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Staff Evaluation of Holtec Design for Portland General Electric's Independent Spent Nuclear Fuel Storage Installation (ISFSI)

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I. EXECUTIVE SUMMARY

This is the second report by the Oregon Office of Energy (OOE, also referred to as "the Staff") on the proposed Independent Spent Fuel Storage Installation (ISFSI) at Portland General Electric (PGE) Company's Trojan Nuclear Plant (TNP). On January 27, 1999, the Staff issued the first report supporting PGE's proposed ISFSI plan,[1] and on April 6, 1999, the Oregon Energy Facility Siting Council (EFSC or "the Council") approved the plan. On July 11, 1999, after pre-operational testing, fuel transfer from the Spent Fuel Pool (SFP) to the ISFSI began. However, shortly after fuel movement to the first PWR Basket, operations were suspended due to the formation of hydrogen bubbles and rust particles in the water in the Cask Loading Pit. PGE investigated, and the results lead to the selection of a new vendor, Holtec International (Holtec).

On October 26, 2001, PGE forwarded a proposed revision to the Safety Analysis Report (SAR) describing the changes to ISFSI components resulting from the change in vendor. Holtec will supply Multi-Purpose Canisters (MPC)[2], constructed entirely of stainless steel, to replace Sierra Nuclear Corporation's (SNC or BNFL/SNC) PWR Basket. Holtec will also supply ancillary equipment needed to transport the MPCs and to prepare loaded MPCs for long-term storage. The loaded MPCs would be stored within concrete casks originally supplied by SNC in 1999. PGE wishes to begin fuel loading in November 2002 and complete it by October 2003.

The Staff reviewed the revised ISFSI SAR for PGE's TNP. As with the first report, the purpose of this review is to:

1. Evaluate whether PGE's ISFSI revised 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 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.

In addition, a primary focus of this report is on the changes brought about by the use of the new MPCs and Transfer Cask, and in particular the impact of the changes on heat transfer, structural integrity, potential for leakage, and radiation. Where possible, we verified that the differences between the two systems were not significant, and the conclusions for the original design remain applicable to the new. Where significant differences exist, we reviewed the Holtec analyses in detail, similar to our original review effort. Most of the descriptions from our January 27, 1999 report that involve only the SNC design have been deleted, since they are no longer applicable.

PGE must eventually remove spent fuel from the SFP in order to complete decommissioning and to facilitate eventual shipment of the fuel to a federal repository, when one becomes available. OAR 345-26-390 currently authorizes fuel storage in an ISFSI under the SNC design approved in 1999. By submitting its revised SAR, PGE requested rulemaking to authorize use of the new Holtec design.

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[1] OOE Staff Evaluation of PGE's Independent Spent Fuel Storage Installation (ISFSI), January 27, 1999
[2] The term "multi-purpose" refers to the canisters' suitability for both storage and transportation.

The Staff's review since the issuance of the revised SAR has included an extensive review of the SAR and its 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 a PGE QA surveillance at the facilities of the MPC manufacturer, U.S. Tool & Die (UST&D); and several meetings with the U.S. Nuclear Regulatory Commission (NRC) and the Council.

In our review of important-to-safety calculations, the Staff 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 computer codes, such as FLUENT or ANSYS, the Staff verified that the computer codes were NRC-approved. For calculations used in support of statements in the SAR, the Staff verified that the SAR accurately reflected the calculation, its assumptions, inputs, and results.

The Staff found that the revised 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 (10CFR72). The SAR includes an accident analysis demonstrating that the risk to the public is acceptably low.

Activities related to the 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. Activities related to the contingency unloading and transfer of the fuel from an MPC back to the SFP are described in LCA-246. Fuel transfer activities at the SFP are governed by PGE's existing NRC license under Title 10 Part 50 of the Code of Federal Regulations (10CFR50), and, therefore, are not described in detail in the SAR. However, the Staff reviewed the SAR, LCA-237, and LCA-246, and recommends amending EFSC rules based on these analyses.

As with the previous report, 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) the Staff's evaluation of PGE's Safety Analysis, (4) the Staff's evaluation of LCA-237 and LCA-246, (5) an assessment of PGE's compliance with the OARs, and (6) the Staff's conclusions and recommendations.

In summary, the Staff concludes that the proposed ISFSI, utilizing the Holtec stainless steel MPCs, meets the criteria set forth in the revised OAR 345-26-390. The Staff further concludes that measures have been taken to assure that fuel transfer activities described in LCA-237 will be performed safely. The Staff 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 ISFSI PROJECT

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. The process of dismantling the plant and removing all residual radioactive contamination is called "decommissioning." In January 1995, pursuant to NRC and EFSC regulations, PGE submitted PGE-1061, Trojan Nuclear Plant Decommissioning Plan, to the NRC and OOE. The Council approved the plan in March 1996. The NRC approved it in April 1996.

The Trojan Decommissioning Plan calls for the removal of all radioactive and contaminated components and material on the site. Much of decommissioning is completed as of September 2002. However, the Trojan SFP and its associated water supply systems cannot be removed or decommissioned while spent fuel is stored in the pool. Nothing can be done that might endanger the pool, its water supply, or its associated systems.

To allow decommissioning to be completed, PGE intends to remove the fuel from the SFP and place it in dry storage casks. The casks would be air-cooled using natural air circulation, and they would be located on the Trojan site outside the existing Fuel Building. This would allow the complete decommissioning of all buildings and structures related to plant operations. In March 1996, PGE submitted a 10CFR72 license application to the NRC for construction and operation of an "Independent Spent Fuel Storage Installation" (ISFSI).

In their ISFSI Safety Analysis Report (SAR) submitted on March 26, 1996, PGE demonstrated compliance with EFSC and NRC regulations, provided an analysis of projected radiation exposure to the public under normal and accident conditions, and identified the vendor that would construct the ISFSI. The spent fuel storage system described in the 1996 SAR was designed by Sierra Nuclear Corporation (SNC)[3]. The SNC design used a fuel canister (the "PWR Basket") with a stainless steel shell and carbon steel internals. SNC proposed to coat the internals with a proprietary coating to prevent rust.

On January 27, 1999, the Staff issued a report supporting PGE's proposed ISFSI plan.[4] On March 31, 1999, the NRC issued Trojan ISFSI License No. SNM-2509. And on April 6, 1999, the Council approved the plan. On July 11, 1999, after pre-operational testing, fuel transfer from the SFP to the first PWR Basket began. However, shortly after fuel movement began, hydrogen bubbles began to appear in the water, the water became cloudy with fine rust particles, and visibility was lost. A total of eight fuel assemblies had been moved. At this point, the fuel transfer was suspended. The water in the Cask Loading Pit was cleaned, visibility was restored, and the fuel assemblies were moved back to the SFP. PGE began an investigation into the causes for the hydrogen and rust.

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[3] SNC was subsequently acquired by British Nuclear Fuels Limited (BNFL)

[4] OOE Staff Evaluation of PGE's Independent Spent Fuel Storage Installation (ISFSI), January 27, 1999

On January 28, 2000, PGE informed the Council that their investigation had shown that the hydrogen and rust were the result of interaction between the carbon steel internals of the PWR Basket and the boric acid of the SFP water. The coating on the internals had failed to adhere properly, and PGE was evaluating alternate coatings. (See the following section for a more detailed description of the problem, investigation, and the Staff's review.)

Not satisfied with a number of issues regarding SNC and their design, PGE began a search for a new vendor. On March 9, 2001, PGE informed the Council that they had signed a contract with Holtec International (Holtec) to supply a Multi-Purpose Canister (MPC) with stainless steel internals to replace the SNC PWR Basket. On October 26, 2001, PGE issued a Proposed Revision to the Safety Analysis Report describing the extensive design changes to the ISFSI, including the new MPC, Transfer Cask, and Lift Yoke designed by Holtec. The other major component, the Concrete Casks, which are an SNC design, remains.

PGE proposes to begin spent fuel transfer from the SFP to the ISFSI in November 2002, and to finish by October 2003.

B. PWR BASKET COATINGS FAILURE

As previously stated, PGE experienced problems during the initial fuel loading, which began on July 11, 1999. Hydrogen bubbles began to appear in the water in the Cask Loading Pit, and the water became so cloudy with fine rust particles that visibility was lost. This resulted in the suspension of loading operations and cleanup of the water. When sufficient clarity returned, PGE removed the fuel from the PWR Basket and returned it to storage locations in the SFP. An investigation into the cause of this began immediately, and Corrective Action Request (CAR) 99-034 was written. Initial analysis indicated that the coating on the interior basket assembly of the PWR Basket had failed, thus exposing the carbon steel to the corrosive boric acid environment of the SFP water. Analysis of the other two PWR Baskets stored at Trojan indicated that both appeared to be in various stages of coatings failure.

In August 1999, BNFL Fuel Solutions Corporation, the parent company of SNC, established an Event Review Team (ERT) to do a root cause analysis of the problem. Their final report was issued on November 19, 1999, "Unanticipated TranStor™ PWR Basket Reaction to Borated and Demineralized Water." Using sophisticated root cause investigation techniques, the ERT concluded that the hydrogen and rust were the sole result of the interaction between the carbon steel internals of the PWR Basket and the boric acid of the SFP water. The coating used on the internal basket structure had not been properly applied, and had failed to adhere in places. The report cited several programmatic and technical issues that led to the coating failure. It also identified several corrective actions, the primary one being to discontinue use of the coating and find another more suitable to this application.

The Staff reviewed the report and found it generally thorough; however, we had some questions, which were put into a letter to PGE on December 13, 1999.[5] The concerns about the coating application and testing are made moot by the replacement of the PWR Baskets with all-stainless MPCs. However, we asked why more attention had not been paid to the issue of schedule pressure. We noted the statement in the report that testing on some coatings "could not be supported in time to support basket fabrication." It appeared to us that some of the errors that led

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[5] OOE, A.Bless to L.Dusek, Event Review Team, December 13, 1999

to the incident were errors of haste, and the failure to address this in the report was a missed opportunity. We asked PGE to address how in the future they would attempt to not be unduly influenced by scheduling pressures, either real or perceived.

In response to OOE's concern, PGE employed an independent company, Pilot, to "review the ERT Report, determine the root cause for failures within PGE control to correct, and establish corrective actions."[6] The Pilot report, "Root Cause Analysis of the ISFSI Delay," determined that the cause of the coating failure was due to "management-related issues" including "schedule pressure."[7] Other root causes included management being not fully informed of the level of risk, possible consequences, or technical issues involved; the lack of clear employee roles and responsibilities, the lack of controls for a poor-performing vendor, and the failure to utilize coating experts or resources to resolve concerns. The Staff has verified that all of the corrective actions identified in CAR-C99-0034, Rev. 1, have been satisfactorily resolved.

C. DRY SPENT FUEL STORAGE

1. Current Spent Fuel Storage

There are 779 intact fuel assemblies currently in the SFP. There are also 11 partial fuel assemblies and 1 fuel rod storage container, which contain intact, suspect, or damaged fuel rods. There are also 1 skeleton fuel assembly without rods, 1 fuel rod storage rack, and several Process Cans containing mostly fuel debris, which came from fuel cladding failures that occurred between 1979 and 1982 when fuel pellets were released into the primary coolant system, and bottom nozzles. The damaged assemblies and Process Cans will go into 22 Failed Fuel Cans or Damaged Fuel Containers, which in turn will each fit into a cell within one of the MPCs. The total of 779 intact fuel assemblies and 22 Failed Fuel Cans or Damaged Fuel Containers will occupy 801 cells in the 34 MPCs and Concrete Casks on the ISFSI Storage Pad.

The spent nuclear fuel 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 8.5 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 intact and damaged 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.

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[6] PGE, CAR 99-0034, Rev.1, PWR Basket Coatings Failure, July 6, 2000
[7] ibid.

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. However, by the end of 2002, all Trojan fuel will have cooled in the SFP for nearly ten years. The decay heat has diminished to the point that air cooling is sufficient.

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 less than 1 kW of heat (about the same as ten 100 watt light bulbs), and most assemblies produce considerably 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 a relatively low driving force in an ISFSI MPC to create a release and dispersion of radioactive materials. The maximum internal pressure in an ISFSI MPC is calculated to be 72.4 psia (57.7 psig); 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 MPCs for which the probability of leakage is extremely low. The structural integrity of these MPCs was a key element in OOE's review.

2. 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.[8] Experience with the storage of spent fuel in dry casks is extensive and growing. Some of the nuclear power plant sites already using dry cask storage include Dominion Resources' Surry plant, Progress Energy's H. B. Robinson plant, Duke Energy's Oconee plant, Public Service of Colorado's Fort St. Vrain plant, CMS Energy's Palisades plant, Constellation Energy's Calvert Cliffs plant, Xoel Energy's Prairie Island plant, First Energy's Davis-Besse plant, Wisconsin Electric Power's Point Beach plant, Georgia Power's Plant Hatch, Exelon's Dresden plant, Entergy Nuclear's Fitzpatrick and ANO plants, and Energy Northwest's Columbia Generating Station. These facilities use a variety of dry storage technologies, including vaults, and several types of metal casks, Concrete Casks, and baskets or MPCs.

3. Status of a Permanent Federal Repository

Permanent disposal of spent fuel from nuclear plants is ultimately the responsibility of the Federal Government. OAR 345-26-390 states that onsite spent fuel storage in an ISFSI is an interim measure only. Onsite storage is needed until there is a federally licensed, centralized storage facility. Congress established the program for geologic disposal of nuclear waste in the Nuclear Waste Policy Act of 1982. This act directed the U.S. Department of Energy (DOE) to

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

find a site, determine its suitability as a repository for nuclear waste, then build and operate the underground facility, also called a geologic repository.

In 1987, Congress passed the Nuclear Waste Policy Amendments Act of 1987. This directed DOE to study Yucca Mountain, Nevada, as the site for the repository. Congress 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. This assessment was completed in December 1998. It included a design concept for the facility, an assessment of system performance relative to the geological characteristics of the site, and cost estimates for the remaining development work, construction, and operation. Based on this assessment, DOE concluded that work should proceed to support a decision on whether to recommend the site. Following this, DOE prepared a draft Environmental Impact Statement in 1999.

In February 2002, Secretary of Energy Spencer Abraham recommended the Yucca Mountain site for the President's approval. Immediately following this, the President sent the recommendation to Congress for their approval, which took place in July 2002. Later this year, DOE will submit a license application to the NRC. If approved by the NRC, construction and operation will proceed. If DOE's schedule estimates are met, the earliest possible date for accepting nuclear waste would be in 2010. The Government's Accounting Office estimates that opening of the site will not occur until 2015. Certainly, judging from the project milestones to date, delays may be expected, plus there remains extensive public and governmental review processes.

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 considered it prudent to require a 40-year life.

D. 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.

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 Trojan's needs until the expiration of its NRC 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 stated:

> 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 to set 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 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 an ISFSI license 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."[9] 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. The SAR is the primary factual basis for the NRC's decision to issue an ISFSI license; however, the NRC may (and usually does) issue "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 10CFR72. 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 10CFR72 on March 26, 1996. The NRC issued public notice in the Federal Register on April 25, 1996,[10] including notice of the

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[9] 10CFR72.24
[10] Federal Register, Volume 61, No. 81, p. 18448

public comment period. The deadline to intervene in the NRC licensing process was May 28, 1996. No one filed to intervene. After the problems with the PWR Basket coating and the selection of a new vendor, PGE filed a revised SAR on October 26, 2001. This proposed revision to the SAR was subsequently updated on July 25, 2002, in response to an NRC RAI, and then again by PGE letter dated August 22, 2002, and September 4, 2002. The NRC expects to complete its review of the revised SAR in October 2002.

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 contained in ORS 469.594(3) 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,[11] 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."[12] 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."[13]

OOE and NRC work toward separate conclusions concerning the proposed ISFSI. In some cases, the Staff may recommend rules go beyond those imposed by the NRC. However, EFSC rules cannot 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. Since the selection of Holtec as the new vendor for the MPCs, the Staff has participated in two meetings between NRC, PGE, and Holtec representatives at NRC headquarters in June 2001 and February 2002. Within the past year, a new NRC policy has provided greater access for OOE Staff to participate in NRC sponsored meetings. The NRC occasionally sends representatives to EFSC meetings regarding Trojan, and the two organizations exchange RAIs and PGE's responses to them.

In May 1996, the NRC invited OOE's site inspector to visit Entergy Nuclear's ANO plant in Arkansas along with an NRC inspection team to witness a mock loading of a dry spent fuel storage basket. In July 2002, the OOE site inspector joined an NRC team that was inspecting preparations for fuel loading at the Columbia Generating Station in Washington.

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 operational issues and concerns specific to this region (for example, seismic concerns).

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[11] Memorandum of Understanding between the NRC and the State of Oregon, January 1980
[12] OAR 345-26-370(1)(c)
[13] OAR 345-26-390(1)(b)

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 expertise in nuclear criticality computer codes. The OOE Staff discussed the criticality calculations with the NRC to verify their method in performing this evaluation. Based on these discussions and on the NRC review, OOE concluded that PGE and their vendor used approved computer codes and applied them properly.

Conversely, OOE can observe PGE's programs and procedures firsthand, through its resident inspector and through frequent site visits by its consultant. Because of its onsite presence, OOE can interview PGE and Holtec personnel directly. OOE reviewed procedures onsite, witnessed receipt inspections, and observed pre-operational testing. Additionally, OOE participated in vendor audits and surveillances performed by PGE's Nuclear Oversight Department.

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 calculational methods, and by direct interviews and observations. This dual review by OOE and NRC resulted in a review that examined PGE's proposed ISFSI from different perspectives.

E. OOE's REVIEW PROGRAM

1. Criteria and Methodology

To approve the proposed ISFSI, the Council must find that 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 Staff reviewed the SAR and supporting documents. These supporting documents included: industry codes and standards; selected NRC documents; numerous vendor documents, including Holtec's HI-STORM LAR and drawings; LCA-237 on spent fuel cask loading in the Fuel Building; LCA-246 on contingency fuel unloading to the SFP; 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 the Staff 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, SNC, or Holtec confirmed that the assumptions and inputs 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 or FLUENT, the Staff 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. In all of the calculations reviewed, the Staff attempted to verify that the results were reasonable and within appropriate acceptance criteria. In a few instances, the Staff were able to compare calculational results with other calculations or empirical data.

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 the calculations selected for review, we asked questions in the following categories:

(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 and within appropriate acceptance criteria?

(4) Computer Codes

Are the computer codes used widely in the nuclear industry or related industries?
Has the NRC approved of each of the computer codes that were used?
Were the computer code appropriately verified and validated by the vendor?
Did PGE QA review the vendors' verification and validation of computer codes?

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, portions of the structural and thermal analyses, and some of the accident analyses. We also confirmed that such calculations were evaluated in depth by the NRC or by their engineering consultant. In addition, OOE verified that PGE QA had audited and found satisfactory BNFL/SNC's and Holtec'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.A of this report.)

2. Requests for Additional Information (RAIs)

With the revised SAR, OOE issued one formal RAI to PGE on April 5, 2002. We based many of our conclusions on PGE's written responses, which were submitted on June 6, 2002, and also on less formal responses given in interviews with Trojan personnel. We also reviewed the NRC's RAI and PGE's written responses to those questions.

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 MPC and components, witnessed test operations of the Vacuum Drying System, and observed various training sessions of ISFSI personnel.

OOE visited the SNC headquarters[14] twice to interview SNC personnel directly, review quality records, and to observe PGE QA surveillances. We include this information, because the Concrete Casks remain an SNC design. In July 2001, we visited the facilities of U.S. Tool & Die (UST&D), the fabricator of the Holtec MPCs, Transfer Cask, and Lift Yoke.

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.

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.

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.

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[14] 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.

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?

Lastly, the primary focus of this report, in comparison with the Staff report issued on January 27, 1999, is on the changes brought about by the use of the new Holtec MPCs and Transfer Cask, and in particular the impact of those changes on heat transfer, structural integrity, potential for leakage, and radiation. Where possible, we verified that the differences between the two systems were not significant; and, therefore, the conclusions for the original design remain applicable to the new. Where significant differences exist, we reviewed the Holtec analyses in detail, similar to our original review effort. Most of the descriptions from our January 27, 1999, report that involve only the SNC design have been deleted, since they are no longer applicable.

III. DESCRIPTION OF THE PROPOSED ISFSI & PROGRAMS

A. ISFSI SITE

1. Location of the ISFSI

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 the remainder of 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 minimizes the need for spent fuel transportation. Concrete Casks will be loaded in the Fuel Building and moved to the ISFSI Storage Pad 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 will be 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.

2. ISFSI Storage Pad

The ISFSI Storage Pad is located in the northeast corner of the Trojan site. The Storage Pad measures approximately 105 by 170 feet and is designed to accommodate up to 36 storage casks (of which 34 will be used), the Transfer Station, and a small storage building. The Storage 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 of the Storage Pad are part of the Trojan site and are under PGE control.

3. ISFSI Boundaries

There are several important boundaries associated with the ISFSI site, which serve a variety of functions. They include:

ISFSI Protected Area**.** This area is designed for security purposes. The Protected Area immediately surrounds the ISFSI and is the area in which federally licensed activities under 10CFR72 will occur. This boundary includes a Protected Area Fence that is from approximately 25 to 34 feet outside the ISFSI Storage Pad and precludes entry by unauthorized persons in accordance with the security requirements of 10CFR73.51.

ISFSI Controlled Access Area**.** This area encompasses the Protected Area. The ISFSI Controlled Access Fence (also known as the Perimeter Fence) varies from 20 to 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 until after the spent fuel is moved onto the ISFSI Storage Pad and dose measurements are recorded.

Restricted Area. This area is defined as the area in which the radiation dose rate is equal to or greater than 2 millirem per hour (mrem/hour), 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.

Controlled Area. This area is within 300 meters from the edges of the ISFSI Storage Pad. PGE has calculated that at this distance, the expected annual dose from fuel storage is 18.4 mrem above background. Federal requirements at 10CFR72.106 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 10CFR72.106). Calculations show that radiation dose from the worst case credible accident would in fact be much less at this distance. 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. Once the ISFSI is in operation, PGE expects to find that actual dose is less than the calculated dose. If that is the case, then the Controlled Area may be reduced in size.

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 10CFR50 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 Exclusion Area for emergency purposes in the Trojan Nuclear Plant Decommissioning Plan. 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 Trojan ISFSI SAR and applies only to 10CFR50 activities.

B. THE VENDORS

1. Sierra Nuclear Corporation

Originally, PGE selected a spent fuel storage system designed and built by Sierra Nuclear Corporation (SNC), which is now called BNFL Fuel Solutions Corporation or BNFL/SNC. The Concrete Casks are the only components from SNC design remaining with the current ISFSI proposal. For these casks, BNFL/SNC performed most but not all of the calculations regarding structural integrity, heat removal, nuclear criticality, radiation, and accident analysis. (Holtec also used the SNC-designed Concrete Casks in several of their thermal and accident analyses.) Facilities using Concrete Casks similar to the ones proposed for Trojan include CMS Energy's Palisades plant in Michigan, Wisconsin Electric Power's Point Beach plant, and Entergy Nuclear's ANO plant in Arkansas. For more information about SNC and the Staff's review, refer to the Staff report of January 27, 1999.

2. Holtec International

Following suspension of fuel loading operations in July 1999, PGE began a search for another vendor. On March 8, 2001, PGE announced to the Council that they had selected Holtec of Marlton, New Jersey, to replace SNC as the principal vendor for design and fabrication of the ISFSI Multi-Purpose Canisters (MPCs) and Transfer Cask. At the EFSC meeting on March 29, 2002, Dr. K. P. Singh, President and CEO of Holtec, made a presentation to the Council explaining the background of his company. According to that presentation, fuel storage is the core business for Holtec, comprising 70% of its income. Their MPC system is the first to be dual certified by the NRC for both storage and transport. Currently, the nuclear power facilities using Holtec MPCs similar to the ones proposed for Trojan are Georgia Power's Plant Hatch, Exelon's Dresden plant, Entergy Nuclear's Fitzpatrick and ANO plants, and Energy Northwest's Columbia Generating Station.

The design that PGE selected for Trojan is a modified version of Holtec's HI-STORM 100 design, which has been certified by the NRC. Holtec is in the process of obtaining an amendment to their Certificate of Compliance for Trojan. In addition to the MPCs and Transfer Cask, other equipment to be supplied by Holtec will include the Lift Yoke for lifting the Transfer Cask and a Vacuum Drying System for removing residual moisture from a wet-loaded MPC.

Additionally, PGE has signed a service agreement with Holtec to do all of the fuel handling, loading, and welding operations on the ISFSI canisters. This represents a major change from the relationship between PGE and SNC. The Staff raised several questions about this arrangement in our RAI to ensure that PGE maintained supervisory responsibility over the loading operations. PGE ensured us that they will maintain overall responsibility for all aspects of the ISFSI loading. For details on shift personnel and responsibilities, see Section V of this report on ISFSI Loading Operations.

3. U.S. Tool & Die

The company subcontracted to Holtec for fabricating the MPCs and Transfer Cask is U.S. Tool & Die (UST&D) in Pittsburgh, PA. UST&D is an N-Stamp holder (N-stamp is a recognized certification for fabricators in the nuclear industry) with 99% of their business with the nuclear industry. They are a major fabricator of both wet and dry fuel storage systems, and their primary customers are Holtec International and Transnuclear. The Staff visited UST&D in July 2001 with PGE QA and Engineering personnel to examine several aspects of their program. The details of that inspection visit are in III.G.4 of this report.

The experiences PGE had on fabrication and quality assurance issues with SNC and their major fabricator ABB Combustion Engineering Nuclear Systems (ABB) have impacted their relationship with Holtec and UST&D. Two significant events at other plants had caused the NRC to question the adequacy of SNC's fabrication controls and quality assurance practices. In May 1996, a hydrogen explosion occurred during welding operations while sealing a PWR Basket at Point Beach; and in April 1997 at the ANO plant in Arkansas, an 18 inch crack was found in the lid-to-shell weld of a PWR basket. The NRC thoroughly investigated both of these incidents, and PGE incorporated the lessons learned into the program at Trojan. Since the switch to Holtec and UST&D and the replacement of the PWR Baskets with the MPCs, these problems have been rendered somewhat moot. However, there remain lessons learned that apply to the current situation. The issue of fabrication controls and quality assurance practices has resulted in PGE stationing a full-time PGE QA representative at UST&D and sending PGE welding engineers periodically to ensure that welding procedures meet PGE specifications. As a result of these efforts, the Staff has concluded that PGE has taken appropriate preventive actions to prevent problems similar to those that occurred in the past.

C. ISFSI STORAGE COMPONENTS & DESCRIPTION OF THE DESIGN CHANGES

The principal loading and storage components of the ISFSI include the MPCs, Transfer Cask, Lift Yoke, Concrete Casks, Failed Fuel Cans, Damaged Fuel Containers, the Fuel Debris Process Can Capsules, Fuel Debris Process Cans, Transfer Station, and impact limiters. There are other components, but these are the major ones. A description of each of these principal components and the major differences between these and the components from the SNC design follows.

1. Multi-Purpose Canister (MPC)

The MPC is a cylindrical stainless steel canister, approximately the same dimensions of the SNC PWR Basket. It is about 15.1 feet tall and 68.4 inches wide. This is 2.4 inches wider than the previous design, a change which prompted a Staff review of the thermal characteristics of the MPC. As a result, we confirmed that this change posed no problem in the passive heat transfer design. The shell of the MPC is 0.5 inches thick, which is less than the 0.75 inches of the PWR Basket, and it weighs slightly more at 78,700 pounds when loaded with dry spent fuel. There will be a total of 34 MPCs for storing spent fuel.

Inside the MPC is an internal sleeve assembly or basket with capacity for 24 fuel assemblies. Twenty of the assembly sleeves (or cells) are designed to hold intact fuel assemblies, and the four slightly larger corner cells are designed to accommodate Failed Fuel Cans or Damaged Fuel Containers. The entire sleeve assembly is made from uncoated stainless steel, unlike the coated, carbon steel fuel racks of the PWR Baskets, which contributed to the problem of hydrogen release and rust in the Cask Loading Pit in July 1999. Boral plates, which are made from aluminum-boron carbide, are attached to the sides of the assembly sleeves to ensure against nuclear fission by absorbing neutrons, and thereby providing an additional safety margin against criticality. After being loaded with spent fuel, vacuum dried, and sealed, the MPCs will be filled with 99.995 % pure helium to a pressure of approximately 30 psig. The helium ensures adequate heat transfer during storage and will provide an inert, non-corrosive atmosphere for long-term fuel integrity. The pressure of the helium is a significant change from the atmospheric pressure of the SNC PWR Baskets. The lack of a pressure differential in the SNC design was one of the major arguments that leakage was not a credible event. The Staff reviewed this design change and found that the argument for non-leakage is still viable for the Holtec design. See Section IV.B.4.d of this report for the details of that review.

Unlike the SNC PWR Baskets, which had two lids totaling 11 inches in thickness, the MPCs have a single stainless steel lid, 9.5 inches thick. This reduction in thickness will cause increased radiation from the top of the storage cask, which is reviewed in Section IV.D of this report on Radiation Protection. The lid will be welded to the MPC shell, dye-penetrant tested, hydrostatic pressure tested, and helium leak tested to ensure an airtight seal. There are two penetrations in the lid for venting and draining during fuel loading operations, and for filling the MPC with helium once the fuel is loaded. These penetrations have cover plates that will be welded, dye-penetrant tested, and helium leak tested. In addition, there are four threaded holes part way through the MPC lid to be used for lifting. Either two high-strength steel cleats are attached to these holes to provide lifting of a loaded MPC with slings, or four eye bolts and slings are attached for lifting just the MPC lid. During normal storage, threaded inserts are installed in the empty holes where the lifting cleats were previously installed.

The other main difference between an MPC and the SNC's PWR Basket is that the MPC has been certified by the NRC as a transportation container.

There are two designations of MPCs used for Trojan: the MPC-24E and the MPC-24EF. The "EF" version will be used in the same way as the "E" version, except that they will also store fuel debris loaded in the Failed Fuel Cans or Damaged Fuel Containers and stored in the corner cells of the MPCs. Both versions are identical from a thermal standpoint. The difference is that the MPC-24EF shell is 1-inch rather than 1/2-inch thick in the vicinity of the lid. This is due to more stringent 10CFR71 transportation requirements on MPCs loaded with fuel debris.

In addition to differences between Holtec's MPC and the SNC PWR Basket, are differences between the generic MPC and the unique design for Trojan. The major one is that the Trojan MPC is 9 inches shorter, which enables it to fit inside the SNC-designed Concrete Cask. Another is that the corner cells for the Trojan MPCs had to be made larger to accommodate the Trojan Failed Fuel Cans, which are slightly larger than the Holtec-designed Damaged Fuel Containers. And a relatively minor change is that the "mouse holes" at the bottom of each cell that are used for convective circulation flow within the MPC have been shortened to accommodate positioning of the Boral plates for the Trojan fuel.

2. Concrete Cask

Once loaded, an MPC will be placed in an SNC-designed Concrete Cask made of reinforced concrete. These casks were designed to the requirements of ACI-349 and constructed to ACI-318. They are approximately 11.3 feet in diameter, 17.6 feet high, with walls 29 inches thick. The internal cavity of each Concrete Cask is formed by a 2-inch thick carbon steel liner and bottom plate, which is coated to promote radiant heat dissipation and to minimize corrosion. With the PWR Basket, there was a 4 inch air space between the inner wall of the Concrete Cask and basket outer wall; however, with the MPC the air gap will be approximately 2.8 inches. This is still larger, however, than the normal Holtec HI-STORM storage system, which uses a gap size of 2.55 inches. Vents at the top and bottom of the Concrete Cask permit natural air circulation, which is the passive method of decay heat removal. The Concrete Cask weighs approximately 292,700 pounds when loaded with a sealed MPC and spent fuel. Weather protection is provided by a coated, carbon steel lid.

3. Transfer Cask

The Transfer Cask is used to transfer a loaded MPC between the SFP and a Concrete Cask and during Transfer Station operations on the ISFSI Transfer Pad. 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 MPC into a Concrete Cask. The Transfer Cask serves as a special lifting device which meets the requirements in NUREG-0612 (1980) and ANSI N14.6 (1993) for movement of an MPC. The Holtec Transfer Cask and the SNC Transfer Cask are similar in design and dimensions. Both are made of carbon steel with lead sandwiched between rolled steel plates on the sides for radiation shielding. There are three significant differences, however.

One difference is that the Holtec Transfer Cask has a water jacket mounted on the outside to provide neutron shielding. Another is an annulus seal on the Holtec model that is placed in the gap between the Transfer Cask and the top of the MPC during fuel loading operations and is designed to reduce contamination of the MPC exterior surfaces from the SFP water.

The third significant difference is that the annular gap between the MPC and the inside diameter of the Transfer Cask is much smaller on the Holtec model, a nominal 0.175 inches versus 0.625 inches. This has been a cause for concern. When manufacturing tolerances and temperature contractions or expansions are added, the gap narrows to less than 1/8 inch. Fabrication irregularities, such as weld crowns on either vessel, could reduce this even further and cause interference between the two during loading operations. PGE requested that Holtec have UST&D manufacture the Transfer Cask to maximize the inner diameter, ensuring a gap of at least 1/8 inch between it and an MPC. They further requested that weld crowns be ground to less than 1/16 inch. To ensure that these requests had been satisfied, the Staff witnessed a portion of the receipt inspection of the Transfer Cask where measurements were taken. The Transfer Cask passed the inspection; however, should any out-of-roundness occur in an MPC or with the normal process of lowering an MPC into the Transfer Cask, it is likely that the epoxy coating of the Transfer Cask will become abraded, creating potential corrosion or contamination problems in the Cask Loading Pit. In response to an RAI question on this issue, PGE responded that the condition of the coating on the Transfer Cask would be inspected after each use and repairs made, if required.

4. Lift Yoke

The Lift Yoke is 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. Because of the new Holtec Transfer Cask, the Lift Yoke is also a new design, although the function remains the same. The previous design used hook-shaped, rigid side arms to engage with the trunnions; whereas, this design uses hydraulically-actuated arms with holes that fit over the trunnions of the Transfer Cask. There is one other difference, which is administrative, that has since been resolved and is described in Section IV.A.3.e of this report. Briefly, the previous submittal stated that the Lift Yoke would be tested to 300 % of design load, but the current submittal requires testing to only 150 % of design load. We inquired and found out that the Lift Yoke was actually successfully tested to 300 % in the fabrication shop.

5. Fuel Debris Process Cans and Process Can Capsules

The Fuel Debris Process Cans are sealed, stainless steel containers, referred to in the SAR as “confinements,” approximately 29 inches long, that hold loose fuel debris which is not confined by the fuel rod zircalloy cladding. Five of these cans fit into one stainless steel Fuel Debris Process Can Capsule, which is approximately 13 feet long. PGE has 42 Fuel Debris Process Cans currently stored in the SFP. Forty of these will go into eight Fuel Debris Process Can Capsules, which, in turn, will go into eight Failed Fuel Cans that are to be located in one of the four oversized corner locations of an MPC. The remaining two Process Cans 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. These are the same Process Cans that were intended for use with the previous design; therefore, there are no changes.

6. Failed Fuel Cans and Damaged Fuel Containers

The Failed Fuel Cans and Holtec-designed Damaged Fuel Containers are stainless steel containers approximately 13 feet long. The Failed Fuel Cans are designed to provide an extra barrier for fuel assemblies with failed cladding and for Fuel Debris Process Can Capsules. The Damaged Fuel Containers are designed to contain damaged fuel assemblies and fuel debris. These containers will be stored in the four oversized corner locations in an MPC. PGE plans to use 22 of these: 8 will contain the 40 Process Cans inside Process Can Capsules, 1 will contain a skeleton fuel assembly without rods, 1 will contain a fuel storage rack, which is a device designed to hold extra fuel rods, 1 will contain a damaged fuel assembly, 10 will contain fuel assemblies with missing rods, and 1 will contain bottom nozzles and the two remaining Process Cans. With one exception, these are the same Failed Fuel Cans that were intended for use with the previous design. The single exception is for a damaged fuel assembly that has one missing fuel rod and one fuel rod replaced with a stainless steel rod. Because the weight of this assembly in a standard Failed Fuel Can would exceed the weight limit for an individual cell within an

MPC, PGE will use a Damaged Fuel Container for this assembly. This container currently counts as 1 of the 22, and PGE may use more than one.

The Failed Fuel Cans and Damaged Fuel Containers are vented to the MPC 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 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."[15] 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 or Damaged Fuel Containers, as designed.

7. Transfer Station

The Transfer Station is located on the ISFSI Storage Pad just west of the storage casks and will be used to transfer: (1) a loaded MPC from one Concrete Cask to another in the event that the original Concrete Cask has been damaged, and (2) a loaded MPC from a Concrete Cask to a shipping container (Transport Cask) for transfer to a federally licensed repository. The Transfer Station will be the only method of transferring fuel from one Concrete Cask to another once the SFP is decommissioned. This is the same Transfer Station that was intended for use with the previous design; however, PGE modified it slightly to accommodate the Holtec Transfer Cask.

8. Impact Limiters

Not used in the ISFSI storage, but important to the loading operation in the Fuel Building and during transfer operations at the Transfer Station are 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, one embedded in the ground floor of the Fuel Building under a Concrete Cask during loading, and one embedded in the ISFSI Storage Pad at the Transfer Station. These are the same impact limiters that were intended for use with the previous design.

D. LOADING PROCESS

The activities inside the Fuel Building are controlled by PGE's existing NRC license under 10CFR50. Therefore, they are not described in detail in the SAR but in a separate PGE documents: License Change Applications LCA-237 and LCA-246. OOE reviewed these documents in the same level of detail as was used to review the SAR. Our review of LCA-237 and LCA-246 is described in Sections V and VI of this report.

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[15]PGE, CPY-001-97, C.P.Yundt to D.Stewart-Smith, Response to RAI, February 6, 1997

The loading process for the Holtec system is very similar to the process described for the SNC system in OOE's 1999 report. The most significant difference is the fact that the Holtec MPC will be filled with helium at a pressure of about 30 psig; the SNC Baskets were filled with helium at atmospheric pressure. Holtec will also use sensitive leak detection based on mass spectrometry to ensure that the welded MPC is leak tight.

E. RADIATION PROTECTION PROGRAM

The radiation protection program is an ongoing PGE program, as it was in 1999. Our detailed review of that program is in our 1999 report. Although the Holtec MPCs are a different design from the SNC PWR Baskets, the PGE radiation protection program was not affected by these design differences. The program, procedures and personnel are essentially unchanged, except for ongoing refinements unrelated to the change in vendor. Therefore, OOE believes the conclusions reached in 1999 remain appropriate.

F. MONITORING AND SURVEILLANCE

1. Temperature Monitoring of the Concrete Casks

a. Technical Specification Requirements for Temperature Monitoring

Since the ISFSI is passive by design and requires no instrumentation to operate, only monitoring instrumentation is required. This system is identical to what was described in our January 27, 1999 report; however, the procedures have since been written and the Technical Specification requirements have been substantially revised.

Section 5.5 of the proposed Trojan ISFSI Technical Specifications requires daily measurements of ambient and Concrete Cask air outlet temperature. This is a reduction from the original requirement to take 12-hour temperature readings, and was done because the original frequencies were deemed not necessary. The Staff concurs with this.

The Technical Specifications also require that action be taken if there is an unexplained air outlet temperature or unexplained difference between ambient and an air outlet temperature. Actions include determining and correcting the cause, and increasing the frequency of temperature monitoring. The Technical Specifications also require action if any air outlet temperature reaches "the program limit." The previous revision required action if any air outlet temperature reached 195 ^{o}F, so we asked PGE to define "the program limit" and how it is determined.

PGE's response was that TIP-17, Thermal Monitoring Program, requires daily verification that the air outlet temperature of any Concrete Cask does not reach or exceed 195°F, which corresponds to the long-term normal bulk concrete operating temperature limit of 225°F. The response stated further:

> Taking actions when any air outlet temperature reaches 195°F should preclude reaching the short-term bulk concrete temperature limit, which is 350°F. Concrete temperature in excess of 350°F could potentially weaken the concrete strength, and tests may have to be performed to evaluate the concrete and to justify continued use of the Concrete Cask.[16]

[16] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

b. Description of Temperature Monitoring System

SAR Section 5 mentions but does not describe the ISFSI instrumentation. The Staff therefore requested information from PGE describing the temperature monitoring devices. They 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."[17] In response to another question, PGE described four methods of taking daily readings of the Concrete Cask air outlets. These include: (1) the use of a data logger (range: from –10 to +240 ^{o}F) in the ISFSI Storage Building, which is a device that reads the outputs of the RTDs (range: from –328 to +900 ^{o}F) mounted on the Concrete Casks and is the normal way that readings will be taken; (2) manually taking readings with a hand-held digital readout device below the data logger; (3) manually taking readings at the junction box on each of the Concrete Casks; and (4) manually taking readings at the air outlets with a reach rod-mounted RTD. The Staff reviewed each of these methods and observed sample readings on the empty Concrete Casks to familiarize ourselves with the system.

PGE does not consider the temperature monitoring system as important-to-safety (ITS) and has stated that they will be purchased as commercial grade items. We asked how they will ensure that commercial grade devices will remain accurate over the 40-year expected life of the ISFSI. PGE responded that the devices will be initially calibrated with a thermometer, which is traceable to the National Institute of Standards and Technology (NIST), in accordance with QA requirements. However, there are no plans to do periodic calibrations on the RTDs. PGE did agree to trend the outlet temperatures to detect any RTD whose output is not consistent with other RTD readings. The trending program is described in the following section on Temperature Monitoring Procedures. OOE considers the designation of "not ITS" and the calibration program adequate for the following 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. NUREG/CR-5560, "Aging of Nuclear Plant Resistance Temperature Detectors," Section 9.1 states: "A significant calibration shift should not occur in an RTD as long as the sensing element is not stressed or contaminated after calibration and the insulation material is kept in place and dry."
- 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.
- The RTDs are the primary means for detecting air vent blockages. A failure in one or more of these would result in the use of adequate alternative methods, including increased visual inspections. Therefore, RTD failure would have no impact on safety.

[17]PGE, CPY-032-96, Response to RAI, June 27, 1996

- The plant will maintain spares in the case of failure for the life of the ISFSI. They currently have ten spare RTDs for use on the Concrete Casks, two spare RTDs for use on the manual reach rod, and nine handheld digital RTD readout platinum thermometers.[18]

c. Temperature Monitoring Procedures

The Staff reviewed PGE's procedures for temperature monitoring. TIP-17, Concrete Cask Thermal Monitoring Program, proceduralizes the Technical Specification requirements. It also describes an accelerated temperature monitoring program when any Concrete Cask outlet temperature reached 180 °F, 15 °F below the Technical Specification limit of 195 °F. At 180 ^{o}F, an alarm sounds at the Central Alarm Station requiring the suspect cask to be inspected and the cask temperature to be recorded at one hour intervals until the air outlet temperature falls below 180 ^{o}F. Should the temperature for any outlet exceed 192 ^{o}F, the ISFSI Specialists are instructed to follow procedure ONI 72-02, ISFSI Concrete Cask Air Outlet Temperature Limit Exceeded. This requires notification of the ISFSI Manager and the NRC, hourly monitoring of the cask air outlet temperatures, auxiliary cooling via a blower directed into the air inlets, initiation of an engineering evaluation of the effects and impact, and initiation of a Corrective Action Report to determine and correct the cause. The Staff considers these procedures satisfactory.

In our previous report, the Staff recommended that a trending program be established that could indicate a problem with a cask or faulty RTD. Experience at CMS Energy's Palisades plant in Michigan had shown that temperatures can vary greatly between the four air outlets due to wind and weather conditions. In response, PGE proposed that the ISFSI Manager will review the temperature data monthly, compare unusual readings to baseline or predicted values, and take appropriate corrective actions. PGE has incorporated these measures into procedure TIP-17.

When our previous report was issued, the Staff had not seen the details of the trending or calibration programs. We therefore recommended two rules: one requiring that these programs be available for inspection prior to fuel loading, and one requiring prior OOE concurrence if PGE wished to reduce the frequency of monitoring air outlet temperatures. The Staff has reviewed all of these programs and now believes that such rules are no longer required.

2. Visual Inspections of the Concrete Cask Air Vents

Section 5.5 of the proposed Trojan ISFSI Technical Specifications requires weekly inspections of the air inlets, and procedure TIP-12, Monitoring of the Concrete Cask System, implements this requirement. The original SAR required daily inspections. Therefore, this represents a significant reduction. The reason for this was to reduce radiation exposure of the ISFSI Specialists. The Staff concurs with the ALARA approach; however, because the estimated time for the concrete of a Concrete Cask with blocked inlets to reach its limit of 350 ^{o}F is 57.1 hours, we asked why the frequency of visual monitoring shouldn't be two days instead of weekly, such that 57.1 hours would not pass without at least one visual inspection? PGE's response was that "Weekly inspections … save on the order of 600 person-mrem exposure annually."[19]

The Staff was also concerned about the frequency of vent inspections in the event that weather or some other phenomena occurred that increased the chances for blocked vents. TIP-12 requires an increase in the frequency of vent inspections in the event of an environmental phenomenon

[18] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002
[19] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

that could result in air vent blockage. We also asked if procedures would reference the expected normal temperatures so that unexpected increases would be noticed? PGE stated that "Heat validation testing of the highest heat load cask (expected to be the third cask loaded) will provide initial actual temperature including the highest expected delta T." These will be used to compare to the daily temperature readings.[20]

Lastly, due to 17.6 foot height of the Concrete Casks, we requested in our previous report information to determine if adequate visual inspections of the air outlets were possible. PGE responded that the stainless steel wire mesh covering the air outlets made it unlikely that foreign objects will be drawn into them; however, ladders or other physical-lifting devices and mirrors will be available to conduct positive visual inspections.

3. Structural Inspection Program

ISFSI SAR Section 9.7.7 describes a Structural Inspection Program that establishes periodic inspections of the concrete surface of the Concrete Casks and the ISFSI Storage Pad. The intent of this program is to ensure that the structural integrity of the concrete is maintained through the establishment of acceptance criteria, degradation evaluation methods, and repair instructions. At the time of our previous report, an implementing procedure had not been written and the term "periodic" in the SAR had not been defined. Since then, TIP-09, Structural Inspection Program, has been developed. The procedure requires a baseline visual inspection prior to loading, and then annually there will be a visual inspection of each loaded Concrete Cask, plus Storage Pad surface, Transfer Pad surface, and Transfer Station by someone familiar with the construction or inspection of concrete and steel structures. Acceptance criteria have also been established. For the Concrete Casks, these consist of defects in any surface approximately one-half inch in diameter or width and a depth greater than one-quarter inch. Exposed reinforcing steel will be noted and repaired, and the inspection will include radiation monitoring at any defect location. In our previous report, the Staff recommended that PGE's commitment to implement a Structural Inspection Program procedure be added to the Council's rules. Having reviewed the procedure and found it satisfactory, we no longer feel that such a rule is required.

4. Concrete Cask Interior Inspection Program

ISFSI SAR Section 9.7.8 describes a Concrete Cask Interior Inspection Program that establishes inspections of the first Cask placed in service at five year intervals. It states that the "interior annulus area and the interior areas of vents shall be inspected to identify degradation mechanisms (not identified in the Safety Analysis Report) that may affect system performance."

Initially, the NRC wanted to impose a requirement that licensees remove the lid from each Concrete Cask every five years to inspect for degradation. PGE argued that this requirement was unnecessary, because the MPCs were fabricated out of stainless steel; degradation of the air flow path would be detected by the proposed surveillance requirements; and, most significantly, such a requirement would increase "the potential risk of unnecessary exposure to those employees performing the inspection."[21] The Staff concurred with this assessment, and the requirement has been reduced to a single cask every five years. Additionally, the selected cask is expected to be the one with the lowest heat load (and lowest dose rate), which will help with ALARA concerns.

[20] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002
[21] PGE, VPN-001-97, Trojan ISFSI (TAC No. L22102) Response to NRC RAI, January 9, 1997

5. Detecting Helium Leakage on the ISFSI Storage Pad

A possible benefit of the higher helium internal pressure in an MPC is the ability to check for leakage at the ISFSI Storage Pad following an event. With only atmospheric pressure in the SNC PWR Basket, the previous argument against measuring for leaks following an event was that a leak would likely be too small to be detectable. Holtec LAR Section 1.2.2.2 states: "The helium backfill ensures adequate heat transfer during storage, provides an inert atmosphere for long-term fuel integrity, and provides the means of future leakage rate testing of the MPC confinement boundary welds."

SAR Section 5.1.1.6 states that "should an off-normal event occur, an inspection for possible damage will be completed within 24 hours. An engineering evaluation will also be required to establish that a component may safely continue to perform the required function." These provisions are incorporated in procedure ONI 72-01, Steps 3.1.6 and 3.1.7. During our review of the previous design, we requested that there be a specific step(s) requiring leakage detection following a significant event affecting the ISFSI. PGE's response was to include in procedure ONI 50-03 a requirement that directs the ISFSI Manager to evaluate the need to perform a helium leak check following a seismic margin earthquake. Should this determination be made, the methods used will include, depending on the event, nondestructive examination of welds and the MPC lid surface, use of a helium sniffer, sampling for radioactive gases or airborne contamination, and monitoring the air outlet temperature of the storage cask for subtle changes in temperature.

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.[22] 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: "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."[23]

The Trojan QA program is essentially unchanged since 1999. The personnel and procedures are unchanged. Our 1999 Staff evaluation included a detailed description of that program and our assessment of it. The description and conclusions in that report remain our position today.

2. PGE QA's Vendor Surveillances

This report focuses on the recent design change involving the switch in primary vendors from BNFL/SNC to Holtec; however, the Concrete Casks are still an BNFL/SNC design. The Staff's analysis of BNFL/SNC and their sub-supplier ABB Combustion Engineering Nuclear Systems (ABB), PGE QA's vendor audits and surveillances, and our own observations of those vendors at their facilities are detailed in our January 27, 1999, report. To evaluate Holtec as a vendor for the design of the MPCs, Transfer Cask, and Lift Yoke, the Staff verified that Holtec was on the PGE "Approved Suppliers List" (ASL) and scheduled for several PGE QA surveillances during

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[22]PGE-8010, Trojan Nuclear Plant Nuclear Quality Assurance Program, Rev. 22, August 13, 1998
[23]ODOE, Review of PGE's Decommissioning Plan for the Trojan Nuclear Plant, January 22, 1996

the design phase of the contract. We reviewed Holtec's written QA program (Holtec International Quality Assurance Manual, Rev.12, January 2, 2001) and found that it met all the requisite criteria of 10CFR72, Subpart G. We also reviewed several audits and surveillances of Holtec, including the following:

> Nuclear Utilities Procurement Issues Committee (NUPIC) conducted an audit in May 2000. The audit team was composed of nine members from six different nuclear utilities with Southern Nuclear acting as lead auditor. This audit served as the initial basis for Holtec to be placed on PGE's ASL. The audit report described serious problems with the way Holtec implemented various commitments of their QA program at a fabrication facility for safety-significant equipment. The concerns related to process controls in welding, document control, and lack of documented procedures. The problems were satisfactorily resolved, and the audit report demonstrated that oversight of Holtec through NUPIC in this instance was effective. The sub-tier fabricator involved has not and will not be used on the Trojan ISFSI.
>
> A PGE surveillance conducted in January 2001 served as a followup to the NUPIC audit in May 2000 to deal with Trojan-specific issues. The report revealed only minor findings.
>
> A PGE surveillance conducted in April 2001 reviewed calculations dealing with dimensions of the MPCs, heat load evaluations of the Concrete Casks, and various design specification reports, one of which dealt with environmental corrosion. There were only minor findings.

Although the adequacy of the actual surveillances or audits could not be assessed from such reports alone, these reports provided verification that PGE's QA program was addressing significant issues with their primary vendor.

In addition to Holtec, we reviewed several audits and surveillances of their primary fabricator, UST&D, including the following:

> NUPIC conducted an audit at the UST&D facility in Pittsburgh, PA, in June and July 2000 with Commonwealth Edison as lead auditor. This audit resulted in 9 findings, and revealed that UST&D required "enormous improvement in the areas of Special Processes and Corrective Action Program to effectively implement its QA Program elements." Some of the findings appeared significant. For example, the calibration for a welding rod oven was three months overdue; welding was permitted without an approved procedure; and witness points in procedures were bypassed after this had been identified as a problem. According to followup correspondence, all of these findings were satisfactorily resolved.
>
> A PGE surveillance conducted in May 2001 reviewed primarily process controls in welding and non-destructive examinations (NDE). There were six significant findings, the result of which was a statement by PGE to the management of UST&D that "fabrication of PGE items not commence at USTD" until these items were satisfactorily resolved. Subsequent correspondence showed that all were resolved.

3. OOE Staff Observation of a PGE Surveillance of UST&D

In addition to reviewing PGE's audit and surveillance reports, two OOE Staff members observed a PGE surveillance performed by a joint QA and engineering team at UST&D's fabrication shop in Pittsburgh, PA, in July 2001. The objectives of our participation were to assess the adequacy of UST&D as a vendor of MPCs and the adequacy of PGE's oversight activities. To meet these objectives we toured the facility and observed fabrication in progress, interviewed key personnel, reviewed selected UST&D fabrication and quality assurance practices, reviewed PGE's oversight program, including audits, surveillances, and use of onsite representatives.

Our document reviews included:

1. Selected portions of the UST&D Quality Assurance manual and procedures
2. Selected nonconformance and corrective action reports, including follow-up actions
3. Corrective actions taken on NUPIC and PGE audit and surveillance findings
4. Involvement of UST&D QA personnel in the fabrication process
5. Procedure for and selected samples of shop travelers / manufacturing process sheets
6. Procedure for establishing witness and hold points during fabrication

The Staff observations are documented in an OOE report kept on file and summarized here. In general we found UST&D's quality program to be effectively implemented. We made some observations regarding their nonconformance (NCR) reporting program. Several NCRs, which had been documented in the May 2001 PGE surveillance, had yet to be fully resolved. Another finding was that when training was prescribed as a corrective action, it was not always adequately documented. A further issue was that although anyone at UST&D could write an NCR, the practice was to leave this function to QC personnel, thereby limiting the involvement of shop personnel. We also found several documentation issues with nonconformances, such as not indicating whether a problem was an isolated event or had generic implications.

We found that several of UST&D's responses to the NUPIC audit findings and to PGE surveillance findings could not be adequately verified; particularly in areas of disagreement. Some of the responses lacked detail or did not address all aspects of the finding. In the areas where UST&D agreed with a finding, which was most cases, they appeared to provide sufficient information to form the bases of adequate closure. In all cases, the responses were investigated and subsequently accepted by the HUG (Holtec Users Group) representative. Additionally, PGE followed up on the NUPIC findings and found sufficient corrective action to close them.

At the end of the surveillance, we provided our comments and observations to the UST&D management. PGE stated that our observations would be added to theirs for appropriate followup and completion. In conclusion, the Staff was satisfied with UST&D's program, and it appeared that PGE is applying an appropriate level of oversight with UST&D to ensure that the quality requirements of the ISFSI components are met.

4. PGE QA Onsite Inspection at UST&D

As with BNFL/SNC and ABB, PGE has assigned a fulltime resident inspector at UST&D. In response to a question early in our review about QA/QC involvement in the manufacturing process for the ISFSI components, PGE responded that "Total surveillance time is expected to be approximately 50% of the fabrication time."[24] This estimate appears as though it is being met. In addition, PGE reviews the surveillances of the Holtec Users Group (HUG). Its significant to note that PGE does not participate in HUG, but chooses to perform an independent oversight program. We believe that this represents a strength in the PGE oversight program.

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[24]ibid.

5. PGE QA Receipt Inspections

The Staff observed PGE QA's receipt inspections on the first MPC, the Transfer Cask, and the Lift Yoke. Our observations regarding the receipt inspections are detailed in Sections IV.B.4.c, IV.B.5.b, and IV.B.6.b of this report. We found that PGE's receipt inspections were thorough, and that numerous problems were being discovered. For example, the Lift Yoke and one of the MPCs was received with incorrect documentation, wood waste was found in one MPC, several of the first "drag tests" to determine if the internal basket cells are correctly sized failed, and there were problems with the welds on the spacers that go into the basket cells. As a result of such problems, PGE QA decided to perform 100% inspections of everything fabricated at UST&D that is considered important to safety. This also raised concerns about the amount of effective oversight that Holtec was performing on UST&D, and whether the focus of both vendors was more on schedule than quality. The Staff was satisfied that all of these issues were being adequately followed and resolved by PGE.

6. Conclusion

The Staff has extensively reviewed PGE's QA involvement in the ISFSI project. PGE's QA program meets all regulatory and procedural requirements, and their program of reviews, audits, surveillances, and other activities appears to go into sufficient depth to assure of satisfactory implementation of the QA program elements. 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 the OOE Staff has reviewed, there has been and continues to be a high level of PGE QA involvement in all critical onsite activities for the ISFSI project.

We have reviewed the QA program of Holtec, PGE's primary contractor, and UST&D, the major fabricator of ISFSI components. In both cases, their QA programs appear adequate to ensure that the ISFSI is designed and constructed in compliance with EFSC rules. We have observed that audit and surveillance findings are being satisfactorily resolved. In addition to PGE's audits and surveillances, PGE has assigned a fulltime inspector at UST&D to monitor fabrication activities.

IV. ANALYSIS OF THE ISFSI DESIGN

A. VALIDATION OF COMPUTER CODES

1. Review Criteria for the Use of Computer Codes

One review topic that covered every analysis was the verification and validation (V&V) of computer codes used as analytical tools. Briefly, verification refers to the effort to ensure that a computer software program has been appropriately written and approved for the task for which it has been designed. Validation, sometimes referred to as benchmarking, refers to the testing of a computer software program to ensure that it is appropriate and capable of performing the task for which it has been designed.

The Staff 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 computer codes were used, the Staff attempted to resolve the following questions:

What computer software codes were used in the ISFSI analyses?

Are these computer codes used widely in the nuclear or related industries?

Were the computer codes appropriately verified and validated (or benchmarked) by the vendor?

Did PGE QA review the vendors' verification and validation of computer codes?

Has the NRC approved of each of the computer codes used in the ISFSI analyses?

2. PGE and Vendor Validation of Computer Codes

The SAR and other applicable documents refer to the following computer software codes used in various analyses in the ISFSI design, all of which are used widely in the nuclear industry. Those identified in brackets are ones that were used with the SNC/BNFL design:

> Criticality analysis: Monte Carlo N-Particle Version 4a (MCNP4a) [SCALE-PC Modular Code, Version 4.1]
>
> Shielding and dose calculations: Monte Carlo N-Particle Version 4a (MCNP4a) [QAD-S, QAD-CGGP]
>
> Spent fuel gamma and neutron source strengths: SAS2H and ORIGEN-S modules of the SCALE 4.4 system [OCRWM computer database]
>
> Thermal-Hydraulic stress analysis: FLUENT Computational Fluid Dynamics (version 4.48) [ANSYS and COBRA-SFS]
>
> Drop accidents: LS-DYNA

Our earlier review of the SNC design determined that PGE's oversight of the use of vendor computer codes was satisfactory. The results of that review are contained in our January 27, 1999 report.

For the change to the Holtec design, the Staff asked PGE if the software codes used in this application had been appropriately validated and benchmarked in Holtec's QA program, and had this been verified by PGE's Nuclear Oversight. PGE responded:

> The software codes have been appropriately validated and benchmarked in the Holtec QA program. PGE Nuclear Oversight conducted three separate surveillances of design control

activities at Holtec, including review of Holtec software verification and validation (V&V) functions. Specifically Supplier Surveillance SSP 01-06 reviewed V&V of FLUENT and MCNP software codes used for PGE's project by Holtec.[25]

The Staff has gained confidence in PGE's QA audits and surveillances and accepts this response. Additionally, we reviewed the above referenced PGE Supplier Surveillance report and confirmed PGE's review of Holtec's V&V process. We also verified that each of the Holtec calculations that used a computer code stated that the code had been validated under the Holtec QA program. However, we found that PGE's QA did not verify the LS-DYNA software program, which was used in the drop analysis of an MPC into a Concrete Cask. This is a significant drop analysis, because it generates the largest deceleration (g-loading) on the fuel and MPC shell. PGE QA auditors had appropriately select a sample size to verify, and this particular software code had not been selected. However, because of the significance of this analysis, PGE agreed to review Holtec's V&V of this code during their next surveillance of Holtec and before loading fuel.

3. NRC Approval of Computer Codes

In accordance with OOE's memorandum of understanding with the NRC, the Staff asked the NRC during our initial review of the SNC designed system about their review of computer codes. NRC representatives stated that they review computer codes for important-to-safety analyses in all license applications for ISFSIs. They review all input and output data. A multi-discipline team, including contractors, then review the data and all assumptions used. The NRC also verifies 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 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.

For the review of the Holtec design, the Staff asked PGE if each of the software codes mentioned by Holtec or PGE in the applicable submittals to the NRC regarding the Trojan ISFSI had been approved for use by the NRC. PGE responded that "The software codes used in support of the Trojan ISFSI are the same as those approved by the NRC through the Holtec HI-STORM 100 certification process." We followed this with a specific question to NRC representatives at a meeting with the NRC on February 7, 2002, regarding the FLUENT code used in the thermal analysis. Had the NRC approved the use of FLUENT and did that approval include the validation and benchmarking of the code? An NRC spokesperson stated that it had been approved, and that a national lab had performed the validation and benchmarking.

4. Conclusion

Based on the results of our reviews of PGE audit and surveillance reports, our witnessing of PGE QA reviews of vendor validation and verifications, and our discussions with the NRC, we have concluded that the use of the computer codes referenced in PGE's SAR is acceptable.

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[25] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

B. 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. The structural analysis for loading operations is described in LCA-237 and reviewed in Section V.E of this report. For normal ISFSI operations, the Staff 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?

Have the welding and structural problems with the previous design been resolved?

Are PGE and Holtec correct in claiming that leakage from an MPC is not a credible event?

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

Section 3.4 of the SAR and NRC regulations at 10CFR72.3 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 (ITS) components is significant because NRC regulations at 10CFR72.122 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, MPC, Fuel Debris Process Can Capsule, Failed Fuel Can, Damaged Fuel Container,Transfer Cask, Transfer Station, Transfer Station Pad and impact limiter, and MPC Lift Cleats.

In response to a question regarding the ITS status of the Lid Retention System, designed to keep the MPC lid in place when a loaded but unsealed MPC is being transported in the Transfer Cask, PGE responded that it is designated ITS, but is not listed separately, because it is considered to be a part of the Transfer Cask.[26] In response to a similar question regarding the Lift Yoke, PGE responded that the fabrication of the Lift Yoke is consistent with ITS standards, but that the ITS designation is limited to Part 72 applications. Since the Lift Yoke is not used at a Part 72 facility (e.g., the Transfer Station) to lift a loaded MPC, it has not been so designated. The Staff got a similar response regarding the Fuel Building Crane. We also verified that there has been significant Nuclear Oversight (QA) involvement in crane maintenance and modification work. NRC inspectors have also found PGE's crane maintenance program acceptable.

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

3. Principal Design Criteria for the Major Components

a. ASME Boiler and Pressure Vessel Code

The design basis for the MPC, Transfer Cask, Failed Fuel Cans, Damaged Fuel Containers, and Fuel Debris Process Can Capsules, as described in Section 3 of the SAR, is the ASME[27] Boiler and Pressure Vessel Code. The ASME Code is the accepted standard for materials, design, fabrication, examination, and testing of steel confinement casks in the nuclear industry. A detailed discussion of the application of the ASME codes for the ISFSI was included in our 1999 report.

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 that the allowable stress intensities (S_m) for the important-to-safety components were based on conservatively determined temperatures, and that load combinations for the MPC were appropriately added before they were compared to the ASME service level limits.

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.

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[26] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002
[27] ASME: American Society of Mechanical Engineers

4. Structural Evaluation of the MPCs

a. MPC Design Internal Pressure

SAR Section 3 provides the load combinations and design strengths for the MPCs. The Staff verified these design requirements by comparing them with Holtec's License Amendment Request LAR 1014-1[28] and the requirements of ASME Section III, Subsection NB. We also reviewed the Holtec calculation package, "Thermal-Hydraulic Calculations for Trojan ISFSI Completion Project," to determine the effect on the MPCs of maximum accident pressure. We verified that the assumptions were appropriate and conservative, the analytical methods were appropriate; and the results were reasonable. For maximum MPC pressure, HI assumed 100% of all fuel rods fail and release 30% of all fission gas to the inside of the MPC. 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 109.5 psia, and the resulting stresses were within those allowed by the ASME Code. Based on these calculations, PGE concluded and OOE concurs that the MPC is conservatively designed to withstand any potential internal pressures.

b. MPC Welding Concerns

The clearance between the MPC and Transfer Cask, as previously stated, will be less than 1/8-inch. This raised concerns about the welding procedures with the MPC. The fabricator of the SNC PWR Basket had problems with some distortion or "hour-glassing" on the prototype baskets. As a result, the welding procedures had to be modified. We asked PGE if this potential problem been addressed with Holtec or UST&D. PGE responded as follows:

> The Holtec design has a significantly less welding of internal components to the shell than the previous PWR Basket. This reduces the potential for " hour-glass" problems. Also, UST&D is very conscious of the potential for distortion when welding MPC shells and clearly understands the implications of being out of tolerance. To minimize shell distortion of any type, including "hour-glassing," the following three-step approach is implemented:
>
> a. The weld prep is designed to allow welding from inside and outside of the shell.
> b. The first side of the weld is completed from the inside of the MPC shell. This weld is then back-gouged from the outside of the shell. A copper backing bar is used to help remove heat and control distortion.
> c. The weld is completed from the outside of the shell.
>
> Working from two sides allows the outside welding to help counteract any distortion caused by the welding begun on the inside of the shell. The copper backing bar assists in removing heat from the welded joint to also help control distortion. This approach has been found to be easily repeatable and very successful in minimizing MPC distortion in the many MPC shells fabricated to date.[29]

c. MPC Receipt Inspections

The Staff reviewed receipt inspection package for the first MPC, and found it satisfactory, with the exception of the "drag tests," which are used to determine if the internal basket cells are

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[28] Holtec International, License Amendment Request, LAR 1014-1, Rev. 2, October 2001
[29] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

correctly sized so that fuel assemblies will fit. In March 2002, we witnessed "drag tests" on the first MPC to arrive onsite, and two of the three tests we observed failed. PGE had only planned to test four cells with each MPC; however, the results of these tests and additional failed cells on the other two MPCs prompted several corrective actions. Holtec and UST&D were notified so that these problems could be corrected at the fabrication shop. The drag tests at UST&D, which had previously been conducted in a horizontal orientation, were changed to a vertical orientation; the test gauges at each facility were configured identically; PGE revised their testing program to include 100% of all cells; and repair procedures using a hydraulic jack were instituted. Subsequently, there have been few drag test failures.

In addition to the regular cells, the Staff noted that the dimensions on the drawings indicated that the fit between the Failed Fuel Cans and the corner cells would be tight. Since there had been no plans to perform "drag tests" on these cells until just prior to loading, we asked PGE to check the fit-up of the Failed Fuel Cans in the MPCs early enough in the process so that there would be sufficient time without a delay in loading for necessary repairs. PGE responded:

> A test fit of a Failed Fuel Can in each of the four oversized corner cells will be performed on each MPC in accordance with HPP-1135-130, Trojan ISFSI Completion Project MPC Off Load and Rigging Procedure. A spare Failed Fuel Can will be used as a test fit can during MPC receipt inspection. Checking the corner cell Failed Fuel Can fit up as part of the MPC receipt inspection routine will allow sufficient time to implement corrective actions (as needed) to avoid fuel load delays. In addition, 100 percent drag testing of all normal cells is being performed on-site.[30]

We also asked PGE about checking for out-of-round conditions with the MPC, since any such condition would have the potential of causing interference with the Transfer Cask. PGE responded:

> …an out of roundness dimensional check is performed during the initial receipt inspection in accordance with Holtec procedure HPP-1135-130… This allows the team to evaluate the condition of the canister and take any corrective action necessary prior to commencing lid fit-up. At the top and bottom of the MPC, any out-of-roundness is minimized by the presence of the lid and baseplate.[31]

d. Non-Leakage Credibility of the MPC

(1) Higher Pressure in the MPC

The major concern with the change to the Holtec MPCs has been the question of leak tightness. The SNC design, in addition to reliance on multiple welds and non-destructive testing of those welds, also relied on a lack of differential pressure across the boundary of the PWR Basket to ensure that leakage was non-credible. The internal PWR Basket pressure was essentially atmospheric. In other words, there was no differential pressure or driving force that could create a leak. As a last defense, if a leak did occur, there was available a sealed overpack in which to place the PWR Basket and thereby stop the leak. Neither of these two characteristics are relevant with the new design. The pressure in the MPC is much higher, and there is no overpack.

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[30] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002
[31] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

The MPC maintains an internal pressure of 36.4 – 72.4 psia (21.7 – 57.7 psig) with a design maximum pressure for normal operations rated at 100 psig. This is a differential of about 7.8 times atmospheric pressure. In spite of this, PGE and the designer, Holtec, consider a leak to be less than credible. ISFSI SAR Section 3.3.2.1 states:

> The MPC is designed to be leak tight under all normal, off-normal, and accident conditions of storage. Based on this design and the very low probability that a flawed confinement boundary weld would be performed, pass NDE inspection, and subsequently leak, the leakage of an MPC is not considered a credible event.

(2) NRC's Probability Study on Leakage

The OOE Staff reviewed the bases for this assertion, which are described in several documents. One is a probability study performed by the NRC and Pacific Northwest National Laboratory, entitled "Estimated Risk Contribution for Dry Spent Fuel Storage Cask."[32] This paper uses the PRODICAL computer code to evaluate the risk to the public of the likelihood a leak due to a faulty weld on a stainless steel canister. It uses the HI-STORM 100 MPC as a model, which is essentially identical to the ones proposed for use at Trojan. The purpose behind the study is a proposal to use as many as 4000 of these cask systems at an interim storage facility called the Private Fuel Storage project which may be established on the Skull Valley Reservation in Utah.

This study includes qualitative arguments (expectation of welding, fabrication, and inspection standards) that a leak will not occur. It also uses the PRODIGAL computer code to assess the probability of leakage under both normal and accident conditions. For example, for the MPC lid-to-shell weld, the major field weld and the most significant pressure boundary weld, the probability for a flaw equal to 87% of the weld thickness due to fabrication was determined to be an extremely low 5.8×10^{-9}. One of the assumptions governing this study is that dye-penetrant examinations, which are to be used at Trojan, have "an effectiveness of 99% in detecting small surface-breaking defects." The study also recognizes that for a failure to occur in the lid-to-shell boundary there must be through-wall cracks in both the lid-to-shell and closure ring-to-shell welds. The probability for that occurrence is determined to be 9.7×10^{-6}. For the accident analysis the study determines that the probability for a leak forming in the lid-to-shell weld due to the bottom end drop accident is 5.2×10^{-6}, also extremely low.

The Staff questioned some of the study's basic assumptions. The paper states, for example, that "these methods do not address less significant welding flaws such as porosity that have little or no potential to degrade structural integrity, but do have the potential to provide a pathway for leakage. The assumption in the present work was that such non crack-like flaws will not grow under service conditions." Another concern by the Staff had to do with the fact that the paper appeared only concerned with the immediate post-fabrication condition of the welds and not with the effects of long-term storage conditions. There is no discussion in the paper about whether or how many such flaws could become through-wall cracks after 20 plus years of being exposed to various environmental conditions. The only comment on this was that "Since the HI-STORM MPC is constructed from stainless steel materials with an inert internal environment of helium gas, RES staff does not postulate any viable active degradation mechanisms which could cause welding flaws to grow in the MPC."

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[32] C. Santos, F.A. Simonien, M.T. Kirk, L. Abramson, J. Guttman, E. Hackett, Estimated Risk Contribution for Dry Spent Fuel Storage Cask

In response to our concerns, PGE provided various proprietary technical papers written by Holtec Staff on the durability of the MPC canisters. One paper[33] dealt with the acceptable flaw size in the MPC lid-to-shell weld, and concluded that through-weld cracks in austenitic stainless steel welds are improbable due to the fracture toughness of the material. It stated that the fracture stress intensity of a 50% through-weld crack develops a fraction of the material's fracture toughness. In other words, the ability of the material to resist crack propagation is significantly greater than the driving potential of a crack. It further listed the crack propagation mechanisms for austenitic stainless steel as hydrogen embrittlement, cyclic fatigue, and stress corrosion cracking. The first two are not credible due to the lack of hydrogen and low level of cyclic fatigue on the ISFSI pad. The third, stress corrosion cracking, requires stress plus an inimical environment (oxygen plus a halide, which is a binary salt formed with metal and either fluorine, chlorine, iodine, bromine, or astatine). It concluded that the stresses on the MPC compared to the fracture toughness of the material are slight, and internal corrosion is not possible because of the inert helium environment. A second paper[34] dealt with the resistance of Holtec cask systems to corrosion in environmental conditions, particularly dissolved halogen in the atmosphere. This second paper concluded that the austenitic stainless steel canisters are extremely resistant to industrial atmosphere and acidic media.

A third paper[35] dealt with the leak tightness of the MPCs, describing possible failure mechanisms versus the loadings and conditions to which the MPCs would be exposed. It evaluated internal and external environments, various environmental phenomena, temperatures and pressures, and cyclic loadings against possible failure modes, and concluded that "the establishment of a thru-wall leakage path in the enclosure vessel cannot be demonstrated based on physical, metallurgical and mechanical factors that govern the design of a Holtec MPC." While we could not duplicate these studies, the Staff concluded that the conclusions were reasonable.

(3) Leak Tightness Testing on MPC Mockup

In addition to these papers, Holtec also performed leak tightness testing on an MPC mockup on March 22, 2002, at Entergy Nuclear's Fitzpatrick plant. In PGE's response to an RAI question about this test, they stated: "The as-tested leak rate of the MPC was 2.97×10^{-8} atm cc/sec, which is less than the leak tight acceptance criterion by nearly an order of magnitude." PGE went on to state: "It should be noted that this 'measured' leakage was, in all likelihood, actually residual helium left in the MPC test cavity from the test instrument calibration since, under near vacuum conditions, it is impossible to remove all residual helium due to the lack of a fluid medium." The Staff reviewed Holtec's report[36] on this test. Although the test arrangement, criteria, and assumptions appeared to be reasonable and conservative, and the test was witnessed by NRC representatives, it did not appear to be a realistic assessment of the conditions of an actual MPC. It would not be unreasonable to assume that the fabrication of the test vessel was performed under closer scrutiny than that of a production MPC. Also, the test only considered the MPC lid-to-shell weld, and not the shell-to-baseplate, longitudinal, and circumferential welds. Therefore, while the test was a successful empirical model, it did not appear to represent a definitive argument for non-leakage credibility.

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[33] Holtec International, DS-213, Rev. 2, Acceptable Flaw Size in MPC Lid-to-Shell Welds, February 23, 1999
[34] Holtec International, DS-252, Rev. 1, Resistance of Holtec Cask Systems to Corrosion in Environmental Conditions, March 26, 2001
[35] Holtec International, DS-256, Rev. 3, On the Leaktightness of HI-STAR/HI-STORM MPCs, July 7, 2001
[36] Holtec International, HI-2022850, Rev. 0, Summer Report on MPC Leak Tightness Test, April 10, 2002

(4) Hydrostatic Pressure Testing of the MPC Welds

Supporting evidence for non-leakage of the MPCs is, however, provided in the two field tests that will be performed on each MPC. The first is the hydrostatic pressure test of the MPC welds. LCA-237[37], Section 4, and ISFSI SAR Section 5.1.1.2 state: "Once the MPC lid weld is completed, the MPC is refilled with borated water and hydrostatically tested to at least 1.25 times the maximum normal design pressure of 100 psig. This pressure is held for 10 minutes with no observable leakage as the acceptance criteria."

The Staff had one concern with this test. All of the MPC welds are subjected to this hydrostatic pressure test; yet, during the test it is impossible to observe leakage from the axial and circumferential seam welds and from the shell-to-baseplate weld, because the MPC is situated within the Transfer Cask. Because of this limitation, PGE has taken exception (SAR Table 4.2-1a) to ASME Code Section NB-6000 for these welds, specifically Section NB-6224, which requires observable leakage as the test criterion. As an alternative, PGE states that these welds are examined volumetrically (either by radiographic RT or ultrasonic UT testing) and helium tested in the fabrication shop. While this alternative appears acceptable, the Staff questioned in our RAI why, in addition to observing leakage during the field test, PGE couldn't also monitor test pressure, such that were test pressure to decrease, they would have evidence of a leak, even if it occurred in a hidden location. PGE responded: "After stability is confirmed, a visual inspection of the weld area is performed. In addition, the test pressure is monitored and if it were to decrease below the minimum during the test, the test would not be valid." Subsequently, we learned that neither the loading procedure, FHP 50-03, nor Holtec's hydrostatic test procedure required monitoring pressure for leakage. In fact, Holtec's procedure kept the test pump running to maintain test pressure. In a meeting with PGE representatives on June 13, 2002, PGE committed to stopping the pump when test pressure was reached and using decaying test pressure as an additional means of determining leakage.

(5) Helium Leakage Testing

The second field test is the helium leakage test performed shortly after the hydrostatic pressure test. LCA-237, Section 4, states:

> After successful dye penetrant testing [of the lid-to-shell weld], approximately 20 gallons of water are removed from the MPC, with the water being replaced with helium during the draining. The helium is pressurized to a nominal test pressure of 90 psig and the MPC lid weld is helium leak tested. The maximum permissible MPC leak rate is $\leq 5\times10^{-6}$ atmosphere-cubic centimeters per second (atm-cc/sec) based on a pressure differential of one atmosphere across the confinement boundary.

The Staff reviewed the acceptance criteria for this test. Atm-cc/sec refers to a "mass-like" leak rate and represents a correlation between the measured volumetric leak rate and the leak rate you would have if there was a 1 atm pressure differential across the crack. Holtec License Amendment Request (LAR)[38] Section 7.3.3.1, p.7.3-3, defines it "as the rate of change of the pressure-volume product of the leaking fluid at test conditions." This depiction of leakage rate takes into account the fact that leakage rate varies as the internal pressure varies. According to

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[37] PGE, Proposed Revision 5 to License Change Application 237, Enclosure I to VPN-033-2001, November 15, 2001

[38] Holtec International, HI-STORM LAR 1014-1, Rev. 2, October 2001

PGE, the limit of $\leq 5x10^{-6}$ atm-cc/sec corresponds to a volumetric leak rate of $\leq 1x10^{-5}$ cc/sec, which is ten times the leak rate limit applied to the SNC PWR Baskets and is based on the 10CFR71.51leakage limits imposed on transportation packages. Were this leakage rate to occur, no more than 2.5% of the helium would leak out in 20 years.

The Staff reviewed the requirements of ANSI N14.5-1997, "Leakage Tests on Packages for Shipment." We confirmed by applying the equations in the standard that the leakage rate limit was conservative. We also confirmed that PGE and Holtec intend to meet the requirements of the standard. Furthermore, in response to an RAI question about the accuracy of the test, PGE stated that the Mass Spectrometer Leak Detector (MSLD) that will be used will be calibrated with a 10^{-8} atm cc/sec range NIST traceable leak standard followed by the MSLD (with sniffer probe attached) System Calibration with a 10^{-5} atm cc/sec NIST traceable leak standard.

(6) NDE Testing of Welds

There is extensive non-destructive testing of the MPC welds. In addition to the fabrication shop testing previously mentioned, there will be extensive dye-penetrant testing of the field welds. For the MPC lid-to-shell weld, dye-penetrant testing will be performed on the root, final, and intermediate weld passes, at each approximately 3/8-inch of weld depth. Another dye-penetrant test will be performed on the lid-to-shell weld following the hydrostatic pressure test. Dye-penetrant tests will also be performed on each pass of the Vent and Drain Port Cover Plate welds followed by helium leak testing. And lastly, a dye-penetrant test will be performed on the final pass of the Closure Ring weld, which is a redundant seal over the other lid welds.

(7) Conclusion

In conclusion, the NRC's probability study and various Holtec position papers provide sound qualitative, probabilistic, and engineering arguments that a leak from a faulty weld in an MPC on the ISFSI Storage Pad is not credible. The leak tightness test conducted at Fitzpatrick provides questionable empirical evidence; however, the testing on every production MPC, including volumetric and dye-penetrant examinations, the field hydrostatic pressure tests, and helium leak tests provide reasonable assurance that leakage is not a credible event.

5. Structural Evaluation of the Transfer Cask

a. Design Criteria for the Transfer Cask Trunnions

The Transfer Cask trunnions on the new Holtec design are different than those analyzed under the SNC design; however, the requirements remain the same. The American National Standards Institute (ANSI) publishes nationally recognized standards for lifting devices. According to ANSI N14.6-1993, 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 trunnions are considered special lifting devices and are designed and fabricated to the requirements of ANSI N14.6-1993 and NUREG-0612 (1980). For these devices, the Staff 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 8.2 for yield strength and 10.2 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.

ISFSI SAR Section 4.7.3.1 states that the trunnions are tested to 300 % of their maximum design load. We asked PGE about the test that was performed, whether the test conditions adequately represented actual load conditions, whether it had been witnessed or verified by PGE personnel, and if the trunnion welds and critical areas were NDE tested following the load test. PGE responded:

> Two load tests are performed on the Transfer Cask that confirm the lifting capacity of the lifting trunnions and the structural load path of the entire cask. One test is at 300% load (lifting trunnion test) and one test is at 150% load (cask load path test). The 300% test involves engaging the lifting trunnions with a load test fixture and applying a static load between the trunnions and top flange. The lifting trunnions are visually inspected and the accessible trunnion block welds are liquid penetrant inspected.
>
> The stress state in the cask during an actual lift is more accurately represented by the 150% load test than the 300% load test, since in the 150% load test the entire cask load path (rather than just the trunnion) is tested. A fixture with a center jacking mechanism and two lift arms is used to apply a load equivalent to 150% of the design load to the inside surface of the cask bottom doors. The lift arms are similar in configuration to the cask lifting yoke arms and are attached to the lifting trunnions as they would be during a lift in the actual field conditions. The load from the jacking mechanism is transmitted through the bottom doors, the bottom door attachment mechanism, the Transfer Cask structure, and the lifting trunnions, via the fixture and the lift links. Again the lifting trunnions are visually inspected and the accessible trunnion block welds are liquid penetrant inspected. The welds in the structural load path were visually and liquid penetrant examined and the doors were checked for proper clearance and operation.
>
> These load tests performed at Holtec's fabricator, US Tool and Die, were observed by PGE Nuclear Oversight's on-site surveillor. The load tests were performed using approved procedures and calibrated equipment and were appropriately documented. PGE ISFSI Project and Plant Engineering personnel provided guidance for what to focus on during witnessing of these tests.[39]

Based on the design criteria and this response, the Staff concludes that the design of the Transfer Cask trunnions are adequately conservative.

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[39] PGE letter VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

b. Transfer Cask Receipt Inspections

(1) Dimensional Checks of the Transfer Cask

The Staff inspected the Transfer Cask and witnessed dimensional checks performed by PGE. We also reviewed the receipt inspection package for the Transfer Cask to verify that the final dimensions were in accord with PGE's concerns about the width of the annulus gap. We verified that the inside diameter was machined to maximize the tolerances and ensure that an MPC would fit inside the Transfer Cask without interference. We also reviewed the nonconformances and found that following machining at UST&D, an MPC mockup gage did not successfully pass through the area between the trunnion blocks, requiring the removal of additional material. Even with this, the fit will be tight, particularly if an MPC is slightly out-of-round or if the weld crowns are not sufficiently reduced. This remains a concern; however, it is a scheduling concern rather than a safety one. Finite element analyses demonstrated that the removal of the additional material, approximately 3/16-inch, did not adversely impact the strength of the materials.

(2) Transfer Cask Load Tests

The Staff reviewed records of the 300% load test in the receipt inspection package. It appeared that two of the welds on the trunnion blocks, which hold the trunnions and provide for the lifting of the Transfer Cask, might not have been tested, which would appear to be in violation of ANSI N14.6, which requires that following the load tests, "critical areas, including all load-bearing welds, shall be subjected to nondestructive testing." The records indicated that "accessible areas" of the trunnion and trunnion block were tested with NDE methodology, but all the welds were not specified. We brought this to PGE's attention. They concurred with our finding, but stated that all of the critical areas and welds would be checked prior to loading, as described in the following section "Periodic Testing and Inspection of the Transfer Cask and Lift Yoke." The Staff is satisfied with this response.

We reviewed the results of the load test equal to 150% of the maximum service load required by ANSI N14.6-1993 Section 6.2.1. Maximum service load is defined as "The maximum load that a special lifting device can lift as established by load testing." The results appeared satisfactory.

A concern with both of these load tests was that there was no indication that the trunnions had been removed and the threads examined for damage or distortion. Holtec's procedure HSP-113, Rev. 2, "Trunnion and Support Lug Load Test Procedure," states only that "After testing, the accessible portion of the trunnions … will be visually examined to verify no deformation, distortion or cracking has occurred." There was also no requirement for NDE in this procedure. We identified this concern to PGE, and they concurred. The trunnions were found to turn freely, indicating that there was probably no damage; however, they had not been thoroughly examined. PGE committed to a close examination of the trunnions during the re-qualification prior to loading. This item remains open and will be verified by the Staff prior to loading.

(3) Periodic Testing and Inspection of the Transfer Cask and Lift Yoke

ANSI N14.6-1993 Section 7.3.1.a states for "Load-bearing members with increased stress design factors for handling the critical load … annual testing shall be conducted in accordance with 6.3.1, except that the device shall be subjected to three times the weight of that the device is to support." ANSI N14.6-1993, Section 6.3.1, states: "Each special lifting device shall be subjected annually (period not to exceed 14 months) to either of the following: a) A test load equal to 150 % of the maximum service load… b) … dimensional testing, visual inspection, and nondestructive testing of major load-carrying welds and critical areas…." Since the loading is expected to last a year, PGE has committed to complying with option b, and to conduct the first test prior to loading, so that there will not have to be an interruption of the loading process. The test will include all critical areas and welds, including those that may not have been tested at UST&D following the 300 % load test. The same applies to the Lift Yoke. This item remains open and will be verified by the Staff prior to loading.

ANSI N14.6-1993 Sections 6.3.6 and 6.3.7 require that special lifting devices be visually inspected for indications of damage or deformation prior to each use and at intervals not to exceed three months. These inspections were not part of the loading procedure; however, PGE committed to including them in a procedure. They would include, at a minimum, a visual inspection for gouges, cracks, deformation or other indications of damage. The same applies to the Lift Yoke. This item remains open and will be verified by the Staff prior to loading.

(4) Gap Flush Connection

The Gap Flush System connection on the Transfer Cask is a 1-inch pipe welded to the side of the bottom plate of the Transfer Cask with a hole drilled through the base plate. The plate and the pipe are carbon steel, and the inside of both are uncoated and will be exposed to the boric acid of the Cask Loading Pit. This will be a source of rust and possibly a small amount of hydrogen through the loading operation. Both the Staff and PGE consider this to be small and possibly insignificant; however, given the situation with the previous PWR Baskets, we suggested that PGE analyze the extent to which this connection will be a source of rust and hydrogen so that it doesn't unnecessarily delay the loading operation. PGE agreed. We asked if there were any other potential exposures of uncoated carbon steel, and PGE stated that they would address this issue. This item remains open and will be verified by the Staff prior to loading.

6. Structural Evaluation of the Lift Yoke

a. Design Criteria for the Lift Yoke

The Lift Yoke is to be used in conjunction with the Fuel Building Crane to lift the Transfer Cask during the loading operation. It is also used with the mobile crane to lift an empty Transfer Cask to the top of the Transfer Station. As with the Transfer Cask trunnions, the Lift Yoke is also considered a special lifting device as described in NUREG-0612 and ANSI N14.6. The Lift Yoke does not require doubled safety factors or load paths for single failure proof handling systems; however, PGE states that it was designed with doubled safety factors. Loads that satisfy the alternative evaluation criteria of NUREG-0612 Section 5.1 need not meet the doubled safety factor requirement. These 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 10CFR100 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.

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, and the design criteria are adequately conservative.

b. Inspection of the Lift Yoke

After inspecting the Lift Yoke and reviewing the receipt inspection package, the Staff reviewed with PGE the testing criteria. The Staff's understanding is that the Lift Yoke is considered, in accordance with ANSI N14.6-1993, "a special lifting device designed with increased stress design factors." This requires that the device be load tested to "three times the weight that the device is to support." However, ISFSI SAR Section 4.7.3.4 states that "the Lifting Yoke is tested to 150 percent of its maximum design load…." This is a change from the previous submittal, which required a 300 % load test. We submitted a question on this issue, and PGE responded that the Lift Yoke was actually tested to 300 % at UST&D, and that Nuclear Oversight representatives reviewed the records of the test. Furthermore, PGE questioned the extent of non-destructive examination (NDE) performed on the Lift Yoke. The requirement in ANSI N14.6-1993 is to subject "critical areas" to NDE testing. However, according to reviewed correspondence and interviews, UST&D did not designate any "critical areas," and, therefore, did no NDE testing following the load test. According to PGE's response to our question, "PGE performed NDE of additional areas of the Lift Yoke at Trojan with successful results."[40] Furthermore, as stated in Section IV.B.5.b (3) above, PGE will conduct "dimensional testing, visual inspection, and nondestructive testing of major load-carrying welds and critical areas" in accordance with ANSI N14.6-1993, Section 6.3.1, prior to loading. This item remains open and will be verified by the Staff prior to loading.

7. Design Criteria and Structural Evaluation of the Concrete Casks

The Concrete Casks did not change as a result of the switch to Holtec. OOE's 1999 report contains a detailed evaluation of the concrete casks' performance and compliance with guidelines of the American Concrete Institute (ACI), whose guidelines are generally accepted in the civil engineering field where concrete specifications are required. The major concern described in the 1999 report was the potential effect of high temperatures on concrete strength. Since PGE was originally authorized to implement dry spent fuel storage in 1999, the fuel has had an additional two years to cool.

In 2002, OOE reviewed Corrective Action Report CAR C-02-0007, which reported damage to Concrete Cask #12 when the air pad that was moving it on the ISFSI Storage Pad ruptured. A portion of the lower part of the cask fractured; however, the damage was slight and easily repaired. The long-term corrective action will be to use 2-inch air pads instead of 3-inch air pads for normal movement, and to ensure prior to use that the air pads are not seriously abraded. This appeared to be satisfactorily resolved.

8. Structural Evaluation of Other ISFSI Components

a. Air Pad System

An air pad system will be used to move a loaded Concrete Cask from the Fuel Building to the ISFSI Storage Pad and to the Transfer Station for transfer of an MPC into a Transport Cask for off-site shipping. 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 to enable lateral movement with a modified fork lift truck. The system requires a smooth surface to transport the casks, and PGE

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[40] ibid.

has constructed such a concrete roadway between the Fuel Building and the Storage Pad. After positioning a Concrete Cask in the desired location on the Storage Pad, the air pads are depressurized and removed.

OOE requested additional information about the air pads and the procedure to be taken if one failed during transport. Specifically, LCA-237 Section 5.3.4, states that "In order to prevent the inner concrete temperature from reaching the long-term operation limit, the amount of time that the air pad system can be inserted into the air inlet openings will be limited to 20 consecutive hours…." This is an increase from 12 hours in the previous submittal. In response to a question regarding this, PGE stated that:

> The increase from 12 hours to 20 hours is the result of additional cooling of the spent fuel, a decrease in the licensed thermal rating of the MPCs, and a refined thermal analysis, indicating the air pads could remain installed for up to 57.1 hours without reaching the short-term temperature limit of 350ºF [for the inner Concrete Cask].

PGE stated in response to an earlier RAI question that potential problems with the air pad system are limited, such as a ruptured air hose or leaking air pad, and could be repaired in situ.[41] Earlier this year, an air pad ruptured while in use, causing damage to Concrete Cask #12. CAR C-02-0007 was written to document the problem. The damage was not structural and was easily repaired. As a result, the long-term corrective action will be to no longer use 3-inch air pads, except for special circumstances, since the drop from 3 inches will cause more damage than a drop from 2 inches. In the future, PGE will use only 2-inch air pads, and they will make sure that the pads are not seriously abraded.

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. 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 Storage Pad provides an unsheltered, outdoor place for the storage of the Concrete Casks. The Staff observed construction of the Storage Pad, including placement of steel rebar and concrete pouring and testing. Since PGE has not modified the Storage Pad since 1999, the conclusions in OOE's 1999 report remain valid.

c. MPC Lift Cleats and Slings

The Staff also reviewed the use of lift cleats and slings for the MPCs, which is a different system to that used on the previous design. Each MPC is lifted using two lift cleats that are bolted (two bolts each) into the MPC lid. SAR Section 4.7.4.4 states: "…the minimum safety factor for each MPC lift cleat is 1.04 over and above the 6 and 10 safety factors suggested by ANSI N14.6." We asked PGE if the slings were capable of holding the maximum weight, including dynamic loads, in the event that one of them failed, i.e. single-failure proof? PGE responded:

> The slings are designed in accordance with ASME B30.9 and NUREG-0612 to lift twice the weight of a loaded MPC, including a 10% dynamic load factor. This "enhanced safety

[41] PGE, CPY-032-96, C.P.Yundt to D.Stewart-Smith, Response to RAI, June 27, 1996

factor" design approach is recognized in the regulatory guidance as equivalent to having redundant loads paths (i.e., single failure proof). In addition, these slings were load tested at the manufacturer to 200 percent of rated capacity in accordance with ASME B30.9.[42]

The SAR does not describe the thread engagement or torque requirements for the lift cleats onto the MPC lid. We asked PGE, and they responded that this was covered in Holtec procedure HPP-1135-690.

9. 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 during the SNC design phase, PGE responded that the design life of the cask system is 40 years. PGE also provided a table listing ISFSI important-to-safety components, materials, functions, and the effects of potential degradation mechanisms. These 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 concurred with this analysis. After the switch to the Holtec system, the Staff verified that the MPC was also qualified for a 40-year life.

OAR 345-26-390(4)(j) also requires that PGE's plan for an ISFSI discuss the options available if the expected lifetime is reached and no Federally licensed permanent disposal or storage facility is available. In its 1999 report, OOE recommended that the Council find PGE in compliance with this rule, based on information in PGE letter CPY-032-96. Notwithstanding actions taken by Congress the US Department of Energy in 2002 regarding Yucca Mountain, OOE considers the 1996 letter CPY-032-96 to still be the appropriate basis for compliance with this rule.

b. Corrosion

Corrosion could be a major impediment to a 40-year life for ISFSI components; however, all of the important-to-safety components are either inherently resistant to corrosion or protected. The contents of the MPC are in an inert, helium environment. 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 MPCs. The entire MPC, with the exception of the Boral panels, is fabricated from austenitic stainless steel (304), which is a significant difference from the previous SNC design. The carbon steel liners of the Concrete Casks have corrosion resistant coatings and are separated from contact with the MPCs by ceramic tiles to prevent galvanic corrosion. The Concrete Cask lids and the Transfer Station are also coated. The Transfer Cask and Lift Yoke are coated with an epoxy-based material suitable for borated water service. When not in use and during the years prior to shipping the MPCs to a federal repository, the Transfer Cask, Lift Yoke, Lift Cleats, and other components will be stored in a manner to prevent corrosion and other degradation. For these reasons, PGE concludes, and OOE concurs, that the ISFSI is satisfactorily designed for a 40-year life.

[42] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

c. Coatings for Corrosion Protection

The primary reason for the change to the Holtec design was a failure of the coating on the SNC-designed PWR Basket internals during loading operations in July 1999. The MPCs have no coatings, and the coating used on the Transfer Cask is an epoxy-based material with a proven record in borated water. Therefore, the Staff believes the previous problems will not recur.

d. Storage of the Transfer Cask and Other ITS Components

We asked PGE if there was a program for storing the Transfer Cask and other Important-To-Safety components, such as the MPC Lift Cleats and Removable Valve Operating Assemblies, for the 40-year design life of the ISFSI. We reviewed procedure TIP-10, The Transfer Cask and Concrete Cask Handling and Storage Program, revision 4, which states that "The Transfer Cask and Lifting Yoke shall be stored in a manner to protect against damage or degradation." There was no mention of the other items that will require adequate storage, and no mention of measures that would be taken to ensure that uncoated carbon steel would be protected. For example, the inside of the Transfer Cask water jacket is uncoated carbon steel, which will require a rust preventative to ensure that the Transfer Cask will be usable when it is time to place the MPCs in Transport Casks. This item remains open and will be verified by the Staff prior to loading.

C. 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. The thermal analysis for loading operations is described in LCA-237 and reviewed in Section V.F of this report. For normal operations, the Staff 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?

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

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 the spent fuel is transferred to the MPC shell. Air in the annulus, the space between the Concrete Cask liner and the MPC shell, is heated by the shell temperature. This heating causes a reduction in the density of the air, which then rises through the annulus and exits the four outlets near the top of the Concrete Cask. Cooler, ambient air is then drawn into the annulus via the four inlets at the bottom of the cask due to the lower pressure caused by the rising, warmer air, and the cycle is repeated.

For the steel components (MPC, MPC basket internals, Fuel Debris Process Can Capsules, Failed Fuel Cans, Damaged Fuel Containers, and Transfer Casks), the principal thermal criteria come 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 MPC 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 MPC 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 are based on reports issued by Pacific Northwest Laboratory. Those reports, and the limits adopted by the NRC, are described in detail in OOE's 1999 report. The change to a Holtec design did not result in a change to the recommendations in that report.

3. Thermal Analysis of the MPC

a. Thermal Analysis of the MPC in a Concrete Cask

As stated previously, the design for cooling the spent fuel in the casks is passive and relies on the natural circulation of air through the 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, for the original design, in calculations TI-030, Concrete Cask Air Flow Analysis; TI-033, Concrete Cask Thermal Analysis; and TI-034, Concrete Cask Thermal Transient Analysis. After the design change to the MPCs, Holtec performed a thermal analysis of the new MPC in the Concrete Cask, calculation HI-2012676, Thermal-Hydraulic Calculations for Trojan ISFSI Completion Project. This calculation determined the projected normal, off-normal, and accident temperatures that are listed in Table 4.2-12 of the SAR. All of these temperatures have been revised because of this new calculation, most in a conservative (lower) direction.

The SAR for SNC casks referenced heat loads of 24 kW and 26 kW per cask. This was due to the BNFL/SNC design limits. Their shipping cask, which was to be used under 10CFR71 to transport a basket to an off-site disposal, is rated for 24 kW; and their storage system, which was to be used under the 10CFR72, is rated for 26 kW. Therefore, these were the numbers used in the SNC analyses. For the Holtec MPC, all analyses were based on Trojan specific design limits rather than generic or license limits. PGE calculated in early 1999, based on fuel loading patterns, that the highest heat load from any cask would not exceed 16.5 kW. In 2002, even this number is conservative.

The Staff reviewed the thermal analysis calculations using the review criteria and methods described in Section II.D.1 of this report and found them satisfactory. We compared the Holtec analysis with the previous calculations, and found, in all cases, that the revised results were expected based on the new inputs. Specifically, the Holtec analysis assumed actual Trojan fuel parameters rather than generic parameters, including 9-year rather than 5-year cooled fuel, and 17.4 kW maximum heat load in an MPC rather than 26 kW for the SNC design. The 17.4 kW is still conservative, given that the actual maximum heat load is less than 16.5 kW. Other major assumptions included: neglecting the heat dissipation through the top and bottom of the MPC, thereby maximizing the rate of heating the annulus air; minimizing the internal pressure and heat convection by the helium, and overstating flow resistance of the air passages in the Concrete Cask. We confirmed that the smaller annulus size with the MPC in the Concrete Cask (as compared to the SNC PWR Basket) was accounted for in the analysis. Further confidence was gained due to the methodology in this calculation having been used for three other established

ISFSIs; and, as stated previously, the computer code, FLUENT, was approved by the NRC and validated by a national lab. It was also reviewed by PGE's QA. In addition, we also verified the thermal properties of the materials listed in Table 4.2-13 of the SAR.

The Staff did question one aspect of this calculation. ISFSI SAR Section 4.2.6.4, states that "Second order external effects, such as wind and interaction with surrounding casks, are bounded by these conservative assumptions and are therefore neglected." We questioned whether this was a conservative assumption, given that the "hypothetical cylinder," which is the air space around a Concrete Cask, is large enough to encompass other casks. PGE responded that the "hypothetical cylinder" was modeled to reflect back all of the radiated heat, and, therefore, was a conservative assumption. The calculation also modeled the "most disadvantageously placed cask," which is the one placed at the center of the array and is therefore the one subjected to maximum radiative blockage.

As a result of this review, the Staff was able to conclude that all temperatures are well-below allowable limits and the cask system will have an adequate, passive heat-removal capability.

b. Thermal Analysis of an MPC in the Transfer Cask

The same Holtec calculation, HI-2012676, was used to determine the maximum projected temperatures for off-normal conditions with an MPC in the Transfer Cask. We did not review the inputs and assumptions for this calculation; however, we compared the results with the previous analyses and found them reasonable given the design changes. All of the expected temperatures are well-below the allowable limits.

4. Arrangement of the Concrete Casks on the ISFSI Storage Pad

The HI-STORM LAR requires that from a thermal standpoint, "regardless of the size of the ISFSI, the casks should be arrayed in such a manner that the tributary area for each cask (open ISFSI area attributable to a cask) is a minimum of 225 ft^2." The Staff asked PGE if this guidance applied to the existing SNC casks, since the currently planned arrangement is for each cask to have available a total 225 ft^2, i.e. those on the interior of the array, which includes the space occupied by the cask. PGE had replied to an earlier RAI question on this subject, before the current design change, that "The separation between casks on ISFSI storage pad is not a critical dimension." PGE's response was that the guidance did apply, and that the total area of 225 ft^2 did include the area of the cask itself. They stated that the Holtec guidance was for their generic design, which included a maximum design heat load of 29 kW. The Trojan casks have a maximum design heat load of 17.4 kW, and more likely will have less.

D. 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. The radiation protection analysis for loading operations is described in LCA-237 and reviewed in Section V.G 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?

Was empirical data available from other plants?

What measurement and monitoring is planned?

2. Principal Design Criteria

The ISFSI design uses the thickness of the Concrete Cask walls and steel liner to provide the major portion of 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 the steel lid of the MPC to provide shielding.

3. Applicable Limits

The NRC limits on radiation from the ISFSI are found at 10CFR72.104:

> 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 in 10CFR72.106, 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 Controlled Area boundary at 300 meters from the Storage Pad.

The NRC regulations applicable to ISFSI loading and storage operations for personnel exposure are contained in 10CFR20. They are the same regulations that apply to operating plants, and which have applied to Trojan during decommissioning.

The Council 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)[43]
- 5 rem Committed Dose Equivalent (CDE) to the thyroid
- 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 the Council have used a more restrictive 500 mrem limit for dose to the whole body.

[43] The terms TEDE and CDE are as defined in 10CFR20. The allowed skin dose is higher because radiation to the skin has less somatic effect than radiation to the whole body.

The 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 for PGE employees and contractors, including a requirement that dose be kept ALARA.

4. PGE Dose Projections

a. Direct Radiation from ISFSI Normal Storage

The change from the SNC PWR Baskets to the Holtec MPCs created some significant changes in dose projections. The Holtec dose projections are higher than the SNC projections. SAR Table 7.4-1 shows that for the Concrete Cask top surface, the estimated dose rate has increased from 158.5 to 332.2 mrem/hour. The estimated dose rate on the side surface has increased from 19.1 to 20.4 mrem/hour. The increase on the top surface is significant, because this results in increased dose rates measured at a distance, due largely to "sky shine," which is the amount of radiation reflected back toward the ground from particles in the air. With the SNC design, PGE projected that the dose rate to any "real person" (someone who is on the site 2080 hours a year, or 40 hours per week for 52 weeks) would be 46 mrem/year at 100 meters. With the Holtec design, however, SAR Section 3.3.5.3 states that the dose rate at 100 meters from the pad is estimated to be 0.18 mrem/hour, which is (0.18 x 2080 hour/year) approximately 374 mrem/year. This increase from 46 to 374 mrem/year at 100 meters is a concern. The limit on total dose to a member of the public, according to 10CFR20.1301, is <100 mrem/year, and the closest structure that will be routinely occupied when the ISFSI is operational is the Trojan Central Building (TCB), the east wall of which is approximately 100 meters from the stored fuel.

We asked PGE for an explanation of the increase. They responded that the reasons for it had not been quantified, but that they were due mainly to the use of a different calculational method, different parameters used in the calculations, higher determined Inconel impurity levels in the non-fuel portion of the fuel assemblies, and, most significantly, the difference in thickness of the steel lid of the MPC over that of the PWR Basket (9.5 inches vs. 11 inches). The Manager of Trojan's Radiation Protection Program explained that a major difference in the calculation (HI-2012749, Shielding Evaluation) was that Holtec used the design basis maximum source term (42,000 MWD/MTU and 9 years cooling) for the entire ISFSI, of which only a small percentage of the fuel meets. And, as with the SNC design, this calculation did not take credit for the loading pattern on the ISFSI Storage Pad. It assumed that all casks would have the maximum design heat load of 17.4 kW. For these reasons, the actual dose rate at 100 meters is expected to be much less than 374 mrem/year. However, we asked PGE to comment on the potential effects of this increase on members of the public. They responded that the definition of "members of the public" does not include:

> …PGE employees, contractors, vendors, and persons who enter the site to service equipment or to make deliveries. However, the definition includes persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant. There are no "members of the public" who work within the Industrial Area. All individuals who work within the Industrial Area, regardless of whether or not they enter the RCA, are considered and will be trained as occupational radiation workers. Their doses are governed by 10 CFR 20.1201.
>
> After termination of the 10 CFR 50 license, it is expected that the Controlled Area boundary will be moved to encompass only the minimum area that would result in an annual dose to a "member of the public" of 0.1 rem. … It is not known if the Trojan Central Building will

> be within or outside the Controlled Area boundary. In any case, individuals who are classified as "members of the public" who work within the Controlled Area boundary will be limited to an annual dose of no more than 0.1 rem.

Since this response was received, it was determined that the TCB will definitely lie within the Controlled Area boundary, set at 300 meters from the ISFSI Storage Pad. For PGE employees, this will not be a problem. 10CFR20.1502 states that adults must wear dosimetry when they are likely to receive in one year a dose in excess of 10% of the limits in 10CFR20.1201(a). The limit for radiation workers is 5 rem/year, so 10% of this is 500 mrem/year, which when divided by 2000 hours/year (the use of 2080 hours/year is a recent change) results in a dose rate of 0.25 mrem/hour before dosimetry is required. The group that raises the most concern is pregnant women. 10CFR20.1502 states that pregnant women must wear dosimetry when they are likely to receive a dose in excess of 10% of the limits in 10CFR20.1208. The limit for declared pregnant women is 0.5 rem for the entire pregnancy, and 10% of this is 50 mrem. When divided by the available working hours in 10 months, this equates to 0.03 mrem/hour when dosimetry would be required, less than the projected 0.18 mrem/hour. We posed this concern to PGE, and they responded that "Declared pregnant women working within the TCB will be issued TLDs if actual TCB dose rates confirm conditions of 10 CFR 20.1502(a)(3) could be met." The Staff is satisfied with this response.

In addition, the Staff has required in OAR 345-26-390(6)(b) a radiation monitoring program that will cover areas of the site outside the ISFSI Access Controlled Area and within the Industrial Area, including the Trojan Central Building. This program will be performed during ISFSI loading through placement of all casks on the pad. We verified that procedure FHP 50-03, Loading and Placing Concrete Cask into Storage, includes a requirement to measure the Industrial Area dose rates each time a loaded Concrete Cask is placed on the Storage Pad. As part of this program, PGE has located three survey points on the east side of the TCB. In addition, Industrial Area surveys are performed quarterly in accordance with the Trojan routine survey schedule, governed by procedure, RP-114, Radiological Protection Routine Schedule. In addition to complying with OOE rules, these surveys will provide a comparison to the predicted values of ISFSI SAR Table 7.4-1.

Dose from the ISFSI will naturally decrease with distance. At the Controlled Area boundary (300 meters from the Storage Pad), the dose rate is projected to be 18.4 mrem/year, based on an occupancy of 2080 hours per year. For comparison, background radiation at the Trojan site is approximately 7 μrem/hour (about 61 mrem/year). The nearest public highway, U.S. Highway 30, is approximately 1000 meters from the Storage Pad. Individuals at this distance would receive virtually no exposure above background.

b. Effluents from the ISFSI

PGE calculated the dose from an undetected helium leak. They assumed a leak rate of 5.0×10^{-6} atm-cm^3/sec, which is the largest leak rate allowed by the Technical Specifications. Using the assumptions and calculational methods recommended by the NRC, they calculated a dose of 0.11 mrem to a hypothetical person standing at the Controlled Area boundary for 2080 hours (1 year). This dose rate is well within EPA Protective Action Guidelines.

Under normal storage conditions, there is no credible exposure from effluents. The fuel and fission products are confined by the MPC, which is sealed and tested to ensure a leak-tight seal. The only credible potential for a release is from residual contamination of particulate matter that adhered to the surface of an MPC during the loading process. As described in SAR Section 7.1.3 and in response to an RAI question, PGE will attempt to limit this possibility by coating ISFSI components with filtered, borated water prior to immersion in the Cask Loading Pit, injecting filtered water into the annulus between the Transfer Cask and MPC, and rinsing again with filtered water as the components are removed from the Cask Loading Pit. Then, before loading the MPC into the Concrete Cask, there will be an attempt to measure the amount of loose contamination by surveying the MPC lid and the accessible areas of the sides, which will include the top three inches at most. According to procedure FHP 50-03:

> If loose contamination levels exceed 10^{-4} µCi/cm^2 (22,000 dpm/cm2) beta-gamma or 10-5 µCi/cm^2 (2200 dpm/cm2) alpha, then an evaluation will be performed to determine if decontamination of the MPC is required.

It is unclear what means would be used to decontaminate an MPC exterior, since their ability to do so is limited by the geometry of the MPC inside either the Transfer Cask or Concrete Cask. In response to an RAI question on this, PGE responded that in addition to rinsing, there were additional practices available, such as using chemical solutions that have been procured and are available. The limit for exterior contamination of 10^{-4} µCi/cm^2 is based on remaining within the 5 mrem limit of OAR 345-026-0390(4)(f). The determination of this limit is discussed in detail in Section IV.D.6.a of this report.

A positive sign that contamination levels will likely be low is the experience with the prototype SNC PWR basket. In response to a question about the Gap Flush System, PGE responded:

> In our previous loading preparation a prototype PWR basket was used to support training. Its use included submerging it, inside the Transfer Cask, in the Cask Load Pit with Gap Flush operating. Subsequent contamination survey results showed the contamination levels were well below the Technical Specification limits.
>
> Holtec's MPC/Transfer Cask combination has a smaller annulus gap than the previous design, includes an annulus seal at the top of the MPC (which the previous design did not have), and will be used with the same Gap Flush System and injection rate. Based on these factors no further testing is planned, but during pre-operational testing an MPC will be placed in the Cask Load Pit and submerged and subsequently have a contamination survey conducted which will verify the adequacy of the Gap Flush System.

c. Occupational Personnel Exposure

The SAR projects occupational exposure during two phases of the ISFSI project: fuel loading and long-term storage surveillance. The highest potential for personnel exposure will be during the fuel loading phase. During this phase, welders and fuel handlers will work close to the MPC 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 114 person-rem. The projected figures per cask are listed in SAR Table 7.4-3. 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 29 person-rem for eventual transfer to the shipping containers (Transport Casks).

During long-term fuel storage, ISFSI personnel will receive the most exposure during weekly cask inspections, temperature readings that have to be taken at the casks, and radiation surveys. PGE projects occupational dose at about 1.6 person-rem/year. This is down significantly from the previous projection of 5 person-rem/year. The reason for this dramatic reduction is mainly due to change from daily to weekly inspections of the air inlet and outlet vents. The estimate on annual dose estimate for these surveillances is conservative when compared with actual operating experience at CMS Energy's Palisades plant in Michigan, where the annual dose for surveillance of 13 casks is only about 120 mrem/year.

ISFSI Specialists and security personnel are expected to accrue on the order of 1 rem/year (1000 mrem/year), which will require them to wear dosimetry when entering the ISFSI Restricted Area. The Staff asked if shielding or other ALARA measures were being considered to ensure that no personnel exceeded the TNP administrative limit of 1 rem/year. PGE responded that the Trojan Radiation Protection Program, TPP 20-02, requires that personnel be administratively prevented from reaching the TNP administrative limit of 1 rem/year. "If they approach the administrative limit, ALARA measures will be taken to prevent them from reaching or exceeding the federal limit."[44]

d. 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 review of the accident analyses is described in Section IV.F 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 credible accidents and natural events, with one exception, 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 MPCs, as described in Section IV.B of this report. The MPCs and Concrete Casks have adequate structural integrity to preclude increased radiation after credible events, including earthquakes, volcanic eruptions, floods, and mishandling events. The one exception involves a tornado-driven missile that damaged a Concrete Cask and removed some of the concrete shielding. Using assumptions and methodology approved by the NRC and reviewed by the Staff, the SAR shows in Section 8.2.4 that the worst case tornado driven missile would potentially remove 5.69 inches of concrete from the Concrete Cask wall localized in the area of impact. The increased dose rate at the surface of the cask near the damaged area is estimated to be 103.0 mrem/hour, compared to a normal rate of 20.4 mrem/hour. The cask could be repaired by filling the damaged area with grout. PGE estimated the collective dose to the repair crew at approximately 0.238 person-rem, assuming two technicians working for about 30 minutes. The Staff considers this acceptable, particularly in light of the low incidence of tornadoes for this area and the 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.

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, in part, for the purpose of selecting the Controlled Area boundary distance from the Storage Pad, and for demonstrating

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[44] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

that under no conditions would members of the public receive significant radiation from the ISFSI. The analysis, described in SAR Section 8.2.1, postulates that all of the fuel in one MPC suffers a total cladding failure and that 30% of the available fission product gases are released to the environment, even though there is no known event that could initiate this kind of failure. The 30% figure is based on NRC Regulatory Guide 1.25, and the assumptions and methods used in the analysis come from NRC interim staff guidance. The calculated doses at the 300 meter Controlled Area boundary, as shown in SAR Table 8.2-2, are less than 1.2 mrem/30 days whole body, thereby well within the EPA PAGs and 10CFR72.106.

As noted in Section III.A.3 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, the Council adopted OAR 345-26-390(6)(a), which requires PGE to demonstrate, within six months after terminating the NRC operating license, evidence of its ability to exercise control of personnel access to the Controlled Area in the event of an accident.

5. Dose Calculations

In our Staff report of January 27, 1999, we described the methodology used to determine the dose projections for the Trojan ISFSI. Those projections have changed due to the design change, calculations by a different vendor, and fuel that is older by three years. However, the differences between the two sets of results were expected and not that significant.

In the 1999 Staff evaluation, we described the methods used to project dose rate and the methods we used to review the dose calculations. We also noted that experience from other plants (notably the CMS Palisades Plant and Entergy's ANO plant) indicate that actual dose rates are likely to be lower than predicted. As noted above, most of the shielding is provided by the concrete cask, which remains unchanged from 1999. The dose due to "sky-shine" from above is projected to increase because the Holtec shield lid is 9 inches thick, compared with the 11 inch thickness of the SNC design. With that one difference taken into account, OOE believes the conclusions regarding dose calculations in the 1999 report remain valid, and OOE recommends retaining the rule requiring systematic dose rate measurements as successive casks are loaded.

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 MPCs may have some residual contamination on their surfaces when removed from the pool. This exterior contamination is the only source of effluent release from the ISFSI. Since there is no effective way to calculate the possible exposure from a release of this exterior contamination, PGE worked backwards from an artificial dose to a reference person of 2.5 mrem, which is half the limit in OAR 345-26-390(4)(f). Such a dose would result from a 989.8 μCi of ^{60}Co spread over the total surface area of a maximum possible 36 MPCs (there will be 34 at TNP). The allowable contamination would, therefore, be 10^{-4} μCi/cm^2, and this is the limit that PGE has chosen. The Staff believes that this is an acceptable and conservative approach. There will be 34 rather than 36 casks; it is unlikely that there will be significant contamination on the exterior surfaces of the MPCs; and the calculation assumes that all the contamination is released at one time in a single plume, and that it all escapes from the Concrete Casks.

The Staff's only area of concern is what would occur if the MPCs could not be decontaminated to less than 10^{-4} μCi/cm^2. As previously stated, PGE's ability to do so is severely limited due to the geometry of an MPC inside either the Transfer Cask or a Concrete Cask. PGE representatives stated in interviews that if a cask could not be decontaminated to less than 10^{-4} μCi/cm^2, they would perform an evaluation to determine an acceptable limit. A procedure is planned that will include criteria for that evaluation, and the Staff will review that procedure to confirm that it meets the acceptance criteria of OAR 345-26-390(4)(f).

b. Impact on Final Decommissioning Survey

The radiation dose rate at 100 meters from the ISFSI will exceed background by up to an estimated 0.18 mrem/hour (374 mrem/year, based on an occupational scenario of 2080 hours per year). This exceeds both the OAR 345-026-0370(2)(a) decommissioning criteria of 5 μrem/hour and the OAR 345-026-0370(2)(h) criteria of 25 mrem/year. PGE hopes to complete decommissioning the Trojan site by 2005, and the decommissioning area includes areas that are approximately 100 meters from the ISFSI. The Staff requested additional information from PGE regarding their ability to demonstrate compliance with OAR 345-26-370(2)(a). Our concern was that in areas outside the ISFSI perimeter fence but inside the Trojan Industrial Area, PGE would be unable to show compliance with EFSC decommissioning criteria because of direct radiation from the ISFSI. PGE stated[45] that they will use shielded detectors for the final survey. Shielded detectors will verify an absence of residual contamination from Trojan plant operations, but will not "see" the direct radiation from the ISFSI. Therefore, these detectors should verify that the site meets the EFSC decommissioning criteria. Direct radiation from the ISFSI is considered unrelated to decommissioning.

In August 2002, PGE began its final radiological survey of the exterior of the plant's Control and Auxiliary buildings, whose north walls face the ISFSI site. PGE will perform these surveys in accordance with its License Termination Plan, which the NRC has approved. These surveys will demonstrate whether or not the exterior of the plant buildings meet the NRC and EFSC standards for unconditional release, and any radiation measured there after the ISFSI is installed can be attributed to the fuel rather to residual contamination associated with the plant. OOE considers this a satisfactory means of demonstrating that the ISFSI will not affect the final plant survey.

c. Area Radiation Monitoring

PGE intends to use Thermoluminescent Dosimeters (TLDs) to monitor direct radiation from the ISFSI. A TLD is a device that collects radiation continuously, similar to the way film collects and records light. A TLD does not provide an instantaneous reading of dose rate. Instead, the TLDs are removed and the data is collected in an offsite laboratory. The results show the cumulative radiation collected at the site of the TLD over a period of time. PGE proposes to process the TLDs quarterly, which is consistent with the practice at the other ISFSI installations. PGE will place 8 environmental monitoring TLDs along the ISFSI perimeter fence, and 8 additional TLDs within the radiologically controlled area.

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

The Staff 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 Storage Pad, so that precise data on the dose rate from the casks will be available immediately. Persons located within 100 meters of the facility will be ISFSI personnel, and their exposure will be monitored and controlled in accordance with 10CFR20. The accident analyses show that the dose consequences from postulated events are small and will not affect members of the general public. Therefore, TLDs with quarterly data collection will 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 our review of the SAR, the dose calculations and the inputs and assumptions to those calculations, and the experience at other plants, the Staff concludes 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.

E. 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. The criticality analysis for loading operations is described in LCA-237 and reviewed in Section V.H.1 of this report. The Staff 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 10CFR72.236(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.

SAR Section 3.3.4.1 states:

> Subcritical conditions are to be maintained by MPC fuel basket geometry and the use of Boral. The MPC fuel basket will establish fuel assembly spacing. The design will assume a fuel assembly enrichment equal to or greater than the maximum initial fuel assembly

enrichment that will be stored (3.56 wt% U^{235}). No credit will be taken for burnup or fuel assembly control inserts. Boral is used as a neutron absorbing material in the MPC fuel basket design, and is credited in the criticality analysis for dry storage conditions.

Even though the SNC-designed PWR Baskets had Boral panels similar to those in the MPCs, credit was not taken for the boral in the SNC design. It is credited in the new design, however, since the MPC is licensed for transportation, and therefore has to include a flooded condition in the analysis, although the effect of these panels on reactivity under dry conditions is small.

Boral is a mixture of boron carbide and aluminum alloy. The boron in the boron carbide is a neutron absorber, and thus helps prevent criticality by interrupting chain reactions in the fission process. According to PGE and Holtec, Boral will remain effective for the 40-year design life of the ISFSI. We asked PGE to provide evidence of Boral's efficacy over the design life of the ISFSI. They responded that:[46]

- The relatively low neutron flux in the MPC does not result in significant depletion.
- The boron content assumed in the criticality analysis is 75% of the minimum allowed.
- The materials used to fabricate the panels is resistant to corrosion and other degradation, plus the inert atmosphere inhibits corrosion.
- The sheathing used to affix the Boral to the cells is qualified to remain in place under all conditions.

We then asked if PGE had verified the design specifications of Holtec's purchase of Boral. The response was that the procurement specification, purchase order, and design drawings specifying the loading were reviewed and approved by PGE. In addition, the material certifications from the supplier were included with the document packages submitted to PGE with each MPC.[47]

3. Criticality Evaluation

The Staff had previously reviewed the criticality analysis performed for the SNC-designed PWR Baskets, which is described in our January 27, 1999, report. The criticality analysis for the Holtec-designed MPC used different software, some different assumptions, and, as previously mentioned, took credit for the Boral. The fuel was assumed to have 3.7 wt% U^{235} initial enrichment instead of a design maximum 4.2 wt% (actual is 3.56 wt%), and no credit was taken for burnup, as in the previous analysis. The result for the SNC calculation was a maximum k_{eff} of 0.4077, and for the Holtec calculation a maximum k_{eff} of 0.3311, including uncertainties. We verified that the input assumptions were conservative, and we verified that the MCNP4a code used by Holtec was appropriately validated, as described in Section IV.A of this report. Based on this and the comparison with our previous review, the Staff concludes that the spent fuel will remain well below the 0.95 criticality criterion under all dry storage conditions.

F. ACCIDENT ANALYSIS

1. Review 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

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[46] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002
[47] PGE, VPN-074-98, S.M.Quennoz to D.Stewart-Smith, December 3, 1998

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.D of this report, and postulated accidents involving fuel loading operations are discussed in Section V.I of this report.

The Staff reviewed the SAR accident analyses and selected calculations used in support of these analyses in accordance with the review methodology described in Section II.E.1 of this report. The Staff also 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.D 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, and 25 mrem to any other organ as a result of exposure to direct radiation from the ISFSI.[48] 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.[49]

In addition, the EPA Protective Action Guidelines (PAGs) 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 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

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[48] 10CFR72.104(a)
[49] 10CFR72.106(b)

The acceptance criterion for maintaining the spent fuel in a subcritical condition is described in Section IV.E 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 decelerations in a postulated accident with limits that the fuel and the MPCs have been shown to be able to withstand. The most widely accepted limits of deceleration for fuel, that is, before damage occurs to the fuel cladding, are 82g in a vertical drop accident and 63g in a side drop accident. For Trojan Westinghouse fuel the numbers are somewhat higher. This comes from a study by Lawrence Livermore National Laboratory (LLNL), UCID-21246, which has come under question recently by the NRC. This study and the controversy about it is described in Section V.E.3. According to the Holtec LAR, the design limit for the MPC is 60g in any orientation within a Concrete Cask and 45g in any orientation within a Transfer Cask. Therefore, for any accident scenario where the impact results in g-forces that are less than these limits, it is reasonable to assume that confinement will be maintained and damage will not occur to the fuel.

2. MPC Handling Event

SAR Section 8.1.1.1 describes an off-normal handling event that consists of a lateral impact of the MPC against the inside of the Concrete Cask during transfer of the cask to the Storage Pad. The Staff evaluated this event scenario as part of our previous review effort and found the results satisfactory. A similar event scenario was evaluated for the loading operation and is described in greater detail in Section V.I.4 of this report. In both cases, the resulting deceleration is 17.5g, which is below the design basis deceleration limits of 45g for an MPC in the Transfer Cask or 60g in a Concrete Cask. The Staff did not review the revision of this analysis using the MPC in place of an SNC PWR Basket, because the two systems react similarly in this scenario.

Another handling event that could occur at the Transfer Station is referred to as an MPC overlift that results in lifting the Transfer Cask. This occurs when the MPC is lifted while in the Transfer Cask, and the top of the MPC comes into contact with the Transfer Cask lid and both units lift. The concern is whether the lift cleats and slings attached to the MPC and the Transfer Cask lid and bolts could hold the weight without risk of failure. This accident is not analyzed in the SAR. Rather, PGE analyzes for the drop accident. Therefore, we asked PGE for additional information about the lift components. They responded that the Transfer Cask lid and bolts are designed to support the weight of the empty Transfer Cask with its water jacket filled in the event of an MPC overlift. All the rigging between the MPC and the Fuel Building crane hook is designed with twice normal safety factors with respect to the MPC loaded weight. Also, the combined weight of the loaded MPC and the empty Transfer Cask is below the 125 ton rating of the Fuel Building crane main hook.

Therefore, in the event of an overlift condition in the Fuel Building, the lifting devices could handle the load without risk of failure. For an overlift condition at the Transfer Station, a load cell will be used to stop the mobile crane when the Transfer Cask top lid is impacted by the MPC.[50]

3. Off-Normal and Accident Thermal Analysis

As part of our previous review, the Staff examined SAR Section 8.1.2 and PGE calculation TI-033, TranStor™ Concrete Cask Thermal Analysis, Rev.1, March 20, 1996, which describes the conditions of sustained high and low ambient temperatures and the blockage of air inlets. Our review is described in our January 27, 1999, report. Since then, the design change from the SNC PWR Baskets to the MPCs has resulted in changes to the input parameters for this analysis. One is a reduction in the size of the annulus. The other is a decrease in the assumed heat load. The original calculation assumed the maximum design basis heat load for the TranStor™ system of 26 kW. The current calculation assumes 17.4 kW, which is still conservative, since the actual heat load for the "hottest" cask will be less than 16.5 kW. As a result, the air outlet temperature for the condition of one-half of the inlets blocked decreases, and the inner concrete temperature of a cask remains well below the limit of 300 ^{o}F. The Staff did not review the revised analysis, since the heat load would have a dominant effect and has moved in a conservative direction.

SAR Section 8.2.7 describes the condition involving the full blockage of air inlets. Again, the primary change due to the MPCs is the reduction in assumed heat load. Another change to the calculation was an assumed ambient temperature of 100 ^{o}F rather than 75 ^{o}F. Since both of these changes are in the conservative direction, the Staff did not review the revised calculation. The result is an expected increase in the time before the inner concrete of a cask reaches the maximum recommended temperature, 350 ^{o}F. The time went from 31.5 hours to 57.1 hours.

SAR Table 4.2-12 does not give the maximum fuel clad temperature at 57.1 hours or at what time the short-term limit for the maximum fuel clad temperature would be reached with all inlets blocked. In response to an NRC RAI question during the previous review, PGE calculated that a total loss of air flow with all inlets and outlets blocked would allow approximately 156 hours before the fuel clad reached 1058 ^{o}F[51]. This figure would be substantially greater with the lower heat load.

4. Concrete Cask Overturning Event

The Staff reviewed a Concrete Cask overturning event as described in SAR Section 8.2.3. 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. The Staff did a separate calculation and reached the same conclusions. The weight used in the calculation was 290,000 pounds, which has since been revised to 292,700 pounds; however, this change had no significant impact on the analysis.

In considering a cask overturning event, the Staff 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 Storage Pad to withstand a hypothetical overturning event. As a result, the Staff concludes that damage due to cask overturning is not a credible concern.

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[50] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

[51] PGE, CPY-001-97, C.P.Yundt to D.Stewart-Smith, Response to RAI, February 6, 1997

5. Tornadoes

The Staff reviewed PGE's tornado accident analysis presented in SAR Section 8.2.4 and in calculation PGE01-10.0203-06[52] in accordance with the guidance in Section II.E.1 of this report. One input to this analysis has changed. The weight of a loaded Concrete Cask is estimated to be 292,700 pounds rather than the original 290,000 pounds; however, this change had no significant impact on the analysis.

The calculations and the Staff's recommendations are essentially unchanged from the 1999 report. PGE calculated that, in a design basis tornado, 5.69 inches of concrete could be removed by objects striking the concrete cask. Since these casks are the SNC casks approved in 1999, this is the same result documented in the 1999 report. Side surface dose rates would increase from 20.4 to 103.0 mrem/hr. 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.B.7 of this report. The Staff reviewed PGE's earthquake accident analysis presented in SAR Section 8.2.5 and in calculation PGE-01-10.02.03-06 in accordance with the guidance in Section II.E.1 of this report. One input to this analysis changed slightly. The weight of a loaded Concrete Cask is estimated to be 292,700 pounds rather than the original 290,000 pounds; however, this change had no significant impact on the analysis.

7. Volcanic Eruptions

SAR Section 2.6.6 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." The Staff 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.[53] They superimposed the ashfall that tracked northeast of Mt. St. Helens in May 1980 on a trajectory towards Trojan. The Staff also reviewed a 1995 report from the U.S. Geological Survey 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."[54] 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

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

[53]PGE, CPY-032-96, C.P.Yundt to D.Stewart-Smith, Response to RAI, June 27, 1996

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

> 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.

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. [55]

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, the Staff concluded that PGE's analysis and response plans in the event of significant ash fall are satisfactory. However, the Council in 1999 recommended that equipment required to unblock the air inlets and outlets be maintained onsite or available within 24 hours of notification that a significant ashfall from a volcanic eruption is likely to occur. OOE believes that recommendation continues to apply.

8. MPC Drop into Concrete Cask or Transport Cask

The postulated drop of an MPC into a Concrete Cask or Transport Cask results in the largest impact deceleration of any accident, a drop of more than 20 feet, and has the greatest potential to breach the confinement, cause fuel damage, and release radioactivity. The Staff reviewed the original analysis of an SNC PWR Basket drop into a Concrete Cask and confirmed PGE's conclusions that the impact limiter embedded in the pad under the Transfer Station maintained the drop decelerations less than the allowable values and that the accident did not result in a radiological release or an increase in k_{eff} to more than 0.95.

In the revised SAR, proposed revision 2, PGE did not describe a Trojan-specific analysis of the deceleration forces that would occur in the postulated accident involving an MPC drop into a Concrete Cask. PGE took the position that this scenario is not credible due to utilization of the increased safety factors of NUREG-0612. However, the SAR did describe a generic study

[55] PGE, CPY-007-97, C.P.Yundt to D.Stewart-Smith, Revisions to Responses to RAI #2, March 17, 1997

performed by Holtec, wherein an MPC was assumed to drop vertically from a height of 25 feet onto a concrete target without the use of an impact limiter to absorb energy. Even though this scenario is not Trojan-specific, the description bounds any postulated accident that could occur at Trojan.

The Staff raised several questions about this generic analysis. The SAR lists five assumptions, one of which states: "The MPC top lid-to-shell weld joints are explicitly modeled with full recognition of the discontinuity stresses that are expected to develop at these locations. The material behavior of the weld joints is assumed to be the same as the MPC shell material." On the face of it, these two sentences appear to conflict. The heat affected zone of a weld has different properties than the base material. PGE responded that "The model makes no distinction between the heat-affected zones of the welds and the contiguous base metal. Both regions are defined as having the same stainless steel properties." They stated also that this was consistent with the analyses provided in the HI-STAR 100 FSAR, which is certified by the NRC, and that the peak deceleration was below the design basis MPC deceleration of 60g.[56] They also stated that a Trojan-specific calculation would be performed.

The Staff asked about buckling or deformation of the MPC in the generic analysis. Proposed Revision 2 of the SAR Section 8.2.13.3.3 stated that the shell is plastically deformed; however, at a meeting with the NRC, PGE, and Holtec on February 7, 2002, a NRC representative asking about this analysis said that since the model treats the contents of the MPC as a solid, no buckling or deformation is possible. We submitted this question to PGE. In response, they said that Holtec would perform a Trojan-specific analysis. "In the new analysis, the enclosure vessel, the fuel basket, and the stored fuel assemblies are individually modeled as deformable bodies with elastic material properties. Thus, the contents are free to deform under the impact load." They stated also that "Since the contents of the MPC are no longer modeled as a rigid solid cylinder, a portion of the impact energy is now absorbed by the MPC in the form of elastic deformation energy."[57]

In July 2002, the Staff reviewed the new calculation, HI-2022881, Finite Element Analysis of the MPC Drop Accident Postulated at the Trojan Transfer Station. In accordance with the review criteria in Section II.E.1, we verified that the assumptions and inputs of the new analysis were conservative, that the stresses experienced by all components were significantly less than the failure stresses, and that the analysis used a methodology that had been approved by the NRC. It was performed using the LS-DYNA finite analysis code. The calculation package states that Holtec performed a verification and validation of this code, but we learned that Holtec's V&V had not been checked by PGE QA. Because of the significance of this calculation, we requested (as described in Section IV.A.2 of this report) that PGE QA (Nuclear Oversight) review this code on their next surveillance at Holtec prior to loading operations. PGE agreed.

The results of the new calculation show that the maximum deceleration is $\leq$ 54g, which is less than the acceptance criteria of 60g for the MPC, and that there is no buckling of the MPC. The integrity of the MPC, i.e. the confinement, remains intact and there is no radiological release or other adverse radiological consequences. The calculation also concludes that fuel damage does not occur. The limits for fuel damage have in the past relied on a Lawrence Livermore National Laboratory study (UCID-21246), which established a maximum limit for the most vulnerable

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[56] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002
[57] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

intact fuel of 82g in a vertical orientation. Section V.E.3 of this report describes questions about the validity of this study; however, even with this uncertainty, the calculated 54g is so much less than the LLNL limit, that damage to the fuel in such a drop scenario is highly unlikely. This is further substantiated by the fact that the Trojan fuel evaluated using the guidelines of the LLNL study would not be damaged with an impact of up to 100g. Therefore, 54g appears conservative.

The Staff had questions concerning the difference between the two designs (SNC and Holtec) with regard to this accident scenario. The original analysis for the SNC design resulted in a deceleration of approximately 75g, the new one only 54g. We asked why the large difference, given that the geometries and weights are so similar? Was the previous result of 75g valid with the inputs that were used? PGE responded that the difference was due to the methodologies. The original result had been valid; however, that calculation involved a relatively simple energy balance equation. The peak deceleration and the crush depth were determined by balancing the potential energy of the falling PWR Basket with the energy absorbed by the impact limiter through a series of small incremental displacements. The new analysis used a sophisticated non-linear, finite element analysis.[58] We verified through interviews with PGE representatives that the engineers who performed and reviewed the original analysis also reviewed and accepted the new one.

Our final concern had to do with the actions PGE might anticipate in the event of such an accident. PGE responded that Trojan procedures require the completion of an inspection by Engineering personnel within 24 hours of such an event, initiation of an engineering evaluation to determine the affected ISFSI components' ability to safely continue to perform their function, and the initiation of a Corrective Action Request to track and document the event evaluation and corrective action. We reviewed the procedures and confirmed the required actions.

PGE stated in response to another of our questions that "the potential scenario of an MPC drop into a Concrete Cask or Transport Cask is addressed with redundant conservatism." They cited the design of the lifting components and the use of safety factors; the generic analysis which showed that even when energy absorption from the impact limiter was not assumed, the confinement remained intact; and the Trojan-specific analysis which showed the impact deceleration to be less than 60g and no fuel damage. Considering all of these factors, the Staff agrees that loss of confinement and fuel damage are highly unlikely.

9. Conclusion

As stated at the beginning of this section, we reviewed each of the event analyses described in the SAR Section 8 and the supporting calculations; however, we focused on the bounding analyses or analyses that involved issues or concerns requiring further resolution by us or PGE. For most of these analyses, the design change involving the replacement of SNC PWR Baskets with MPCs did not impact the analysis in a non-conservative direction. In some of these instances, the Staff did not review the revised analysis. For the postulated drop of an MPC into a Concrete Cask or Transport Cask, we did review the new analysis. In all cases, 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. Lastly, the events or accidents analyzed by PGE appeared to encompass all probable or credible accident scenarios.

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[58] Calculation HI-2022881, Finite Analysis of the MPC Drop Accident

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, the Staff also reviewed the associated License Change Applications (LCAs), including LCA-237, Spent Fuel Cask Loading in the Fuel Building. This review is critical to establishing a basis for the Council'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 pressurized line break during drying and helium pressurization of a loaded MPC are evaluated in LCA-237, but not the SAR. Further, OAR 345-26-390(4)(f) requires the Council 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.

In March 1996, PGE submitted a 10CFR72 license application to the NRC for construction and operation of an ISFSI. Then in May 1996, PGE submitted the original version of LCA-237 in support of their license application. In May 1997, a license condition that prohibited bringing casks into the Fuel Building was removed from the license by License Amendment 196. That prohibition had been added in 1983 because the potential for a heavy load drop in the SFP had not been analyzed for a shipping cask.

LCA-237 has undergone six revisions. The first four were based on the original design of the ISFSI, using the SNC/BNFL system. Proposed Revision 5 was issued November 15, 2001, following the switch to Holtec as the principal vendor. And on July 31, 2002, PGE submitted proposed Revision 5a, in response to questions asked by the NRC. The final revision when issued will be called Revision 5.

In the proposed revisions of LCA-237, PGE determined that the fuel transfer process using the Holtec design do not involve new types of accidents not previously analyzed. LCA-237 demonstrates that the consequences of postulated accidents are well below the EPA Protective Action Guidelines. The Staff has submitted three formal RAI packages on issues related to the LCA—August 6, 1996, January 9, 1997, and April 5, 2002—submitted many informal written questions, and conducted numerous interviews; and we concur with PGE's determination that the accidents described in LCA-237 are well below the level of the EPA Protective Action Guidelines.

B. ISFSI LOADING SHIFT PERSONNEL AND RESPONSIBILITIES

PGE signed a service agreement with Holtec to do all of the fuel handling, loading, and welding operations on the ISFSI. The Staff raised several questions about this arrangement in our RAI to ensure that PGE maintained supervisory responsibility over the loading operations. PGE responded with a list of key supervisory positions and responsibilities.[59]

> <u>Manager, Operations</u>: The PGE Manager, Operations, reports to the General Manager, Trojan, and is responsible for performance of pre-operational and startup testing, ISFSI loading, and maintaining personnel trained and qualified in accordance with the Certified Fuel Handler Training Program for fuel handling operations.
>
> <u>Shift Manager</u>: The PGE Shift Manager reports to the PGE Manager, Operations and is responsible for all Trojan 10 CFR 50 licensed activities.
>
> <u>Shift Coordinator</u>: The PGE Shift Coordinator reports to the PGE Manager, Operations and is responsible for providing shiftly oversight of ISFSI fuel loading, MPC sealing, and transporting of loaded Concrete Casks to the Storage Pad. This position also provides assistance as necessary to ensure coordination of Holtec and PGE activities.

We also asked about approval responsibility for procedures, even those written by Holtec for Holtec employees and contractors during the loading. PGE responded that there are two groups of procedures: PGE fuel handling procedures (FHPs) and Holtec site specific procedures (HPPs). The FHPs are developed, revised, and approved using the Trojan procedure control process; the HPPs are reviewed and approved by PGE prior to use. Revisions to the HPPs are also reviewed and approved by PGE.

<u>C. TRAINING OF HOLTEC AND CONTRACT WORKERS</u>

The Staff requested PGE to provide a description of the training that would be given to Holtec and contract workers during the ISFSI loading operations. PGE responded as follows:

> A PGE Shift Coordinator will be assigned to each loading crew to provide management oversight and to coordinate Holtec activities with the Plant Operations crews. The Shift Coordinator will complete the Certified Fuel Handling Training and ISFSI loading training, which will include all equipment and procedures used to load and move an MPC to the ISFSI Pad. The Plant Operations Staff will receive ISFSI Overview Training provided by the Operations Trainer. The ISFSI Specialists will be requalified in accordance with the new storage system requirements. Radiation Protection Technicians will be trained on revisions to the procedures and programs that are affected by the new storage system.
>
> Holtec Project Procedure HPP-1135-300, Trojan ISFSI Completion Project Training Program, describes the training requirements for Holtec personnel. Prior ISFSI experience is a consideration but not a requirement of contractors and employees used by Holtec for the Trojan fuel loading. To the extent possible (i.e., considering the number of people in the industry with cask loading experience and their availability), workers with experience are chosen for the Trojan team. Holtec is attempting to use resources that were involved in the cask loadings at Hatch and Dresden in preparing for the Trojan cask loading.
>
> The Holtec International ISFSI Training Program consists of many facets. When project personnel report to the site they are trained in the General Employee Training (GET)

[59] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

programs conducted for many years by the PGE plant staff for staff workers and contract workers. The project team personnel will receive additional ISFSI and specialized training as detailed below.

GET I: This program consists of training in the following areas:

Specific plant information including security measures required by the site for workers

Fire protection measures implemented at the plant

Safety program measures required for plant work

Nuclear Quality Assurance measures required for plant work activities

Plant site Emergency plan requirements

GET II: The radiation worker training program consists of the following:

Worker fitness for duty program requirements

Radiation protection fundamentals for rad worker activities

Radiation protection fundamentals – site specific

OSHA Training:

OSHA training concerning industrial hazards present at the plant and how to mitigate them under various scenarios of work conditions. This includes working with asbestos, lead, silica and various other substances that may be present at some work locations.

ISFSI Program Training (classroom and on-the-job): This program consists of the following:

Overview training in the overall ISFSI operation

Specific training in the equipment preparation for fuel loading activities

Specific training in the actual fuel loading cycles and operations

Crane operator training

Rigging and lifting operations training

Specific ISFSI support ancillary equipment training

Certified Fuel Handler training for required personnel

The Staff found the personnel training program satisfactory; however, there is no mention in the SAR about the use of mockups for training. SAR Section 9.2.3.1.1 states: "PGE is constructing several MPC weld mock-ups and a full size, full weight dummy MPC." This section describes fitup and testing to be performed, but not specifically for training. PGE responded that procedure HPP-1135-300 requires the full size, full weight dummy MPC to be used for load handling training outside the spent fuel pool. They stated also that "For underwater dry run activities, an actual certified MPC will be used in the Cask Load Pit for training purposes including fuel shuffle and lid fitup."[60] In addition, partial size mockups will be used for training on lid welding and vacuum drying.

[60] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

D. SYNOPSIS OF SPENT FUEL LOADING

LCA-237 describes the sequence for loading spent fuel into the MPCs and the MPCs into the Concrete Casks. Briefly, the sequence is as follows:

1. Preparation of the MPC and Transfer Cask

An empty MPC and Transfer Cask are brought into the Fuel Building. The Transfer Cask is moved by the Fuel Building Crane and Lift Yoke to the 93 elevation (top floor) of the Fuel Building and placed in the empty Cask Wash Pit. The Lift Yoke is removed. Then an MPC is moved by the same crane and placed into the Transfer Cask. An annulus seal for the gap between the MPC and Transfer Cask is installed. The Transfer Cask assembly is then moved by the crane and Lift Yoke, suspended over the Cask Loading Pit near the SFP, and filled with borated water. The Gap Flush 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 MPC.

2. Loading Fuel at the Cask Loading Pit

The Transfer Cask is lowered to the impact limiter in the Cask Loading Pit and the Lift 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 and Damaged Fuel Containers are then loaded into the MPC using the Fuel Handling Bridge Crane. This operation is directed by Certified Fuel Handlers. When the loading is completed, the MPC lid with a drain pipe attached is lowered onto the MPC. The Lift Yoke is attached to the Transfer Cask trunnions. The loaded Transfer Cask is lifted to the top of Cask Loading Pit, where the Transfer Cask is washed to remove potential contamination, the MPC Lid Retention System is engaged to secure the MPC lid during lifting and moving of the loaded Transfer Cask, and the Gap Flush System is removed.

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 Transfer Cask exterior is decontaminated, the water jacket is filled to provide additional shielding, and the MPC Lid Retention System is removed. Approximately 50 to 120 gallons of water are removed from the MPC, and the MPC shell-to-lid weld is completed using an automated welding system, then dye-penetrant tested. The MPC is then re-filled with borated water and hydrostatically tested to 125 psig for 10 minutes. Next, the MPC is drained of approximately 20 gallons of water, and the shell-to-lid weld is pressure tested with 90 psig helium. The MPC is blown dry with helium, and the residual moisture removed by a vacuum drying system to a stable internal vacuum of $\leq$ 3 mm Hg ($\leq$ 3 torr) for a minimum of 30 minutes. Next, the MPC is filled with helium to approximately 30 psig. The vent and drain port cover plates are welded, dye-penetrant tested, and helium tested. Then the closure ring, a redundant seal over the shell-to-lid weld and the vent and drain port cover plate welds, is welded and dye-penetrant tested.

4. Loading the MPC into a Concrete Cask

The loaded Transfer Cask is next moved via the Fuel Building Crane and Lift 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 Lift Yoke is disengaged from the trunnions, the Transfer Cask bottom doors are opened, and the MPC is lowered into the Concrete Cask using slings attached to lifting cleats attached to the MPC lid.

An indirect check of the MPC external contamination levels is performed by surveying the internal wall surfaces of the Transfer Cask. The empty Transfer Cask is then moved to the Cask Wash Pit. A shield ring and steel cask cover plate are installed on top of the Concrete Cask. The loaded Concrete Cask is surveyed for contamination and radiation levels.

5. Transferring the Concrete Cask to the ISFSI Storage Pad

The loaded Concrete Cask is then moved via an air pad system and modified forklift truck to the ISFSI Storage Pad. Once on the Storage Pad, the air pads are removed from the air inlets, the inlet screens are installed, and the cask is hooked up to the temperature monitoring system.

E. STRUCTURAL ANALYSIS DURING ISFSI LOADING

1. Fuel Building Crane Heavy Loads

a. Minimum Distance to SFP

NRC guidance 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. The Staff 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.[61] PGE contends that the 6 foot separation is sufficient because the maximum lift height of 7 inches at the Cask Loading Pit prevents a dropped load from tipping. The Staff concurs with this assessment.

The lift height varies from 4 to 12 inches, depending on where in the sequence the Transfer Cask is. According to LCA-237, a "handling height on the order of 12 inches is required only where it is necessary to lift the Transfer Cask onto or off of such items as the Transfer Cask impact limiter, or load distribution system." During the previous review, the Staff confirmed through PGE calculations that a drop from 15 inches was insufficient to cause a tipover. Calculations with the previous design 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. Since the previous design is similar in size and weight to the present design, the Staff believes that the conclusions of that analysis still apply. As an added precaution, however, the Staff plans to review the testing of electrical interlocks on the Fuel Building Crane prior to movement of the Transfer Cask. The Staff 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 a load, the steps and proper sequence to be followed in handling the load, and defining the safe load path. Lastly, the Staff will witness the first fuel loading and all of the movements of loads.

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[61]PGE, CPY-035-96, C.P.Yundt to D.Stewart-Smith, Response to RAI, September 23, 1996

b. Maximum Lift Height

LCA-237 gives the maximum lift height for a Transfer Cask as "12 inches or less above the floor depending on the location along the safe load path."[62] 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."[63] The Staff considers this acceptable, and will monitor the effectiveness of these measures when fuel loading occurs.

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 re-qualification 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 pre-operational setup and testing phase.[64] 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 and Spent Fuel Pool Crane Testing

The Staff questioned PGE 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 pre-operational testing.[65] This test was successfully performed in April 1997. A portion of this test was witnessed by the 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.

LCA-237 Section 7.2.2 states that "the load handling equipment will be load tested and interlock operation verified prior to use to ensure that they are adequate." The Staff asked PGE what constituted "use" in the sense of how often or at what frequency during the months of the loading process would the Lift Yoke, the Transfer Cask trunnions, and Fuel Building Crane be tested, including mechanical stops and electrical interlocks. PGE responded that load testing for the Lift Yoke and Transfer Cask trunnions was completed in the fabrication shop. Following that, periodic testing will be performed in accordance with ANSI N14.6-1993.[66] Section 6.3.1, states: "Each special lifting device shall be subjected annually (period not to exceed 14 months) to either of the following: a) A test load equal to 150 % of the maximum service load… b) …

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[62]PGE, LCA-237, Spent Fuel Cask Loading in the Fuel Building, Rev 5, November 15, 2001
[63]PGE, CPY-035-96, C.P.Yundt to D.Stewart-Smith, Response to RAI, September 23, 1996
[64]ibid.
[65]PGE, VPN-015-97, S.M.Quennoz to U.S.NRC, Fuel Building Crane Load Testing and Certification for LCA-240 and 237, Rev.1, February 24, 1997
[66] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

dimensional testing, visual inspection, and nondestructive testing of major load-carrying welds and critical areas….” A PGE representative stated that they intend to do (b) just prior to loading operations. Since the fuel loading is not expected to last a full year, PGE does not anticipate having to repeat the dimensional, visual, and NDE testing during the loading process. The Staff intends to review this testing prior to loading.

Regarding the Fuel Building Crane, PGE stated that there is no requirement to do frequent load testing, unless the load rating is in question. They stated that since the test in 1997, “The load paths for both the main hook and auxiliary hooks were load tested following recent installation of load indicating sheave pins.” They stated further that the mechanical stops are not physically tested, but that the limit switches (electrical interlocks) are tested daily when the crane is used.

LCA-237 Section 7.2.1 states that “the fuel handling equipment will be load tested prior to use to ensure that the structural and lifting capabilities are verified to be adequate.” We asked the same question regarding the Spent Fuel Pool Crane: What constitutes “use” in the sense of how often or at what frequency during the months of the loading process will the crane be tested? PGE responded:

> During plant operations, the Spent Fuel Pool Crane was routinely load tested at full rated load prior to refueling campaigns. There is no requirement for continuing load testing. However, if a significant time has passed (over six months) in which no fuel has been moved, operators are required to use the Spent Fuel Pool Crane and fuel handling tool to lift a dummy fuel assembly in the SFP before using the equipment to lift an actual assembly. This operation constitutes a functional test since the weight of the dummy fuel assembly bounds the weight of the heaviest fuel assembly at Trojan. Moving the dummy fuel assembly is a routine operation prior to moving actual fuel assemblies, even for shorter intervals between fuel movements.[67]

PGE also stated that the electrical interlocks of the Spent Fuel Pool Crane are tested daily when the crane is used.

3. Impact Decelerations for Spent Fuel and MPCs

Since the previous submittal, the issue of impact decelerations for drop accidents involving spent fuel has changed significantly. Previous accident analyses in LCA-237 and the SAR for certain drop accidents referenced the results of a study by Lawrence Livermore National Laboratory (LLNL), UCID-21246[68], which gives the maximum allowable decelerations for the most vulnerable fuel assembly (Westinghouse 17 x 17 array) as 82g in a vertical drop accident and 63g in a side drop accident. This study was used to show that drop accidents which produced less than these decelerations would not result in damaged fuel. In March 1998, an analysis at the Calvert Cliffs plant indicated that the LLNL study, which until then had been widely used by the industry, did not substantiate spent fuel integrity at these numbers. There were several reasons for this, but the main ones were that the LLNL study had neglected the weight and stiffness of the fuel pellets, the material properties of irradiated cladding, and also the restraint and lateral support of the fuel basket structure to the fuel assemblies.

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[67] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002
[68] LLNL, Chun, R. et al., , Dynamic Impact Effects on Spent Fuel Assemblies, October 20, 1987

As a result of the analysis at Calvert Cliffs, PGE issued Corrective Action Request (CAR) C-01-0012 to document the problem and determine the appropriate corrective action for Trojan. In that CAR, PGE concluded that the problem did not impact Trojan directly, since the Trojan ISFSI does not rely solely on the UCID-21246 study. However, it may still impact the accident analyses, if not directly then indirectly. The result of the CAR was that PGE would rely on the generic, approved bases provided in the Holtec FSAR and Certificates of Compliance. However, it appears from a review of the Holtec FSAR that they, too, rely on the LLNL study. The current SAR and LCA-237 no longer reference either the LLNL study or its results. However, LCA-237 states:

> As described in Section 3.5 of the Holtec HI-STORM 100 System FSAR, studies of the capability of intact spent fuel rods to resist impact loads indicate that the most vulnerable intact fuel can withstand a 63g deceleration in the most adverse orientation (horizontal impact). Thus, limiting the MPC deceleration to 45g ensures that intact fuel rod cladding integrity is maintained…

Those studies referenced in the Holtec FSAR appear to be the same LLNL studies, since there is no change in the numbers. This brings us full circle to reliance on the increased safety factors specified in NUREG-0612. However, until further studies are done on the impact limits for spent fuel, the Staff believes that limiting the impacts to less than 45g for an MPC in the Transfer Cask or 60g for an MPC in the Concrete Cask, both of which are analyzed for vertical drops only, is sufficiently conservative. Further evidence of the conservative nature of the drop analysis is that the analysis PGE did for Trojan-specific fuel based on the LLNL study was that it could withstand 100g. So, even if the LLNL study is incomplete, the Trojan fuel would most likely be able to withstand the most limiting postulated accident scenario of 60g.

In response to the findings at Calvert Cliffs, the NRC issued Interim Staff Guidance (ISG)-12, "Buckling of Irradiated Fuel Under Bottom End Drop Conditions," in June 1999 to provide guidance to NRC staff to ensure consistent reviews of license change applications that used the methodology of UCID-21246. In response to our questions on this matter, PGE responded that their revised analyses of a postulated cask drop has incorporated the ISG-12 considerations.

> For all drops except for the postulated MPC drop into a Concrete Cask or Transport Cask (that is not considered credible), this reanalysis is accomplished by confirming that the maximum impact deceleration for each postulated MPC drop is less than the design deceleration for the MPC[69], then using the Holtec HI-STORM SAR Section 3.5 analysis to conclude that fuel rod integrity is maintained. The Holtec HI-STORM SAR Section 3.5 analysis uses the UCID-21246 methodology and incorporates the ISG-12 additional considerations, such that the analysis takes into account the weight of the fuel pellets.

PGE's conclusion is that "there is no reasonable likelihood of significant damage to spent fuel in the unlikely event of a drop event." Based on what we have been able to determine through our reviews of all this material, the Staff concurs with this conclusion.

4. Impact Limiters

Impact limiters are used on the 93 foot level of the Fuel Building, in the Cask Loading Pit, near the Fuel Building hoistway, and on the 45 foot level under the hoistway to ensure that the forces

[69] The design deceleration for the MPC in the Transfer Cask is 45g, and for the MPC in the Concrete Cask is 60g.

in a postulated drop accident of a loaded MPC and Transfer Cask are within the design limits. We requested additional information about these devices, including: What material is used? What are the dimensions? Are they considered important-to-safety? Are they purchased commercial grade? And, how are their energy absorption characteristics verified?

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. We verified that PGE's Nuclear Oversight (QA) personnel witnessed the tests to ensure that the impact limiters perform as designed. Vendor certification of material properties are required to verify the energy absorption characteristics.[70] 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.

F. THERMAL ANALYSIS DURING ISFSI LOADING

1. Maximum Fuel Cladding Temperature

Part of the following discussion on the use of the short-term, high temperature limit (1058 ^{o}F) for fuel cladding is repeated from our January 27, 1999, report, partly to provide continuity to our review effort, partly because of changes in parameters and new information, and also because the licensee and Holtec make the same argument as with the previous design in favor of a short-term, high temperature limit, which the Staff finds controversial. We continue to take issue with the use of this temperature limit due to conflicts in the data and the interpretation of the data upon which it is based; however, our conclusion remains the same—that the fuel cladding will remain substantially below the limit, and, therefore, the fuel cladding is not in jeopardy of fracturing during the loading operations.

The Staff reviewed PGE and Holtec's thermal analysis of an MPC in the Transfer Cask loaded with spent fuel to ensure that temperature limits would not be exceeded during the loading process. The Staff verified that this thermal analysis considered the worst case scenario from the standpoint of heat transfer. For example, the analysis considered both the MPC with a helium atmosphere and in a vacuum. In response to an RAI question from our earlier review, 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.[71] We concurred with this response, and believe it remains valid with the new design.

Vacuum drying operations during the loading and sealing of spent fuel assemblies in the MPCs 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. OOE's 1999 evaluation included an extensive discussion of the maximum allowable temperature during vacuum drying operations. Although OOE expressed concern over the NRC's choice of maximum allowable temperature, we considered the maximum projected temperature during vacuum drying at Trojan to be acceptable, based on PGE-specific calculations that had

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[70]PGE, CPY-035-96, C.P.Yundt to D.Stewart-Smith, Response to RAI, September 23, 1996
[71]ibid.

conservative assumptions regarding the heat output of Trojan's spent fuel. The full details of that discussion are contained in the 1999 report.

The current analysis by Holtec puts the maximum temperature of the fuel in a vacuum as 659 ^{o}F. This is based on maximum heat load for an MPC of 17.4 kW, which is very conservative. The reasonably bounding heat load for an MPC is likely to be less than 15 kW. We reviewed the inputs and assumptions to the Holtec calculation (HI-2012725, Rev.2, Computation of the Peak Cladding Temperature During Vacuum Drying of Trojan Fuel), and found them reasonable, although intentionally not extremely conservative in order to get a more realistic number. For example, the analysis did not assume an adiabatic situation, but included heat losses from the MPC to ambient via natural convection and radiation heat transfer. It also employed a helium conductivity reduction of about 5% in the near vacuum state, whereas the actual conductivity would be closer to 1% for a vacuum of 3 torr. And the analysis assumed no paint on the Transfer Cask, thereby giving the surface a conservatively higher emissivity. The maximum temperature for the 17.4 kW fuel in a vacuum condition for 24 hours was determined to be 520 ^{o}F, and the asymptotic temperature (that is, no time limit) was 659 ^{o}F. For the reasonably bounding heat load of 15 kW, the maximum temperature at 24 hours in a vacuum was less than 500 ^{o}F, and the asymptotic temperature was 609.6 ^{o}F. The Staff concludes that these results are reasonable.

Additionally, we expect that the vacuum drying system (VDS) is designed to dry an MPC long before the fuel reaches these maximum temperature. OOE Staff observed vacuum drying on a mock Holtec fuel canister at the Columbia Generating Station in Washington state, and observed that it reached a strong vacuum very rapidly. The system we observed in Washington was supplied by Holtec and is the same design that Trojan will use.

In conclusion, based on our review of thermal calculations, the conservatism built into the thermal analyses, and PGE's revised numbers for fuel cladding temperature, the Staff concurs with PGE's finding that "there will be no significant degradation of the fuel cladding due to vacuum drying conditions."

2. Time for the Water in the MPC to Boil prior to Vacuum Drying

The Staff reviewed the issue of boiling in an MPC prior to it being vacuum dried. LCA-237 Section 5.3.2 states:

> The time required to complete MPC lid sealing, hydrostatic testing, MPC lid-to-shell weld leakage testing, and MPC cavity blowdown is administratively limited as described in ISFSI SAR Section 5.1.1.2 to ensure that the water inside the MPC does not boil prior to removal of the water from the MPC.

This section refers to Holtec's FSAR for the HI-STORM 100, July 2000, which provides a methodology in Section 4.5.1.1.5. The ISFSI SAR states that "The heat output contents (fuel and other materials) of the MPC is first calculated and a heat-up rate established. The heat-up rate is then used to determine the administratively controlled limit (in hours) to ensure that the water in the MPC does not boil." LCA-237 Section 5.2.5.1 and LCA-246 Section 5.2.5.2 state: "A calculation shows that approximately 55 hours of heat-up are required for the water in the MPC loaded with the bounding design decay heat load to reach boiling." This is significantly different than the original SNC calculation, which determined a time-to-boil of 37 hours.

The Staff went through the process described in the Holtec FSAR. We used the methods of Section II.E.1 of this report to verify the accuracy of the process, and we confirmed the estimate of 55 hours. The primary reasons for the difference between this and the SNC result are heat load and slightly different masses. We also confirmed that 55 hours is a conservative number, since it was determined with a heat load of 17.4 kW. Actual maximum decay heat will most likely be closer to 14.5 kW. Therefore, the time-to-boil will be significantly greater.

The Staff reviewed procedure FHP 50-03, which is the governing procedure for the loading operations, and found references to the actions required when the administrative time limit for boiling was reached. However, there was no consideration of the time that it would take to complete preparations for those actions, i.e. set up the process to establish circulation cooling, before the administrative time limit passed. The procedure requires that circulation cooling be "immediately available," but this phrase is defined as the system not being tagged out and available within 96 hours. We presented this to PGE at a meeting on June 13, 2002, and they agreed to revise the procedure such that the administrative time limit included the set-up time to establish circulation cooling, such as Time-To-Boil minus 2 hours.

G. RADIOLOGICAL SHIELDING AND CONTROLS

1. Radiological Control Measures During Loading

PGE's Radiation Protection Program is described in Section III.E of this report. Section 7 of the SAR describes the Radiation Protection measures that will be in place for the ISFSI, and we describe our review of those measures for normal operations in Section IV.D of this report. The review criteria and acceptance limits for normal operations stated in Section IV.D remain the same for loading operations.

Table 7.4-3 in the SAR gives the estimated personnel exposure doses during loading operations, including moving fuel, decontaminating the MPCs and Transfer Cask, welding the MPC lid, and draining and vacuum drying. PGE projects that loading all 34 casks and moving them to the ISFSI Storage Pad will require an exposure of approximately 114 person-rem. The estimates are based on a combination of industry experience and Trojan specific calculations. 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 from the source, and shielding. Each of these factors can vary greatly from plant to plant. Time of exposure is heavily influenced by procedures and administrative controls, which can also 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 component design. A significant difference between the SNC and Holtec designs is that the MPC has less lid shielding than the SNC PWR Basket. Conversely, the Holtec Transfer Cask has a water jacket for neutron shielding; the SNC Transfer Cask had none.

The most significant factor for loading operations will be PGE's administrative controls instituted in their ALARA job reviews prior to conducting loading activities. Based on the performance of PGE's radiological practices during the Large Component Removal Project and other decommissioning activities, including ALARA job reviews, the Staff is satisfied that the radiological control measures for loading operations will meet ALARA requirements. In addition, however, because of the service agreement with Holtec to perform the fuel handling, loading, and welding operations, and the uncertainties with which that will be handled, the OOE site inspector will attend selected pre-job briefings to verify that ALARA measures are covered, and will review exposure records over the course of the loading operations.

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."[72] We verified that this policy is already part of PGE procedures.

2. Loading of Fuel Debris

Section 3.1.1.3 of the previous SAR revision stated that the fuel debris would not exceed 7.5 kg per SNC PWR Basket, and gave limits of 10 kg of fissile material per basket for the TranStor™ Shipping Cask and an additional 20 Curies of plutonium to meet the offsite transportation requirements of 10CFR71.63. This was deleted from the current proposed revision to the SAR; however, the current proposed Technical Specifications limits fuel debris to 7.5 kg of fissile material per MPC and 20 Curies of Plutonium per MPC. The mass of fuel debris is a small fraction of the mass of an intact fuel assembly, it is less reactive, and it will be stored in Failed Fuel Cans or Damaged Fuel Containers, separate from intact fuel. Therefore, it is unlikely to cause thermal, structural, or shielding problems no matter how the fuel debris is arranged within the Failed Fuel Cans or Damaged Fuel Containers. Also, SAR Section 5.1.1 states that fuel debris has been "visually inspected to verify that each ... item conforms to the established classification criteria."

In 1997, the fuel debris was collected and stored in 42 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 go into corner locations in the MPCs 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.

H. CRITICALITY EVALUATION FOR LOADING OPERATIONS

PGE's criticality analysis for storage of spent fuel in a dry condition is described in SAR Section 4.2.7. The Staff's review is provided in Section IV.E of this report. The review criteria for criticality during loading operations is the same as that described in Section IV.E. Briefly, the fuel is to be maintained in a subcritical condition under all circumstances. The measure of criticality, k_{eff}, must be less than 0.95.

The criticality analysis for loading operations is described in LCA-237 Section 5.1.1. The parameters used in this latter analysis were identical to that used in the dry storage conditions except that the MPC is filled with pure water. This is a conservative approach, since the water that will fill an MPC will be the same as that used in the SFP (2000 ppm boron or greater). The k_{eff} in this scenario is bounded by the results for the transportation accident, which results in a k_{eff} of 0.9377, which is below the limit of 0.95. As with the analysis for dry storage, this analysis used the NRC-approved and verified MCNP4a computer code. Also, as described in Section IV.A.1 of this report, PGE QA reviewed the Holtec's validation and verification (V&V) of the

[72] ibid.

code. The Staff concurs that the approach taken is conservative and provides reasonable assurance that the fuel will remain below the criticality criterion under all loading conditions.

Section 5.1.1.2 of the SAR describes the potential for boiling in an MPC after it has been removed from the Cask Loading Pit. One of the concerns with boiling was the unknown impact this could have on criticality. As part of our 1999 review, we asked PGE to address this concern. They responded that the SNC design "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."[73] Discussions with NRC representatives also confirmed PGE's analysis that boiling was not a factor in the criticality analysis. The new Holtec design is similar in this regard. Therefore, these comments remain valid.

I. ACCIDENT ANALYSIS OF ISFSI LOADING OPERATIONS

1. Review Criteria

The review criteria for the accident analysis is the same as that described in Section IV.F.1 of this report. The Staff sought to verify that credible accident scenarios had been thoroughly evaluated by PGE and their primary vendor, including so-called "off-normal" and "accident events," and that hypothetical and non-credible events that could result in the maximum potential impact were also evaluated. As described in Section IV.F.1, the principal acceptance criteria for evaluating postulated accident scenarios involve three distinct elements: radiation protection, subcriticality, and confinement.

Briefly, the radiation protection criteria for normal and off-normal operations is to limit the annual dose equivalent to any individual beyond the controlled area to 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ. The criteria for a design basis accident is to limit the dose to less than 5 rem to the whole body or any organ. In addition, the EPA PAGs limits set the limits for offsite dose resulting from an accident to 1 rem to the whole body, 5 rem to the thyroid, and 50 rem to the skin. The criteria for subcriticality is to maintain the fuel in a subcritical condition under all circumstances. The measure of criticality, k_{eff}, must be less than 0.95. Lastly, confinement must be maintained in all scenarios and fuel damage prevented. This requires that any impact of a loaded MPC be less than 45g in a Transfer Cask and 60g in a Concrete Cask.

2. Drop Accidents During Loading Operations

PGE's drop accident scenarios involving an object dropped onto an MPC loaded with spent fuel include the following dropped objects: a fuel assembly, MPC lid, Lift Yoke, Transfer Cask lid assembly, MPC lift rigging, and Concrete Cask shield ring or lid. The fuel assembly drop is described in LCA-237 Section 5.2.1.1, and is postulated to occur while the MPC is inside a Transfer Cask and submerged in the Cask Loading Pit. The design change from the SNC Basket to the Holtec MPC had no effect on this postulated accident, which is bounded by the existing analysis in the TNP Defueled SAR (DSAR). That analysis demonstrated that such an accident would not result in a radiological release that exceeds regulatory limits or a k_{eff} of 0.95. That analysis assumed six months of decay time for the spent fuel. By the end of 2002, the spent fuel will have a minimum of nearly ten years decay time. Because the accident is bounded by the DSAR and the present fuel is much less radioactive, the Staff did not review the analysis..

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[73]PGE, CPY-007-97, C.P.Yundt to D.Stewart-Smith, Revisions to Response to RAI #2, March 17, 1997

The Staff reviewed PGE's original analysis of the drop of the Lift Yoke onto a loaded and unsealed SNC PWR Basket using the review methods described in Section II.E.1 of this report, and concurred with the findings described in the calculations and in LCA-237 Section 5.2.1.3. As with the SNC PWR Basket, this scenario was postulated to occur after the MPC lid is in place in the MPC, since the Lift Yoke is used after the lid is lowered onto the MPC to lift the Transfer Cask out of the Cask Loading Pit. The original analysis considered the impact of the Lift Yoke on the 8-inch thick PWR Basket shield lid, and determined that it would not deform sufficiently to damage the spent fuel. In the current proposed revision to LCA-237, PGE cites the guidance of NUREG-0612 regarding the design of lifting components to claim that this scenario is not credible. Consequently, they no longer analyze the accident. However, were this accident to occur, damage would remain limited to the lid, which is now 9.5 inches thick instead of 8 inches thick, and the fuel would most likely be unaffected.

As with the previous revision of LCA-237, PGE did not analyze the drop of an MPC lid onto the spent fuel in an MPC in the Cask Loading Pit, citing the guidance of NUREG-0612 and ANSI N14.6, plus load testing and administrative controls. PGE states in LCA-237 Section 5.2.1.2, "…the guidance of NUREG-0612 is being used in the design of the lifting components to minimize the possibility of this event to the degree that the event is not considered credible." If the effects of a drop accident are not going to be followed, NUREG-0612 requires that the license follow the single-failure proof guidelines of the regulatory guide; however, for the previous analysis, the NRC had granted an exemption to those requirements. For this analysis, the lifting components are such that the single-failure proof guidelines will be followed. We asked PGE to describe the lift of the MPC lid.

> The lid lifting hoist rings and the rigging are designed with safety factors that are designed to lift twice the weight of the lid and associated rigging equipment in accordance with NUREG-0612 and ANSI B30.9 for "lifting devices that are not specifically designed" (i.e., commercial rigging). In addition, the lift equipment was load tested to 200% of the rated capacity of the equipment. NUREG-0612 allows either redundant load paths or enhanced safety factors to demonstrate an equivalent single-failure-proof design for these components. The lid lifting rigging is redundant in that four separate hoist rings and associated rigging are used for each lid and only two are necessary to support the lid.[74]

For the previous analysis, we asked if the drain line, which is to be attached to the MPC lid (at that time the PWR Basket shield lid) while it is suspended over the fully loaded MPC, could be dropped onto fuel in the MPC and had that been analyzed? PGE responded that such a drop had not been analyzed but that they would secure it with a safety line to prevent such an occurrence. The Staff found this satisfactory. Since that time we have verified that the attachment of the safety line (called a "tagline") is described in FHP 50-03.

Postulated drops of a loaded Transfer Cask described in LCA-237 include 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 reviewed each of these accident scenarios in our January 27, 1999 report, and agreed with PGE's conclusion that by limiting the drop height and using impact limiters, these postulated drops did not result in damage to the intact fuel or exceed the EPA Protective Action Guidelines. In 2002, we examined calculations TI-140, Rev.2, Failure Modes

[74] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

and Effects Analysis, and TI-137, Rev.1, Holtec Transfer Cask Hypothetical Drops, Fuel Building and ISFSI Transfer Station. The latter calculation concludes that the differences between the weights and geometries of the two systems are small; and, therefore, the previously calculated drop decelerations remain valid.

For the Holtec design, the limiting deceleration is 45g due to the Transfer Cask, whereas for the SNC design the limiting deceleration was 82g due to the limit on the fuel. The MPC alone has a limit of 60g. The SNC PWR Basket is more robust probably due to its thicker shell. Although the likelihood of a drop is remote, the revised analysis had to ensure that a postulated drop does not result in exceeding 45g. Because of the thinner shell, we asked PGE about the likelihood of buckling and/or deformation of the MPC in a postulated drop. PGE responded that the buckling stability of the MPC shell was evaluated under a 60g axial load, with satisfactory results. (See the discussion in Section IV.F.8 regarding buckling and deformation.) Therefore, a 45g hypothetical drop would not cause significant buckling. "In addition, the stresses in the MPC due to a 45g hypothetical drop accident have been demonstrated to meet ASME Service Level D stress limits. Therefore, the integrity of MPC confinement boundary is safely maintained in the aftermath of a Transfer Cask drop."[75]

The other major drop accident scenario involves the drop of a loaded MPC into a Concrete Cask. The consequences of this accident are bounded by a postulated MPC drop accident into either a Concrete Cask or a Transport Cask at the Transfer Station. Our review of that accident scenario is described in detail in Section IV.F.8 of this report.

The Staff's review of the drop accident scenarios associated with loading operations 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 well below the EPA PAG limits.

LCA-237 Section 5.2.3.1 also addresses the issue of a breach in the Cask Loading Pit liner or SFP liner. This event would have the same causes, likelihood, and consequences for either the SNC or Holtec design. Therefore the analysis described in our 1999 report remains valid.

3. Tipover Accidents During Loading Operations

The Staff 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 such events were bounded by the consequences of the drop accidents analyzed, which developed much greater impact forces. 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.

The issues related to mishandling events are the same for the Holtec design as they were for the SNC design. In our January 27, 1999 report, OOE recommended that PGE procedures include

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[75] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

acceptance criteria for centering a canister in a Concrete Cask initially, provisions for determining when a visual examination is required and the circumstances that would require re-centering, and corrective actions to re-center a canister. PGE stated that the procedures would be modified to address these concerns. This recommendation relative to the use of Holtec MPCs is still valid; and, as with the previous report, the Staff will confirm these procedural changes prior to loading. Additionally, the Staff recommends that visual verification of centering an MPC 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

This section has not changed significantly since our 1999 review. Most operational events would involve installed equipment such as cranes or procedures that are specific to PGE, and these would not change as a result of the Holtec design.

a. Loss of Electrical Power

Loss of electrical power is addressed briefly in LCA-237 regarding its impact on welding, MPC draindown, and vacuum drying. PGE actions during a loss of electrical power would be essentially the same for either the Holtec or SNC design. Therefore our recommendations from 1999 remain valid. If anything, a loss of power is less of a concern now, because the fuel has had an additional 2 years to cool.

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:

> The fabrication specification for the PWR basket requires drag testing each 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.[76]

The Staff accepts that this response is still valid for the current design with the Holtec MPCs. We plan 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 are evaluated in LCA-237 involving the rupture of a pressurized line. One is the rupture of the discharge line on the MPC while it is being blown dry. The second rupture involves the helium supply line. Both events have the same likely causes and consequences regardless of whether PGE uses a SNC or Holtec canister. Both events are considered unlikely because of the relatively low pressures involved, as compared to the design pressure ratings of the subject piping and flexible hose lines.

6. Conclusion

Fuel loading operations will be performed in accordance with PGE's existing operating license under 10CFR50. 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

[76] PGE, CPY-032-96, C.P.Yundt to D.Stewart-Smith, Response to RAI, June 27, 1996

safety questions in procedures. In our 1999 evaluation, we concluded that postulated accidents during fuel loading would not result in radiation exposure greater than the EPA Protective Action Guidelines, which form the basis for PGE's EFSC-approved emergency plan. We also concluded that fuel handling operations would not result in dose from effluents to any person in an unrestricted area in excess of 5 mrem, as required by OAR 345-026-0390(4)(f). These conclusions are not affected by the change from SNC to Holtec canisters.

VI. CONTINGENCY FUEL UNLOADING

A. DESCRIPTION OF LCA-246 AND THE UNLOADING PROCESS

PGE developed LCA-246, Spent Fuel Cask Loading in the Fuel Building Contingency Fuel Unloading to the Fuel Pool, Rev.2, for the scenario during loading operations and before the SFP is made inoperable where a defect is identified in or damage occurs to an MPC prior to placement in a Concrete Cask and being moved to the Storage Pad. Such a scenario might occur if, for example, there was helium leakage following the lid welding operations or the dose rate was higher than anticipated. Were such an event to occur, the process of unloading would be similar to the loading process, but in reverse.

The MPC inside the Transfer Cask would be moved to the DAS. The Transfer Cask top lid and MPC lift cleats would be removed, then portions of the closure ring would be removed to allow access to the vent and drain cover plates. These plates would be removed and the vent and drain piping connections attached. Next, the MPC atmosphere would be sampled. If greater than 200 ^{o}F, it would be cooled by connecting the helium recirculation cooling system and circulating cooled helium through the MPC cavity or by cooling the MPC exterior with air or water. Once the temperature of the cavity was below 200 ^{o}F (to prevent boiling and thermal stresses), the MPC would be flooded with water through the drain line. With cooldown complete, 50 to 120 gallons of water would then be removed, and a cutting system installed on top of the MPC to cut the MPC lid away from the MPC shell. Next, the lid retention system and annulus seal would be installed, the Gap Flushing System connected, and the Cask Loading Pit filled. Then the Transfer Cask would be lowered into the Cask Loading Pit, and the MPC lid removed. Finally, the fuel would be removed and stored in the Spent Fuel Pool following fuel handling procedures.

B. OOE'S REVIEW OF CONTINGENCY UNLOADING

The accident analyses described in LCA-246 are identical to and bounded by those described in LCA-237, which are reviewed in Section V of this report. Areas where problems unique to the unloading could occur are in weld cutting and the use of the helium recirculation cooling system. The Staff didn't find any significant concerns in the weld cutting. The equipment to be used is similar to the welding apparatus, and we understand that there will be training in this prior to the commencement of loading operations.

With regard to the helium recirculation cooling system, LCA-246 Section 5.1.2 states: "If not controlled, the extremely rapid cooldown rates to which the hot MPC internals and fuel cladding could be subjected during reflooding of the MPC cavity could potentially result in uncontrolled thermal stresses and failure in the structural members." However, no rate of reflood is given. The previous revision to the LCA stated: "Analyses have been performed which demonstrate that injection flow rates less than 8 gpm do not result in unacceptable fuel clad stress…." These statements were deleted. Additionally, the HI-STORM LAR, July 2001, states for this event that the minimum flow rate of the cooling water is specified by formula. We asked PGE if the rate of reflood would be specified? They responded that once the MPC cavity is cooled to less than 200 ^{o}F, "the water reflood can safely proceed without concern for rapid metal quenching that occurs in an agitated (i.e. boiling) heat transfer regime. Specification of a minimum re-flooding rate and MPC pressure monitoring is not necessary."[77] We should note also that the cooling water will

[77] PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

enter via the drain port and be directed to the bottom of the MPC, thus causing minimal shock to the fuel by rising from the bottom. The Staff accepts PGE's response; and we will review the procedure for this process, FHP 50-08, prior to loading operations.

The NRC requires a "dry-run" demonstration of ability to cool down and unload a fuel canister in case of accident. PGE anticipates that helium cooldown will not be required, since the temperature of the helium inside the MPC is likely to be less than 200 ^{o}F. However, if the analyses demonstrate that a temperature equal to or greater than 200 ^{o}F is likely for any MPC, PGE must demonstrate the ability to cool the helium inside the MPC to the point where PGE can transfer fuel safely. We have recommended a rule to this effect, which is included as part of OAR 345-0390(6)(c). The Staff believes PGE can meet this requirement either by:

1. Providing an analysis showing that helium temperature will be acceptably low without further cooling. The analysis must be reviewed by either the NRC or OOE; or
2. Demonstrating a helium cooldown system (OOE witnessed such a test at the Columbia Generating Station in Washington State on July 17, 2002.); or
3. Demonstrating the ability to cool the exterior of the MPC and verify that the helium temperature is low enough by taking a bulk measurement.

Any of these methods would be acceptable, but OOE must see one of them prior to the first MPC loading.

VII. 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 then consider PGE's compliance with the EFSC spent fuel storage rule, OAR 345-26-390.

A. EFFECT OF ISFSI ON DECOMMISSIONING

OOE's January 22, 1996 report "Review of PGE's Decommissioning Plan for the Trojan Nuclear Plant" stated that PGE's Decommissioning Plan complied with all requirements of OAR 345-26-370. 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 our 1999 review of the ISFSI using the SNC design, OOE concluded that the SNC design met the requirement of OAR 345-026-0390(4), and that the ISFSI would not adversely affect PGE's continued ability to meet the Council's Decommissioning rules. To approve the change to the Holtec design, the Council must find that the design change will not invalidate the conclusions in that 1999 report, or otherwise adversely affect PGE's ability to comply with the decommissioning rules. The relevant decommissioning rules are as follows:

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).*[78]

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 the change from SNC to Holtec.

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 the change from SNC PWR Baskets to Holtec MPCs that would increase the chances for contamination of the site. The MPCs will be sealed, welded shut, and pressure-tested to ensure an airtight seal. Once sealed, the MPCs will be backfilled with helium to a pressure of approximately 30 psig. The MPCs will, in turn, be stored in reinforced Concrete Casks, which will not be airtight but will preclude gross damage to the MPC. The SAR also provides analyses demonstrating that the MPC is designed to withstand off-normal events such as seismic events and handling accidents. The Staff's review of the structural analysis, including the non-leakage credibility, and the accident analyses are described in Section IV this report. Based on our review of the SAR, the Staff concludes that the canister design precludes recontamination of the site due to fuel damage.

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[78] This rule is superceded by OAR 345-026-0370(2)(h), which adopts new site release criteria consistent with more current NRC regulations at 10 CFR 20.1402. However, the reasons why the change from SNC to Holtec design will not affect compliance with the site release criteria apply equally to the newer criteria.

The Council's 1999 approval of the ISFSI with the SNC design was also based on findings that external contamination of the steel canister was not significant. The Council also recommended radiological monitoring requirements. These considerations are not affected by the change from SNC to Holtec.

Exposure Rate - OAR 345-26-370(2)(b): *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.*

This rule is superceded by OAR 345-026-0370(2)(h), which adopts the most recent NRC requirement for final site release criteria at 10 CFR 20.1402. The applicable NRC release criterion is 25 mrem/year above background.

The direct radiation dose rate at 100 meters from the stored fuel has been estimated at approximately 374 mrem/year (assuming occupancy of 2080 hours/year). This exceeds the EFSC standard for decommissioning. In our 1999 ISFSI evaluation, OOE raised concern regarding PGE's ability to meet the decommissioning criteria, taking into account the direct radiation from the ISFSI. PGE responded by conducting preliminary surveys of the ISFSI area with no spent fuel present. PGE also described a method for shielding the instruments used in the final radiation survey, so survey technicians could distinguish radiation due to the ISFSI from residual contamination due to plant operations.[79]

Both of these measures were described in detail and found to be acceptable in OOE's 1999 evaluation of the SNC-designed ISFSI. The change from SNC to Holtec canisters does not change the approach used in the 1999 evaluation; and, therefore, OOE believes that the change in design will not adversely affect PGE's ability to meet the decommissioning criteria of OAR 345-26-370(2)(b) or 345-026-0370(2)(h).

Radioactive Waste Removal - OAR 345-26-370(2)(c): *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. The Staff reviewed the plan and found that the removal of spent fuel from the Trojan SFP would not adversely affect PGE's ability to remove non-fuel waste. Nothing about the change from SNC to Holtec canisters changes this conclusion. Therefore, the proposed design change does not adversely affect PGE's ability to comply with OAR 345-26-370(2)(c).

Effluent Monitoring - OAR 345-26-370(2)(d): *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 change from SNC to Holtec canisters does not affect PGE's ability

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

or responsibility to monitor effluents. 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 changes to these plans will continue to require Council approval.

Environmental Radiological Monitoring - OAR 345-26-370(2)(e): *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.*

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 ISFSI SAR. As stated above, the details of PGE's monitoring plans may change, subject to Council approval, as site conditions change. PGE has requested, and received, permission to reduce its radiological monitoring plan because most of the radioactive material at the site has been removed. However, these changes reflect progress in decommissioning, and the Staff considers them acceptable.

Removal or Control of Hazardous Wastes - OAR 345-26-370(2)(f): *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. The Staff reviewed those provisions as part of our review of the Decommissioning Plan, and found them acceptable. The Holtec design does not involve more or different hazardous wastes when compared with the SNC design. In fact, the Holtec MPCs are stainless steel and require no protective coating, which eliminates a possible source of hazardous waste.

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 345-026-370(2)(a) through (f), or with the criteria at OAR 345-026-0370(2)(h).

B. COMPLIANCE WITH OAR 345-026-0390

In November 1995, the Council amended OAR 345-026-0390, which sets forth general criteria that any spent fuel storage installation must meet. In approving the SNC-designed ISFSI in 1999, the Council found that the ISFSI met the criteria of OAR 345-026-0390. To approve the Holtec design, the Council must conclude that the change to Holtec canisters does not affect PGE's compliance with OAR 345-026-0390, or invalidate the findings made in 1999. The criteria are as follows:

345-26-390(1) Purpose:

(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), USDOE must ultimately take title to and possession of the spent fuel and high level waste.

At this time, there is no Federal repository, and no one can predict with certainty when there will be one. However, the ISFSI is an interim measure, and is designed to facilitate shipment offsite. The Holtec MPC is designed for transport as well as storage. The acronym "MPC" stands for "multi-purpose canister," and one of the purposes is as a transport canister. We expect that Holtec will receive a federal shipping license under 10 CFR 71 on a schedule consistent with fuel loading. If, however, Holtec does not have the shipping license by the time transfer of all spent fuel from the pool to the ISFSI is complete, current rules require PGE to provide the Council with alternative shipping plans, including cost estimates. We conclude that the proposed use of Holtec MPCs 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 10CFR61 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.

Proposed Revision 2a to the SAR, Sections 3.1.1 and 3.1.2, states that there are 779 intact fuel assemblies currently in the SFP, plus 11 partial fuel assemblies and 1 fuel rod storage container, which contain intact, suspect, or damaged 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. The NRC has concluded that leakage from the Holtec MPC is unlikely enough to be considered "not credible." This precludes radiological discharges from the ISFSI, and satisfies subpart (2)(a) of this rule.

In evaluating the original SNC design, the Staff reviewed the projected off-site radiation exposures following accidents and natural events. We concluded that PGE had selected and analyzed the worst credible accidents, and that the projected consequences of those accidents 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 at the Controlled Area boundary from any accident or natural event were well under the limits of the EPA's Protective Action Guidelines for off-site protective actions, as required by subpart (2)(b) of this rule. The Holtec MPC is designed to withstand the same accidents and events as the SNC PWR Basket was; therefore, the design change does not invalidate our previous findings.

PGE will not store any radioactive material other than that generated during Trojan plant operations. This satisfies subpart (2)(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 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, as revised for the use of Holtec MPCs, addresses the design and operation of the proposed storage casks, and compliance with OAR 345-26-390(2) is discussed above.

(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 revised 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. The Staff reviewed the accident analyses 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 345-026-0390, subparts (2)(a), (2)(b) and (2)(c). Chapter IV of this report describes the Staff's review of the safety analyses 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.

OOE's 1999 Staff evaluation concluded that the ISFSI complied with this requirement. The conclusion was based largely on the SNC-designed Concrete Cask, which PGE will now use in conjunction with the Holtec MPC. In terms of ability to withstand a seismic event, the Holtec MPC is equivalent to the SNC PWR Basket. Therefore, our 1999 conclusion still applies.

(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 10CFR72.106 (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.

OOE's 1999 Staff evaluation concluded that radiation exposure from the ISFSI following the Seismic Margin Earthquake and other credible accidents would remain well within EPA Protective Action Guidelines. The revised SAR shows that, with the Holtec MPC, projected radiation exposure following a design-basis accident remains within those guidelines.

At an EFSC meeting in May 1997, Council members expressed concern over the ability to make repairs promptly, particularly after a major earthquake or volcano eruption. In 1999 the Council adopted a rule requiring pre-staging of repair materials and contractual arrangements for prompt repair services. OOE has not proposed a change to that rule.

The other accidents analyzed in the SAR 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.

None of these accidents were expected to produce increased consequences because of a change from SNC to Holtec canisters. For drop events, the forces predicted in Holtec's analysis are different from those predicted in the SAR for the SNC design. However, the SAR shows that the MPC is designed to withstand the forces predicted from an event involving a cask drop. Our review is described in more detail in Section IV of this report.

(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.

The Council, in approving the ISFSI with the SNC design in 1999, found that the ISFSI site meets this rule. The change to Holtec MPCs is a change to the canister, not the site. Therefore, the ISFSI site continues to meet this rule.

(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.

The proposed rule approves a change to the steel canister. That design change does not affect plans for monitoring, security, or emergency planning. PGE has made certain changes to its security and radiological monitoring plans since 1999; however, the security changes were in response to the events of September 11, 2001, and the radiological monitoring changes were in response to reductions in the amount of radioactive material at the site. Both sets of changes were independent of the design change from SNC Baskets to Holtec MPCs.

(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 10CFR20.1003 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.

Our 1999 Staff evaluation included a detailed review of fuel transfer activities, and concluded that this requirement was met. The procedures and ancillary equipment for the Holtec MPC are slightly different from those used with the SNC design, but the steps in the fuel transfer process remain essentially unchanged. Both canisters are designed to be leak-tight, and NRC has found that leakage from the Holtec MPC is unlikely enough to be considered “not credible.” Therefore, the Trojan ISFSI using the Holtec design remains in compliance with this rule.

(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.

In our 1999 Staff evaluation, we concluded that the ISFSI would not adversely affect PGE’s financial ability to decommission the site. The problems with the SNC PWR Baskets and the delay that followed caused PGE to store fuel in the SFP for two years longer than scheduled. PGE has stated its intention to complete fuel transfer to the ISFSI and complete decommissioning by 2005. Moreover, most of the contaminated material at Trojan has already been removed, and the reactor building has been surveyed and shown to meet the site release criteria. This indicates that the change in design will not impede PGE’s ability to complete decommissioning.

(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 10CFR20 (effective March 1, 1994), including a program to maintain personnel radiation exposure As Low As Reasonably Achievable (ALARA) as that term is defined in 10CFR20.

As described in the revised SAR and in Section IV of this report, PGE has committed to using the same radiation protection procedures that apply to decommissioning generally. These are largely the same procedures that were in use when the plant was operating. OOE and NRC have observed PGE decommissioning activities throughout decommissioning and have found that they meet NRC radiation protection requirements at 10CFR20, including the requirement to maintain personnel exposure “ALARA.” Those practices would be followed regardless of whether PGE uses the SNC or Holtec design; and, therefore, this part of the rule is satisfied.

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 a federal rule, but it is clearly consistent with the ALARA principle. Since this capability was not required by federal rules, EFSC in 1999 adopted a rule requiring this capability, with local temperature monitoring used as backup. OOE does not propose to change that rule at this time.

(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 OAR 345-026-0370(2)(h). The change from SNC to Holtec design has not affected PGE’s ability to meet any of these decommissioning criteria.

(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.

OOE’s 1999 Staff evaluation described why the ISFSI met the requirement for a 40-year design life. The design change being considered now, in 2002, is the change from a combination stainless and carbon steel SNC PWR Basket to the all-stainless Holtec MPC. The other storage components of the ISFSI are unchanged. If anything, the all-stainless MPC is expected to be less susceptible to corrosion over a 40 year time period.

OOE also observed PGE Quality Assurance activities related to testing and fabrication of the MPCs, including an inspection of U.S. Tool & Die, where the MPCs are fabricated. As in 1999, OOE considered the Quality Assurance activities in place to provide reasonable assurance that the MPCs will perform as designed.

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 OOE’s 1999 Staff evaluation and were found acceptable at the time. There is nothing about the change from SNC to Holtec canisters that would require a different answer to this question.

(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. The change from SNC to Holtec canisters further facilitates eventual shipment, because the Holtec MPC is designed and licensed for shipping under 10 CFR 71.

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 change from SNC to Holtec canisters for use in the ISFSI meets 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. The Staff therefore recommends rules allowing fuel storage in an ISFSI utilizing the Holtec MPC as described in the final Revision 2 to the ISFSI SAR, as may be amended pursuant to NRC requirements.

VIII. CONCLUSIONS

A. SUMMARY

Based on our review of the revision to the ISFSI SAR and associated documents, the Staff considers that the proposed design change to the ISFSI, including the change to Holtec canisters, ensures adequate structural integrity, heat removal capability, leakage protection, and radiation protection under both 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 conclude that the contingency unloading process described in PGE LCA-246 can be performed safely. Section VII of this report lists recommended findings of compliance with EFSC requirements in OAR 345 Chapter 26. The recommendation to approve the change in ISFSI design is based on PGE commitments and assumptions identified in the SAR and associated correspondence. We therefore recommend a proposed rule, described in the following section, approving the ISFSI subject to specific requirements.

B. EXPLANATION OF PROPOSED RULE CHANGES

In letter VPN-049-2001, dated November 15, 2001 from Stephen Quennoz to David Stewart-Smith, PGE asked the Council to amend its rules to approve the use of Holtec MPCs in the ISFSI. PGE also asked EFSC to streamline the version of OAR 345-026-0390(6) adopted in 1999. Specifically, PGE asked EFSC to eliminate some requirements, either because they did not apply to the Holtec design or because they were one-time requirements that PGE had met.

After reviewing the revised SAR and associated technical material and PGE's request, OOE recommends the following changes for the reasons given.

<u>OAR 345-026-0390(6)</u>: Change the references to the ISFSI SAR, LCA-237, and LCA-246 to reflect the most current revisions of those NRC licensing documents.

<u>Delete old OAR 345-026-0390(6)(b) through (6)(i), plus (6)(k) and (l)</u>: These subparts to OAR 345-026-0390(6) require plans for remote temperature monitoring, radiation surveys during ISFSI loading, radiological monitoring, contingency plans after an accident or natural event, a procedure to unload damaged canisters, a license to ship fuel offsite, structural inspection, reliability of temperature monitoring, and procedural controls to ensure that PGE keeps all commitments it has made to EFSC or OOE. PGE asked EFSC to streamline the rule by combining some of these requirements into a single subpart (b) that lists required programs and requires PGE to maintain these programs as described in this September 2002 Staff report. OOE recommends this change because the September 2002 Staff report provides a sufficient basis for PGE to implement the required programs and for OOE to inspect them.

<u>New subparts (c), (d) and (e)</u>: These new subparts are streamlined requirements involving contingency plans following an accident, the license to ship fuel offsite using the Trojan canister, and the need for PGE to obtain Council approval before making changes that would invalidate the assumptions or conclusions that the Council relied on in approving this rule.

<u>New subpart (f)</u>: This subpart is similar to the current subpart (j) and concerns the reporting requirements. The only change from the existing wording is from annual to biennial (once every two years) reporting.

Subpart (g): This subpart is similar to the current subpart (m), and concerns temperature monitoring and air vent inspections. The only substantive change is that, at PGE's request, OOE recommends replacing the daily inspection of the air vents with a weekly inspection requirement. The Staff agrees that a weekly inspection is sufficient, because the spent fuel has had additional time to cool and because the latest analysis shows that the fuel would not reach an unacceptable temperature during the increased time.

New subpart (h): This is a new rule that requires PGE to require contractors to adhere to OOE requirements and the commitments PGE has made to OOE. Staff proposes this new requirement, because PGE intends to contract with Holtec for fuel loading operations.

C. TEXT OF PROPOSED CHANGE TO OAR 345-026-0390(6)

OOE proposes to delete rule language that is shown as bracketed [] and in *italics*. OOE proposes to add language in the text that is shown in **bold type**.

Spent Nuclear Fuel Storage

(6) The Council approves the plan, as may be amended under Part [*(i)*]**(e)**, for an Independent Spent Fuel Storage Installation (ISFSI) as described in the ISFSI Safety Analysis Report (SAR) (PGE-1069)**, Revision 2,** [*and*] License Change Application LCA-237**, Revision 5, and License Change Application LCA-246, Revision 2**. In addition to the criteria in OAR 345-026-0390(2)(a), (b), and (c), and (4)(a) through (k), the plan is subject to the following criteria:

(a) Controlled Area Boundary: Within six months after terminating the NRC operating license, under 10 CFR 50, PGE shall submit to the Oregon Office of Energy for approval evidence of its ability, in the event of an accident**,** to exercise control of personnel access to the Controlled Area as described in the SAR, as may be amended. This evidence may include such factors as any lease or contractual agreements with tenants at the site, administrative controls, or the results of a drill.

[*(b) Capability for Remote Temperature Monitoring: 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: Prior to loading the first basket, PGE shall provide the Office 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: Notwithstanding OAR 345-026-0390(4)(e), PGE shall continue the Radiological Effluent Monitoring Program and Radiological Environmental Monitoring Program required under OAR 345-026-0330 until PGE obtains the approval of the Oregon Office of Energy that PGE has implemented acceptable programs, as described in the ISFSI SAR in effect on the date of adoption of this criterion, to monitor the radiological impact of the ISFSI.

(e) Staging of Emergency Equipment: Prior to loading the last basket, PGE shall submit for Council approval a plan for maintaining equipment onsite and having equipment available within a reasonably short time to respond to credible accident scenarios.

(f) Replacement of Cask: Prior to loading the last basket, PGE shall submit for Council approval a plan for construction of new concrete casks.

(g) Unloading Procedure: Prior to loading the first basket, PGE shall demonstrate 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 TranStorä cask has not been received at the time the final fuel assembly is transferred from the SFP to the ISFSI, then PGE shall submit for Council approval 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 shall establish and submit to the Office a process, subject to Office concurrence, for the control of changes to written commitments made to the Office during the Office's review in support of the Council's approval of the ISFSI. The process shall be implemented by the next regularly scheduled Council meeting following the adoption of this rule.

(A) This process shall include provisions for:

(i) identifying and tracking the implementation of commitments;

(ii) controlling proposed modifications to commitments including criteria defining when prior notification to the Office is required; and

(iii) reporting and record keeping requirements.

(B) PGE shall not make a change that significantly reduces the scope or effectiveness of the commitment control process without prior Office concurrence.

(C) PGE must submit to the Office a proposed change to equipment or administrative controls described in the SAR, LCA-237, or Technical Specifications, or implemented under the above process, if the change may:

(i) Invalidate the basis for any finding of compliance with any of the requirements of OAR 345-026-0390(4); or

(ii) Result in a significant reduction in the scope, surveillance frequency or effectiveness of administrative controls.

(D) Prior Office concurrence is not required when the proposed change would not have the above effects, is required for compliance with the regulations or orders of the U.S. Nuclear Regulatory Commission or is necessary to protect the health and safety of the public when there is insufficient time to obtain prior Office concurrence.]

(b) Programs: PGE shall establish and maintain programs for Temperature Monitoring and Air Vent Inspection, Radiation and Environmental Monitoring, and Structural Inspection that are consistent with maintaining exposures to ionizing radiation As Low As Reasonably Achievable (ALARA) and with the assumptions and conclusions in the OOE "Staff Evaluation of Holtec Design for Portland General Electric's Independent Spent Nuclear Fuel Installation (ISFSI)" of September 20, 2002.

(c) Contingency Plans: Prior to loading the first canister, PGE shall have and demonstrate procedures to remove spent fuel from a previously loaded canister and safely return the fuel to the Spent Fuel Pool (SFP). PGE need not demonstrate procedures previously demonstrated during pre-operational testing in 1999, provided those procedures involve substantially the same operations and utilize substantially the same equipment. Before loading the last canister, PGE shall submit for Council approval a plan for maintaining equipment onsite and having equipment available within a reasonable time

period to respond to credible accident scenarios and a plan for construction of new concrete casks.

(d) Shipping License: If the 10 CFR 71 shipping license for the canister has not been amended to permit use of the Trojan canister at the time the final fuel assembly is transferred from the SFP to the ISFSI, then PGE shall submit for Council approval alternative plans and cost estimates for shipping spent fuel offsite prior to taking action that would preclude future use of the SFP.

(e) Changes to Commitments: PGE may make changes to the ISFSI as described in the plan without prior Council approval providing such changes do not reduce commitments or change the assumptions and conclusions in the OOE "Staff Evaluation of Holtec Design for Portland General Electric's Independent Spent Nuclear Fuel Installation (ISFSI)" of September 20, 2002. In the event that the proposed changes would reduce commitments or change the assumptions or conclusions of the OOE "Staff Evaluation of Holtec Design for Portland General Electric's Independent Spent Nuclear Fuel Installation (ISFSI)" of September 20, 2002, PGE shall obtain prior concurrence from OOE. Prior OOE concurrence is not required when the proposed change would not have the above effects, is required for compliance with the regulations or orders of the U.S. Nuclear Regulatory Commission or is necessary to protect the health and safety of the public when there is insufficient time to obtain prior OOE concurrence.

[*(j)*]**(f)** Reporting requirements: Within one year of the first cask loading and [*annually*] **biennially** 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 345-026-0380 may be used to satisfy this requirement.

[*(k) Structural Inspection Program: Prior to loading the first basket, 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 Office concurrence.*

(l) Reliability of Temperature Monitoring Equipment: Prior to loading the first basket, 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 the temperature data sufficient to identify anomalous trends that could indicate degraded instrumentation.]

[*(m)*]**(g)** Frequency of Temperature Monitoring and Air Vent Inspection: Prior to loading the first [*basket*]**canister**, PGE shall implement a program for thermal monitoring that includes readings of air outlet and ambient temperatures. The program shall include temperature surveillances [*every 12 hours*] **daily**, 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 [*every 24 hours*] **weekly**. PGE may reduce these surveillance frequencies with Office concurrence. Extensions of up to 25 percent of individual surveillance intervals may be applied to accommodate minor variations in work scheduling.

(h) Contractors: PGE shall require contractors who perform portions of the ISFSI loading, storage, or transporting operations to adhere to all applicable provisions of OAR 345-026-0390.

D. PROPOSED POST-APPROVAL OOE INSPECTIONS

The following is a list of open items and proposed action items for the Staff. 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.

- PGE committed to a close examination of the trunnions during the re-qualification prior to loading. This item remains open and will be verified by the Staff prior to loading. {Section IV.B.5.b(2)}
- PGE committed to performing "dimensional testing, visual inspection, and nondestructive testing of major load-carrying welds and critical areas" of the Transfer Cask and Lift Yoke prior to loading. The test will include all critical areas and welds, including those that may not have been tested at UST&D. This will include, at a minimum, a visual inspection for gouges, cracks, deformation or other indications of damage. This item remains open and will be verified by the Staff prior to loading. {Sections IV.B.5.b(3), IV.B.6.b, and V.E.2}
- PGE committed to analyzing the extent to which the Gap Flush connection on the Transfer Cask and other potential exposures of uncoated carbon steel will be a source of rust and hydrogen during the loading operation. This item remains open and will be verified by the Staff prior to loading. {Section IV.B.5.b(4)}
- The program for the storage of all ITS components, including the measures that will be taken to ensure that uncoated carbon steel will be protected item remains an open item and will be verified by the Staff prior to loading. (Section IV.B.9.d)
- PGE committed to developing a procedure that will include criteria for evaluating the actions required to decontaminate an MPC. The Staff will review that procedure to confirm that it meets the acceptance criteria of OAR 345-26-390(4)(f). (Section IV.D.6.a)
- The Staff will review the testing of electrical interlocks on the Fuel Building Crane prior to movement of the fuel-loaded Transfer Cask. The Staff 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 a load, the steps and proper sequence to be followed in handling the load, and defining the safe load path. (Section V.E.1.a)
- The Staff will witness the first fuel loading and all of the movements of loads. (Section V.E.1.a)
- The Staff will monitor the effectiveness of the administrative measures to control the height limit of 12 inches or less above the floor on lifting the Transfer Cask during fuel loading.

These measures should include flagging or tape or equivalent attached to the equipment plus personnel on or near the floor to verify clearance from obstructions and the height limit. (Section V.E.1.b)

- The Staff will 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.E.1.c)
- The Staff 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. The Staff will also review exposure records over the course of the loading operations. (Section V.G.1)
- The Staff 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)
- The Staff will confirm prior to loading that procedures contain acceptance criteria for centering a canister in a Concrete Cask initially, provisions for determining when a visual examination is required and the circumstances that would require re-centering, and corrective actions to re-center a canister. (Section V.I.4)
- The Staff will review the procedure FHP 50-08 on the helium recirculation cooling prior to loading operations. (Section VI.B)

Prior to loading, the Staff will review PGE’s analysis that helium recirculation cooling is not necessary, a method of demonstrating the helium recirculation cooling process, or the ability to cool the exterior of the MPC and verify that the helium temperature is low enough by taking a bulk measurement. (Section VI.B)

- The Staff plans to witness selected pre-operational tests.

APPENDIX A — 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 10CFR20.1003 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."[80]

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 an MPC. In addition to storage, the cask provides radiation shielding, protects MPCs from postulated environmental events, such as earthquakes and tornadoes, and provides passive heat transfer from an MPC to the environment.

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

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.

dose: The total amount of radiation received by an individual. Dose is normally expressed in terms of rem or millirem.

dose rate: Radiation dose delivered per unit of time (rem/hour or mrem/hour).

DSAR: Defueled Safety Analysis Report.

EPA PAG: Environmental Protection Agency Protective Action Guide.

Failed Fuel Can or Damaged Fuel Container: 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 canister. The Failed Fuel Can or Damaged Fuel Container allow for water drainage and vacuum drying during MPC closure.

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, placed a Failed Fuel Can, and stored in one of the four oversized corner storage locations of an MPC. 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 placed in a Failed Fuel Can and stored in one of the four oversized corner storage locations of an MPC. 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. There will be no GTCC waste stored at the Trojan ISFSI.

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.

HI (Holtec International): The manufacturer of the Transfer Cask and MPCs to be used in the Trojan ISFSI.

important-to-safety: 10CFR72.3 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.

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 (or other license holder) to the NRC to request a change in an existing license.

Lift 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. It will also be used with a mobile crane on the ISFSI Storage Pad to lift the empty Transfer Cask to the Transfer Station platform.

mrem: Millirem, one thousandth (1/1000) of a Rem. A Rem is a measure of radiation based on the amount of biological damage it will do to the human body.

MPC: Holtec International's multipurpose, stainless steel canister, which for Trojan will store up to 24 spent nuclear fuel assemblies in a honeycombed fuel basket. The MPC is the confinement boundary for storage conditions, and is placed inside a ventilated Concrete 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.

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.

PWR Basket: The BNFL/SNC designed stainless steel canister for storing up to 24 spent nuclear fuel assemblies inside a ventilated Concrete Cask. These were originally planned for use at Trojan, but they have been replaced by Holtec's multipurpose canisters (MPC).

Quality Assurance (QA): 10CFR50, 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 which 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 measure of radiation based on the amount of biological damage it will do to the human body. A mrem is one thousandth (1/1000) 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 10CFR72.

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 10CFR72.

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 (BFS) 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. BNFL/SNC is the designer of the Concrete Cask to be used at Trojan . 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."[81]

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 (BNFL/SNC) ISFSI cask system for storage and transportation of spent nuclear fuel. The only component designed by BNFL/SNC that will be used for the Trojan ISFSI is the Concrete Cask.

TNP: Trojan Nuclear Plant.

Transfer Cask: A steel vessel used to lift and transport an MPC in the Fuel Building and to support an MPC at the Transfer Station. The Transfer Cask also provides radiation shielding.

trunnions: The load-bearing components of the Transfer Cask that mate with the Lift Yoke and provide a means for the Fuel Building Crane to lift the Transfer Cask.

Transfer Station: A structure designed for transferring an MPC 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 MPC transfer operations.

VSC-17 and VSC-24: Ventilated storage cask systems manufactured by Sierra Nuclear Corporation (BNFL/SNC). The only component designed by BNFL/SNC that will be used for the Trojan ISFSI is the Concrete Cask.

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

10CFR50: 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.

10CFR72 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.

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[81] ibid.

APPENDIX B — REFERENCES

The following is a partial list of the documents reviewed, referenced, or used as a basis for OOE's review of the Holtec Design for PGE's ISFSI. This list contains the major documents that were reviewed extensively or quoted in this report. It does not contain all of the PGE and vendor audit and surveillance reports, calculational packages and procedures, or numerous other documents that were reviewed for this report. See also the documents referenced in the January 27, 1999 Staff evaluation.

ANSI N14.6-1978, Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More

ANSI N14.6-1993, Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More

ASME, Boiler and Pressure Vessel Code, Section II, Materials, July 1, 1992

ASME, Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, July 1, 1992

Federal Register, Volume 61, No. 81

Holtec International, DS-213, Rev.2, Acceptable Flaw Size in MPC Lid-to-Shell Welds, February 23, 1999

Holtec International, DS-252, Rev.1, Resistance of Holtec Cask Systems to Corrosion in Environmental Conditions, March 26, 2001

Holtec International, DS-256, Rev.3, On the Leaktightness of HI-STAR/HI-STORM MPCs, July 2001

Holtec International, Final Safety Analysis Report for the HI-STORM 100 System, Rev.0

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Holtec International, HI-STORM LAR 1014-1, Rev.2, October 2001

Holtec International, License Amendment Request, LAR 1014-1, Rev.2, July 2001, October 2001

LLNL, Chun, R., et al., , Dynamic Impact Effects on Spent Fuel Assemblies, October 1987

Memorandum of Understanding between the NRC and the State of Oregon, January 1980

NRC, Code of Federal Regulations, Title 10, Part 72

NRC, NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, July 1980

NRC, NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, February 1996

NRC, Part 72 Statements of Consideration, September 29, 1995

ODOE, Review of PGE's Decommissioning Plan for the Trojan Nuclear Plant, January 1996

OOE, A.Bless to L.Dusek, Event Review Team, December 1999

OOE, OAR 345-26-370 and OAR 345-26-390

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

PGE, CAR 01-0012, Postulated Drop of a Transfer Cask, November 2001

PGE, CAR 99-0034, Rev.1, PWR Basket Coatings Failure, July 2000

PGE, CPY-001-97, C.P.Yundt to D.Stewart-Smith, Response to RAI, February 1997

PGE, CPY-007-97, C.P.Yundt to D.Stewart-Smith, Revisions to Response to RAI #2, March 1997

PGE, CPY-032-96, C.P.Yundt to D.Stewart-Smith, Response to RAI, June 1996

PGE, CPY-035-96, C.P.Yundt to D.Stewart-Smith, Response to RAI, September 1996

PGE, CPY-093-96, C.P.Yundt to D.Stewart-Smith, Response to RAI Regarding the Final Survey Plan for the Trojan ISFSI Site, November 1996

PGE, LCA 72-02, Trojan ISFSI Proposed Revisions 2 through 2b to the License, Technical Specifications, and Bases

PGE, LCA 72-02, Trojan ISFSI Proposed Revisions 2 through 2c to the Safety Analysis Report (SAR)

PGE, Proposed Revision 5 and 5a to License Change Application 237

PGE, Proposed Revision 2 and 2a to License Change Application 246

PGE, VPN-001-97, Trojan ISFSI (TAC No. L22102) Response to NRC RAI, January 9, 1997

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

PGE, VPN-033-2001, Enclosure I, Proposed Revision 5 to License Change Application 237

PGE, VPN-033-2001, Enclosure III, Proposed Revision 2 to License Change Application 246

PGE, VPN-033-2002, Enclosure II, Proposed Revision 2a to the Trojan ISFSI Technical Specifications and Bases

PGE, VPN-033-2002, Enclosure III, Proposed Revision 2a to the Trojan ISFSI Safety Analysis Report (SAR)

PGE, VPN-035-2002, PGE Responses to OOE RAI, June 6, 2002

PGE, VPN-074-98, S.M.Quennoz to D.Stewart-Smith, December 3, 1998

PGE-8010, Nuclear Quality Assurance Program for Trojan Nuclear Plant, Revs.20 and 22

Santos, C., et al., Estimated Risk Contribution for Dry Spent Fuel Storage Cask

USGS, Open-File Report 95-497, Volcanic-Hazard Zonation for Mount St. Helens, Washington, E.W.Wolfe and T.C.Pierson, 1995