue of having approved their QA program and reviewed calculations. We reviewed SNC's procedures and verified that PGE Nuclear Oversight (QA) auditors had reviewed SNC's validation program. Part of a January 1997 surveillance at SNC headquarters in which OOE staff members participated was to verify that error notices from computer software vendors had been properly handled by SNC. Three discrepancies were found by PGE auditors in the use of error notices. None had significant consequences, but because of this surveillance, interviews, and responses to our inquiries about software validation, we were able to determine that computer software codes were appropriately validated and that PGE was exercising sufficient QA oversight over SNC's validation program.

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[41] PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-001-97, question 30, February 6, 1997

[42] ibid.

In addition, we reviewed PGE procedures and calculations and interviewed PGE staff to make sure that PGE personnel had reviewed the assumptions and design inputs for every calculation that used the listed computer codes. We also verified that PGE staff members also reviewed the results to ensure that they were reasonable and that acceptance criteria were met.

To help further check the accuracy of computer codes, we requested empirical performance data. PGE representatives reported that thermal and radiological performance of vertical, ventilated concrete casks had been satisfactorily demonstrated by full-scale testing of a VSC-17 Ventilated Concrete Cask as described in the Pacific Northwest Laboratory May 1992 report PNL-7839, Performance Testing and Analyses of the VSC-17 Ventilated Concrete Cask.[43] The VSC-17 is a smaller version of the SNC cask, designed to hold 17 assemblies. As part of this testing, the COBRA-SFS computer code was used to predict cask thermal performance. Results showed the code to be accurate to within 20 ^{o}F.

We requested additional information demonstrating that testing on a 17 assembly cask was adequate to show that the computer codes would accurately predict temperatures in a 24 assembly cask. We cited differences in the heat load, the configuration of the fuel, and geometry differences between the VSC-17 and the TranStor™ design proposed for Trojan. PGE responded by stating that the computer code is conservative in predicting thermal performance for a 24 assembly system. They stated that the heat transfer for the TranStor™ system is greater because of its larger size, because the TranStor™ cask has a larger vent area, and because the TranStor™ basket has a much higher thermal conductivity due to the presence of poison material containing aluminum.[44] They did not give a precise estimate for the margin of safety provided by these computer calculations. However, the qualitative reasons given in this response were plausible, and the PNL report appears to support this conclusion. It states that "...the approximate temperatures of the commercial cask may be extrapolated from those obtained from the VSC-17." It also concludes that the "COBRA-SFS is an effective code that can be used to accurately predict temperatures in spent fuel dry storage systems."[45]

Based on the tests described in PNL-7839 and the improved heat transfer characteristics described above, OOE believes that the computer codes used by SNC to predict peak temperatures are sufficiently reliable and conservative.

b. NRC Review of Computer Codes

In accordance with OOE's memorandum of understanding with the NRC, we asked the NRC about their review process on these codes. NRC representatives stated during a telephone conference that they review computer codes for important-to-safety analyses in all license applications for ISFSIs. They reviewed all input and output data. A multi-discipline team, including contractors, then reviews this data and all assumptions used. The NRC also verified the adequacy of the manuals and supporting documents for these computer codes. For selected analyses, they independently verify the calculations using other methods. For other analyses, they use the same computer codes but with inputs

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[43]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-032-96, question 103, June 27, 1996

[44]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-001-97, question 2, February 6, 1997

[45]Pacific Northwest Laboratory, PNL-7839, Performance Testing and Analyses of the VSC-17 Ventilated Concrete Cask, May 1992

independently developed. The NRC further stated that all computer code analyses used in important-to-safety analyses for an SAR must be approved by the NRC for use in the specific or related applications.

Based on the results of our own reviews and our discussion with the NRC, we have concluded that the use of the computer codes referenced in PGE's SAR is acceptable.

4. Thermal Evaluation of ISFSI Components

a. Air Flow and Temperature Analysis

As stated previously, the design for cooling the spent fuel in the casks is passive and relies on the natural circulation of outside air through the air annulus of the concrete casks. The air flow and temperature analysis for the concrete cask is described in Section 4.2.6 of the SAR and in calculations TI-030, Concrete Cask Air Flow Analysis; TI-033, Concrete Cask Thermal Analysis; and TI-034, Concrete Cask Thermal Transient Analysis. A determination was made of the air mass flow in the annulus formed by the basket and concrete cask. This was combined with the axial heat source distribution, which is in direct proportion to the relative fuel burnup, to calculate air temperature as a function of elevation in the annulus. These results were then used in the ANSYS finite element models for calculation of the concrete cask temperatures. The resulting temperatures are listed in Table 4.2-12 of the SAR.

OOE reviewed the thermal analysis calculations using the review criteria and methods described in Section II.D.1 of this report and found them to be satisfactory. As part of that review, we verified that the input assumptions used in the ANSYS software code for thermal analysis of the concrete cask and baskets were conservative. (One problem that we found with inputs to an ANSYS calculation resulted in a detailed look at SNC's calculation reviews, culminating in the "stop work" order discussed in section III.G of this report.) We also verified that all of the thermal properties of the materials listed in Table 4.2-13 of the SAR and used in the ANSYS program were correct. We confirmed that the ANSYS code is a widely accepted industry model for computing thermal analysis, and one that is accepted by the NRC. We also confirmed that the ANSYS code was appropriately validated as described in Section IV.B.3 of this report. Because of this review, OOE staff was able to conclude that the cask system will have an adequate, passive heat-removal capability.

OOE requested additional information about the consequences to cooling heat transfer if the overpack was used. The overpack is a cylindrical shell that can be placed over a basket in the event of a basket leak. Since the air flow analysis described in the SAR did not include the overpack, we were concerned about the impact of the reduction in air flow and cooling capacity that would be caused by using an overpack. PGE responded that the reduction in air velocity due to the overpack is less than 2% for 75 ^{o}F ambient temperature and a 26 kW heat load.[46] Given the dimensions involved, the response appeared reasonable.

b. Basket Heat Load

The SAR makes various references to heat loads of 24 kW per cask, 26 kW per cask, and 1.08 kW per assembly. Some calculations assume a basket heat load of 24 kW and others assume 26 kW per assembly. We requested additional information from PGE to address these apparent inconsistencies. PGE responded that the design limit for the TranStor™ shipping cask, which will be used under the pending 10 CFR 71 license application to transport a basket to an off-site disposal, is 24 kW. The

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[46]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-032-96, question 32, June 27, 1996

TranStor™ storage system to be used under the 10 CFR 72 licensed conditions is rated for 26 kW. The administratively imposed operational limit, therefore, is the lower of these two numbers; and Technical Specification Table 2-1 limits the heat load of the fuel assemblies stored in a basket to 24 kW. The 1.08 kW limit per assembly was determined by dividing 26 kW by 24 assemblies per basket. PGE reports that based on the Trojan specific fuel assembly characteristics and the U.S. Department of Energy's, Characteristics of Potential Repository Wastes, DOE/RW-0184-R1, July 1992, the hottest Trojan fuel assembly has a thermal heat load of 1.05 kW. They conclude, therefore, that the design basis heat load value cannot be exceeded with the spent fuel at Trojan under any fuel loading scheme.[47]

Actual heat load will be less than 24 kW. Based on the fuel loading patterns specified by PGE, the highest heat load from any cask will be approximately 16.5 kW. Because the fuel clad temperature calculations are based on much higher heat loads than are expected, PGE concluded, and OOE concurs, that the peak fuel temperature will be well below the steady state limit of 705 ^{o}F. This conclusion is further supported by a separate calculation based on a 16.5 kW heat load.

5. Use of Other SNC Project Calculations as Inputs to Trojan Calculations

Many of the calculations on which the SAR is based refer to or use the results of SNC VSC-24 calculations from other projects. Several dose calculations, for example, used VSC-24 results scaled by SNC for the Trojan casks. For example, the SAR indicates on p.7- 14 that "...a simple source ratio technique is used to calculate the gamma and neutron dose rates on the Trojan storage cask side using the gamma and neutron dose rates previously calculated for the VSC-24 storage cask side." Other calculations, for example, on the concrete cask thermal analysis and fuel clad temperature limits, also referenced VSC-24 or "WEP" (Wisconsin Electric Power) calculations. Radiation dose calculation TI-005 stated that "All assumptions used in WEP 109.001.24 'VSC-24 Concrete Cask Distance On & Off Site Dose (SKYSHINE Calculation)' apply."

Interviews with PGE representatives and PGE's response to an RAI question indicated that "PGE personnel did not perform a thorough analysis of the VSC-24 calculations." They were reviewed to determine if the results were "reasonable."[48] However, PGE's response to another RAI question stated that "If the calculation is prepared by the vendor under a PGE approved Quality Assurance program, then the calculation is prepared and independently checked per vendor procedures and is also reviewed by PGE per procedure TPP 18-9, 'Trojan Calculations.'"[49] It appears that this was not done for many of the VSC-24 dose calculations used as the bases for Trojan casks calculations. We asked PGE to describe their review of calculations from other SNC projects. PGE responded with a description of their administrative controls, and they also stated that:

> "Because the implementation of TPP 18-9 regarding the verification of inputs and assumptions is somewhat subjective, PGE will review the Trojan calculations to determine which ones specifically use WEP or other calculations as inputs or assumptions and review those inputs and assumptions to determine if they have been appropriately verified, are reasonable, and are accurate."[50]

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[47]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-001-97, question 9, February 6, 1997

[48]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-032-96, question 80, June 27, 1996

[49]ibid., question 92

[50]PGE, letter from C. P. Yundt to D. Stewart-Smith, Revisions to Responses to Request for

PGE completed this review in May 1997. Eight calculations were found that used the results of other SNC project calculations as inputs. PGE reviewed the applicability and accuracy of the input calculations, and no problems were identified. OOE staff reviewed this report and accepts the results.[51]

C. RADIATION PROTECTION ANALYSIS

1. Review Criteria

The radiation protection analysis for normal operation of Trojan's ISFSI design is described in Section 7 (Radiation Protection) of the SAR and reviewed here. The radiation protection analysis for loading operations is described in LCA-237 and reviewed in Section V.E of this report. For normal operations, OOE evaluated the following general areas of concern, some of which are further evaluated in the accident analyses portion of this report.

What are the applicable radiation limits under normal and accident conditions?

What are the dose rates projected under those conditions?

Did PGE use appropriate calculation methods, inputs and assumptions?

Were computer codes widely accepted by the industry and the NRC?

Was empirical data available from other plants?

What measurement and monitoring should be done?

2. Principal Design Criteria

The ISFSI design uses the thickness of the concrete cask walls and steel liner to provide radiation shielding. The air vents are designed so that gamma or neutron radiation does not have a straight line path from the fuel basket to the outside. At the top of the cask, the design relies on steel lids to provide shielding.

3. Applicable limits

The NRC limits on radiation from the ISFSI are found at 10 CFR 72.104:

"During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ as a result of exposure to: (1) Planned discharges of radioactive materials ... (2) Direct radiation from ISFSI operations."

The regulations also require operating restrictions be established to meet ALARA objectives.

NRC limits for radiation in a design basis accident are set forth at 10 CFR 72.106, which requires the ISFSI to have a controlled area, such that any individual located on the boundary of that area will not receive more than 5 rem to the whole body or any organ from any design basis accident. The Controlled Area boundary must be at least 100 meters from the spent fuel, but it can be further away. PGE set the ISFSI Controlled Area boundary at 325 meters from the storage pad.

Additional Information #2 Trojan SAR, CPY-007-97, question 1, March 17, 1997

[51]PGE, memorandum from G.A.Zimmerman to D.C.Gildow, Review of Trojan Calculations Input and References, GAZ-004-97, May 1, 1997

The NRC regulations applicable to ISFSI loading and storage operations for personnel exposure are contained in 10 CFR 20. They are the same regulations that apply to operating plants, and which have applied to Trojan during decommissioning.

EFSC has also established limits for radiation during normal storage and accidents. For normal storage OAR 345-26-390(4)(f) limits the dose from effluents to any member of the public during fuel transfer and storage to 5 mrem. For direct radiation, the NRC limit of 25 mrem per year at the Controlled Area boundary applies.

For accidents, OAR 345-26-390(4)(c) limits the dose outside the controlled area to the EPA Protective Action Guidelines (PAGs). Those limits are:

- 1 rem Total Effective Dose Equivalent (TEDE)[52]
- 5 rem Committed Dose Equivalent (CDE)
- 50 rem CDE to the skin

These limits are for dose received by direct radiation or inhalation. When there is a possible ingestion pathway, PGE and EFSC have used a more restrictive 500 mrem limit for dose to the whole body.

The major emphasis in radiation dose criteria is on dose to the public. However, EFSC rule OAR 345-26-390(4)(h) also requires consideration of personnel dose, including a requirement that dose be kept ALARA.

4. PGE Dose Projections

a. Normal Storage

Once the ISFSI is operational, the closest structure that will be routinely occupied is approximately 100 meters from the stored fuel PGE projects the dose rate to any real person from the ISFSI to be 46 mR/year at this distance, assuming the person is on the site for 2000 hours per year. We consider this acceptable, because a 100 meter distance is well within the Trojan industrial area boundary to the south and west, and the site will be used only for industrial or commercial purposes. To the north and east direct radiation is shielded by the earthen berm. However, OOE notes that once decommissioning is complete, access to the area outside the ISFSI perimeter fence may be unrestricted because PGE's Decommissioning Plan assumes no institutional controls. For example, there are no restrictions planned for personnel who may work in what is now the Trojan Central Building. Because the direct dose in this area may exceed the dose that formed the basis for EFSC's approval of the decommissioning plan, OOE requested a more detailed plan describing radiation surveys planned for normally occupied portions of the Trojan site, and we recommend a rule requiring that results of these surveys be provided to EFSC.

Dose from the ISFSI will decrease with distance. For purposes of complying with NRC regulation. PGE has defined the Controlled Area boundary at a distance of 325 meters from the ISFSI pad. At this distance, radiation dose rate from the ISFSI is projected as 8.5 mR/yr., assuming a person stays at the Controlled Area Boundary for 8760 hours per year. For comparison, background at the Trojan site is approximately 7 μR/hr (about 61mR/year). The nearest public highway, U.S. Highway 30, is approximately 1000 meters from the spent fuel storage pad. Individuals at this distance would receive virtually no exposure above background.

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[52]The terms TEDE and CDE are as defined in 10 CFR 20. The allowed skin dose is higher because radiation to the skin has less somatic effect than radiation to the whole body.

PGE also calculated the dose from an undetected helium leak. They assumed a leak rate of 10^{-4} scc/sec, which is the largest leak rate allowed by Technical Specifications, and also the sensitivity limit for leak detection equipment. Using assumptions and calculation methods recommended by the NRC, they calculated a dose of 9.3mR to a hypothetical person standing continuously at the Controlled Area Boundary for 1 year. This is well within EPA Protective Action guidelines.

Under normal storage conditions, there is no exposure from effluents. The fuel and fission products are confined by the basket, which is welded shut and tested to ensure a leak tight seal. The helium inside the basket is at atmospheric pressure, so that there is no driving force to spread fission products in the unlikely event of a leak. The fuel produces no liquid effluent.

The only potential for particulate release is from residual contamination on the side of the basket. Before loading the basket into the concrete cask, PGE will clean the basket exterior to the extent possible. Also, the coating used for the basket exterior was chosen for ease of cleaning. However, some residual contamination may remain. The SAR states that the allowable limit for exterior contamination will be 10^{-4} $\mu Ci/cm^2$. At this limit, the maximum dose to a person standing 100 meters from the storage pad would be 2.4 millirem. This would be a one-time dose, assuming all 34 casks were uniformly contaminated to the extent allowable, and that all of the exterior contamination is released in the direction of this person. The calculation does not take into account the fact that the basket will not be exposed to the open air, but will be contained in the concrete casks. The 2.4 millirem is well within the OAR 345-026-0390(4)(f) limit.

b. Accident Conditions

As required by OAR 345-26-390(4)(a), the SAR identifies the accidents considered in the ISFSI design and analyzes the dose consequences of each accident or natural event.

Our detailed review of the accident analyses is shown in Section IV.E of this report. For radiation protection purposes, PGE must show that dose to members of the public in an accident will not exceed the EPA Protective Action Guidelines (PAGs) for offsite protective action.

PGE's analysis shows that for most accidents and natural events, there is no release of radioactive material and no increase in direct radiation. This conclusion is based on the structural integrity of the concrete casks and the basket. As discussed in Section IV.A of this report, OOE reviewed the structural analysis and concluded that the basket and concrete casks have adequate structural integrity to preclude increased radiation exposure after most credible events. This includes events such as earthquakes, volcanic eruptions, floods, and any mishandling events involving the Fuel Building Crane.

The one event which could cause increased radiation involves a tornado driven missile. The radiation exposure near the casks would increase if a tornado driven missile damaged the concrete cask and removed some of the concrete shielding. Using assumptions and methodology approved by the NRC and reviewed by OOE, the SAR shows that the worst case tornado driven missile would potentially remove 5.69 inches of concrete from the concrete cask wall. This damage would be localized at the area of impact. The resulting dose rate at the surface of the cask near the damaged area would be 286.5 mrem/hr, compared to original rate of 19.1 mrem/hr. The cask could be repaired by filling the damaged area with grout. PGE estimated the collective dose to the repair crew at approximately 0.826 person-rem, assuming two technicians working for about 30 minutes. OOE considers this acceptable, based on four factors:

- The relatively low collective dose needed for repair
- The low incidence of tornadoes in this area

- The very low probability that a missile with the worst case mass, shape and material would strike a cask in the precise orientation needed to produce the maximum damage.
- The fact that dose rate will decrease with time.

PGE also analyzed a hypothetical accident assuming that 100% of the fuel rods in a basket fail and release 30% of the available fission product gases. There is no known event that could cause this type of failure. For the release of fission products to ten environment, PGE used assumptions and methods directed by NRC interim staff guidance. The calculated dose was 22.5 mR to a hypothetical person standing at the Controlled Area Boundary for 30 days. This is well within EPA Protective Action Guidelines.

c. Bounding Accident

In addition to the accidents considered pursuant to OAR 345-26-390(4)(a), PGE also analyzed the dose consequences of one non-credible accident. This event was used only for the purpose of selecting the Controlled Area boundary, and for demonstrating that under no conditions will members of the public receive significant radiation from the ISFSI. The analysis postulates that all of the fuel in one basket suffers a total cladding failure and 30% of the fission product inventory is released to the environment. The 30% figure was based on NRC Regulatory Guide 1.25.

In this analysis, the fission products are assumed to be released directly to the atmosphere, ignoring the confinement provided by the sealed steel basket and the 31 inch thick concrete cask (steel liner and concrete combined thickness). The analysis shows that under these hypothetical conditions, and assuming worst case wind and weather conditions, the dose would not exceed the EPA PAGs beyond a distance of 325 meters.

There is no known event that could initiate this kind of failure, and an event of this kind is not credible. This calculation was not used to meet any NRC requirement, and in fact is not described in the SAR. However, it provides a basis for choosing 325 meters as the distance to the Controlled Area Boundary.

As noted in Section III.A of this report, the Controlled Area boundary is not marked by a fence or other physical barrier. PGE will control this area by maintaining the ability to exclude people in an emergency. This capability is part of the basis for the ISFSI Emergency Plan, which was approved separately by the Council on November 14, 1997. Because there is no physical boundary, and because we anticipate reductions in Trojan security after Decommissioning is complete, OOE proposes a rule requiring PGE to demonstrate ability to exclude unauthorized people during an emergency.

d. Personnel Exposure

The SAR projects occupational exposure during two phases of the ISFSI project: fuel loading and surveillance. The highest potential for personnel exposure is during the fuel loading phase. During this phase, welders and fuel handlers will work close to the basket and transfer cask.

Dose during the long-term storage phase will be the result of required surveillance activities. Measures to maintain dose ALARA will apply to both the loading phase and the storage phase, in accordance with OAR 345-26-390(4)(h).

PGE projects that the total personnel exposure for loading all 34 casks and moving them to the ISFSI storage pad is estimated at approximately 100 person-rem. This estimate is based partially on operating experience at other plants and partially on site specific estimates of dose rate and time requirements. PGE estimates another 17 person-rem for eventual transfer to shipping containers.

During long-term fuel storage, PGE projects occupational dose at about 5 person-rem per year. ISFSI personnel will receive this exposure during the course of daily cask inspections, temperature readings, and radiation surveys. The annual dose estimate for these surveillances is very conservative when compared with operating experience at Palisades Nuclear Plant in Michigan. Annual dose for surveillance of thirteen casks at Palisades is about 120 mR/year.

5. Calculation Review

a. Calculation Methodology

The dose calculation for the ISFSI is the product of several separate calculations. These include calculations of :

- Source Term (radiation from spent fuel casks)
- Shielding for the cask sides, top, and air vents
- Dose from a single cask at various distances
- Composite dose from 34 casks

SNC used a variety of calculation techniques in their dose projections. They used computerized tables for the "source term," two different computer codes for the shielding calculations, and a combination of computer codes and manual calculations to arrive at the composite dose from 34 casks. In many cases, the dose projection was based on the results of previous calculations for VSC-24 systems, with scaling factors applied to account for the difference between Trojan fuel and fuel at other utilities.

The source term calculations show the radiation produced by each fuel assembly. They were computed using a database of radionuclides compiled by the U.S. Department of Energy's Office of Civilian Reactor Waste Management (OCRWM). OOE confirmed that the OCRWM database is an NRC accepted method of computing the radionuclide content of spent fuel.

The dose from each fuel assembly depends on the assembly age, its burnup, and its initial enrichment. OOE verified that PGE used conservative assumptions for age, burnup and enrichment.

Shielding calculations show how effectively the cask shields personnel from radiation. For the dose at the air outlets and cask sides, SNC used calculations previously performed for the VSC-24 cask. SNC multiplied the VSC-24 calculation results by a scaling factor to account for the difference between Trojan fuel and fuel from previous utilities. This was appropriate because the geometry of the VSC-24 casks and Trojan casks are similar for the outlets and cask sides.

SNC performed new calculations for the air inlets because the air inlet vents at Trojan are a different design from previous casks. They used a Monte Carlo type computer code to simulate the behavior of radiation and neutrons traveling through the cask and vents. OOE verified that this code (MCNP) is widely accepted by the industry and by the NRC. OOE reviewed the documentation associated with these calculations to ensure SNC used appropriate inputs and assumptions and to ensure SNC calculations were reviewed by PGE.

To calculate the dose at a distance from a single cask, SNC used the results of VSC-24 calculations, scaled for Trojan fuel. OOE reviewed the inputs and assumptions for these calculations, and ensured that SNC calculations were reviewed by PGE's Radiation Protection department.

SNC calculated the dose from all 34 casks by using the results for the dose from a single cask. The casks will be arranged in a six by six array, as shown on Figure1 (Appendix A). The calculation of dose from the 34 cask array assumes that interior casks are partially shielded by exterior casks.

The ISFSI is designed with hottest fuel in the northeast corner of the site, and less radioactive fuel in the southwest corner. This minimizes personnel exposure because Trojan buildings are located south and west of the ISFSI site, while the north and east sides of the ISFSI are shielded by an earthen berm. Projected radiation dose is up to 46 mrem per year at a distance of 100 meters from the ISFSI, based on an "occupational" year (2000 hours). Highest doses would be to the north and east. However, the area to the north and east is not occupied, it is under PGE control, and the calculation does not include the effects of shielding from the earthen berm. Therefore, OOE considers this dose projection acceptable and conservative.

b. Key Assumptions

The calculation of exposure from the ISFSI assumes the casks are arranged with the lower dose fuel on the outside rows of the six by six array. OOE reviewed initial calculations and could not verify that they matched the description in the SAR. However, a subsequent calculation was performed, based on loading patterns designed by PGE with increased design input from RP. The loading pattern described in the final update to the SAR is appropriate because it minimizes exposure to ISFSI personnel and other personnel involved with decommissioning.

A second key assumption is that radiation from the casks on the interior of the ISFSI is completely shielded by the casks on the exterior. OOE reviewed dose rate measurements from Palisades Nuclear Plant in Michigan, which also has casks arranged in rows. The Palisades data indicates that the outer row of casks shields only about 90% of the radiation from the back rows. Therefore, we asked PGE to revise their exposure projection. Based on NRC comments, PGE revised the dose calculations to account for the contribution from interior casks. Because of the complexity of the calculation, PGE proposes a program of successive measurements during ISFSI loading to verify the results. We find this acceptable, but recommend a requirement that this program be finalized prior to fuel loading.

Another assumption is that individual baskets are loaded with less radioactive fuel towards the perimeter to reduce occupational exposure. OOE could not verify that this assumption was true. Therefore, PGE submitted revised loading patterns with the less radioactive assemblies placed at the perimeter. This is consistent with ALARA requirements.

PGE also assumed that calculations performed for Wisconsin Electric Power (WEP) are applicable at Trojan. OOE accepts this assumption because the dose calculations contain sufficient information to show that the WEP design and the Trojan design are similar in terms of radiation shielding, particularly at the upper half of the casks. We also verified that PGE used a more detailed and Trojan specific calculation to model the cask lower half, because of design differences between the WEP design and the Trojan design.

We conclude that the assumptions in the dose calculations are conservative and properly reflect the actual design.

c. Data from Other Utilities

To help assess the reliability and conservatism of PGE's dose projections, OOE reviewed radiation data from the Palisades plant in Michigan, the Point Beach Plant in Wisconsin and the Arkansas Nuclear One (ANO) plant.

Data from Point Beach show that annual radiation exposure at a distance of 100 meters is well below 25 mrem above background. Point Beach takes quarterly readings. In the quarter with the highest reading, they recorded 3.86 x 10 -3 mrem/hr. For a 2000 hour year (occupational scenario), this equals approximately 8 mrem per year above background. Point Beach also made measurements at 117 feet, 141 feet, and 255 feet. At 117 feet, the dose rate in the highest calendar quarter was approximately 19 mrem above background per year, based on 2000 hours/year. Actual dose rates at Trojan will be different, because the ISFSI layout is different, the fuel is different, and because Trojan is storing more assemblies than Point Beach. Therefore, the purpose of this review was not to use Point Beach dose measurements as an estimate for dose rate at Trojan, but to compare actual dose rates at Point Beach with predicted dose rates. A comparison of measured versus calculated dose rates indicates that the measured dose rates were generally lower than the calculated dose rates.

PGE also received radiation data from Palisades.[53] Actual gamma dose rates at the surface of the casks at Palisades were approximately one third of calculated dose rates. However, actual gamma dose rates at the top of the cask were higher than predicted by approximately a factor of two. Neutron doses at the side of the transfer cask and the top of the basket agreed closely with predicted values, but neutron doses at the surface of the concrete cask were higher than predicted. This suggests that actual neutron doses at the cask surface are difficult to predict with accuracy. However, this is acceptable because under PGE's radiation protection program, PGE will measure dose rate as the casks are loaded as a check on their dose rate calculations.

ANO provided information on occupational exposure during preparations and loading. We received personnel exposure information for loading and preparation of two casks. In both cases, the personnel exposures received at ANO were lower than the exposures projected by PGE. In some cases, the personnel dose accumulated during cask loading at ANO was less than half of the dose projection in the Trojan SAR. Actual exposures at Trojan will be different from those at ANO because of differences in fuel burnup, the layout of the work area, and design differences between the Trojan and ANO casks. (For example, the ANO baskets use a shield lid constructed with RX-277 neutron shielding material, while the Trojan casks have an all-steel shield lid.) However, the TranStor™ design and the SNC VSC-24 have many similarities as detailed in Section 7.3.2.2.1 of the SAR. Moreover many of the dose calculations for Trojan were performed by utilizing results from previous VSC-24 calculations and applying a simple source ratio technique, also as described in Section 7.3.2.2.1 of the SAR. Therefore, OOE believes the dose rates observed at ANO provide some indication of expected doses at Trojan.

6. Other Review Issues

a. Release of External Contamination

There is no path for a release of gaseous or particulate contamination from the ISFSI. However, the basket may have some residual contamination on its surface when it is removed from the pool. This exterior contamination is the only source of effluent release from the ISFSI. As described in Section IV.C.4.a of this report, PGE calculated the exposure to a person 100 meters away if all contamination from all 34 casks were released at one time. The result is a one time exposure of 2.4 mrem.

OOE could not find a basis for the 10^{-4} $\mu Ci/cm^2$ limit used by PGE, or for the 2.4 mrem value. PGE staff stated in interviews that if a cask could not be cleaned to less than 10^{-4} $\mu Ci/cm^2$, then PGE would perform an evaluation to determine an acceptable limit. A procedure is planned which will include criteria for that evaluation. OOE will review that procedure to confirm that it meets the 5 mrem acceptance criteria of OAR 345-26-390(4)(f).

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[53] PGE letter CPY-007-97 from C. P. Yundt to D. Stewart-Smith, "Revisions to Response to Request for Additional Information #2" March 17, 1997.

b. Impact on Final Decommissioning Survey

The radiation dose at 100 meters from the ISFSI will exceed background by up to 46 mrem per year, based on an occupational scenario of 2000 hours per year. This exceeds the OAR 345-026-0370(2)(a) decommissioning criteria of 5 μR/hr, and it also exceeds the more recently adopted OAR 345-026-0370(2)(h) criteria of 25 mR/year. However, PGE intends to complete decommissioning the Trojan site by the year 2001. The decommissioning area includes areas that are approximately 100 meters from the ISFSI. OOE requested additional information from PGE regarding their ability to demonstrate compliance with OAR 345-26-370(2)(a). OOE's concern was that in areas that are outside the ISFSI perimeter fence but inside the Trojan Industrial Area, PGE would be unable to show compliance with EFSC decommissioning criteria because of dose from the ISFSI. PGE stated[54] that they will use shielded detectors for the final survey in 2001. The detectors will verify that there is no residual contamination from Trojan Plant operations, but they will not "see" the direct radiation from the ISFSI. Therefore, these detectors would verify that the site meets the EFSC decommissioning criteria. The direct radiation from the fuel casks would be considered unrelated to decommissioning.

OOE requested Council guidance on this issue. At a meeting of the EFSC Trojan Subcommittee on April 11, 1997, OOE and PGE explained the relationship between measurements of residual contamination from plant operations and measurements of direct exposure from the ISFSI. At its meeting of April 18, 1997, the Council received a recommendation from the subcommittee that 25 mrem above background at a distance of 100 meters be considered an acceptable rate of dose from the ISFSI, and that the use of shielded detectors to distinguish between contamination from the plant and direct radiation from the ISFSI be considered an acceptable method of showing compliance with OAR 345-26-370(2)(a). The Council accepted the subcommittee's recommendation, but proposes a rule requiring a program of dose measurements outside the ISFSI perimeter fence.

In 1998, PGE performed radiological surveys of areas within the Trojan Industrial Area that will be affected by direct radiation from the ISFSI. Specifically, PGE surveyed maintenance areas and also the earthen berm that surrounds the ISFSI pad to the north and east. The survey included gamma radiation scans, fixed point gamma exposure rate measurements, and soil samples. Results showed that these areas meet the decommissioning criteria, and that any dose rate at these locations after fuel loading is due to the ISFSI, not residual contamination from plant operations.[55]

c. Radiation Monitoring

PGE intends to use Thermoluminescent Dosimeters (TLDs) to monitor direct radiation from the ISFSI. PGE will place 8 environmental monitoring TLDs along the ISFSI perimeter fence, and 8 additional TLDs within the radiologically controlled area. A TLD is a device that collects radiation continuously. It does not show an instantaneous dose rate. The data from a TLD is collected in the lab, and shows the cumulative radiation seen by that TLD over a period of time. PGE proposes to process the TLDs quarterly. This is consistent with the practice at the other SNC cask installations.

OOE considers the use of TLDs acceptable for several reasons. First, radiation exposure data from the casks will be taken before they are placed on the ISFSI pad, so that precise data on the dose rate from the casks will be available immediately. Also, the exposure rates at distances of 100 meters or greater

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[54]PGE letter Yundt, C.P to D. Stewart-Smith, Response to Request for Additional Information Regarding the Final Survey Plan for the Trojan ISFSI Site, CPY-093-96, November 21, 1996

[55] PGE letter VPN-074-98 S.M. Quennoz to David Stewart-Smith, December 3, 1998

are a small percentage of natural background, so that a short-term change in dose rate might be due to natural background fluctuations. The personnel located within 100 meters of the facility will be ISFSI personnel, and their exposure will be monitored and controlled in accordance with 10 CFR 20. The accident analyses show that dose consequences from postulated events are small and will not affect members of the general public. Therefore, TLDs with quarterly data collection would still provide sufficient indication in the unlikely event of an unexpected increase in radiation. And, accident analyses show that there is no credible event that could cause a significant exposure to the general public, even in the event of a major natural event such as an earthquake.

We also discussed this with the NRC in a telephone conference on October 16, 1996. The NRC indicated that the use of TLDs , as proposed by PGE, meets Federal radiation monitoring requirements. Also, PGE procedures include a radiation survey after a natural event such as an earthquake or tornado. For these reasons, we conclude that PGE's proposal to use TLDs with quarterly data collection constitutes sufficient monitoring.

7. Conclusion

Based on its review of the SAR, the dose calculations, the inputs and assumptions, experience at other plants, and Council guidance, we conclude that the ISFSI, as designed, will meet the Council criteria for radiation exposure to workers and members of the public under normal and accident conditions.

D. CRITICALITY ANALYSIS

1. Review Criteria

The criticality analysis for normal operation of Trojan's ISFSI design is described in Section 4 (Installation Design) of the SAR and reviewed here. The criticality analysis for loading operations is described in LCA-237 and reviewed in Section V.F.1 of this report. For normal operation, OOE evaluated one area of concern: Will the spent nuclear fuel remain subcritical under all credible conditions during its projected life, including normal, off-normal, and accident conditions?

2. Principal Design Criteria

The regulatory requirements pertaining to criticality are contained in 10 CFR 72.236(c), which requires that the cask system be designed to be subcritical under all credible conditions. NRC guidance in NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, provides the basic acceptance criteria that the multiplication factor (k_{eff}), including all biases and uncertainties at a 95% confidence level, cannot exceed 0.95 under all credible normal, off-normal, and accident conditions. The multiplication factor k_{eff} is a measure of the reactivity of a system. A k_{eff} of 1.0 represents criticality, and a number less than 1.0 represents a subcritical condition. The parameters that determine k_{eff} in a dry storage condition are initial enrichment, burnup, moderation, poisons, and geometry.

3. Criticality Evaluation

PGE and SNC calculated a maximum k_{eff} of 0.4077, which is well within the acceptance criteria. Their criticality analysis was performed using the SCALE-PC Modular Code, Version 4.1. Since OOE does not have the resources to validate this software code, we reviewed the input assumptions and found them to be conservative. For example, initial enrichment was assumed to be 4.2 wt% (weight percentage) U^{235} with no credit taken for burnup, whereas the actual fuel has a maximum initial enrichment of 3.56

wt% U^{235} and varying amounts of burnup. Both of these assumptions would cause the calculation to result in a higher than actual k_{eff}. Also making this determination for the dry storage condition conservative was that it did not take into account the neutron poison plates (Boral), which are included in the basket design to meet the more stricter requirements for loading operations when water is in a basket or for transportation.

OOE also confirmed that the SCALE-PC Modular Code was appropriately validated as described in Section IV.B.3 of this report. We verified that PGE personnel independently reviewed the assumptions, design inputs, results, and conclusions.[56] We also confirmed that the NRC had independently analyzed the criticality evaluation. Based on our review and on consultation with the NRC analysis, OOE concludes that the spent fuel will remain well below the 0.95 criticality criterion under all dry storage conditions.

E. ACCIDENT ANALYSIS

1. Review Process and Acceptance Criteria

In accordance with OAR 345-26-390(4)(a), Section 8 of the SAR includes various accident analyses that identify specific off-normal events and postulated accidents. Off-normal events are those that would be expected to occur infrequently, on the order of once during any calendar year of operations. Accident events or scenarios consist of infrequent or postulated accidents that might occur over the lifetime of the ISFSI. They also include hypothetical events that are not expected to occur but are postulated because their consequences may result in the maximum potential impact on the immediate environment.

The events described in Section 8 of the SAR include natural phenomenon (such as weather, tornadoes, and earthquakes), handling events, radiological events, blockage of air inlets, fuel failure, heat load, overturning, pressurization, explosions, fires, and a collapse of the cooling tower. We reviewed each of the event analyses; however, this section focuses on the analyses that are considered bounding (that is, they encompass other accident scenarios) or analyses that involved issues or concerns requiring further resolution. Our review of radiological accidents is described in Section IV.C of this report, and postulated accidents involving fuel loading operations are discussed in Section V.F of this report.

OOE staff reviewed the SAR accident analyses and the calculations used in support of these analyses in accordance with the review methodology described in Section II.D.1 of this report. OOE considered the following additional questions:

Do the calculational results demonstrate that radiation exposures resulting from a postulated accident are maintained within acceptable limits?

Do the SAR analyses encompass of all the credible off-normal and accident scenarios?

The principal acceptance criteria for evaluating postulated accident scenarios involve three distinct elements:

(1) Radiation Protection

Radiation dose limits are defined in Section IV.C of this report. There are two categories of limits. During normal operations and anticipated occurrences, the annual dose equivalent to any individual located beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid,

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[56] PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-032-96, question 22, June 27, 1996

and 25 mrem to any other organ as a result of exposure to direct radiation from the ISFSI.[57] During any design basis accident, an individual located at or beyond the Controlled Area boundary must not receive a dose greater than 5 rem to the whole body or any organ.[58]

In addition, the EPA Protective Action Guides have set potential offsite dose levels at which actions should be taken to protect the health and safety of the public. The guidelines are limiting values based on the sum (TEDE) of the effective dose equivalent resulting from exposure to external sources and from the committed effective dose equivalent (CDE) incurred from the significant inhalation pathways during the early phase of an event. The EPA PAG dose limits are 1 rem to the whole body (TEDE), 5 rem to the thyroid (CDE), and 50 rem to the skin (CDE).

(2) Subcriticality

The acceptance criterion for maintaining the spent fuel in a subcritical condition is described in Section IV.D of this report. The spent fuel must be subcritical under all credible normal, off-normal, and accident conditions. The measure of criticality, k_{eff}, must be less than 0.95.

(3) Confinement

The important-to-safety systems must demonstrate that they will reasonably maintain confinement of the radioactive material under credible accident conditions. One way this is measured is by comparing the impact accelerations in a postulated accident with limits that the fuel and baskets have been shown to be able to withstand. The SAR gives the maximum accelerations for the intact fuel assemblies in the basket or concrete cask as 82g vertical and 44g horizontal. If the impact accelerations of a postulated accident scenario remain under these numbers, then we can be reasonably sure that confinement will be maintained. (The source of these impact accelerations is discussed in Section V.C.3 of this report.)

2. Basket Handling Event

Section 8.1.1.1 of the SAR describes an off-normal handling event that consists of a lateral impact of the basket against the inside of the concrete cask during transfer of the cask to the ISFSI pad. OOE staff evaluated this event scenario and found the results satisfactory. A similar event scenario was evaluated for the loading operation and is described in greater detail in Section V.F.4 of this report. In both cases, the resulting deceleration is 17.5g, which is below the limit of 44g for a horizontal impact.

The SAR stated that the stresses caused by a lateral impact were determined by "scaling the results of the shipping cask drop analyses." We verified that this was one of the eight calculations that PGE reviewed to ensure that the other SNC project calculations used as inputs or assumptions to Trojan calculations were appropriate, reasonable, and accurate. As stated in Section IV.B.5 of this report, PGE completed that review in May 1997. OOE has reviewed it and accepted the results.

3. Off-Normal and Accident Thermal Analysis

Section 8.1.2 of the SAR and PGE Calculation TI-033, TranStor™ Concrete Cask Thermal Analysis, Rev.1, March 20, 1996, describe the conditions consisting of sustained high and low ambient temperatures and the blockage of air inlets. OOE requested additional information concerning this analysis. For the condition with half the air inlets of the concrete cask blocked, the analysis assumed an ambient temperature of 75 °F rather than a higher temperature, such as the steady-state severe

[57] 10 CFR 72.104(a)

[58] 10 CFR 72.106(b)

temperature condition of 100 ^{o}F. PGE responded that both the blockage of half the air inlets and an ambient temperature of 100 ^{o}F are considered off-normal conditions, and that in accordance with NRC guidance, two off-normal or accident conditions do not have to be considered simultaneously.[59] Although this is not the most conservative set of assumptions, we concluded that the calculation was acceptable because of its overall conservatism, the assumption of 26 kW as a heat load (versus the expected maximum of 16.5 kW), and the fact that PGE procedures will require air vent blockages to be cleared promptly.

Section 8.2.7 of the SAR describes the full blockage of air flow. As originally submitted in January 1996, the SAR contained analyses which assumed that all air inlets are blocked, but no analysis was submitted for the case where both the inlets and outlets are blocked simultaneously. In response to an NRC RAI question, PGE performed a revision to calculation TI-034, which assumed a total loss of air flow with all inlets and outlets blocked. PGE stated that "Even under this hypothetical accident condition, the concrete temperature remains below the 350 ^{o}F limit for up to 31.5 hours, and the fuel temperature remains below the 1058 ^{o}F limit for over 156 hours."[60] OOE considers this acceptable because the assumption that all air inlet and outlets are blocked for 30 hours seems highly conservative.

4. Concrete Cask Overturning Event

Section 8.2.3 of the SAR describes a concrete cask overturning event. This event is considered a "beyond design basis accident," since there is no known causal factor for such an event. The analysis presented in the SAR concludes that impact accelerations (g forces) are less than the maximum allowable. OOE reached the same conclusions, although we had minor concerns with the conservatism of some of the inputs. For example, the centers of gravity and the weight of the cask used in the analysis were not the most conservative figures available. The weight used in the calculation was 290,000 pounds. This does not take into account the weight of an overpack, which results in a total weight of 297,616 pounds. Such minor differences, however, were more than offset by other conservative assumptions. OOE considered the analysis satisfactory.

In considering a cask overturning event, OOE notes two conservative factors. First, the SAR shows by analysis that the cask will not overturn even during the worst case Seismic margin Earthquake. Further, the NRC required PGE to redesign the casks and ISFSI pad to withstand a hypothetical overturning event. OOE concludes that damage due to cask overturning is not a credible concern.

5. Tornadoes

We reviewed PGE's tornado analysis presented in Section 8 of the SAR and in calculation PGE01-10.0203-06[61] in accordance with the guidance of Section II.D.1 of this report. The design basis and calculation assumptions for the tornado accident scenario appear conservative. We verified that the design for the concrete cask assumes the most severe meteorological conditions of NRC Regulatory Guide 1.76. Table 8.2-3 of the SAR presents the design basis comparison between the requirements of

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[59]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-001-97, question 28, February 6, 1997

[60]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-001-97, question 28, February 6, 1997

[61]PGE, Calculation PGE01-10.02.03-06, TranStor™ Concrete Cask Tornado, Flood, Earthquake, and Explosion Analysis, Rev.1, December 19, 1996

Region III (Regulatory Guide 1.76), where Trojan is located, and the actual cask design. For example, the design requirement maximum wind speed for Region III is 240 mph. However, the actual design is capable of withstanding a 360 mph wind.

We verified that the maximum wind loadings are insufficient to move or overturn the casks. The overturning force of the wind is approximately one-fifth the resisting moment of the cask. We also verified by calculation that the wind loadings in terms of stresses in the concrete casks are within the allowable limits.

We verified that the design basis missiles generated by a tornado met the requirements specified in NUREG-0800.[62] The missiles include a 3960 pound car traveling at 126 mph and an 8 inch armor piercing shell also traveling at 126 mph or 185 fps. We verified that the formula used to predict penetration depth into the cask side was the industry standard and was validated in tests conducted by Sandia Laboratories in 1977.[63] We also verified that the SAR prediction of 5.7 inches penetration into the concrete cask side for the 8 inch diameter armor piercing shell was accurate. For comparison purposes, the ISFSI concrete cask walls are 29 inches thick and made of 4000 psi concrete. We did not verify the analysis of the ability of the cask closure plate to withstand the impact of an armor piercing shell; however, the methodology is similar and the references are identical.

Additionally, we compared the results presented in the SAR to the tests conducted by Sandia Laboratories. The Sandia tests most comparable to the SAR analysis included using as missiles a 1500 pound utility pole; a 743 pound, 12 inch diameter pipe; and a 8 pound piece of 1 inch rebar. The utility pole was driven at 205 fps into an 18 inch thick concrete panel rated at 3770 psi and produced no penetration and only backface cracking. The 12 inch diameter pipe was driven at 202 fps into a 24 inch thick concrete panel rated at 3795 psi. It penetrated less than 7 inches and produced only backface cracking. The rebar was driven at 303 fps into an 18 inch thick concrete panel rated at 3650 psi. It penetrated 3.6 inches and produced no backface damage.

OOE also reviewed PGE's analysis of the combined effect of tornado winds and missile impact, and we agree with the conclusion that overturning of the cask under such conditions will not occur.

PGE's analysis concludes that a tornado event does not result in the release of radiological material to the environment. We agree with this conclusion based on the evidence presented in the analysis. PGE estimates, however, that radiation dose rates will increase substantially in the design basis missile impact in which 5.7 inches of concrete is removed. The SAR states that shielding materials will be onsite to mitigate the consequences of this event until repairs can be affected.

In summary, the conservatism of the calculational results, the thickness of concrete and steel, plus the low probability of an impact of a tornado missile demonstrate that PGE's accident analysis in this area is conservative and adequate.

6. Earthquakes

The Seismic Margin Earthquake (SME) and the structural analysis of the cask design for earthquakes is discussed in Section IV.A.3.d of this report. OOE staff reviewed the accident analysis presented in Section 8 of the SAR and reviewed calculation PGE-01-10.02.03-06 on the earthquake analysis in accordance with the guidelines presented in Section II.D.1 of this report.

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[62] NRC, NUREG-0800, Standard Review Plan, Rev. 2, July 1981

[63] EPRI NP-440, Full-Scale Tornado-Missile Impact Tests, July 1977

We requested additional information concerning the application of peak ground accelerations in the calculation. Use of the 100/40/40 method, described in Section IV.A.3.d of this report is not as conservative as the common analytical method of applying all loads simultaneously. However, in accordance with the SME study, the peak horizontal acceleration of 0.38g represents the maximum expected horizontal ground motion acceleration at the site in any direction. Therefore, applying all loads simultaneously is unrealistic. PGE's calculation determined the geometric mean of two orthogonal accelerations in the horizontal direction. The resulting acceleration of 0.41g is greater than 0.38g and is, therefore, a conservative assumption for the horizontal acceleration.

The SME study gives a peak vertical acceleration of 0.25g. Therefore, using the 100/40/40 method, the vertical acceleration was calculated as 40% of 0.25g. The resulting Factor of Safety for an SME event was 1.3, which means that the force to keep the cask upright was 1.3 times greater than the force required to overturn the cask. Therefore, the analysis showed that a cask will not overturn in an SME event. However, even if 100 % of the vertical acceleration was used, the Factor of Safety would still be approximately 1.1, with the result that the cask would not overturn. The analysis is, therefore, appropriately conservative. We also verified that the resulting shear forces and moments due to an earthquake are less than the structure design capabilities.

PGE's analysis concludes that an SME event does not result in the release of radiological material to the environment. OOE agrees with this conclusion based on the evidence presented in the analysis.

7. Volcanic Eruptions

Section 2.6.6 of the SAR states that “If Mount St. Helens were to have another tephra eruption similar to the May 18, 1980 eruption, only directed towards the ISFSI ... then the expected ashfall accumulation would be about 1.8 inches.” We requested additional information concerning PGE's assertion on the level of expected ashfall. PGE responded that their analysis was based on the heaviest ashfall from the 1980 eruption at a distance of 34 miles, which is the distance from the Trojan site to the volcano.[64] They superimposed the ashfall that tracked northeast of Mt. St. Helens in May 1980 on a trajectory towards Trojan.

OOE also reviewed a 1995 report from the U.S.G.S. that stated "a conservative approach to hazards assessment requires us to assume, until there is specific evidence to the contrary, that the next eruption will be explosive and as large as or larger than the eruption of May 18, 1980."[65] This report also indicated, however, that the likelihood of significant ashfall in the direction of the Trojan plant would be small. The U.S.G.S. report stated that:

> “Westerly winds prevail; thus, significant tephra accumulation from a single eruption is more likely east than west of Mount St. Helens. The calculated probability that ten or more centimeters (four or more inches) of tephra (ash fall) from a large eruption will fall as far as 60 km (40 mi) directly east of Mount St. Helens is 20 percent; the probability that such an eruption would deposit ten or more centimeters (four or more inches) 60 km (40 mi) directly west of Mount St. Helens is less, between 1 and 2 percent.”

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[64]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-032-96, question 14, June 27, 1996

[65]USGS, Open-File Report 95-497, Volcanic-Hazard Zonation for Mount St. Helens, Washington, Edward W. Wolfe and Thomas C. Pierson, 1995

Due to this report, the large ash drifts in Yakima and Spokane following the 1980 eruption, and the difficulty in predicting volcanic events and weather conditions, we requested additional information regarding PGE's readiness to respond in the event of a worst case eruption. Specifically, we asked what sort of equipment and personnel would be required to unblock the air vents in the concrete casks if a significant ashfall were to occur and in approximately what amount of time could this be accomplished. PGE responded that they had analyzed the consequences of significant ashfall at Trojan. Ash could be ‘‘vacuumed, blown with air, washed with water, or swept out of the air inlets.’’ Their analysis indicated that the following equipment and material would be needed for ash removal: portable high pressure pumps and hoses, high pressure sprayers, powered man lifts, powered sweepers, front end loaders, dump trucks, portable lights and communication systems, fuel reserves, equipment fuel and air filters, snow shovels, personnel protection equipment (masks, goggles, gloves), and portable sanitary facilities. PGE stated that:

> "…a response team could be staffed by PGE Line Crew and Maintenance personnel. The Response Team would be organized well in advance of any potential ash fall. It is estimated that should there be a significant ash fall and the ISFSI air vents become blocked, the ash could be removed under unfavorable conditions in approximately five hours by a Response Team of six individuals, with three working and three in standby, and alternating every half-hour or hour." [66]

Because Trojan is not in the path of the prevailing winds from Mt. St. Helens and considering the low probability of a amount of significant ashfall and the high probability of several days or weeks advance warning of volcanic activity, we concluded that PGE's analysis and response plans in the event of significant ash fall are satisfactory. However, OOE recommends that the equipment required for credible accident scenarios, including volcanic ashfall, should be maintained onsite. Equipment to unblock the air inlets and outlet should be maintained onsite or available within 24 hours of notification that a significant ashfall from a volcanic eruption is likely to occur.

8. Conclusion

As stated at the beginning of this section, we reviewed each of the event analyses described in Section 8 of the SAR. We focused, however, on those analyses that are considered bounding (that is, they encompass other accident scenarios) or analyses that involved issues or concerns requiring further resolution by us or PGE. Overall, the results of PGE's accident analyses appear reasonable, acceptable methodologies were used, the assumptions were appropriately conservative, and the results fell within acceptable guidelines or limits. Also, the events or accidents analyzed by PGE appeared to encompass all probable or credible accident scenarios.

[66] PGE, letter from C. P. Yundt to D. Stewart-Smith, Revisions to Responses to Request for Additional Information #2 Trojan SAR, CPY-007-97, question 29, March 17, 1997

V. ISFSI LOADING OPERATIONS

A. OOE REVIEW OF LCA-237

For purposes of determining compliance with OAR 345-26-390(4)(a), the SAR was considered to be the "safety analysis and report identifying the specific accidents considered in the design of the facility and demonstrating compliance with criteria (2)(a), (b) and (c)." Therefore, OOE findings related to radiation dose, structural integrity, and thermal performance of the spent fuel storage system are based primarily on the SAR. Also, the accidents considered in determining compliance with OAR 345-26-390(2) and (4) are primarily those described in the SAR.

However, the SAR primarily addresses long-term storage and contains little detail about the process of transferring fuel from the SFP to the storage casks. Therefore, in order to fully assess the installation of the Trojan ISFSI, OOE also reviewed several associated License Change Applications (LCAs), including LCA-237, Spent Fuel Cask Loading. This review is critical to establishing a basis for the EFSC's evaluation of the proposed ISFSI plan, because LCA-237 evaluates significant evolutions and accident scenarios not evaluated in the SAR. For example, cask drop accidents during the loading of spent fuel and an accident involving a vacuum line break during drying and helium pressurization of a loaded basket are evaluated in LCA-237 but not the SAR. Further, OAR 345-26-390(4)(f) requires EFSC to consider activities related to transfer of spent fuel from the pool to an interim storage facility. A detailed review of LCA-237 was necessary to meet this rule.

PGE submitted LCA-237 on May 29, 1996. It has undergone two revisions since then, Revision 1 issued January 16, 1997, and Revision 2 issued April 16, 1998. The LCA requests the NRC to approve an amendment to the 10 CFR 50 license Facility Operating (Possession Only) License No. NPF-1 to allow loading and handling of spent fuel casks in the Trojan Fuel Building. PGE's current license prohibits movement of a spent fuel assembly shipping cask into the Fuel Building. That prohibition was added in October 1983 because the potential for a heavy load drop in the SFP had not been analyzed for a shipping cask.[67]

In LCA-237, PGE determined that the fuel transfer process involved certain new types of accidents not previously analyzed. However, LCA-237 demonstrates that the consequences of these new types of accidents are well below the EPA Protective Action Guides. OOE submitted two RAIs on issues related to the LCA — one August 6, 1996, and another January 9, 1997 — and we concur with PGE's determination that the accidents described in LCA-237 are well below the level of the EPA Protective Action Guides.

B. SYNOPSIS OF SPENT FUEL LOADING

In LCA-237, PGE describes the sequence for loading spent fuel into the baskets and the baskets into the concrete storage casks. Briefly, the sequence is as follows:

(1) Preparation of the Basket and Transfer Cask

[67]On May 19, 1997, the NRC issued amendment 196 to PGE's license. This removed the prohibition against placing a fuel cask in the Fuel Building, but it does not allow PGE to load that cask with fuel.

An empty basket and transfer cask are brought into the Fuel Building. The transfer cask is moved by the Fuel Building Crane and lifting yoke to the 93 elevation (top floor) of the Fuel Building and placed in the empty Cask Wash Pit. The lifting yoke is removed. Then a basket is moved by the same crane and placed into the transfer cask. Radiation shielding shims for the gap between the basket and transfer cask and a cask lid assembly are installed. The lid assembly or retaining ring allows loading of the basket but prevents the basket from being inadvertently lifted out of the transfer cask. The transfer cask assembly is then moved by the crane and lifting yoke to a position over the Cask Loading Pit near the SFP and filled with borated water. The gap flushing system is attached to the transfer cask to flush borated water through the gap to minimize contamination on the transfer cask and the outside of the basket.

(2) Loading Fuel at the Cask Loading Pit

The transfer cask is lowered to the impact limiter in the Cask Loading Pit and the lifting yoke is removed. The Cask Loading Pit is flooded with borated water and the gate between the Cask Loading Pit and the SFP is opened. Spent fuel assemblies and failed fuel cans are then loaded into the basket using the Fuel Handling Bridge Crane. This operation is directed by Certified Fuel Handlers. When the loading is completed, a shield lid with a drain pipe attached is lowered onto the basket. The lifting yoke is then attached. The loaded transfer cask is lifted from the Cask Loading Pit, and the gap between the transfer cask and basket is drained.

(3) Decontamination, Welding, Draining, and Vacuum Drying

The loaded transfer cask is moved to the Decontamination and Assembly Station (DAS) on the 93 foot elevation. Here the cask exterior is washed to remove potential contamination and the Vacuum Drying System (VDS) is attached to the basket through connections in the shield lid. Approximately 75 gallons of water are removed from the basket, and the shield lid is welded to the basket shell using an automated welding system. The basket is then filled with water and hydrostatically tested to 15 psig for 10 minutes with zero leakage. The basket is drained of approximately 75 gallons once again, and pressure tested with 15 psig helium. A third test of the shield lid is a dye penetrant examination. Next, the basket structural lid is welded into place and dye penetrant examinations are performed on those weld passes. The borated water is then pumped out of the basket, and residual moisture is removed by blowing dry air through the basket via the basket drain, vent line, and VDS. Multiple evacuations are then used to achieve a stable internal basket vacuum of 3 mm Hg for a minimum of 30 minutes. The basket is flushed with helium and the vacuum drying process is repeated. Next, the basket is filled with helium to atmospheric pressure. Finally at the DAS, the two lids are welded together through the valve access ports and the cover plates in the structural lid are welded on and dye penetrant tested.

(4) Loading the Basket into a Concrete Cask

The loaded transfer cask is next moved via the Fuel Building Crane and lifting yoke to the Fuel Building hoistway where it is lowered and placed on top of a concrete cask that has been positioned in the Fuel Building bay at the 45 foot elevation (ground floor). The lifting yoke is removed, the transfer cask bottom doors are opened, and the basket is lowered into the concrete cask using slings attached to hoist rings in the structural lid. The transfer cask is then moved back to the Cask Wash Pit where the interior is checked for loose contamination. A shield ring and steel cask cover plate are installed on top of the concrete cask.

(5) Transferring the Concrete Cask to the ISFSI Pad

The loaded concrete cask is then moved via an air pad system and modified forklift truck to the ISFSI pad. Once on the storage pad, the air pads are removed from the air inlets, the inlet screens are installed, and the process is complete.

C. STRUCTURAL ANALYSIS DURING ISFSI LOADING

1. Fuel Building Crane Heavy Loads

a. Minimum Distance to SFP

NRC guidance document NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, July 1980, states that a minimum distance of 15 horizontal feet should be maintained between the crane load block and the SFP. Because the Cask Loading Pit is only 9 feet by 12 feet and is adjacent to the SFP, PGE will maintain a minimum distance of only 6 feet. We asked what additional provisions PGE would take to ensure that a drop into the SFP would be extremely unlikely. PGE responded that the only provision they plan to adopt in addition to the guidelines of Section 5.1.1 of NUREG-0612 is to use lifting slings that are rated for twice the sum of the static and maximum dynamic loads.[68] PGE contended that the 6 foot separation was sufficient because the maximum lift height of 6 inches prevented a dropped load from tipping. We concurred with this assessment.

PGE subsequently revised the maximum lift height to 15 inches. Because of this, we confirmed through PGE calculations that a drop from 15 inches is still insufficient to cause a tipover. PGE calculations show that it would take more than twice this height, in addition to having the center of gravity vertically over the lower edge of the transfer cask, to create the conditions for a tipover. OOE concurs. As an added precaution, however, OOE plans to review the testing of mechanical stops and electrical interlocks on the Fuel Building Crane prior to movement of the transfer cask. OOE will also review the procedures required by NUREG-0612, Section 5.1.1, which will cover load handling operations for heavy loads that are or could be handled over or in proximity to spent fuel. These procedures should include identification of required equipment, inspections and acceptance criteria required before movement of load, the steps and proper sequence to be followed in handling the load, and defining the safe load path.

b. Maximum lift height

Revision 2 of LCA-237 gives the maximum lift height for a transfer cask as 15inches or less above the floor depending on the location along the safe load path."[69] It states that this lift height will be procedurally controlled. We asked if any physical controls would be used. PGE's response was that "Flagging or tape or equivalent may be attached to the load handling equipment to assist the crane operator in maintaining life heights. Personnel on or near the floor will verify clearance from obstructions and the height of the load."[70] OOE considers this acceptable, but will monitor the effectiveness of these measures when fuel loading occurs.

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[68]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-035-96, question 4, September 23, 1996

[69]PGE, License Change Application (LCA) 237, Spent Fuel Cask Loading in the Fuel Building, Revision 2, April 16, 1998, p.19

[70]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-035-96, question 18, September 23, 1996

c. Operator Training

We asked about the specifics of the crane operator training. How soon prior to load movement will it occur? Will it adhere to the guidelines of ANSI B30.2-1976? Will the training include movement along the "safe load path"? Will the safe load path be defined in procedure and clearly marked on the floor?

PGE responded that the initial qualification training and annual requalification training does comply with the standard, pre-job briefings will go over the safe load path, the safe load path will be defined in procedure and marked on the floor, and training will be incorporated into the preoperational setup and testing phase.[71] OOE considers this response satisfactory.. OOE verified that the safe load path is defined in procedures and will verify that pre-job briefings held during the project cover this subject material.

2. Fuel Building Crane Testing

OOE questioned when the Fuel Building Crane was last load tested and to what extent would it be load tested prior to loading spent fuel. In February 1997, PGE committed to performing a 125% rated load test of the Fuel Building Crane during preoperational testing.[72] This test was successfully performed in April 1997. A portion of this test was witnessed by OOE site inspector and NRC representatives. The test included operational testing, brake testing, and a complete inspection of the crane's structural elements, including ultrasonic inspection of the crane rails. In addition, an NRC inspector with experience in cranes conducted an inspection of the crane at that time. The results were satisfactory.

3. Impact Accelerations for Spent Fuel and Baskets

LCA-237 gives the maximum accelerations for the intact fuel assemblies in the basket as 82g vertical and 44g horizontal. This means that in a vertical impact, such as dropping the transfer cask with a loaded basket, with an acceleration of less than 82g, the spent fuel assemblies would remain intact. OOE requested information from PGE regarding the bases of these numbers, since they are used as the benchmarks in evaluating accident analyses for loading operations. PGE responded that the figures for the storage baskets are design parameters of 124g vertical and 44g horizontal. The figures for the fuel assemblies are 82g vertical and 63g horizontal and come from a study performed by Lawrence Livermore Laboratory.[73] Therefore, the limiting accelerations are the minimum numbers in each direction: 82g vertical and 44g horizontal. OOE reviewed the Lawrence Livermore Laboratory study and confirmed the impact accelerations as provided.

4. Impact Limiters

Impact limiters are used in and on the side of the Cask Loading Pit, near the Fuel Building hoistway, and on the ground floor under the hoistway to ensure that the forces in the fuel and basket in a postulated drop accident are within the design limits described above. We requested additional information about these devices, including: What material is used as impact limiters? What are the dimensions? Are they considered important-to-safety? Are they purchased commercial grade? How are their energy absorption characteristics verified?

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[71] ibid., question 6

[72] PGE, letter from S.M.Quennoz to U.S.NRC, Fuel Building Crane Load Testing and Certification for LCA-240 and 237, Revision 1, VPN-015-97, February 24, 1997

[73] Lawrence Livermore Laboratory, Chun, R. et al., , Dynamic Impact Effects on Spent Fuel Assemblies, October 20, 1987

PGE responded that the material is a rigid polyurethane foam that varies in density from 3 to 30 lbs/ft^3 according to the energy absorption requirements of the particular accident scenario. They are considered important-to-safety. Therefore, PGE has required the supplier to do specific performance tests in order to demonstrate that the impact limiters are suitable for use under PGE's QA program, and PGE Nuclear Oversight Department (QA) personnel witnessed the tests to ensure that the impact limiters perform as designed. Vendor certification of material properties will be required to verify the energy absorption characteristics.[74] PGE also furnished the dimensions of the impact limiters, which were used to evaluate the postulated drop accidents described in Section V.F of this report. OOE considers this approach satisfactory.

D. THERMAL ANALYSIS DURING ISFSI LOADING

OOE reviewed PGE and SNC's thermal analysis of a basket in the transfer cask loaded with spent fuel to ensure that temperature limits would not be exceeded during the loading process. OOE verified that this thermal analysis considered the worst case scenario from the standpoint of heat transfer. For example, the analysis considered both the basket with a helium atmosphere and in a vacuum. In response to an RAI question, PGE responded that the vacuum condition was bounding, resulting in the highest fuel temperatures, because helium has better heat transfer properties, regardless of the pressure.[75] We concur with this response.

Vacuum drying operations during the loading and sealing of spent fuel assemblies in the baskets present the greatest structural (thermal and pressure) stresses to the spent fuel rods under normal conditions. Fuel cladding, the zircalloy metal that surrounds the fuel pellets and forms the fuel rods, is subject to degradation at elevated temperatures. Vacuum conditions are the most unfavorable for heat transfer, and therefore result in the highest temperatures. The greatest concern, according to the industry literature, is fuel pin degradation caused by strain in the cladding. This is referred to as "clad creep," and is defined as plastic deformation in the cladding caused by sustained loading or stress at elevated temperatures. The principal variables that cause clad creep are temperature and the differential pressure across the cladding. An increase in either of these increases the likelihood of clad creep. Because of these concerns, OOE performed an extensive review of the calculations, the SAR, LCA-237, supporting documentation, and the governing industry standards for temperature limits during vacuum drying.

The applicable industry standards are those listed in the NRC's Standard Review Plan NUREG-1536. These limits are based on experimental data from national laboratories both in the U.S. and Europe. NUREG 1536 and the PGE SAR rely largely on data reported in Pacific Northwest Laboratory reports PNL-6364 and PNL-4835.

These two PNL reports contain differing recommendations for short-term temperature limits. PNL-6364 recommends temperature limits that would ensure no more than 0.1% strain (in other words, limiting the plastic deformation of the physical dimensions of fuel cladding to no more than 0.1%). These temperature limits are time dependent. Higher temperatures may be allowed if the duration will be short, and lower temperature limits are recommended if the duration will be longer. The PNL 6364 temperature recommendations also vary with type of fuel. For Trojan fuel these recommendations equate to a temperature limit of 842 ^oF for 8 hours, or 816 ^oF for 24 hours.

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[74]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-035-96, question 14, September 23, 1996

[75]ibid., question 1

The PNL 6364 recommendations also assume that the fuel has hoop stress less than 70 MPa. Hoop stress is a measure of the stress in the cladding, and is primarily due to the internal pressure of the fuel rod. PGE fuel includes some fuel rods with hoop stress greater than 90 MPa.

PNL 4835 describes test data from lab experiments and contains recommended short term temperature limits up to 1058 ºF. This higher limit is used by the NRC in NUREG 1536 and was also used by the PGE SAR. The NUREG also states that this limit may be lowered if PGE's fuel rods have hoop stresses exceeding the rods that were tested in PNL-4835.

PNL-4835 does not clearly state the maximum fuel cladding stress at which fuel rods were tested to 1058 ^{o}F. Some fuel was tested with hoop stresses as high as 140 MPa, which far exceeds the calculated stress on Trojan fuel. PGE therefore believes that the limit of 1058 ^{o}F applied to their fuel. However, OOE confirmed in interviews with PNL representatives test results at 1058 ^{o}F apply to fuel with hoop stresses only as high as 75.7 MPa.[76] Further, NUREG-1536 states that the short-term temperature "limit may be lowered for fuel with hoop stresses exceeding the rods that were high temperature tested...." Since some of PGE's fuel rods exceed the stress of the tested rods, OOE believes PGE should use a lower limit. NUREG-1536, however, does not provide a methodology to determine a lower limit.

The above discussion is a summary only. OOE submitted these issues to the NRC in greater detail in a letter on April 29, 1998[77]. We also discussed them with NRC representatives in a telephone conference on June 11, 1998. The NRC stated that a short-term limit of 1058 ^{o}F was so conservative that these issues were not a concern. In a written response, dated November 24, 1998, the NRC stated that the staff would consider our comments for the next revision to NUREG-1536.[78]

The issue of short term temperature limit at Trojan was particularly important because peak temperatures for Trojan fuel during the vacuum drying phase have been recalculated several times over the course of this review. Calculated temperatures have ranged from 801 ºF to 888 ºF for a cask with heat load of 26 kW. Also, OOE discovered errors in one calculation, which resulted in a detailed look at SNC's calculation review process. This attention culminated in the PGE "stop work" order described in section III.G of this report.

The most recent calculations indicated peak temperatures of 888 ºF under worst case conditions. This temperature is well within the NUREG 1536 recommendation, but it is clearly higher than the more conservative recommendations of PNL 6364.

PGE pointed out that their thermal analyses assumed a basket heat load of 26 kW. The actual maximum heat load, however, for their fuel was conservatively predicted to be no more than 16.5 kW. The actual cladding temperatures would therefore be much less than the predicted 888 ^{o}F. Therefore, in August 1998, OOE issued a letter requesting PGE to perform the additional calculation to determine the maximum fuel cladding temperature for a basket with a realistic heat load. We also asked them to

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[76]Einziger, Robert, et al., High Temperature Postirradiation Materials Performance of Spent Pressurized Water Reactor Fuel Rods Under Dry Storage Conditions, Nuclear Technology, April 1982

[77]OOE letter, Adam Bless to Dennis G. Reid, Spent Fuel Project Office, NRC, OOE's Review of PGE's Basis for Spent Fuel Cladding Temperature Limits, April 29, 1998

[78] NRC letter Timothy J. Kobetz to Adam Bless, Response to Inquiry Concerning NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, November 24, 1998

evaluate the consequences of having fuel with hoop stresses in excess of 70 MPa, the maximum hoop stress used in PNL-6364.

PGE completed this calculation and submitted their response on October 21, 1998. They stated that the "calculated peak cladding temperature for vacuum drying operations in the transfer cask is 682 ^{o}F."[79] This value is substantially below the NUREG-1536 limit of 1058 ^{o}F, below the conservative PNL-6364 limit of 816 ^{o}F, and even below the long-term limit of 705 ^{o}F. OOE reviewed the input assumptions to the calculation and concurred with the results. PGE further stated that, "Even though the cladding hoop stresses in some Trojan spent fuel may exceed the 70 MPa value ..., the calculated maximum potential vacuum drying temperature for the 16.5 kW case demonstrates that there is sufficient margin to ensure that there will be no significant degradation of the fuel cladding due to vacuum drying conditions." This lower maximum temperature demonstrates that fuel pin failure through pin hole leaks or gross ruptures are extremely unlikely. OOE concurs with this finding.

Additionally, PGE's Vacuum Drying System (VDS) is designed to dry a basket long before the fuel reaches its maximum temperature. Unlike the vacuum drying systems used at other SNC client facilities, PGE's VDS has all welded connections. OOE staff observed the VDS in operation during training, and we were impressed with its performance. PGE representatives estimated that this new VDS will be able to dry a basket in approximately 10 to 16 hours, depending on the heat load. They predicted that they will meet the more conservative recommendations of PNL-6364; that is, less than 816 ^{o}F in 24 hours.

In conclusion, based on our review of selected thermal calculations, the corrective actions taken as a result of an error in one calculation, the conservatism built into the thermal analyses, our review and observation of the VDS, and PGE's revised numbers for fuel cladding temperature, OOE concludes that peak clad temperatures will be within even the more conservative PNL recommendations. OOE concurs with PGE's finding that "there will be no significant degradation of the fuel cladding due to vacuum drying conditions."

E. RADIOLOGICAL SHIELDING AND CONTROLS

1. Radiological Control Measures During Loading

We asked about the radiological controls for the loading operations, including the loading of fuel, decontaminating the baskets and transfer cask, welding the shield and structural lids, and vacuum drying.

Table 7.4-3 in the SAR gives the estimated personnel exposure doses during these activities. These estimates are based on a combination of industry experience and Trojan specific estimates. However, radiation dose rates at other plants may not be indicative of the situation at Trojan. Dose varies with source strength, time of exposure, distance to the source, and shielding. Each of these factors can vary greatly from plant to plant. Exposure time depends greatly on procedures and administrative controls, which will vary. Distance is dependent on the physical geometry of a plant's layout, and the layouts of most U.S. nuclear plants are unique. Shielding varies with administrative controls and with the cask

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[79] PGE letter Quennoz, Stephen M., to David Stewart-Smith, OOE, Response to OOE Open Items on Trojan ISFSI, October 21, 1998

design. SNC's previous model, VSC-24, had neutron shielding material built into the basket shield lid, whereas the TranStor™ shield lid is constructed without this material (see Section V.E.3 below).

PGE indicated that SNC's industry-based information was to be used for estimating purposes only, that ALARA job reviews prior to loading activities will be more accurate and provide the necessary limits.[80] Based on the performance of PGE's radiological practices during the Large Component Removal Project, including ALARA job reviews, we are satisfied that the radiological control measures for loading operations will meet ALARA requirements. In addition, however, the OOE site inspector will attend selected pre-job briefings to verify that ALARA measures are covered, and will review PGE exposure records over the course of the fuel loading effort.

In response to an RAI question about the status of the Fuel Building ventilation during loading operations, PGE responded that "...the Fuel Building ventilation will be operating during spent fuel loading and handling to minimize the amounts of particulate radioactivity released from the Fuel Building to the environment."[81] We verified that this policy is already part of PGE procedures.

2. Loading of Fuel Debris

Section 3.1.1.3 of the SAR gives the limits on the contents of fuel debris in the baskets as 10 kg of fissile material per basket for the TranStor™ Shipping Cask and 20 curies of plutonium to meet the offsite transportation requirements of 10 CFR 71.63. A revision in October 1998 added that "A criticality analysis was performed and determined that the quantity of fuel debris contained as fissile material will not exceed 7.5 kg per basket." Also, a recent change to the Technical Specifications limits fuel debris to 7.5 kg per basket. Section 4.2.7 of the SAR states that this amount of fuel debris is not as reactive as an intact fuel assembly, nor will it cause thermal, structural, or shielding problems no matter how it is distributed within the failed fuel can. Section 4.2.7 also states that the amount of fuel debris per basket "will be administratively controlled," and Section 5.1.1 states that fuel debris will be "visually inspected prior to loading to verify that each ... item conforms to the established classification criteria."

We asked PGE about the controls that would be placed on this material. How and at what time in the loading process would the weight and curie limits be determined, and what amount would go into each can? PGE responded that "the entire amount of fuel debris will be well below the limits of 20 Ci plutonium and 10 kg U (uranium), even if all debris were packaged in one can."

Following our inquiries, the fuel debris was collected and stored in 41 fuel debris process cans which were in turn placed in 8 fuel debris process can capsules with one remaining can to go into a failed fuel can. The process cans and can capsules are currently stored in the SFP and will later go into corner locations in the baskets during fuel loading. The fuel debris collection process was completed in December 1997. The OOE site inspector observed portions of this process and concluded that adequate administrative controls were in place. From these observations and the information provided to us on the analysis of the debris, we have concluded that the fuel debris is not a criticality concern or a significant radiological hazard.

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[80] PGE letter CPY-032-96 from C.P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, question 87, June 27, 1996

[81] ibid., question 76

3. Neutron Shielding of the Shield Lid

The highest working dose rates will affect the welders and the non-destructive testing (NDE) inspector(s) during the welding of the shield and structural lids. According to the SAR, the TranStor™ shield lid is constructed from plates of stainless steel. However, another SNC design, the shield lid of the VSC-24 basket, consists of steel plates sandwiching a neutron shielding material. A comparison between the two designs shows that the neutron dose rate for the Trojan TranStor™ design is almost five times that of the VSC-24. We requested additional information from PGE about their choice of a design with less neutron shielding on top of the basket.

PGE responded that SNC changed the shield lid design from the VSC-24 to develop a canister that was similar to the DOE Multi-Purpose Canister (MPC), to reduce the basket length and weight, and to facilitate the manufacturing process for the shield lid. According to SNC, the MPC did not include neutron shielding; the reduction in length and weight was required to meet a design goal of 125 tons; and the fabrication process was made simpler by replacing 2 inches of RX-277, which required baking, with 0.5 inches of stainless steel. PGE also stated that the higher doses would be offset by temporary neutron shielding placed on the shield lid during welding and vacuum drying operations.[82] OOE considers this response acceptable.

F. ACCIDENT ANALYSIS OF ISFSI LOADING OPERATIONS

1. Criticality Evaluation During Loading Operations

Our review of PGE's criticality analysis for storage and transportation of spent fuel is described in Section IV.D of this report. Although the criticality analysis for the loading operations considered different parameters and assumptions than the analysis for storage and transportation, it was bounded by the analysis described in Section IV.D. The result of both analyses are that the spent fuel in the basket remains subcritical under all credible conditions.

Section 5.1.1.2 of the SAR describes the potential for boiling in a basket after it has been removed from the Cask Loading Pit. One of the concerns with boiling was the unknown impact that this could have on criticality. We requested additional information from PGE to address this concern. PGE responded that "the TranStor™ system uses Boral, a neutron poison, and the criticality analysis for the fuel loading process takes credit for the Boral, thus eliminating the criticality concern with boiling in a basket."[83] (The criticality analysis for dry storage conditions does not take credit for the Boral; however, PGE takes credit for Boral during loading operations when there is water in the basket and in the 10 CFR 71 analysis involving the transportation of spent fuel.) Further discussions with NRC representatives confirmed PGE's analysis that boiling was not a factor in the criticality analysis.

As stated in section IV.D of this report, the criticality analysis is based on fuel enrichment of 4.2%, which is higher than the maximum actual enrichment at Trojan. The analysis is based on fresh fuel, although all fuel assemblies will have at least one year of burnup. The water in the pool is assumed to have no soluble boron, whereas the actual boron concentration of water in the SFP and Cask Loading Pit

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[82] PGE, letter from C. P. Yundt to D. Stewart-Smith, Revisions to Response to Request for Additional Information #2, CPY-007-97, question 4, March 17, 1997

[83] PGE, letter from C. P. Yundt to D. Stewart-Smith, Revisions to Response to Request for Additional Information #2, CPY-007-97, question 10, March 17, 1997

is required to be at least 2000 ppm (parts per million). These three factors make the criticality calculation highly conservative. We conclude that fuel will remain well below the criticality criterion under all loading conditions.

2. Drop Accidents During Loading Operations

PGE's drop accident scenarios included the following objects being dropped onto a basket loaded with spent fuel: fuel assembly, shield lid, structural lid, lifting yoke, transfer cask lid assembly, basket lift rigging, and concrete cask shield ring or lid. Of these, PGE limited their analyses to the bounding scenarios. Drops of the structural lid, transfer cask lid assembly, basket lift rigging, and concrete cask shield ring or lid are all bounded by the drop of the lifting yoke onto a basket.

We reviewed PGE's analysis of the drop of the lifting yoke onto a loaded basket using the review methods described in Section II.D.1 of this report, and we concurred with the findings described in the calculations and in LCA-237. The lifting yoke would be postulated to drop after the eight inch shield lid was in place in the basket. Damage would be limited to the shield lid, and the fuel would be unaffected.

A significant load drop that was not analyzed was that of a shield lid onto a loaded basket in the Cask Loading Pit. LCA-237 stated that the shield lid drop was not postulated to occur and that "... the guidance of NUREG-0612 is being used to minimize the possibility of this event to the degree that the event need not be considered credible." However, given the fact that the lifting yoke is considered to be the most serious drop of an object onto a loaded basket and that this drop is mitigated by the shield lid, we asked PGE why the consequences of a shield lid drop were not analyzed. They responded by quoting LCA-237 that "Load testing requirements, and administrative controls (i.e., procedures, training, maintenance, inspections) will also be implemented. Implementing these design features and controls makes the possibility of a shield lid drop extremely small in accordance with NUREG-0612. Therefore, the consequences of a shield lid drop need not be determined." Our understanding of NUREG-0612 is that if the effects of a drop are not to be analyzed, then the single-failure-proof guidelines must be followed, including double safety factors for the slings. PGE responded that the NRC had granted an exemption to this part of the NUREG. We reviewed and confirmed this; however, we also confirmed that the safety measures in place made a shield lid drop extremely unlikely. The lifting devices will have a safety factor greater than 10 times the rated load. In addition, the crane was tested to 125% of its rated load, which is far in excess of the shield lid lift. OOE therefore accepts PGE's assessment.

Additionally, regarding the shield lid, we inquired about the installation of the drain line. The drain line is a pipe that is threaded into the shield lid and extends to the bottom of the basket. According to the draft loading procedure, this drain line is installed while the basket is suspended over the Cask Load Pit. We asked: Could the drain line be dropped onto fuel in the basket and has that been analyzed? PGE responded that such a drop had not been analyzed but that it will be secured with a safety line to prevent such an occurrence. OOE finds this satisfactory.

Postulated drops of a loaded transfer cask included the scenarios where it is dropped into the Cask Loading Pit, the Cask Wash Pit, the Fuel Building hoistway, and onto the Fuel Building floor. We looked at each one of these accident scenarios, because these drops appeared to be the most limiting in terms of potential damage to the fuel and potential radiological releases. OOE reviewed PGE Calculation TI-026, Evaluation of the Transfer Cask Drop Inside the Trojan Fuel Building, Rev.0, which covered all of the postulated scenarios for a transfer cask drop. The most limiting of these was the drop into the Fuel Building hoistway. The assumption used was that a loaded transfer cask drops from above the 93 foot level to an impact limiter at the 45 foot level. The result of this postulated drop is that the impact acceleration is 17 g, which is far less than the 82 g limit. OOE confirmed these results. This drop scenario causes no damage to the intact fuel. The postulated radiological release from damage that may occur to suspect or failed fuel was well below the EPA Protective Action Guide of 1 rem whole body for the early phase of an event.

The other major drop accident scenario considered involved a basket drop into a concrete cask. We reviewed this analysis and confirmed PGE's conclusions that the impact limiter in the floor of the Fuel Building would maintain the drop decelerations to less than the allowable values and that the drop would not result in a radiological release or an increase in k_{eff} to more than 0.95.

OOE's review of these drop accident scenarios confirms that the accidents analyzed encompassed all probable or credible scenarios, the results were reasonable, acceptable methodologies were used, assumptions were appropriately conservative, and the results fell within acceptable guidelines or limits. Also, projected radiation doses at the site boundary were within the limits of EPA PAGs.

LCA-237 additionally addresses the issue of a tear or breach in the Cask Loading Pit liner or SFP liner. In the case of the Cask Loading Pit, an impact limiter will be installed in the bottom to prevent damage in the event of a drop. However, should either liner be punctured, the leakage will be collected by telltale drains, and the size of the leak will be limited to a maximum of 44 gpm. It would take approximately four hours for the water level to decrease to 10 feet above the top of the fuel assemblies in the Cask Loading Pit, which is still enough water to provide an adequate heat sink for the fuel and adequate shielding for personnel in the area. According to PGE, four hours is sufficient to take appropriate corrective actions. OOE concurs.

3. Tipover Accidents During Loading Operations

OOE reviewed the transfer cask tipover scenarios described in LCA-237, including a tipover near the Cask Loading Pit, the Cask Wash Pit, the Fuel Building hoistway, and during concrete cask transfer operations. All of these were tipover events were bounded by the consequences of a drop accident. The impact forces involved were much less than those for a cask drop. OOE found that PGE's tipover analyses appeared to encompass all probable or credible accident scenarios.

4. Mishandling Events During Loading Operations

Mishandling events during fuel loading are bounded by the drop accident scenarios. The analysis of a crane mishandling event, for example, showed maximum accelerations well below those projected in the analysis for the drop of a loaded transfer cask.

PGE's analysis of mishandling events did not appear to encompass all probable or credible accident scenarios. OOE requested additional information about the effect on the cooling capability of the cask system if a basket was not centered during loading into a concrete cask or if it shifted to one side during movement with the air pad system, thereby blocking one side of the vertical ventilation channel. PGE responded that:

> "A thermal analysis has not been performed for the scenario where the basket would be immediately next to the concrete cask steel liner. If a basket is next to the concrete cask steel liner, the basket would be over a large portion of one of the two air inlets and would be in front of a large portion of two of the four air outlets near the top of the basket... If shifting of the basket within the concrete cask is suspected, then action would be taken to re-center the basket within the cask... Possible corrective actions would include visual examination of the basket lid to determine if it is

still centered. If required, the basket would be re-centered by moving the cask to the Fuel Building crane bay or to the ISFSI Transfer Station."[84]

In response to a different question about the effects on the natural circulation air flow due to improperly centering the basket, PGE stated that the air flow up the annulus between the basket and concrete cask is "relatively insensitive to the basket being exactly centered."[85] The air flow calculation shows a decrease of less than 2% when the overpack, which is 2 inches larger in diameter than the basket, is placed in the concrete cask. OOE also reviewed PGE Calculation TI-030, Concrete Cask Air Flow Analysis, Rev.2, which describes the postulated scenario of half the inlets being blocked and indicates that the outlet temperature would increase 18 ^{o}F. This approximates the situation if the basket were to shift off-center.

PGE stated that they will rely on procedures for centering the basket and for taking corrective actions in the event that a basket is loaded off-center or becomes off-center during movement to the ISFSI pad. OOE recommended that the procedures also include acceptance criteria for centering, provisions for determining when a visual examination is required, the circumstances that would require re-centering, and corrective actions to re-center a basket. PGE stated that the procedures will be modified to address these issues, and OOE will confirm this prior to loading. Additionally, OOE recommended that visual verification of centering a basket in a concrete cask be aided, i.e., with a visual scope or mirrors, as appropriate, for ALARA purposes.

5. Operational and Support System Events

a. Loss of Electrical Power

Loss of electrical power is addressed briefly in LCA-237 regarding its impact on welding, basket draindown, and vacuum drying. PGE's analyses demonstrate that loss of electrical power during the loading operation will not result in fuel cladding temperatures exceeding the short-term limit. Further, if a loss of power occurs while water is in the basket, the analysis shows that it will take at least 30 hours to reach boiling, which should be enough time to restore power or to provide other means of cooling. Should boiling occur, however, the fuel cladding will not exceed the short-term temperature limit nor experience significant strain that could cause long term cladding degradation. OOE concurs with these conclusions.

We asked PGE what emergency measures were planned should electrical power be lost during movement of fuel. PGE responded that "If a fuel assembly were being moved at the time of the loss of power, the fuel assembly would be suspended more than 9.5 feet underwater, which limits the dose rate at the water surface to no more [than] 2.5 mrem/hr. ... Vertical motion of the suspended fuel assembly would be stopped by the brake on the electrical hoist. If the suspended fuel assembly has cleared the spent fuel racks, then it could be moved east or west by using the manually operated trolley. This movement may be desired as a precaution if the suspended fuel assembly is over another fuel assembly in the spent fuel pool racks."[86] Neither PGE's response nor LCA-237 addressed the condition of a loss of electrical power to the Fuel Building Crane. From discussions with PGE representatives, however, we

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[84]PGE, letter from C. P. Yundt to D. Stewart-Smith, Supplemental Response to Request for Additional Information, CPY-040-96, question 95, July 31, 1996

[85]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-032-96, question 71, June 27, 1996

[86]PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for Additional Information, CPY-032-96, question 109, June 27, 1996

learned that loss of electrical power to this crane results in failure “as is.” Both cranes have holding brakes that continue to grip the load in the position they were in when power was lost.

b. Stuck Assembly During Loading Operations

Due to industry experience with bowed fuel assemblies becoming stuck, we asked PGE what measures would be in place to prevent and recover from such an occurrence. They responded that “The fabrication specification for the PWR basket requires drag testing each [storage] cell with a full length gauge to ensure straightness and squareness.... The corner storage cells are slightly larger and could accommodate bowed fuel assemblies. Procedures will be written for spent fuel transfer to the ISFSI, including recovery from abnormal events and accident conditions.”[87] OOE accepts this response, but plans to review the procedures prior to cask loading operations to verify that they contain provisions for handling bowed fuel assemblies.

c. Rupture of a Pressurized Line

Two cases were evaluated in LCA-237 involving the rupture of a pressurized line. One was the discharge line on the pump used to drain the basket. This analysis assumed that all the water drains onto the Fuel Building floor at the 93 foot elevation, which results in negligible dose at the site boundary due to a lack of airborne radioactivity. OOE verified the assumptions of radioactivity in the water and agreed with the PGE's conclusions. The second rupture involved the inert gas supply line and assumed that the basket was pressurized to 25 psig. Again, OOE verified the assumptions on surface contamination levels and agreed with PGE's conclusion that the resulting off-site dose would be well below the EPA Protective Action Guide of 1 rem whole body for the early phase of an event. Additionally, this accident scenario was made even less likely by the all-welded connections of PGE's VDS design. The vacuum drying systems used to date at all other SNC client locations have used threaded connections.

d. Communications to Prevent Accidents During Loading

As a result of an OOE RAI question regarding communications, PGE stated that seismic monitors and audible control room alarms would be functional during loading operations. Communications will be established between fuel handlers and the control room, and the on-shift supervisor will be able to communicate with the fuel handler located at the SFP via independent circuits.[88] Subsequently, PGE responded to a further question about control room alarms by stating that:

> “...there are no audible alarms associated with the control room components of the seismic monitoring instrumentation systems. ... local ground motion from a seismic event that is sufficient to trigger existing control room indication of the event would be immediately apparent to the fuel moving crew during loading operations. The most timely and appropriate action for the fuel moving crew upon sensing an event is to immediately place any suspended load in a safe position and then secure from fuel loading operations until the effects of the event can be evaluated. Therefore, during spent fuel cask loading operations, sufficient seismic monitoring instrumentation will be maintained in service to facilitate the evaluation of the effects of a seismic event so as to permit such timely follow on action as may be appropriate. This requirement will be incorporated into the spent fuel cask loading procedures.”

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[87] ibid., question 110

[88] PGE, letter from C. P. Yundt to D. Stewart-Smith, Response to Request for AdditionalInformation, CPY-035-96, question 29, September 23, 1996

OOE plans to review the applicable procedures prior to the beginning of loading operations to ensure that actions for a seismic event are addressed.

We also asked if procedures would preclude fuel handling while the transfer cask was being moved. PGE responded that "Fuel Handling Procedures will administratively control fuel movement and cask movements such that during cask movements near the spent fuel pool, no fuel will be moved except to mitigate hazardous situations."[89]

6. Conclusion

Fuel loading operations will be performed in accordance with PGE's existing operating license under 10 CFR 50. All EFSC rules in OAR 345 Division 26 will apply, just as they would during any normal fuel handling operation. PGE committed to addressing many of OOE's operational safety questions in procedures. OOE has reviewed PGE's safety evaluation for the fuel handling operations and found that it meets existing requirements. In particular, PGE has analyzed potential fuel handling events and external events and has demonstrated that all postulated events are bounded by the Defueled Safety Analysis Report. Postulated accidents during fuel loading will not result in radiation exposure greater than the EPA protective Action Guidelines, which form the basis for PGE's EFSC-approved emergency plan. Fuel handling operations will not result in dose from effluents to any person in an unrestricted area in excess of 5 millirem, as required by OAR 345-026-0390(4)(f).

[89] ibid., question 30

VI. COMPLIANCE WITH EFSC RULES

In order to adopt rules approving an ISFSI, EFSC must find that the proposed ISFSI complies with OAR 345-26-390. This rule has safety standards that apply generally to any spent fuel storage facility, and requires compliance with other EFSC Decommissioning rules.

In this chapter, we first consider the ISFSI's effect on PGE's ability to meet the EFSC decommissioning rule, OAR 345-26-370. We will then consider PGE's compliance with the EFSC spent fuel storage rule, OAR 345-26-390.

A. EFFECT OF ISFSI ON DECOMMISSIONING

OAR 345-26-390(4)(i) states that the ISFSI must not adversely affect PGE's ability to meet the EFSC Decommissioning standards in OAR 345-26-370(2), subparts (a) through (f).

In its February 6, 1997, response to OOE questions, PGE stated that:

> " The Trojan Nuclear Plant Decommissioning Plan (D-Plan, PGE 1061) is the controlling document for decommissioning activities and for demonstrating compliance with Oregon Administrative Rules OAR-345-26-370(2). Section III of Appendix A of the D-Plan is an evaluation of compliance with OAR 345-26-370(2). The SAR does not supersede the D-Plan but rather complements it and attempts to demonstrate how compliance with OAR 345-26-390 is achieved..."

In its "Review of PGE's Decommissioning Plan for the Trojan Nuclear Plant" , dated January 22, 1996, OOE concluded that PGE's Decommissioning Plan complied with all requirements of OAR 345-26-370(2), including subparts (a) through (f). For the SAR, OOE reviewed each of these rules and verified that none of the conclusions contained in that January 1996 report are changed by the ISFSI.

Acceptable Surface Contamination - OAR 345-26-370(2)(a): *The plan contains criteria for the free release of materials and the area as specified in Table 1 (as excerpted from NRC Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," June 1974).*

This rule contains limits for loose and fixed surface contamination following decommissioning. Section 4.2 of the PGE Decommissioning Plan contains final release criteria that match the criteria specified in EFSC's rule. That plan would remain applicable and would not be affected by construction of an ISFSI.

PGE's Decommissioning Plan calls for decontamination of the plant after the fuel has been stored in the ISFSI. Once the fuel is in the ISFSI, there is no interaction between the ISFSI site and the rest of the plant site. There is nothing about storage of fuel in the TranStor™ casks that would interfere with or preclude the successful decontamination of the rest of the Trojan site (the portion outside the ISFSI.)

OOE asked if contamination from the ISFSI basket could re-contaminate the site, after the Decommissioning Plan is completed. The SAR shows that recontamination from the fuel is not credible. The fuel will be contained in steel canisters ("baskets"), three-quarter inches thick. The baskets will be sealed, welded shut, and pressure tested to ensure an airtight seal. Once sealed, the baskets will be backfilled with helium at atmospheric pressure, so that there will be no outward pressure that might cause contamination to leak outside the baskets. The baskets will in turn be stored in reinforced concrete casks, which will not be airtight but which will preclude gross damage to the basket. The SAR also provides analyses demonstrating that the basket is designed to withstand off-normal events such as seismic events and handling accidents. OOE's review of these accident analyses is described in Section IV this report. Based on our review of the SAR, OOE concurs that the cask design precludes recontamination of the site due to fuel damage.

We also considered the possibility that loose contamination on the outside of the basket could contaminate the plant site after the Decommissioning plan is complete. Section 8.1.3 of the SAR contains a limit of 10^{-4} μCi/cm^{2} beta-gamma contamination on the exterior surface of the basket. This is a very low level of contamination and should preclude recontamination.

Further, PGE must submit an environmental radiological monitoring plan as required by OAR 345-26-330. PGE's plan is described at sections 7.5 and 7.6 of the SAR. PGE will routinely survey for contamination at the cask air inlets and outlets, and will perform quarterly surveys in accessible areas of the ISFSI. OOE concurs that this would be an effective way to determine if any unanticipated contaminants were escaping from the storage casks.

Based on the use of sealed baskets, the low level of contamination on the basket exterior, and the monitoring requirements, we conclude that the ISFSI will not adversely affect PGE's ability to meet the free release criteria contained in the Decommissioning Plan.

<u>Exposure Rate - OAR 345-26-370(2)(b)</u>: *After decommissioning, the exposure rate at one meter from all surfaces in the facility buildings and outdoor areas shall be 5mR/hr or less above the background radiation level. Background radiation is defined in OAR 345-01-010.*

The PGE Decommissioning Plan describes PGE's Final Decommissioning Survey and commits PGE to meeting this rule. The ISFSI is an integral part of that plan.

PGE calculations show that there will be areas that are within the Trojan site but outside the ISFSI perimeter fence where direct radiation from the stored fuel exceeds 5 μR/hr. On an annual basis, direct radiation at these locations would exceed 25 mR/year. For example, direct radiation at 100 meters from the stored fuel has been estimated at 46 mR/year (assuming occupancy of 2000 hours/year). This exceeds the EFSC standard for decommissioning. We asked PGE how they can meet the EFSC standard, in view of the dose contribution from the ISFSI. PGE's response (see PGE letter from C. P. Yundt to D. Stewart-Smith, November 21, 1996) states that PGE will meet the EFSC decommissioning criteria in two ways.

First, PGE performed a radiological survey of the proposed ISFSI Site. The survey was designed according to the guidance in NRC NUREG 5849, the same methodology used for the Final Site Survey after decommissioning. PGE performed this survey from August 5 through October 17, 1996. The results are shown in PGE's "Final Survey Report for the ISFSI Site," dated 10/30/96. They show that the area surrounding the proposed ISFSI already meets the EFSC free release criteria for both soil contamination and direct radiation.

OOE inspected this survey. We reviewed the survey plan prior to its implementation, and inspected the survey while in progress. OOE's inspections were augmented by the Radiation Control Section of the Oregon Health Division, which took confirming measurements and soil samples. OOE reviewed the final survey report and concurs with it. The PGE survey of the ISFSI site was also approved by the NRC. PGE repeated portions of this survey in 1998 with similar results.

Based on this survey, PGE concludes, and we concur, that the land area surrounding the proposed ISFSI meets the EFSC criteria for free release. Also PGE has implemented positive control over ingress and egress, to prevent contamination of this area during decommissioning. (see PGE letter VPN-046-96 to NRC dated August 13, 1996).

The second way is through the use of shielded detectors during the Final Release Survey. Once the Decommissioning Plan is complete, PGE must perform a Final Survey to prove that the Trojan Plant site meets state and federal criteria for free release. This survey is currently scheduled for the year 2001.

For areas near the ISFSI, PGE will use shielded detectors. These detectors will "see" any residual contamination from Trojan plant operations, but will not see the direct contribution from the stored Spent Fuel. This will prove that all residual contamination from plant operations has been removed, and that the only dose contribution is "shine" from the stored fuel. The dose contribution from the spent fuel will vanish when and if the fuel is transferred to a federally licensed permanent spent fuel site. OOE concurs that this is an acceptable approach.

On October 15, 1998, EFSC adopted alternate final survey criteria at OAR 345-026-0370(2)(h). The new criteria are consistent with the latest NRC regulations at 10 CFR 20. The new criteria require doses of 25 mR/year or less from all sources. PGE submitted new information to OOE, describing recent radiological surveys that show compliance with the new decommissioning criteria and account for the direct radiation from the ISFSI.[90] OOE concurs with the approach in this letter.

We conclude that the proposed ISFSI would not adversely affect PGE's ability to meet the decommissioning criteria of OAR 345-26-370(2)(b) and (h).

<u>Radioactive Waste Removal - OAR 345-26-370(2)(c)</u>: *The plan must contain provisions that require removal from the site of all radioactive waste as defined in ORS 469.300 on a schedule acceptable to the Council. Spent nuclear fuel and other radioactive materials that must be disposed of in a federally approved facility may be stored on the site until such a federally approved facility will take the fuel and these radioactive materials.*

PGE's plan for removal of radioactive waste is fully described in its Decommissioning Plan, as approved in March 1996. OOE reviewed the plan in detail and found that the removal of spent fuel from the Trojan SFP would in no way adversely affect PGE's ability to remove non-fuel waste. In fact, dry fuel storage would facilitate removal of non-fuel waste, because it would allow PGE to remove some contaminated components that are now needed for SFP maintenance and cooling. OOE concludes that the proposed ISFSI would not adversely affect PGE's ability to comply with OAR 345-26-370(2)(c).

<u>Effluent Monitoring - OAR 345-26-370(2)(d)</u>: *The plan must contain an acceptable program for monitoring and controlling effluents to ensure compliance with applicable state and federal limits. This program may be incorporated by reference, if it has previously been approved by the department.*

PGE met this rule by providing an acceptable Radioactive Effluent Controls plan in accordance with OAR 345-26-330. The details of PGE's radiological and effluent monitoring plans may be modified as site conditions change. However, OAR 345-26-330 will continue to be in effect, and it will continue to require Council approval.

Section 7.6 of the SAR describes PGE's plans for radiological monitoring. PGE plans to monitor for direct radiation with quarterly data collection, with surveys for contamination as needed. OOE concurs that the ISFSI will produce no effluents during normal storage. Radiological surveys can adequately detect any radiological releases following accidents or natural events. PGE's ISFSI plans in no way interfere with performance of its current effluent monitoring plan for decommissioning. Therefore the proposed ISFSI will not adversely affect PGE's ability to comply with the Council's effluent monitoring requirements.

<u>Environmental Radiological Monitoring - OAR 345-26-370(2)(e):</u> *The plan must contain a program for radiological monitoring to ensure the environment is not being adversely affected. This program may be incorporated by reference if it has previously been approved by the department.*

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[90] PGE letter VPN-074-98 from S.M. Quennoz to David Stewart-Smith, December 3, 1998

PGE complied with this rule by submitting its Radiological Environmental Monitoring Program and Offsite Dose Calculation Manual, in accordance with OAR 345-26-330. The "plan" referred to in this rule is the Decommissioning Plan, not the SAR. As stated above, the details of PGE's monitoring plans may change, subject to EFSC approval, as site conditions change. The SAR states that PGE will monitor radiation from the ISFSI by using TLDs with quarterly data collection. OOE finds this sufficient, as discussed in section IV.C of this report. PGE's plans for ISFSI monitoring do not interfere in any way with PGE's performance of its current radiological monitoring plan for decommissioning. OOE therefore concurs that the proposed ISFSI would not adversely affect PGE's ability to comply with EFSC's Environmental Radiological Monitoring requirements.

<u>Removal or Control of Hazardous Wastes - OAR 345-26-370(2)(f)</u>: *The plan must contain provision for removal or control of hazardous wastes that are consistent with applicable federal and state regulations.*

The Decommissioning Plan contains provisions for hazardous waste removal. OOE reviewed those provisions as part of our review of the Decommissioning Plan, and found them acceptable. Any hazardous materials used in the construction of the ISFSI would be handled and disposed of in accordance with the provisions of the Decommissioning Plan. OOE concludes that the ISFSI will not adversely affect PGE's ability to properly dispose of hazardous wastes.

In summary, OOE has determined that the proposed ISFSI would not adversely affect PGE's ability to comply with any of the requirements of OAR 345-26-370(2)(a) through (f), or with recently adopted criteria at OAR 345-026-0370(2)(h).

B. COMPLIANCE WITH OAR 345-26-390

In November 1995, the Council amended OAR 345-26-390, which sets forth general criteria that any Spent Fuel Storage installation must meet. OOE's conclusions regarding PGE's proposed ISFSI and its compliance with those criteria are as follows:

<u>345-26-390(1) Purpose:</u>

(a) Storage of spent nuclear fuel and related radioactive material and waste at a nuclear power plant is an interim measure; otherwise utilities and residents of Oregon would face the financial burden of maintaining, operating, and safeguarding the on-site storage facilities indefinitely;

(b) The purpose of this rule is to cooperate with the federal government in accordance with Oregon's siting policy in ORS 469.310 to ensure the safety of interim on-site storage and to ensure spent nuclear fuel and related radioactive materials and waste will not be an undue financial burden to utilities or people of Oregon.

The ISFSI is consistent with this rule. It is designed as interim storage. As stated in the Trojan Decommissioning Plan, the ISFSI will include the capability to transfer spent fuel from a storage cask to a shipping cask for shipment directly to an offsite repository. Moreover, under the terms of the "Standard Contract for Disposal of Spent Nuclear Fuel and/or High Level Radioactive Waste" executed between PGE and the U.S. Department of Energy (USDOE), DOE has the responsibility, following commencement of operation of a repository, to take title to and possession of the spent fuel and high level waste as expeditiously as practicable upon the request of PGE.

At this time, there is no Federal repository, and no one can predict with certainty when there will be. However, the PGE proposed ISFSI is clearly designed as an interim measure, and is designed to facilitate shipment offsite. PGE has selected a cask that is designed for shipment as well as storage. The vendor, SNC, has applied to the U.S. Nuclear Regulatory Commission (NRC) for a license to use this design as a shipping container under 10 CFR 71. Should the NRC not issue SNC's 10 CFR 71 certification by the time PGE begins cask loading, PGE has committed to investigating alternatives, such as the availability of other licensed shipping casks, prior to cask loading (PGE 2/6/97 letter CPY-001-97 from Paul Yundt to D. Stewart-Smith, Response to RAI question 24). The likelihood that SNC will not eventually receive a 10 CFR 71 shipping license is small. Nonetheless, OOE required PGE to state possible contingency actions. PGE has committed to having a procedure for fuel unloading, in case transfer to different baskets is required. If the Spent Fuel Pool has been decommissioned and the SNC baskets are still not licensed for shipping, then PGE states that a dry fuel transfer technology is available.[91] This would allow PGE to transfer the fuel to baskets approved for shipping. If SNC has not received its 10 CFR 71 shipping license by the time transfer of all spent fuel from the pool to the ISFSI is complete, OOE proposes a rule requiring PGE to provide the Council with alternative shipping plans, including cost estimates. We conclude that the proposed ISFSI meets part (1) of this rule.

<u>OAR 345-26-390(2) Capacity and Safety Standards</u>: *Storage of spent nuclear fuel shall be limited to a maximum of 791 complete and partial fuel assemblies; and storage of containers with nuclear fuel materials. Storage of spent nuclear fuel and related radioactive material and waste not eligible for disposal as low-level radioactive waste at a land disposal site (as defined in 10 CFR 61 in effect on June 15, 1995, herein after referred to as "Greater than Class C waste") at the site of a nuclear power plant by a Site Certificate holder which has executed a contract with the United States of America pursuant to the Nuclear Waste Policy Act, shall be deemed a permitted use of the site pending transfer of spent nuclear fuel to the U.S. Department of Energy provided that:*

(a) Storage facilities are designed to maintain discharges within the limits specified in applicable licenses authorized under the Atomic Energy Act of 1954, as amended, and permits under the National Pollutant Discharge Elimination System;

(b) Storage facilities are designed such that in case of accidents off-site radiation exposures will not exceed the Environmental Protection Agency Protective Action Guidelines (October 1991) for off-site protective actions; and

(c) The facility may not be used to store any spent nuclear fuel or radioactive materials and wastes other than that generated or used in the operation of the facility.

The SAR, Section 3.1.1, states PGE would store 780 intact fuel assemblies, 10 partial assemblies, and one storage container with intact and failed fuel rods. This is consistent with the 791 assemblies allowed by this rule.

The ISFSI is designed to limit discharges, as this rule requires. As described in the SAR the fuel would be placed in sealed containers that are designed to ensure that no radioactive gas, liquid, or solid waste effluents are released from the ISFSI during storage. OOE has reviewed the cask design, which confines the spent fuel in welded steel canisters ("baskets") which are pressure tested to ensure a leak-tight seal and designed to ensure no damage during off-normal events. In the unlikely event that a cask is damaged, the design provides for an overpack that would provide a leak-tight container for the basket. OOE concurs that the ISFSI is designed to prevent solid, liquid and gaseous discharges. This satisfies subpart (a) of this rule.

OOE also reviewed the projected off-site radiation exposures following anticipated accidents and natural events. We concluded that PGE has selected and analyzed the worst credible accidents, and that the

[91] PGE letter VPN-065-98 from S.M. Quennoz to David Stewart-Smith, October 21, 1998

projected consequences of those accidents would bound any anticipated accident or natural event at the ISFSI. We also concluded that PGE's dose projections were performed using appropriate methods and conservative assumptions. In all cases, the projected exposures from the worst case accidents were well under the limits of the EPA's Protective Action Guidelines for off-site protective actions. We further found that the accident with the greatest projected exposure ("bounding accident") was a non-credible event (a purely hypothetical event with no known means of happening.) Section IV of this report discusses our review of PGE's accident analyses and dose projections in greater detail. The largest credible accident is damage to the cask due to a tornado driven missile. Radiation exposures to the public following such an event would be a small fraction of the EPA PAG's. We conclude that the proposed ISFSI is designed so that the radiation dose at the Controlled Area boundary from any accident or natural event would be less that the EPA Protective Action Guidelines for off-site protective actions, as required by subpart (b) of this rule.

PGE will not store any radioactive material other than that generated during Trojan plant operations. This satisfies subpart (c) of this rule.

(3) Approval of Alternative Spent Nuclear Fuel Storage:

Spent Nuclear Fuel shall be stored only in the Trojan Spent Fuel Pool (SFP) or in an interim storage facility approved by the Council. Storage of spent nuclear fuel in any facility other than the Trojan Spent Fuel Pool shall require the prior adoption of rules by the Council allowing the specific type of proposed facility.

PGE has complied with this rule by submitting the SAR for Council review and by requesting a rulemaking to allow construction. Loading of fuel into the dry storage casks will be contingent on adoption of new EFSC rules.

(4) The Council may approve by rule a plan by the owner for storage of spent nuclear fuel or other related radioactive materials and wastes in an interim storage facility other than the SFP, and in doing so the Council may impose criteria in addition to those set forth in this rule. After approval of any such proposal the nuclear installation operator may proceed with movement of spent nuclear fuel and related materials and waste from the Trojan Spent Fuel Pool to the approved interim storage facility. Any such plan must address the design and operation of storage casks and meet the criteria in subsection (2) of this rule and the criteria below:

The plan required by this rule is the SAR, PGE 1069. The SAR does address the design and operation of the proposed storage casks, and compliance with OAR 345-26-390 part (2) is discussed above. OOE has proposed additional rules to ensure safety, as described in Section VII of this report.

(a) A proposal for an interim spent fuel storage installation (ISFSI) facility, including casks used for holding spent fuel and other radioactive materials and wastes, other than the Trojan Spent Fuel Pool shall include a safety analysis and report identifying the specific accidents considered in the design of the facility and demonstrating compliance with criteria (2)(a), (b) and (c) above.

The SAR meets this rule because it does analyze the consequences of specific accidents and natural events. The accidents and events are identified and analyzed in Section 8 of the SAR. OOE reviewed the accident analysis and concurs that PGE has identified and analyzed a complete and bounding set of possible events, including seismic events, severe weather, mishandling events, and loss of air cooling from all causes. As described above, the safety analyses in the SAR demonstrate compliance with the safety criteria of OAR 345-26-390 subparts (2)(a), (2)(b) and (2)(c). Chapter IV of this report describes OOE's review of the safety analysis in detail.

(b) The accident analysis shall include a Seismic Margin Event based on the "Seismic Margin Earthquake Study for the Trojan Site," submitted by PGE to the U.S. Nuclear Regulatory Commission and the Oregon Department of Energy on May 27, 1993. The facility shall be designed such that in the event of the Seismic Margin Earthquake, anticipated damage to spent nuclear fuel or containers will not preclude acceptance of spent nuclear fuel and related radioactive material at a Federally licensed disposal or storage facility, or release spent nuclear fuel, particulate matter or Greater Than Class C waste into the environment.

The PGE Safety Analysis Report does include an analysis of the consequences from the "Seismic Margin Earthquake" (SME) defined in the May 1993 report. The SAR demonstrates that there would be no damage to the fuel, the basket, or the concrete cask as a result of the ground motions predicted from the worst case earthquake. The stresses on these components due to the worst case earthquake are much less than the forces analyzed for other events, such as crane mishandling events. The SAR also demonstrates that the cask would not tip over in a worst case accident. We reviewed PGE's demonstration that the cask would not tip over during the Seismic Margin Earthquake, and we concur. Moreover, PGE has shown by analysis that the cask would not be damaged if it did tip over, and PGE has redesigned the casks and storage pad to preclude damage in a tipover event. OOE concludes that the SAR meets this requirement.

(c) The facility shall be designed such that in the event of the Seismic Margin Earthquake or any accident considered in the safety analysis required by subpart (a) of this rule, projected radiation exposure rates due to effluents and direct radiation shall not exceed the Environmental Protection Agency Protective Action Guidelines (October 1991) for off-site protective actions outside the interim storage facility controlled area as defined in 10 CFR 72.106 (June 15, 1995). The plan for the interim spent fuel storage facility shall demonstrate the capability to restore post-accident radiation exposure rates outside the interim storage facility controlled area to the levels permitted during normal facility operations.

Chapter 8 of the Safety analysis describes the accidents PGE considered in compliance with subpart (a) of this rule, and Chapter IV.E of this report describes OOE's review of those accident analyses in detail. PGE has shown that no credible accident would release radiation or contamination in excess of the EPA's Protective Action Guidelines for offsite protective actions.

The EPA Protective Action Guidelines are also used in support of PGE's emergency plan, as approved by the Council. The EPA protective action guidelines for purposes of evacuation and sheltering during a nuclear accident are:

- 1 rem Total Effective Dose Equivalent (TEDE)**,**
- 5 rem Committed Dose Equivalent (CDE) to the thyroid, and
- 50 rem CDE to skin.

These dose limits are for dose received either by direct radiation or by inhalation. When there is a possible ingestion pathway, PGE and EFSC have used a more restrictive 500 mrem limit for dose to the whole body or to any organ.[92]

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[92]The terms TEDE and CDE are as defined in Title 10 Chapter 20 of the Code of Federal Regulations (10 CFR 20). The allowed skin dose is higher because radiation to the surface of the skin has much less somatic effect than radiation to the whole body.

PGE's safety analysis shows that no credible accident would cause a breach of the basket. For seismic events, the analysis shows that the cask would not tip over during the maximum credible earthquake. PGE also performed a cask tip over analysis, which concluded that if the cask did tip over, there would be local cracking and spalling of the concrete cask, and approximately ½ inch of flattening to the steel basket. Deceleration would not be enough to cause fuel damage. There would be no radiological releases or adverse radiological consequences.

PGE's analysis shows that the worst credible accident, in terms of radiological impact, is a tornado driven missile striking the cask. For this analysis, PGE assumed that a sharp nosed, 275 pound, 8 inch artillery shell strikes the cask at a speed of 126 miles per hour. PGE used empirical correlations developed by EPRI to calculate how much concrete would be removed. PGE calculated that the impact would remove 5.69 inches of concrete. The initial concrete thickness is approximately 30 inches.

There would be no release of radioactive material from such an event, because the steel basket would remain undamaged. However, direct radiation would increase because of the damage to the concrete shielding. The radiation dose rate at the cask surface resulting from the loss of 5.69 inches of concrete shielding is 286.5 mrem/hr for a cask loaded with spent fuel , compared to original rate of 19.1 mrem/hr. PGE calculated that the dose rates at a distance of 100 meters are minimal for the duration of the event. PGE has concluded, and we concur, that the dose rates from the worst case credible accident are within the EPA Protective Action Guidelines.

The SAR also describes the recovery from such an event. PGE would repair the cask by filling the damaged area with grout. PGE has committed to maintaining shielding materials onsite for use in mitigating the direct radiation until repairs are completed. PGE estimates that it could procure the necessary repair materials within 12 hours, and that it could perform the repair in about 30 minutes once the materials are available. Occupational dose would be less than one person-rem.

At an Energy Facility Siting Council meeting in May 1997, EFSC members expressed concern over the ability to make repairs promptly, particularly after a major earthquake or volcano eruption. OOE addressed this concern by recommending a rule requiring pre-staging of repair materials and contractual arrangements for prompt repair services.

The other accidents analyzed in the SAR had little or no safety consequences to the public. These accidents included:

- mishandling of the basket inside the concrete cask,
- loss of air cooling due to blockage of air vents,
- explosion of commercial cargo on nearby rail and barge shipments,
- natural events such as flood and volcano, and
- dropping the cask in preparation for shipping, or while installing an overpack

In all cases, the Safety Analysis showed that the basket would not be breached or damaged, so that there would be no release of radioactive material. For events involving a basket drop, the analysis showed that the forces on the basket would be less than the allowed 124g (for vertical forces) or 44g (for horizontal forces). The SAR also shows that the concrete cask would not be damaged, so that there would no increase in direct radiation at the site boundary. Therefore, radiation doses would already be at pre-event levels.

OOE reviewed PGE's analyses of these events and concurs with PGE's assessment. Our review and conclusions are described in greater detail in Section IV of this report. We further concur that the SAR demonstrates the ability to restore post accident radiation levels to those allowed for normal operations.

(d) The site of the interim spent fuel storage facility shall be selected such that the expected ground motion in a seismic margin event is bounded by the accident analysis required by section 4(b) of this rule. The safety analysis report shall include a review of the seismic margin analysis referenced in 4(b) and shall demonstrate whether the Seismic Margin Event defined above remains the appropriate design basis event for the proposed interim storage facility.

PGE will locate the ISFSI at the northeast corner of the Trojan Site, about 100 meters from the existing Spent Fuel Building. PGE included a detailed discussion of site geology in their original Final Safety Analysis Report for plant operations. Studies on site geology and on the expected ground motion at the site during a worst case earthquake have been regularly updated and reviewed by the EFSC Earthquake Subcommittee.

The SAR describes the drilling program which PGE used to characterize the rock underlying the Trojan site. The SAR describes a program of 59 diamond drill and soil sampling holes totaling about 5200 feet. The records of the location and results of this program show that the rock under the proposed ISFSI site is essentially the same bedrock found under the Fuel Building and Reactor Building. PGE concluded, and we concur, that the expected ground motions at the proposed ISFSI site during a Seismic Margin Earthquake are bounded by the analysis in PGE's May 1993 report, "Seismic Margin Earthquake for the Trojan Site." The peak horizontal ground acceleration predicted in that study was .38 g.

PGE has since updated the Seismic Margin Earthquake study to include data from the 1994 Northridge, California, earthquake. The new data increases the contribution to peak ground acceleration from crustal events. However, the maximum expected acceleration of 0.38g did not change. Therefore, PGE has concluded that the Seismic Margin Earthquake described in PGE's 1993 remains the appropriate design basis for the ISFSI.

OOE reviewed PGE's study in consultation with DOGAMI. DOGAMI's recommendations are included. DOGAMI pointed out that more recent seismic work in this area has refined the conclusions in PGE's 1993 study. However, DOGAMI concurred with PGE's overall conclusion that peak acceleration of .38g is a conservative design basis for the ISFSI site.

(e) Radiation and effluent monitoring programs, security plans, and emergency plans for an interim spent fuel storage facility shall be maintained in accordance with OAR 345-26-330, 340, and 350.

PGE currently has approved programs for effluent monitoring, security, and emergency preparedness in accordance with OAR 345-26-330, 340 and 350. Those programs will remain in effect for the duration of the activities described in PGE's approved Decommissioning Plan (PGE 1061), with reductions in scope subject to EFSC approval.

PGE will perform fuel loading operations under its existing Federal license, in accordance with 10 CFR 50. Therefore, the current EFSC approved plans for radiological and effluent monitoring, emergency planning and security will apply.

Once the fuel is in storage, PGE may modify or reduce the scope of one or more of these plans. PGE has submitted an ISFSI emergency plan for OOE review. (PGE letter VPN-024-97, dated April 3, 1997, from Steve Quennoz to the U.S. Nuclear Regulatory Commission). The Council approved the ISFSI Emergency Plan in November, 1997.

PGE has also submitted an ISFSI security plan for review by EFSC and the NRC. EFSC reviewed and approved the ISFSI security plan in August, 1997.

The SAR states that no radioactive gas, liquid or solid effluents are released from the ISFSI during operation. Therefore, PGE has not proposed routine monitoring for effluents. PGE has based this assessment on the use of steel storage baskets which are sealed, welded shut and pressure tested to ensure a leak-tight seal. Further, the SAR shows that the steel baskets would not breach or leak following any credible accident or natural event. The SAR includes a commitment to routinely monitor the cask inlet and outlet vents for contamination. In the unlikely event that contamination is found, then PGE has committed to investigating for further contamination outside the ISFSI restricted area. PGE will post Thermoluminescent Detectors (TLDs) at the perimeter of and in the Controlled Area near the storage casks. PGE will read these quarterly to monitor radiation levels in the nearby vicinity of the ISFSI. Any unusual quarterly readings would also indicate the unanticipated release of radioactive material, and would require investigation.

We concur with PGE's assessment that the ISFSI will not release effluents during normal storage, and we recommend rules to codify PGE's commitments regarding monitoring for contamination at the cask vents and TLDs at the perimeter of the ISFSI site.

(f) In the absence of any accident considered in the safety analysis required by part (a) of this rule, activities related to transfer of spent fuel or other reactor components from the Spent Fuel Pool to an interim storage facility and subsequent storage and fuel handling activities will not result in anticipated annual radiation dose due to effluents to any member of the public in an unrestricted area to exceed 5 mrem Total Effective Dose Equivalent (TEDE) as defined in 10 CFR 20.1003 as of March 1, 1994. The plan shall provide an estimate of the quantity of the radionuclides expected to be released annually to the environment in liquid and gaseous effluents during normal operation of the ISFSI.

As stated in the discussion of subpart (e) above, the SAR says the ISFSI will release no radionuclides to the environment in liquid or gaseous form during storage operations. We concur with that assessment.

Fuel transfer activities include:

- Placing an empty transfer cask and basket into the Cask Loading Pit,
- Loading the basket with spent fuel,
- Moving the transfer cask and loaded basket to the Cask Wash Pit
- Draining and vacuum drying the basket and welding of the shield lid,
- Pressure testing the basket to ensure an airtight seal, and
- Transfer of the sealed basket to a concrete cask for storage at the ISFSI pad.

None of these activities are expected to release radioactive material. PGE's LCA-237 describes the precautions PGE will take to ensure no release of radioactive material during these steps. Fuel loading operations will take place under at least 10 feet of water. The transfer cask will be washed down upon removal from the Cask Loading Pit to prevent the spread of contamination. PGE will survey the basket, transfer cask and concrete cask before moving the concrete cask out of the Fuel Building to ensure that contamination is not spread to the ISFSI pad. The draining and vacuum drying process for the basket will use filtration equipment to minimize airborne contamination. All fuel loading and basket preparation will be performed in the Fuel Building, which has a monitored ventilation exhaust to ensure that any unexpected radiation release is quantified.

We have reviewed these precautions and concur that they preclude release of radioactive material during fuel transfer activities. This satisfies the requirement to ensure that no member of the public in an unrestricted area receive a radiation dose of 5 mrem TEDE due to effluents, in the absence of a seismic event or accident.

The only source of effluent release during storage operations would be residual contamination on the surface of the basket. In Chapter 8 of the SAR, PGE calculated the dose consequences from an average residual contamination level of 1x E-4 $\mu Ci/cm^2$ on the surface of the baskets. The calculations show that a complete release of all contamination from all 34 baskets would cause a maximum one time dose of 2.4 mrem to a hypothetical person standing 100 meters from the point of release. The analysis does not take credit for the fact that the steel baskets are contained in concrete casks, so that actual releases would likely be lower than analyzed. Also, PGE's SAR describes contamination of 1.0×10^{-4} as a hold point for further evaluation, not as an absolute limit. Therefore, OOE recommends a requirement that PGE document the evaluation criteria in approved procedures prior to fuel loading.

With this qualification, we conclude that the ISFSI will not cause a radiation exposure greater than 5 mrem TEDE to any person outside a restricted area during normal fuel handling and storage operations.

(g) Transfer of spent fuel or other reactor components to a temporary storage facility shall not adversely affect the owner's financial ability to decommission the Trojan site, including the interim storage facility site after the Federal government has accepted high level waste at a Federally licensed disposal facility.

PGE included the costs of building and maintaining an ISFSI in their Decommissioning Plan (PGE-1061), as approved by EFSC on March 11, 1996. In OOE's "Review of PGE's Decommissioning Plan for the Trojan Plant," dated January 22, 1996, we found PGE's cost estimates acceptable. We further determined that ...

"the plan does include an estimate of the funding necessary to complete decommissioning and it appears adequate based on the decommissioning cost estimate. Furthermore, the department recognizes the Oregon PUC's analysis and approval of the funding plan presented in the decommissioning plan."

OOE's Review of the PGE Decommissioning Plan, page 72, shows that PGE estimated $44.7 million for ISFSI construction and decommissioning, and $65.6 million for ISFSI operation and maintenance through the year 2018. PGE listed the costs of ISFSI construction and operation on a year by year basis in Table 5.1-2 of the Decommissioning Plan. All of the decommissioning cost estimates reviewed by EFSC in connection with the Decommissioning Plan were based on PGE's stated intention to transfer the fuel to an ISFSI.

Chapter 9 of the SAR discusses the eventual decommissioning of the ISFSI. PGE states that decommissioning the ISFSI is projected to cost approximately $9.7 million (1997 dollars).

The SAR breaks these costs down further on Table 9.8-1. We reviewed this breakdown with a member of PGE's accounting staff and verified that PGE developed these costs using conservative assumptions about personnel requirements and burial costs. For example, PGE assumed that the steel liners which will line the inside of the concrete casks will be buried as Low Level Waste. This is a conservative assumption, because, barring some failure of the basket, these liners should not be contaminated to the point where they cannot be cleaned and surveyed for free release. We concur with PGE's assessment that $9.7 million is a reasonable estimate of ISFSI decommissioning cost.

Table 5.1-2 of the Decommissioning Plan estimates the cost of ISFSI Decommissioning at $18.5 million for ISFSI decommissioning. This is part of the overall $425 million estimate that PGE provided for total decommissioning cost, and was part of the scope of costs included in the 1995 PGE rate case before the

Oregon Public Utilities Commissioning (OPUC). OOE noted that this is larger than the amount estimated in the ISFSI SAR. The reason for the difference is that the Decommissioning Plan originally assumed that it might be necessary to dispose of GTCC waste.

The Oregon PUC approved recovery of these costs in its order of March 29, 1995, as discussed on p. 73 of OOE's Review of the Decommissioning Plan. We note that the costs associated with construction, operation and eventual decommissioning of the ISFSI were an integral part of the PUC's 1995 rate order. And, we concur with PGE's assessment that $9.7 million is sufficient, considering the cost savings from the reduction in GTCC waste due to the one-piece Reactor Vessel shipment. We therefore conclude the ISFSI will not have an adverse affect on PGE's ability to decommissioning the Trojan Site.

(h) Activities related to transfer, storage and handling of fuel and other radioactive waste shall be performed in accordance with a radiation protection program which complies with 10 CFR 20 (effective March 1, 1994), including a program to maintain personnel radiation exposure As Low As Reasonably Achievable (ALARA) as that term is defined in 10 CFR 20.

PGE describes its radiation protection program in Section 7 of the SAR. The SAR describes the features of the spent fuel storage system which are designed to minimize occupational exposure. A partial list of these features includes the use of non-planar paths for air inlet and outlet vents to minimize radiation streaming, multiple pass welds on all seal closures to provide redundant radioactive material containment, and arrangement of loaded fuel casks on the storage pad such that the more radioactive casks are shielded by less radioactive ones. A key feature is the absence of any mechanical equipment, which minimizes the need for workers to be near the spent fuel casks for extended periods of time. Section IV.C of this report discusses PGE's radiation protection measures in more detail.

OOE notes that the PGE radiation protection program, as implemented in these procedures, is largely the same program in effect during power plant operations. PGE has implemented certain changes as mandated by new NRC regulations at 10 CFR 20. Occupational dose limits allowed by plant procedures are lower than those permitted by 10 CFR 20. PGE has documented its program to maintain personnel exposures ALARA in procedure TPP 14-11. OOE's conclusion that PGE maintains a radiation protection program that complies with 10 CFR 20 is supported by NRC inspection reports. Fuel handling activities are already performed in accordance with PGE radiation protection procedures, as they have been throughout plant operations.

In addition to program controls, the ISFSI has design features to reduce occupational exposure. For example, PGE has installed the capability for remote temperature monitoring. This is not required by any federal rule, but it is clearly consistent with the ALARA principle. Since this capability is neither required by federal rules nor is it described in the SAR, OOE proposes a rule requiring this capability, with local temperature monitoring used as backup.

OOE concurs with PGE's assessment that its radiation protection program implements the requirements of 10 CFR 20, including the requirement to maintain personnel exposures ALARA.

(i) Any temporary storage facility shall not adversely impact the potential for unrestricted use of the site, including the storage facility site, after decommissioning, or the ability of the site certificate holder to comply with the standards of OAR 345-26-370(2)(a) through (f), nor shall it excuse the site certificate holder from any rules of the Council in OAR Chapter 345.

Section VI.A of this report discusses the ISFSI's impact on decommissioning, and concludes that the ISFSI would not adversely impact PGE's ability to comply with OAR 345-26-370(2)(a) through (f), or with the recently adopted criteria in OAR 345-026-0370(2)(h).

Section 9.6 of the SAR discusses plans to eventually decommission the ISFSI. As described in Section 9.6.3 of the SAR, the ISFSI is designed to minimize the decontamination efforts required for decommissioning. The use of sealed, leak-tight stainless steel PWR baskets is expected to preclude contamination of any exterior storage components. As discussed in Section IV of this report, OOE has reviewed the structural integrity of the basket and concurs with PGE's assessment that the basket has the necessary structural integrity to withstand natural events such as earthquakes or tornadoes. Section 8.1.3 of the SAR contains a limit of 10^{-4} $\mu Ci/cm^2$ beta-gamma contamination on the exterior surface of the basket. This is a very low level of contamination and should preclude gross contamination of the concrete cask. The concrete cask is further protected from contamination by a steel liner. It is likely that any contamination of the steel liner could be removed by surface cleaning. However, PGE assumed that the steel liners may become slightly activated, and provided for burial of all steel liners in its approved decommissioning fund.

Once the fuel and casks are removed, PGE has committed to performing a final radiological survey of the ISFSI site, confirming that the site meets Federal and State standards for unconditional release. Therefore, the ISFSI will not adversely affect PGE's ability to restore the site for unrestricted use.

(j) A spent fuel storage facility other than the Spent Fuel Pool shall have a minimum design life of 40 years. The plan for an interim spent fuel storage facility shall demonstrate that the interim storage facility will perform as designed for the required 40 year life and shall describe all testing of storage equipment and materials during design and fabrication. The plan shall discuss the options available if the expected lifetime is reached and no Federally licensed permanent disposal or storage facility is available.

Section 3.3.7.1 of the SAR commits to a minimum design life of 40 years. Further, in its June 27, 1997, response to OOE questions, PGE stated that the vendor, Sierra Nuclear Corporation (SNC) warranted the design for 50 years. PGE has identified the critical components which must remain functional for 40 years, and the degradation methods which would limit component life. Features to increase design life include the stainless steel basket, corrosion resistant coatings which were tested under harsh conditions, temperature limits on the concrete and concrete mix specifications which meet American Concrete Institute (ACI) guidelines, and the inert helium atmosphere in the basket. The steel basket will be shielded from weather by the concrete cask, and will be seated on ceramic tiles inside the cask to prevent metal to metal contact with the steel liner. As stated in Section 4.6 of the SAR, the ISFSI is a dry, above ground system so that cathodic protection in the form of impressed current is not required. The normal operating temperatures are well above ambient air temperatures, so there is no opportunity for condensation on any surfaces. Section IV.A.5 of this report "Design Lifetime" discusses the design lifetime in further detail.

The SAR provides information on testing. Table 4.2-1 of the SAR describes fabrication specifications for the basket and overpack. It specifies that weld will be liquid penetrant or magnetic particle tested and radiographed in accordance with Section V of the ASME Code. Welds performed on the basket lids will be penetrant tested. All baskets will be pressure tested with helium and leak checked to ensure a leak-tight seal. As stated in section 9.2.2 of the SAR, materials used for radiological shielding (such as concrete) will be checked for density. At the NRC's request, corrosion resistant coating materials were laboratory tested and NRC approval of the coating materials was contingent on the test reports.

Section 9 of the SAR also describes operational tests, including testing of transfer cask trunnions at 300% of the design load and testing of the lifting yoke at 150% of design load. PGE will also perform a full scale mock up test of fuel loading procedures, using dummy fuel assemblies. The OOE inspector witnessed similar testing at the Arkansas Nuclear One plant in May 1996 and concurs that it effective tests fuel loading and cask preparation equipment and procedures.

In response to an OOE RAI question, PGE provided supplemental information concerning the options available if no Federally licensed permanent fuel storage facility is available at the end of the ISFSI design life. These options are described in Section IV.A.5 of this report. OOE concurs with PGE's assessment of these options.

(k) To the extent feasible, an interim spent fuel storage facility shall be designed to minimize spent nuclear fuel handling. The plan for an interim spent fuel storage facility shall include the ability to transfer spent nuclear fuel from the interim spent fuel storage facility to a shipping container. Except as required for accident mitigation as described in the Safety Analysis Report, transfer of spent fuel from an interim spent fuel storage installation to new casks or shipping containers must be approved by the Council prior to their removal.

The ISFSI is designed to minimize spent nuclear fuel handling, and is designed to facilitate transfer to shipping containers with minimal handling. Section 3.1.2 of the SAR states that after Concrete Casks are loaded, the only operations will be surveillances to ensure the air vents are not blocked and that outlet air temperature remains acceptable. The loaded baskets are designed to be transferred from the concrete cask to a shipping cask without opening the basket or exposing the fuel. In the unlikely event that the basket is damaged due to accident or natural event, the fuel would not be unloaded, but the entire basket would be placed in a sealed overpack. The ISFSI also includes a transfer station designed to transfer fuel baskets from one concrete cask to another, or from a concrete cask to a shipping container, without fuel handling.

OOE notes that the vendor, SNC, has submitted a Safety Analysis for the TranStor™ Shipping Cask System to the NRC for review under 10 CFR 71. If approved, the basket will be Federally licensed for both shipping (under Part 71) and storage (under Part 72). We conclude that the TranStor™ system is designed to minimize spent fuel handling to the extent feasible.

Conclusion

Based on the information provided in the SAR, the responses to OOE questions, the supporting documentation, PGE License Change Applications, and the conditions recommended for incorporation into EFSC rules, the proposed ISFSI would meet all of the requirements of OAR 345-26-390(1) through (4). Part (5) of the rule is not a construction or operation requirement but a reporting requirement.

We therefore recommend the adoption of rules allowing implementation of the ISFSI as described in the SAR, and of rules specifying any additional requirements for public health and safety.

VII. CONCLUSIONS

A. SUMMARY

Based on review of the SAR and associated references, we conclude that the proposed ISFSI is designed with adequate structural integrity, heat removal capability, and radiation protection. We further find that the ISFSI is designed to ensure these elements under normal and accident conditions. As described in section V of this report, we conclude that the loading process described in PGE LCA-237 can be performed safely. As described in Section VI of this report, we find that the proposed ISFSI meets the EFSC requirements in OAR 345 Chapter 26.

OOE's recommendation to approve the ISFSI is based on certain PGE commitments and assumptions identified in the SAR and associated correspondence. We therefore recommend a proposed rule, approving the ISFSI subject to the following requirements:

B. PROPOSED ADDITION TO RULE OAR 345-026-0390

Spent Nuclear Fuel Storage

(6) PGE may construct and operate an Independent Spent Fuel Storage Installation (ISFSI) substantially as described in the ISFSI SAR (PGE 1069) and License Change Application LCA-237. PGE may transfer spent fuel from the Trojan Spent Fuel Pool to the ISFSI, subject to the following:

(a) Controlled Area Boundary: Within six months of terminating the NRC operating license under 10 CFR 50, PGE shall provide evidence satisfactory to the Council that it has the ability to evacuate unnecessary personnel from the Controlled Area boundary as described in the SAR. This evidence may include such factors as any lease or contractual agreements with tenants at the site or the results of an evacuation drill.

(b) Capability for Remote Temperature Monitoring: To the extent practical, PGE shall implement design features that minimize the time necessary for ISFSI personnel to be in close proximity to the casks while performing required surveillances. These features should include the capability for remote temperature monitoring, with hand-held temperature measuring devices used only as backup for the installed devices.

(c) Radiation Survey Plan During Loading: Prior to loading the first basket, PGE shall provide OOE with a radiation dose rate monitoring program including successive measurements for areas of the site outside the ISFSI Access Controlled Area and within the Industrial Area, including the Trojan Central Building, to be performed during ISFSI loading through placement of all casks on the pad. The results of these surveys shall be provided to the Council within six months of placing the final cask on the ISFSI pad.

(d) Radiation and Environmental Monitoring after Decommissioning: Upon termination of the NRC operating license under 10 CFR Part 50, PGE may terminate the Radiological Effluent Monitoring Program and Radiological Environmental Monitoring Program only after a finding by the Council that appropriate programs have been implemented to monitor the radiological impact of the ISFSI.

(e) Staging of Emergency Equipment: Equipment required for credible accident scenarios should be maintained onsite. Equipment to unblock the air inlets and outlets in the event of a volcanic eruption should be maintained onsite or available within 24 hours of notification that a significant ashfall from a volcanic eruption is likely to occur.

(f) PGE shall demonstrate, at an interval acceptable to the Council, that contractual arrangements are in place to ensure that new concrete casks could be assembled on a timely basis in the unlikely event of severe damage to one or more casks. PGE shall also demonstrate that necessary construction materials are either staged at the site or can be quickly procured by the company.

(g) Unloading Procedure: Prior to loading the first basket, PGE shall have available approved procedures and supporting analyses to allow removing the spent fuel from a previously loaded basket and safely returning the fuel to the SFP, as described in its October 21, 1998, letter from S. Quennoz to D. Stewart-Smith.

(h) Shipping License: If a 10 CFR 71 shipping license for the TranStor™ cask has not been received at the time the final fuel assembly is transferred from the SPF to the ISFSI, then PGE shall provide the Council with alternative plans and cost estimates for shipping spent fuel offsite prior to taking action that would preclude future use of the SFP.

(I) Changes to Commitments: PGE may make changes to equipment or administrative controls described in the SAR, LCA-237 ,or Technical Specifications, or implemented in response to concerns raised in OOE correspondence, without prior notification to OOE, unless the proposed change:

(i) May render invalid prior findings of compliance with any of the requirements at OAR 345-026-0390(4) or (5), or

(ii) Would significantly reduce the scope, surveillance frequency or effectiveness of administrative controls described in the SAR or implemented in response to specific concerns raised in OOE correspondence.

Proposed changes that meet the foregoing criteria shall be submitted to OOE prior to implementation. This notwithstanding, PGE may make changes to ISFSI equipment or procedures without prior OOE concurrence if required for compliance with the regulations or orders of the U.S. Nuclear Regulatory Commission or required to protect the health and safety of the public.

PGE shall maintain records of changes to ISFSI equipment or procedures made without prior OOE concurrence pursuant to this rule. These records shall include a written evaluation that provides the basis for the determination that the change does not meet the foregoing criteria. A report of such changes, and changes made under 10 CFR 72.48, shall be transmitted to OOE annually, on a schedule consistent with required reports to the NRC under 10 CFR 72.48 as in effect January 1, 1999.

(j) Reporting Requirements: Within one year of the first cask loading and annually thereafter, PGE shall provide a written report to the Council on the status of the ISFSI. The report shall include, as a minimum, results of radiation monitoring programs, a summary and breakdown of personnel exposure related to ISFSI maintenance and surveillance activities, a statement of expenses related to maintenance and surveillance activities an estimate of funds available for continuing ISFSI maintenance and surveillance, and a statement of any significant developments regarding the opening of a Federally licensed High Level Waste facility. Other reports submitted pursuant to OAR 345-26-380 may be used to satisfy this requirement.

(k) Structural Inspection Program: PGE shall implement a Structural Inspection Program that establishes acceptance criteria, degradation evaluation methods, repair instructions, and periodic inspections of the surfaces of the concrete casks and the storage pad to ensure that the structural integrity of the concrete is maintained throughout the life of the ISFSI. Changes to this program that reduce its scope require prior OOE concurrence.

(l) Reliability of Temperature Monitoring Equipment: PGE shall implement programs to ensure the continued reliability of the air outlet temperature monitoring devices for the 40 year ISFSI life. Such programs shall include but not be limited to periodic calibration of electronic circuitry associated with the temperature monitoring devices, and periodic evaluation of temperature data sufficient to identify anomalous trends that could indicate degraded instrumentation.

(m) Frequency of Temperature Monitoring and Air Vent Inspections: PGE shall implement a program for thermal monitoring that includes readings of air outlet and ambient temperatures. The program shall include temperature surveillances at least once every 12 hours, with provisions for more frequent measurements if temperatures approach Technical Specification limits. The program shall also include a requirement to check air inlet and outlet vents for blockage at least once every 24 hours. PGE may reduce these surveillance frequencies with OOE concurrence.

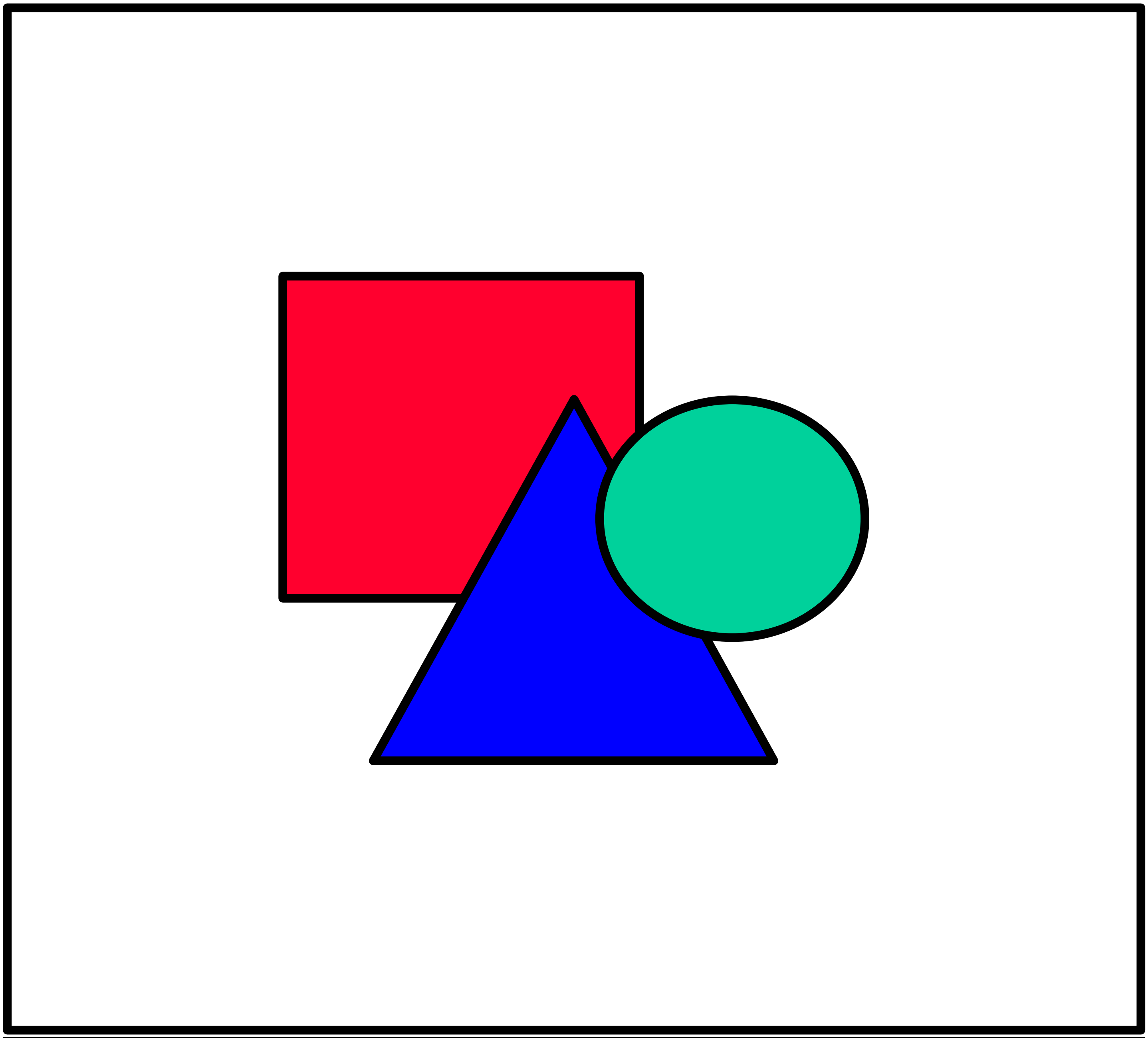
C. PROPOSED POST-APPROVAL OOE INSPECTIONS

The following is a list of proposed action items planned for OOE. It contains draft procedures to be reviewed prior to fuel loading, critical evolutions to be witnessed, and PGE commitments to verify. These items will be completed as our schedule permits.

- OOE will verify that PGE has implemented measures to ensure that temperature monitoring equipment remains functional and reliable over the 40 year design life.
- OOE will review the testing of mechanical stops and electrical interlocks on the Fuel Building Crane prior to movement of the transfer cask. OOE will also review the procedures required by NUREG-0612, Section 5.1.1, which will cover load handling operations for heavy loads that are or could be handled over or in proximity to spent fuel. (Section V.C.1.a)
- OOE plans to monitor the effectiveness of the administrative measures to control the height limit on lifting the transfer cask during fuel loading. (Section V.C.1.b)
- OOE plans to verify on a sampling basis that pre-job briefings held during the ISFSI fuel loading and cask movement include a discussion of the safe load path. (Section V.C.1.c)
- OOE plans to attend selected pre-job briefings and ALARA job reviews during the ISFSI fuel loading to review PGE's ALARA planning and response to unplanned exposures. (Section V.E.1)
- OOE plans to review procedures prior to cask loading operations to verify that they contain provisions for handling bowed fuel assemblies. (Section V.F.5.b)

APPENDIX A — ISFSI SITE PLAN

(May be visible only on the hard copy version)



APPENDIX B — GLOSSARY

ACI: American Concrete Institute. This organization provides internationally accepted codes and standards for the construction and use of concrete structures.

air pad system: A system of four pressurized air pads inserted into the air inlet vent openings under a concrete cask and used to move the casks between the Fuel Building and the ISFSI storage pad.

ALARA: As Low As Reasonably Achievable. ALARA is a philosophy of limiting radiation exposures (both public and occupational) below the limits of the regulations. The NRC defines ALARA in 10 CFR 20.1003 as "making every reasonable effort to maintain exposures to radiation as far below the dose limits in this part [of the regulations] as is practicable consistent with the purpose for which the licensed activity is undertaken, taking into account the state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to the utilization of nuclear energy and licensed materials in the public interest."

ANSYS: A software computer code used in thermal analyses of systems. It is based on a finite element model that reduces complex problems to simpler representations of the problem.

ASME: American Society of Mechanical Engineers. This organization provides internationally accepted codes and standards for components used in the power industry. These codes and standards provide authoritative technical guidelines for safety, reliability, productivity, and efficiency; and they are frequently used as a means of satisfying government regulatory and procurement requirements in the nuclear power industry.

Audit: The PGE QA Program defines audit as "An activity which determines through investigation, review, and objective evidence, the adequacy of, and adherence to, established procedures, instructions, specifications, codes, standards, or other applicable contractual and licensing requirements, and the effectiveness of implementation."[93]

basket (or PWR basket): A stainless steel cylindrical container used to store up to 24 spent nuclear fuel assemblies inside a ventilated concrete cask. The basket provides a confinement boundary and is designed to withstand credible accidents without loss of integrity. See the SAR for more details.

clad creep (strain): Creep or strain is defined as plastic deformation in metals caused by sustained loading or stress at elevated temperatures. Clad creep occurs in the cladding that surrounds nuclear fuel pellets and forms the fuel pins or rods that in turn make up a fuel assembly. The principal variables that cause clad creep to occur are temperature and the differential pressure across the cladding.

An increase in either of these increases the likelihood of clad creep.

Cladding: A metal alloy tube made of zircalloy that surrounds uranium oxide fuel pellets in a fuel assembly. 264 of these metal tubes, known as fuel pins, make up a fuel assembly of the type used at Trojan.

concrete cask: A reinforced concrete, cylindrical container used to store a basket. In addition to storage, the cask provides radiation shielding, protects baskets from postulated environmental events,

[93] PGE, PGE-8010, Nuclear Quality Assurance Program for Trojan Nuclear Plant, Rev. 20, December 12, 1996, Glossary

such as earthquakes and tornadoes, and provides passive heat transfer from a basket to the environment. See the SAR for more details.

Confinement: The ability of a container to prevent release of radioactive material into the biosphere.

Containment: The ability of a container to contain the fuel in an acceptable geometry.

Decommissioning: The process of removing a nuclear facility safely from service and reducing residual radioactivity to a level that permits release of the property for unrestricted use and termination of the license.

DOE or U.S. DOE: The U.S. Department of Energy is the organization within the Federal Government responsible for eventually taking and storing spent nuclear fuel.

DSAR: Defueled Safety Analysis Report.

EPA PAG: Environmental Protection Agency Protective Action Guide.

failed fuel can: A stainless steel container designed to provide a containment boundary for a partial or complete fuel assembly with failed or suspect rods within one of the four oversized corner storage locations of a basket. The failed fuel can allows for water drainage and vacuum drying during basket closure. See the SAR for more details.

fuel debris: fuel with known or suspected defects such as ruptured fuel rods, severed rods or lose fuel pellets and pellet fragments. Fuel debris includes fuel assembly metal fragments such as portions of fuel rods and grid assemblies.

fuel debris process can: A stainless steel container designed to provide a containment boundary for fuel debris during the high temperature steam process to remove water and organic media from fuel debris. Following this process, a process can is stored in a fuel debris process can capsule and stored in one of the four oversized corner storage locations of a basket. See the SAR for more details.

fuel debris process can capsule: A sealed, inerted, steel canister designed to provide both a containment and confinement boundary for fuel debris stored in fuel debris process cans. Up to five process cans may be stored in a process can capsule, which is then stored in one of the four oversized corner storage locations of a basket. See the SAR for more details.

g (force of gravity): A measure of acceleration in terms of the force of gravity. One g is equivalent to 32.2 feet per second squared.

gpm: Gallons per minute.

GTCC: Greater Than Class C Waste. Low Level radioactive waste is categorized as either class A, B, C, or Greater Than Class C (GTCC). The classifications are based on the concentrations of long and short-lived radionuclides present. Classes A, B, and C are suitable for disposal in near-surface disposal sites, but GTCC waste must be disposed of in a geologic repository, such as the one being developed at Yucca Mountain, Nevada.

heat load: The heat output of a fuel assembly or basket measured in kilowatts thermal, and usually expressed as kW.

high-level radioactive waste: Waste that is not eligible for disposal as low-level radioactive waste at a land disposal facility, such as the one operated by U.S. Ecology on the Hanford reservation in Washington. High-level radioactive waste generally refers to spent nuclear fuel.

hold point: A step in a procedure or activity where an inspection, surveillance, or analysis must be completed before the procedure or activity can proceed to the next step. Typically, hold points are put into procedures by QA/QC personnel to facilitate their ability to monitor the process.

important-to-safety: 10 CFR 72.3 defines important-to-safety as those features of the ISFSI whose function is: (1) to maintain the conditions required to store spent fuel or high-level radioactive waste safely, (2) to prevent damage to the spent fuel or the high-level radioactive waste container during handling and storage, or (3) to provide reasonable assurance that spent fuel or high-level radioactive waste can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

impact limiters: Rigid polyurethane foam blocks used to absorb energy from falling objects. They vary in density from 3 to 30 lbs/ft^3 depending on the energy absorption requirements of the particular accident scenario for which they are designed.

ISFSI: Independent Spent Fuel Storage Installation. A complex designed and constructed for the interim dry storage and passive natural convection cooling of spent nuclear fuel and other radioactive materials associated with spent fuel storage. See the SAR for more details.

K_{eff}: A measure of the reactivity of a system. A k_{eff} of 1.0 represents criticality, and a number less than 1.0 represents a subcritical condition.

LCA: License Change Application submitted by PGE to the NRC to request a change in an existing license.

lifting yoke: A special lifting device designed to mate with the transfer cask trunnions and provide a means for the Fuel Building Crane to lift the loaded transfer cask.

NRC or U.S. NRC: The U.S. Nuclear Regulatory Commission is the organization within the Federal Government responsible for the regulation of nuclear power plants.

OAR: Oregon Administrative Rules.

OOE: Oregon Office of Energy.

overpack (or basket overpack): A stainless steel cylindrical container used to provide containment and confinement for a basket in the event that a basket develops a leak. An overpack with an enclosed basket is then stored inside a ventilated concrete cask. See the SAR for more details.

Psia: Pressure measured in pounds per square inch absolute. The absolute pressure of air at sea level is 14.7 psia.

Psig: Pressure measured in pounds per square inch gage. The gage pressure of air at sea level is 0 psig.

PWR: Pressurized water reactor. The Trojan Nuclear Plant used a pressurized water reactor system when it operated.

Quality Assurance (QA): 10 CFR 50, Appendix B, defines quality assurance as "all those planned and systematic actions necessary to provide adequate confidence that a structure, system, or component will perform satisfactorily in service."

Quality Control (QC): 10CFR50, Appendix B, defines quality control as being part of quality assurance and comprising "those quality assurance actions related to the physical characteristics of a material, structure, component, or system which provide a means to control the quality of the material, structure, component, or system to predetermined requirements."

RAI: Request for Additional Information. A formal list of questions submitted by a regulatory body to a licensee or organization that is applying for regulatory approval on a specific topic. Both OOE and NRC submitted multiple RAIs to PGE for information about their application for an ISFSI.

Rem: A unit of radiation exposure. A mrem or mR is one thousandth of a rem.

RTD: Resistance temperature detectors used to monitor the temperatures of the air outlets on the ISFSI concrete casks.

SAR: Safety Analysis Report. The SAR is part of the licensee's application to the Nuclear Regulatory Commission for a license to construct and operate an ISFSI under the provisions of 10 CFR 72.

SER: Safety Evaluation Report issued by the NRC following a review of the licensee's application to construct and operate an ISFSI under the provisions of 10 CFR 72.

SME: Seismic Margin Earthquake. The SME describes the ground motions associated with the maximum potential earthquake that could affect the Trojan site.

SNC or BNFL/SNC: Formerly the Sierra Nuclear Corporation, SNC was bought by British Nuclear Fuels Limited (BNFL) and renamed BNFL Fuel Solutions Corporation in 1998. It is also referred to as BNFL/SNC. BNFL/SNC is the manufacturer of the TranStor™ dry ventilated storage cask system for spent nuclear fuel. They are located in Scotts Valley, California. In this report, OOE uses both designations: SNC and BNFL/SNC.

spent nuclear fuel: Fuel assemblies that have been removed from a reactor following irradiation.

Spent Fuel Pool (SFP): A pool of borated water used to store and provide cooling for nuclear fuel assemblies that have been removed from a reactor.

Stress: The internal resistance experienced in a substance to an applied load. Stress is an internally distributed force that is measured by dividing the applied load by the area over which the stress is effective.

Surveillance: The PGE QA Program defines surveillance as "A documented observation or review of an activity for the purpose of verifying conformance with specified requirements or evaluating their adequacy and effectiveness."[94]

TEDE: Total Effective Dose Equivalent. A measure of radiation dose in rem (roentgen equivalent man) or mrem (thousandths of a rem) to the whole body.

TranStor™: Sierra Nuclear Corporation's ISFSI cask system for storage and transportation of spent nuclear fuel.

TNP: Trojan Nuclear Plant.

transfer cask: A steel vessel used to lift and transport a PWR Basket in the Fuel Building and support a PWR Basket at the Transfer Station. The transfer cask also provides radiation shielding.

trunnions: The load-bearing components of the transfer cask that mate with the lifting yoke and provide a means for the Fuel Building Crane to lift the loaded transfer cask.

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[94]PGE, PGE-8010, Nuclear Quality Assurance Program for Trojan Nuclear Plant, Rev. 20, December 12, 1996, Glossary

transfer station: A structure designed for transferring a basket from a concrete cask to an overpack or to a shipping container. The primary function of the transfer station is to prevent the transfer cask from falling or overturning during basket transfer operations. See the SAR for more details.

VSC-17 and VSC-24: Ventilated storage cask systems manufactured by Sierra Nuclear Corporation.

Yucca Mountain, Nevada: Proposed site of the Federal Government's permanent repository for storing nuclear fuel.

10 CFR 50: Title 10 of the Code of Federal Regulations, Part 50, provides rules promulgated by the Nuclear Regulatory Commission that govern the licensing and operation of a nuclear power facility.

10 CFR 72 Title 10 of the Code of Federal Regulations Part 72 provides rules promulgated by the Nuclear Regulatory Commission that govern the licensing, construction, and operation of an ISFSI.

APPENDIX C — REFERENCES

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