ENERGY NORTHWEST

P.O. Box 968 ■ Richland, Washington 99352-0968

October 30, 2000
GO2-00-182

Docket No. 50-397

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

Subject: **WNP-2 OPERATING LICENSE NPF-21**
REQUEST FOR AMENDMENT
SPENT FUEL STORAGE AND HANDLING

In accordance with the Code of Federal Regulations, Title 10, Parts 2.101, 50.59, and 50.90, Energy Northwest hereby submits a request for amendment for the WNP-2 Operating License. Specifically, Energy Northwest is requesting modification of the spent fuel storage and spent fuel cask handling descriptions as originally evaluated in the WNP-2 Final Safety Analysis Report (FSAR).

Energy Northwest has evaluated the requested FSAR text changes in accordance with 10 CFR 50.59 and determined an unreviewed safety question (USQ) exists. Specifically, there are significant physical differences between the General Electric cask (GE IF-300) originally analyzed in the FSAR and the HI-STORM 100 spent fuel cask system that cause unreviewed changes in procedures and processes previously described and evaluated in the FSAR. Also, the physical description of the WNP-2 spent fuel pool as discussed in the FSAR, does not accurately reflect the configuration as it exists today.

It is the objective of this request to present for your review and concurrence our safety assessment of the proposed FSAR modifications in accordance with 10 CFR 50.59(c) and 10 CFR 50.90. We request your concurrence with our position that this modification is acceptable and preserves the health and safety of the public.

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Additional information has been attached to this letter to complete Energy Northwest's amendment request. Attachment 1 provides a detailed evaluation of the proposed change. Attachment 2 provides marked up pages of the FSAR. Attachment 3 provides an evaluation of the proposed change in accordance with 10 CFR 50.92(c) and concludes that it does not result in a significant hazards consideration. Attachment 4 provides the environmental assessment applicability review and notes that the proposed change meets the eligibility criteria for a categorical exclusion as set forth in 10 CFR 51.22(c)(9). Therefore, in accordance with 10 CFR 51.22(b), an environmental assessment of the change is not required.

This request for an amendment has been reviewed and approved by the WNP-2 Plant Operations Committee and the Energy Northwest Corporate Nuclear Safety Review Board. In accordance with 10 CFR 50.91, the State of Washington has been provided a copy of this letter.

Should you have any questions or desire additional information, please contact PJ Inserra at (509) 377-4147.

Respectfully,

DK Atkinson
(Acting) Vice President, Operations Support/PIO
Mail Drop PE08

Attachments:

1. Evaluation of the Proposed Change
2. Revised Safety Analysis Report
3. Evaluation of Significant Hazards Considerations
4. Environmental Assessment Applicability Review

cc: EW Merschoff - NRC RIV
JS Cushing - NRC NRR
NRC Sr. Resident Inspector - 988C
DL Williams - BPA/1399
TC Poindexter - Winston & Strawn
DJ Ross - EFSEC

STATE OF WASHINGTON)
)
COUNTY OF BENTON)

Subject: Request for Amendment Spent Fuel Storage and Handling

I, D.K. ATKINSON, being duly sworn, subscribe to and say that I am the (Acting) Vice President, Operations Support/PIO for ENERGY NORTHWEST, the applicant herein; that I have the full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information, and belief the statements made in it are true.

DATE 10-30-00, 2000

D.K. Atkinson

D. K. Atkinson
(Acting)Vice President, Operations Support/PIO

On this date personally appeared before me D. K. ATKINSON, to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 30 day of October 2000.

Lori A. Mix

Notary Public in and for the
STATE OF WASHINGTON

Residing at N. Richland

My Commission Expires 3-29-01

Evaluation of the Proposed Change

Introduction

During readiness activities for dry cask spent fuel storage, preparation of a safety evaluation was begun to meet the requirements of 10 CFR 72.212(b)(4). 10 CFR 72.212(b)(4) states in part that: "Prior to use of the general license, determine whether activities related to storage of spent fuel under this general license involve any unreviewed facility safety question or change in the facility technical specifications, as provided under Part 50.59." This safety evaluation found that spent fuel cask handling activities as described in the Final Safety Analysis Report (FSAR) do not reflect actual plant design. This evaluation also found that the WNP-2 SER, NUREG-0892 "Safety Evaluation Report related to the operation of WPPSS Nuclear Project No. 2," did not evaluate the actual facility configuration or specific processes that will be used in support of spent fuel cask handling.

The original FSAR description of spent fuel departure from the spent fuel pool was based on using the GE IF-300 cask. New regulations and requirements for interim storage have changed the spent fuel handling process. At present, we intend to use the 10 CFR 72.214 approved Holtec HI-STORM 100 spent fuel cask system. (The GE IF-300 cask is not included in the list of approved fuel storage casks under 10 CFR 72.214.) There are significant physical differences between the cask originally analyzed in the FSAR and the HI-STORM 100 spent fuel cask system that cause changes in processes previously evaluated in the FSAR and in the SER. Also, the physical description of the facility, as discussed in the FSAR and the SER, does not accurately reflect the configuration as it exists today.

Changes are required to update the FSAR process descriptions of cask loading and handling. Changes are also required to update spent fuel pool features of construction in order to reflect the actual plant configuration.

Summary of the Proposed Changes

The following summarizes unreviewed issues associated with the proposed FSAR text revisions:

Issue 1 - There is only one (1) spent fuel pool, not two (2) separate pools, for spent fuel handling.

The current licensing basis assumes that there are two (2) separate fuel pools: the spent fuel cask loading pool and the spent fuel pool.

This incorrect design concept is perhaps most clearly illustrated by the following statements: "The spent fuel cask storage and loading pool is adjacent to the spent fuel pool, is separated from the fuel pool by 2-foot-thick reinforced concrete walls, and is isolated from the pool by a seismic Category I gate." (Reference: SER Section 9.1.5, 'Overhead Heavy Load Handling Systems,' March 1982);

and, "A separate spent fuel cask loading area, isolated from the fuel storage pool eliminates the potential accident of dropping the cask and rupturing the fuel storage pool." (Reference: WNP-2 FSAR Section 9.1.4.1, Fuel Handling System - Design Basis).

The spent fuel cask loading pit is separated from the spent fuel racks by an approximately 2-foot-thick reinforced concrete wall extending from the bottom of the spent fuel pool to slightly below the top of the spent fuel racks. However, both the spent fuel cask loading pit and the spent fuel racks reside within the common spent fuel pool.

The proposed FSAR change for Issue 1 would eliminate references to separate pools and correct the acceptability statement to read, "Sufficient redundancy is provided in the reactor building crane such that no credible postulated failure of any crane component will result in dropping of the fuel cask and rupturing the fuel storage pool." which is in agreement with the FSAR Chapter 15, 'Accident Analyses' (Reference: WNP-2 FSAR Section 15.7.5, Spent Fuel Cask Drop Accident).

<u>Issue 2 - Contrary to existing FSAR statements, limitations on reactor building crane travel do not preclude transporting the spent fuel cask over the spent fuel pool.</u>

Several sections in the FSAR and the SER presently state that interlocks on the reactor building crane prevent travel over the spent fuel pool. There are no interlocks that inhibit crane movement over the spent fuel cask loading pit area, which is part of the spent fuel pool. The restricted area interlocks prevent crane travel over the spent fuel in the spent fuel racks.

The proposed FSAR change for Issue 2 would read, "Interlocks on the reactor building crane prevent travel over the spent fuel racks."

<u>Issue 3 - Spent fuel cask movement over safety-related equipment.</u>

The FSAR and the SER state that at no time while being transported does the fuel cask pass over any safety-related equipment. However, a safety-related conduit associated with Fuel Pool Cooling level element (FPC-LE-5) is included in the cask load path.

The proposed FSAR change for Issue 3 would read, "At no time while being transported does the cask pass over any safe shutdown equipment."

<u>Issue 4 - The FSAR analyzed using the GE IF-300 spent fuel cask rather than the Holtec HI-STORM 100 spent fuel cask system.</u>

The FSAR and the SER discuss cask loading, handling, and features of construction that do not reflect the selected spent fuel cask system.

The proposed FSAR changes for Issue 4 would accurately describe the spent fuel cask loading and handling using the NRC approved Holtec HI-STORM 100 spent fuel cask system (References: #11 below, and NRC Certificate of Compliance #1014, and Federal Register, May 1, 2000, Page 25241 - 25265)

Change evaluation

Issue 1 - There is only one (1) spent fuel pool, not two (2) separate pools, for spent fuel handling.

Separate pools are not required to eliminate the potential accident of rupturing the fuel storage pool and/or damaging the spent fuel stored in the pool from a cask drop accident because the single-failure proof reactor building crane renders the cask drop accident incredible. As a defense-in-depth measure, crane position interlocks prevent lifting a spent fuel cask over the spent fuel stored in the pool. Additionally, there is an approximately 2-foot-thick reinforced concrete wall that separates the spent fuel cask loading pit from the spent fuel racks providing significant collision protection from any potential lateral movement (swing) of the cask that might impact the spent fuel racks.

The proposed change brings the spent fuel pool description into agreement with the actual plant configuration and the FSAR Chapter 15 accident analysis for a spent fuel cask drop. The WNP-2 reactor building crane is single-failure-proof (Reference: #5 below) and therefore no credible postulated failure of any crane component will result in dropping of the fuel cask and rupturing the fuel storage pool (References #9 & #10 below).

Conclusion: There is no safety significance to this proposed FSAR change.

Issue 2 - Contrary to existing FSAR statements, limitations on reactor building crane travel do not preclude transporting the spent fuel cask over the spent fuel pool.

There are no interlocks to prevent crane travel over the fuel pool - the interlocks prevent travel over the spent fuel racks in the spent fuel pool. For example, there are no interlocks that inhibit crane movement over the spent fuel cask loading pit area, which is part of the spent fuel pool. FSAR Figure 9.1-5 shows the correct relationship between the interlocks (restricted area) and the spent fuel pool.

The proposed change to the FSAR correctly describes the restricted area interlocks which prevent crane travel over the spent fuel racks, not the entire pool area.

Conclusion: There is no safety significance to this proposed FSAR change.

Issue 3 - Spent fuel cask movement over safety-related equipment.

As the cask is moved in and out of the fuel pool, it passes over several cables and conduits supporting plant equipment. They include nonsafety-related cables such as those supplying the refueling bridge, and spent fuel pool temperature indicator FPC-TE-7. Additionally, a safety-related conduit for FPC-LE-5 is included in the cask load path. While a cask drop, which could damage or cut the cable to FPC-LE-5 is not credible, operator error in which the cable is damaged by the cask not clearing the conduit during cask movement may be credible. If the cable were damaged, it might inhibit one train of the automatic isolation signal for the fuel pool cooling system. The automatic isolation of interest occurs on low fuel pool water level,

isolating the Seismic Category I cooling portion of the system from the Seismic Category II cleanup portion of the system. If a crane operator error (single active failure) should damage the cable for FPC-LE-5, automatic isolation on low water level would still occur because a separate, redundant, logic train (from FPC-LE-4) would not be affected and would still be capable of accomplishing the isolation function described in FSAR Section 9.1.3.2.3. The cable for the redundant logic train is not in the cask load path. The cable for FPC-LE-5 also carries a nonsafety-related signal for high/low spent fuel pool water level alarm, which has a redundant analogue signal (undamaged in this scenario) from FPC-LS-4.

To agree with the current plant configuration noted above, the FSAR needs to be changed to read, "At no time while being transported does the cask pass over any safe shutdown equipment." The objectives referenced in RG 1.13, Rev. 1, and the guidelines of NUREG-0612 (to prevent impact by heavy loads with safe shutdown equipment) will continue to be met.

Conclusion: There is no safety significance to this proposed FSAR change.

Issue 4 - The FSAR analyzed using the GE IF-300 spent fuel cask rather than the Holtec HI-STORM 100 spent fuel cask system.

The FSAR description of spent fuel departure from the spent fuel pool was based on using the GE IF-300 cask. New regulations and requirements for interim storage have changed the spent fuel handling process. At present, we intend to use the 10 CFR 72.214 approved Holtec HI-STORM 100 spent fuel cask system. The GE IF-300 cask is not included in the list of approved fuel storage casks under 10 CFR 72.214. There are significant physical differences between the cask originally analyzed in the FSAR and the HI-STORM 100 spent fuel cask system that cause changes in processes previously described and evaluated in the FSAR.

The proposed FSAR change would accurately reflect use of the Holtec HI-STORM 100 spent fuel cask system. Plant procedures will be developed to handle the NRC certified cask in accordance with Technical Specification and license requirements.

Conclusion: There is no safety significance to this proposed FSAR change.

Summary

There are no plant modifications required as a result of this proposed FSAR change. The proposed FSAR text changes correct inaccuracies partly resulting from incorrect original process descriptions. Since that time, there have been significant changes to spent fuel cask handling and design requirements including the necessity for extended dry storage of spent fuel in independent spent fuel storage installations in accordance with 10 CFR Part 72. With the proposed FSAR text changes incorporated, the FSAR will accurately describe actual plant configuration and processes related to spent fuel cask handling and use of the NRC certified Holtec HI-STORM 100 System.

The proposed FSAR change contains information about WNP-2 spent fuel cask handling that has not been previously reviewed and approved by the NRC; however, there is no safety significance to this FSAR change. The FSAR text corrections are in agreement with applicable regulations and no physical alteration to the plant configuration is required.

References:

1) Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," Revision 1, December 1975.

2) NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," May 1979.

3) NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity A-36," July 1980.

4) Unnumbered Generic Letter, "Control of Heavy Loads," December 22, 1980.

5) Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' NUREG-0612," June 28, 1985.

6) NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," April 11, 1996.

7) Supply System Letter, GO2-96-100, "WNP-2, Operating License NPF-21, Response to Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," May 10, 1996.

8) Supply System Letter, GO2-96-141, "WNP-2, Operating License NPF-21, Response to Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment - Additional Information," July 19, 1996.

9) Supply System Letter, GO2-97-064, "WNP-2, Operating License NPF-21, Response to NRC Request for Additional Information Related to Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," April 3, 1997.

10) Supply System Letter, GO2-97-134, "WNP-2, Operating License NPF-21, Additional Information Related to Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," June 30, 1997.

11) Topical Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (NRC Docket No. 72-1014).

Revised Safety Analysis Report

This request for amendment affects the following FSAR sections:

1.8.3
3.8.4.1.1.6
9.1.2.3.3
9.1.4.1
9.1.4.2.1
9.1.4.2.2.1
9.1.4.2.7.1
9.1.4.2.10.2
9.1.4.2.10.3
9.1.4.3
15.7.5
15.7.6

Marked up copies of the FSAR pages affected by this request for amendment are attached as listed below:

1.8-96
3.8-86
9.1-14
9.1-15
9.1-27
9.1-28
9.1-29
9.1-37
9.1-41
9.1-45
9.1-46
9.1-47
Figure 9.1-16
15.7-9

Regulatory Guide 1.13, Revision 1, December 1975

Spent Fuel Storage Facility Design Basis

Compliance or Alternate Approach Statement:

WNP-2 complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

A controlled leakage building is provided enclosing the fuel pool. The building is not designed to withstand extremely high winds, but leakage is suitably controlled during refueling operations. The building is equipped with a ventilation and filtration system which is designed to limit the potential consequences of the release of radioactivity specified in Regulatory Guide 1.25 to those guidelines set forth in 10 CFR 100.

The movement paths of heavy objects such as the reactor pressure vessel head, containment vessel head, and the spent fuel cask are designed not to pass over the spent fuel racks ~~pool~~. Furthermore, the reactor building crane and its auxiliary hoist are prevented by means of interlocks from passing over any of the spent fuel pool except the spent fuel cask area. Bypassing of the interlocks is permitted only during fuel handling and storage operations and is administratively controlled.

The fuel pool is designed so that no pipe break will drain water from the fuel pool.

Specific Evaluation Reference:

See Section 9.1.

A floor connection is provided in the dryer-separator pool that allows the pool to be drained to a level 5 ft 0 in. above the pool floor. The 5 ft 0 in. depth of water is constantly maintained in the pool by means of the 5 ft 0 in. high permanent bottom plug monolithically constructed as part of the pool.

The walls and bottom slab thicknesses of the pools are adequate to provide the required biological shielding.

The spent fuel ~~shipping~~ cask storage area is separated from the new and spent fuel storage area by concrete walls within the fuel pool.

3.8.4.1.2 Radwaste and Control Building

The radwaste and control building is located west of the reactor building and diesel generator building and south of the turbine generator building. See Figures 1.2-13 through 1.2-17, and 3.8-42 for general arrangements and principal features of the radwaste and control building.

The radwaste and control building houses equipment and components for both safety-related and non-safety-related systems.

The radwaste and control building is functionally designed and arranged to provide structural facilities to accommodate safety-related and non-safety-related equipment and components housed therein.

The radwaste and control building is rectangular in shape. It is approximately 209 ft by 160 ft in plan, and 105 ft in overall height measured from the top of the foundation mat at el. 437 ft 0 in.

Part of the building is a cast-in-place reinforced-concrete structure and part is a structural steel-framed structure. The portions of the building that are reinforced-concrete are arranged so as to form a completely enclosed boxlike structure, and are, therefore, comprised of concrete walls on four exterior sides and of top and bottom concrete slabs. The portions of the building that are structural steel framed are made up of steel-framed floors and exterior walls.

The concrete portion of the building is a diaphragm slab/shear wall type structure, supported on a foundation mat which is separated and apart from the mats under each of the other structures in the plant complex. The floor and roof slabs are continuous one-way and two-way slab and beam systems monolithically constructed.

The steel-framed portions of the building consist of conventionally braced framed wall systems with vertical bracing in bays along each wall. The girts and roof purlins are covered with insulated metal siding and insulated metal roof decking, respectively. The steel-framed floors

The high density fuel rack designer, NUS Corporation, analyzed the fuel racks from both a structural and criticality standpoint concerning a 1510-lb object dropped from the surface of the fuel pool. The results indicated that none of the fuel rack damage that might occur in this situation would lead to a criticality problem. Details of these analyses are given in NUS Corporation Technical Reports 5326-FA-01 and G-RA-17 entitled, "Structural Analysis of the WNP-2 Rack and Fuel Assemblies for an Accidental Object Drop Loading Condition," and "Criticality Analysis of Dropped Object Accident for WNP-2 Spent Fuel Storage Racks," respectively.

During shutdown, crane operations over the spent fuel storage pool (when fuel assemblies are stored within) are suspended when operability of less than the required number of onsite and offsite power sources occurs, as defined in the Technical Specifications.

9.1.2.3.3 Spent Fuel and Cask Handling

The 125-ton reactor building crane traverses the full length of the refueling floor level of the reactor building. The design of the refueling floor provides aisles on both sides of the fuel pool for moving components past (and not over) the fuel storage pool. Interlocks on the reactor building crane prevent travel over the spent fuel pool. The interlock-controlled restricted area for crane travel is shown in Figure 9.1-5. The interlocks are bypassed only when it is necessary to operate the crane in the fuel pool area in conjunction with activities associated with fuel handling and storage. During these rare occasions when the interlocks are bypassed, administrative controls are used to prevent the crane from carrying loads that are not necessary for fuel handling or storage and which are in excess of the rack design drop load (one fuel assembly at 4 ft above the top of the fuel rack). See Section 9.1.2.3.2.

Transfer of fuel assemblies between the reactor well and the spent fuel pool is performed with the refueling platform (see Section 9.1.4.2.10.2). The fuel grapple or the auxiliary fuel hoist may be used, depending on the transfer operation.

The grapple and hoist are provided with load sensing and limiting devices designed to the following limits:

	Fuel Grapple (lb)	Auxiliary Fuel Hoist (lb)
Load limiting switch NF400	1250	1000
Load limiting switch NF500	1700	
Load sensing switch NF400	535	485
Load sensing switch NF500	750	

Stall torque for hoist system	3000	3000

The load limiting features of the refueling platform grapple and auxiliary fuel hoist will prevent damage to the fuel racks if a fuel assembly accidentally engages a rack while being lifted. These load limits provide a redundant safety feature since the fuel handling grapple is not lowered below the upper fuel rack and is designed to interface only with the fuel bail. Thus, the possibility of inadvertent direct lifting of the racks with the grapple is precluded.

Guard rails around the spent fuel pool prevent the falling of fuel handling area machinery into the pool. Other objects that could conceivably fall into the pool will not transfer energy amounts exceeding the specified limits of the fuel racks.

The preclusion of accidental dropping of the spent fuel cask on the spent fuel racks is accomplished by incorporating a separate cask storage area in the spent fuel pool, by interlocks on the reactor building crane, and by designing the path of the spent fuel cask to avoid passing over the spent fuel ~~pool~~ racks. The pool cask storage area is separated from the spent fuel ~~pool~~ racks by a wall over which the spent fuel is transferred.

For removal of spent fuel from the plant, a ~~shipping~~ spent fuel cask is lowered into the cask area. Transfer of fuel to the cask is made over the wall between the spent fuel ~~pool~~ racks and the cask storage area. When the main hook of the reactor building crane removes the cask from the cask area, it follows the travel path shown in Figure 9.1-5. In addition, sufficient redundancy is provided in the reactor building crane such that no credible postulated failure of any crane component will result in the dropping of the fuel cask. (See Section 9.1.4.2.2.)

Failure of the gates between the reactor well and the spent fuel storage pool is improbable. However, in the event of this failure, the loss of water from the storage pool into the reactor well would not uncover the stored spent fuel due to the elevation of the weir wall under the gates. This elevation ensures that sufficient water is retained in the pool to cover the spent fuel.

To avoid unintentional draining of the spent fuel storage pool to levels below that required for adequate shielding of the spent fuel, no inlets, outlets, or drains that would normally permit the pool to be drained are provided. Discharge lines extending below the pool water level are designed to prevent any siphon back flow. Two skimmer surge tanks are provided and are sized to accommodate water displacement due to large items being placed into or removed from the spent fuel storage pool.

See Section 9.1.3 for additional evaluation of continuous cooling capabilities of the spent fuel pool cooling and cleanup (FPC) system.

Routine, periodic visual inspection of system components, instrumentation, and alarms are adequate to verify system operability. Likewise, the interconnecting valves between the FPC system and the RCC, SW, and RHR systems are periodically inspected to verify their operability.

9.1.4 FUEL HANDLING SYSTEM

9.1.4.1 Design Bases

The fuel handling system is designed to provide a safe and effective means for transporting and handling fuel from the time it reaches the plant until it leaves the plant after postirradiation cooling. Safe handling of fuel includes design considerations for maintaining occupational radiation exposures as low as reasonably achievable during transportation and handling.

Design classification criteria for major fuel handling system equipment is provided in Table 3.2-1 which lists the safety class, quality class, quality group, and seismic category for the equipment.

receiving area

The cask storage area in the spent fuel pool and the cask handling facilities are designed to accommodate a spent fuel cask of 125 tons. The reactor building crane is designed to transport the spent fuel cask between the cask transport vehicle, the cask storage area in the spent fuel pool, and the cask washdown area without inducing a permanent deformation of any crane element. Adequate safety features are designed into the reactor building crane system and controls to ensure that no credible postulated failure of any crane component will result in the dropping of the fuel cask.

The transfer of new fuel assemblies between the uncrating area and the new fuel inspection stand and/or the new fuel storage vault is accomplished using the reactor building crane or jib crane.

The reactor building crane auxiliary hoist or the refueling floor jib cranes are used to transfer new fuel from the new fuel vault to the fuel storage pool. From this point on, the fuel will be handled by the telescoping grapple on the refueling platform.

The refueling platform is General Class G and Seismic Category 1M from a structural standpoint in accordance with 10 CFR Part 50, Appendixes A and B. Allowable stress due to SSE loading is 120% of yield or 70% of ultimate, whichever is less. A dynamic analysis is performed on the structures using the response spectrum method with load contributions resulting from each of three components earthquake motion being combined by the RMS procedure.

Working loads of the platform structures are in accordance with the AISC Manual of Steel Construction. All parts of the hoist systems are designed to have a safety factor of five, based

on the ultimate strength of the material. A redundant load path is incorporated in the fuel hoists so that no single component failure could result in a fuel bundle drop. Maximum deflection limitations are imposed on the main structures to maintain relative stiffness of the platform. Welding of the platform is in accordance with AWS D14-1. Gears and bearings meet AGMA Gear Classification Manual and ANSI B3.5. Materials used in construction of load bearing members are to ASTM specifications. For personnel safety, OSHA Part 1910-179 is applied. Electrical equipment and controls meet ANSI-ANS C1, National Electric Code, and NEMA Publications No. IC1 and MG1.

The general purpose grapples and the main telescoping fuel grapples have redundant hooks. The main telescoping fuel grapple has an indicator which confirms positive grapple engagement.

The fuel grapple is used for lifting and transporting fuel bundles. It is designed as a telescoping grapple that can extend to the proper work level and in its fully retracted state still maintains adequate water shielding over fuel.

In addition to redundant electrical interlocks to preclude the possibility of raising radioactive material out of the water, the cables on the auxiliary hoists incorporate an adjustable, removal stop that will jam the hoist cable against the platform structure to prevent hoisting when the free end of the cable is at a preset distance below water level.

~~A separate spent fuel cask loading area, isolated from the fuel storage pool eliminates the potential accident of dropping the cask and rupturing the fuel storage pool.~~ Furthermore, limitation of the travel of the crane handling the cask will preclude transporting the cask over any fuel storage ~~pool~~ rack. See Chapter 15 for accident considerations.

Sufficient redundancy is provided in the reactor building crane such that no credible postulated failure of any crane component will result in dropping of the fuel cask and rupturing the fuel storage pool.

9.1.4.2 System Description

Table 9.1-7 is a listing of typical tools and servicing equipment supplied with the nuclear system. The following paragraphs describe the use of some of the major tools and servicing equipment and address safety aspects of the design, where applicable.

9.1.4.2.1 Spent Fuel Cask

The designs of cask storage and dry cask handling facilities are based on a design cask weighing approximately 125 tons and being approximately 17 ft long by ~~7.5~~ 8 ft in diameter.

~~The cask transport vehicle will take the spent fuel cask to the cask offloading area located beneath the equipment hatch in the reactor building and will be equipped with a cask coolingsystem and storage area for the cask yoke. The reactor building crane will transfer the cask to and from the refueling floor area.~~

The following description of the spent fuel cask is based on the licensed Holtec HI-STORM 100 system. That system includes a canister that contains the fuel and a transfer cask that contains the canister. The canister confines the fuel. The transfer cask provides shielding and structural protection of the canister during canister loading or movement.

Overland offsite transportation of the cask will conform to transportation rules and regulations, 49 CFR Part 173.

9.1.4.2.2 Reactor Building Crane

9.1.4.2.2.1 Description. The main purpose of the reactor building crane is to handle the spent fuel cask between the cask ~~transport vehicle~~ receiving area, the cask loading area in the spent fuel pool, and the cask decontamination washdown area. Secondary purposes of the crane include servicing and refueling the reactor pressure vessel, and handling of equipment or parts thereof received or shipped through the loading facility in the reactor building.

The reactor building crane is a single-trolley top-running electric overhead traveling crane with a 125-ton capacity main hoist, and a span of 126 ft ~~1-1/8 in~~ approximately. The general arrangement of the crane in the reactor building is shown in Figures 1.2-11 and 1.2-12. The crane is Class A1 as defined for nuclear fuel handling by the Crane Manufacturers Association of America Specification No. 70 for Electric Overhead Traveling Cranes, (CMAA No. 70). The reactor building crane is designed, fabricated, installed, and tested in accordance with ANSI Standard B30.2, Safety Code for Cranes, Derricks and Hoists, and CMAA Specification No. 70. The crane is Seismic Category I.

Operation of the reactor building crane is from the cab or from the floor by radio control. The crane radio control system uses crystals highly tuned to a narrow frequency band, thereby precluding interferences from other signaling systems. Control at any one time is from one point only.

The structure of the crane bridge consists of welded box type girders with truck saddles and truck frames of welded-steel construction. The trolley side frames, sheave frames, and truck frames are of structural steel-welded construction.

The rated full-load capacities, lifts, and full-load speeds are as follows:

Main Hook

Rated full-load capacity, tons	125
Hook travel, ft	190 ft 0 in.
Hoisting speed, fpm at full load	5.5

Auxiliary Hook

Rated full-load capacity, tons	15
Hook travel, ft	199 ft 8 in.
Hoisting speed, fpm at full load	20

Materials are in accordance with ASTM Standards. The structure is coated with an approved paint.

The cranes are exempted from the NUREG 0612 requirement because the definition of heavy loads at WNP-2 is 1200 lb. Each 500 lb hoist is equipped with a load limiting device to prevent lifting fuel bundles or unauthorized heavy loads. Interlocks on the platform or hoists are not provided since the hoists and platform are controlled administratively and will not be used for reactivity manipulation.

9.1.4.2.6 In-Vessel Servicing Equipment

The instrument strongback attached to the reactor building crane auxiliary hoist is used for servicing neutron monitor dry tubes should they require replacement. The strongback initially supports the dry tubes into the vessel. The in-core dry tube is then decoupled from the strongback and is guided into place while being supported by the instrument handling tool. Final in-core insertion is accomplished from below the reactor vessel. The instrument handling tool is attached to the refueling platform monorail hoist and is used for removing and installing fixed in-core dry tubes as well as handling neutrons source holders and the source range monitor/intermediate range monitor dry tubes.

Each in-core instrumentation guide tube seals by metal-to-metal contact with the guide tube flange seal. In the event that the seal needs replacing, an in-core guide tube seal tool is provided. The tool is inserted into an empty guide tube and sits on the beveled guide tube entry in the vessel. When the drain on the water seal cap is opened, hydrostatic pressure seats the tool. The flange can then be removed for seal replacement.

The auxiliary hoist on the refueling platform is used with appropriate grapples to handle control rods, flux monitor dry tubes, sources, and other internals of the reactor. Interlocks on both the grapple hoists and auxiliary hoist are provided for safety purposes; the refueling interlocks are described and evaluated in Section 7.7.1.13.

9.1.4.2.7 Refueling Equipment

Fuel movement and reactor servicing operations are performed from a platform which spans the refueling, servicing, and storage cavities.

9.1.4.2.7.1 Refueling Platform. The refueling platform is a gantry crane that is used to transport fuel and reactor components to and from pool storage and the reactor vessel. The platform spans the fuel storage and vessel pools on rails bedded in the refueling floor. A telescoping mast and grapple suspended from a trolley system is used to transport and orient fuel bundles for core, storage rack, or ~~shipping~~ spent fuel cask placement. Control of the platform is from an operator station on the main trolley with a position indicating system provided to position the grapple over core locations. The platform control system includes interlocks to

The new fuel inspection stand holds one or two bundles in vertical position. The inspector(s) ride up and down on a platform, and the bundles are manually rotated on their axes. Thus the inspectors can see all visible surfaces on the bundles.

The refueling platform uses a grapple on the telescoping mast for lifting and transporting fuel bundles or assemblies. The telescoping mast can extend to the proper work level, and, in its normal up position, maintains adequate water shielding over the fuel being handled.

The reactor refueling procedure is shown schematically in Figure 9.1-15. The refueling platform moves over the fuel pool, lowers the grapple on the telescoping mast (H), and engages the bail on a new fuel assembly which is in the fuel storage rack. The assembly is lifted clear of the rack and moved through slot (A) and over the appropriate empty fuel location in the core (J). The mast then lowers the assembly into the location, and the grapple releases the bail.

The operator then moves the platform until the grapple is over a spent fuel assembly which is to be discharged from the core. The assembly is grappled, lifted, and moved through slot (A) to the fuel pool.

To preclude the possibility of raising radioactive material out of the water, redundant electrical limit switches are incorporated in the hoist and interlocked to prevent hoisting above the preset limit. In addition, the cables on the auxiliary hoists incorporate adjustable stops that will jam the hoist cable against the hoist structure, which prevents hoisting if the limit switch interlock system should fail. Prior to moving irradiated fuel in the reactor pressure vessel, the reactor is verified to have been subcritical for at least 24 hr. This is consistent with fuel handling accident assumptions described in Section 15.7.4.5.

When spent fuel is to be ~~shipped~~ transferred, it is placed in a cask, as shown in Figure 9.1-16. The refueling platform grapples a fuel bundle from the storage rack in the fuel pools, lifts it, carries it to the ~~shipping~~ transfer cask area of the pool, and lowers it into the ~~cask~~ canister within (M). When the ~~cask~~ canister is loaded, the building crane sets the ~~cask~~ canister cover on the ~~cask~~ canister, and then lifts the cask out of the pool. The cask is then decontaminated and lowered through the open hatchways (P) to the ~~truck or railcar~~ cask transport system at near grade level.

9.1.4.2.10.2.1 New Fuel Preparation.

9.1.4.2.10.2.1.1 Receipt and Inspection of New Fuel. The incoming new fuel shipping container will be visually inspected for shipping damage as the containers are offloaded from the transport vehicle. The fuel is also visually inspected upon opening the container on the refuel floor and then closely examined for debris and or defects and for tolerances against standards in the inspection stand.

9.1.4.2.10.2.1.2 Channeling New Fuel (Non ABB Fuel). New fuel is normally channeled with new channels after the fuel is inspected in the inspection stand. The new channels are visually

b. Control rod drive tests. The control rod drive timing, friction and scram tests are performed;

c. Install inner fuel pool gate;

d. Replace separator and drain dryer-separator storage pool and reactor well;

e. Bolt separator and remove the four steam line plugs;

f. Replace steam dryer;

g. Install reactor vessel head;

h. Decontaminate reactor well;

i. Open drywell vents, install vent piping;

j. Replace slot plugs;

k. Replace vessel studs;

l. Install vessel head piping and insulation;

m. Replace dryer-separator canal plugs;

n. Hydrotest vessel, if necessary;

o. Install drywell head;

p. Install reactor well shield plugs; and

q. Startup tests. The reactor is returned to full power operation. Power is increased gradually in a series of steps until the reactor is operating at rated power. At specific steps during the approach to power, the in-core flux monitors are calibrated.

9.1.4.2.10.3 Departure of Spent Fuel From ~~Site~~ the Reactor Building. Spent fuel is ~~shipped~~ removed from the ~~plant~~ Reactor Building in an NRC licensed shielded cask. The reactor building and cask handling facilities (see Section 9.1.4.2.2 for a description of the crane) are designed to handle ~~the casks the size of the GE IF-300 and larger casks of the future.~~ casks up to 125 tons.

Attachment 2

Holtec HI-STORM 100 System which includes a transfer cask of 125 tons maximum weight with a capacity of 68 BWR fuel bundles.

The following description of the spent fuel departure is based on the licensed GE IF-300 cask of 80 tons loaded weight with capacity for 18 BWR irradiated fuel bundles:

The empty cask will arrive at the plant on a special rail car. Health physics personnel will check the cask exterior to determine if decontamination is necessary. Decontamination, if required, and washdown to remove road dirt would be performed before removal of the cask from the transport vehicle. Administrative controls are provided to maintain secondary containment isolation when the cask transport vehicle is moved into the building.

After the cask has been inspected for damage and prepared for lifting, it is transferred to the reactor operating floor and placed onto the cask decontamination pad where it is prepared for placement into the spent fuel cask loading area, which is adjacent to but separated from the spent fuel pool by a weir, the top of which is at a higher elevation than the top of the fuel stored in the spent fuel pool. The cask provides shielding for a canister contained within the cask. The canister, when sealed, is a confinement for the fuel. The cask is next raised and transferred into the spent fuel cask loading area and placed on the pad provided. The cask lifting yoke is lowered until it is disengaged from the cask trunnions. The closure head is lifted off the cask and is moved onto the cask decontamination pad for storage.

Spent fuel is transferred under water from storage in the fuel pool to the cask using the telescoping fuel grapple mounted on the refueling platform. When the cask is filled with spent fuel, the closure head is replaced on the cask and the lift yoke engaged with the cask trunnions. The loaded cask is raised and transferred to the cask decontamination pad. The canister is sealed, dewatered and inerted. The cask lid is installed and the cask prepared for lifting. The cask is checked by health physics personnel and decontamination is performed with high pressure water sprays, chemicals, and hand scrubbing, as required, to clean the cask to the level required for transport. Smear tests are performed to verify cleaning to offsite transportation requirements. The remaining cask closure nuts are replaced and tightened.

canister

The cleaned cask is replaced on the cask skid mountings, the cask cooling system of the transport vehicle is connected to the cask, and the personnel barrier is replaced. The cask internal pressure and temperature are monitored, and when they are at equilibrium conditions, the cask is ready for shipment to a fuel processing plant. The sealed canister is then transferred to a storage cask. The storage cask and canister are moved to the spent fuel storage area.

9.1.4.3 Fuel Handling Safety Evaluation

Stresses in all structural and mechanical parts of the reactor building crane system are far below the endurance limits for infinite life of the various materials for both the rated crane capacity and the test load of 125% of rated load. All loads to be handled are below rated crane capacity. Therefore, stresses should never reach allowable working stresses. Loads on the structural parts vary but do not reverse. The only critical parts with stress reversals are the rotating parts, and these are provided with single failure protection. Since the crane is to operate under normal

temperature conditions and since the stress levels are below the endurance limits for infinite life, testing of the crane to 125% of rated capacity provides reasonable assurance that the crane will not fail while handling a spent fuel cask.

As described in Section 9.1.4.2.2.2, sufficient redundancy is provided in the reactor building crane so that no credible postulated failure of any crane component will result in the dropping of the fuel cask. Therefore, the consequences of a cask drop accident are precluded. In addition, crane travel over the spent fuel ~~storage area~~ pool is controlled by travel path (Figure 9.1-5) and interlocks that prohibit the crane from traveling over the ~~storage area~~ spent fuel racks. At no time while being transported does the cask pass over any ~~other safety related~~ safe shutdown items. The objectives of Regulatory Guide 1.13 are met. Furthermore, when the crane is carrying the cask over the refueling floor area, the clear distance between the bottom of the cask and the refueling floor is ~~less than the free drop limitations specified in 10 CFR Part 71.~~ minimized by good operating practice. Should any crane failure occur while the crane is moving the cask over the refueling floor area, the crane drop of less than one inch described in Section 9.1.4.2.2.2 prevents the cask from physically contacting the floor. See Section 9.1.4.2.2.2 for safety features which ensure crane stability during tornado and seismic excitation.

Jib cranes MT-CRA-9A and 9B are designed and equipped to meet the Class A1 (standby) service requirements of CMAA Specification #70. Each crane is capable of raising, lowering, holding, and transporting a test work load of 125% of rated work load without damage to or excessive deflection in a part and without inducing permanent deformation in any crane element. Cables are of the nonrotating type to prevent rotation of load during raising or lowering operations.

The following factors of safety are used in design and are based on the maximum stress produced by worst load combinations and the average ultimate strength of the material, unless otherwise stated herein:

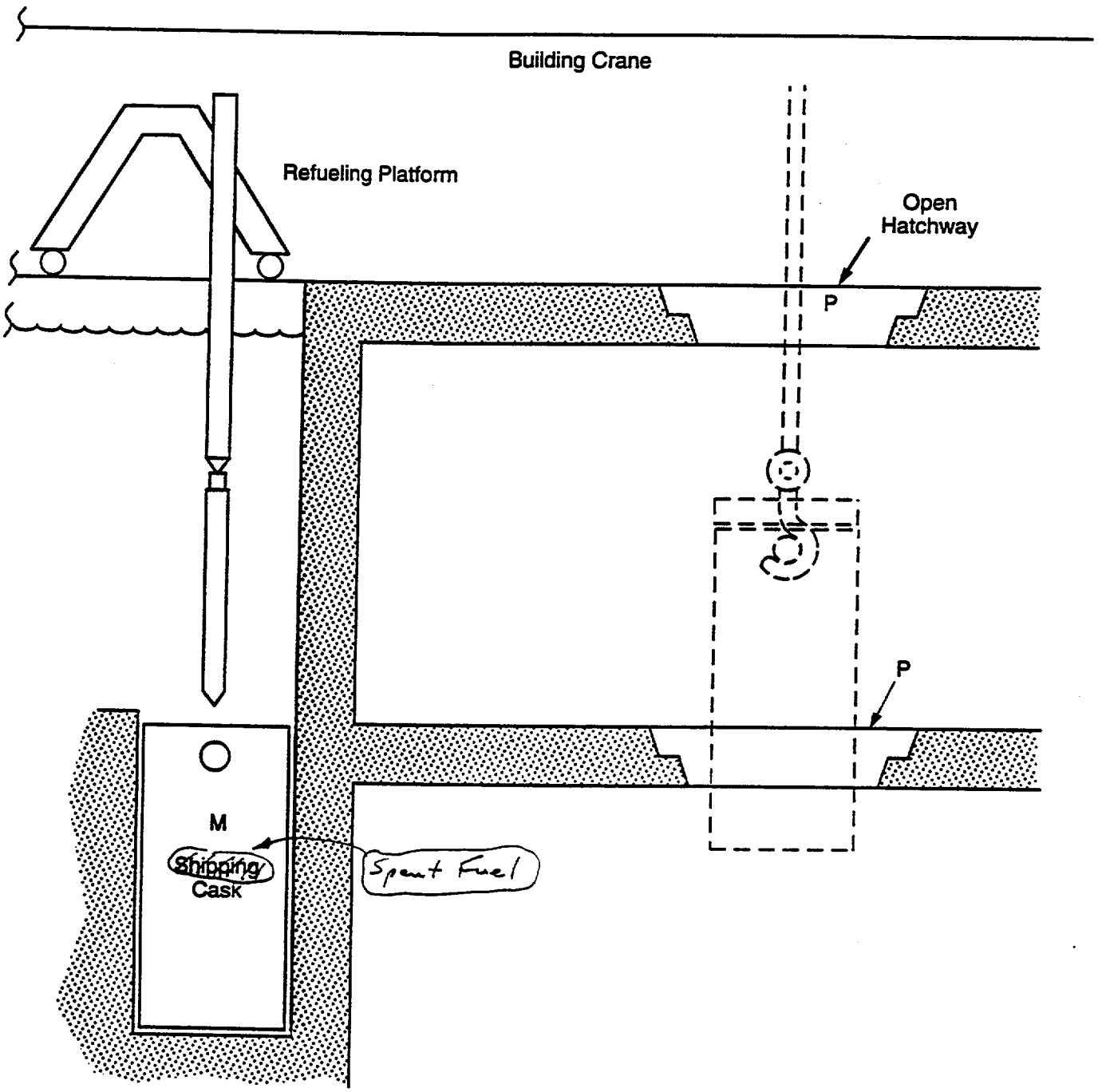
a. All load carrying components except structural members and hoisting ropes are designed to have a minimum safety factor of five;

b. All mechanical parts subject to dynamic strains, such as gears, shafts, drums, blocks, and other integral parts, have a minimum safety factor of five;

c. All hoisting ropes are designed to have a minimum safety factor of five based on the published breaking strength of the rope;

d. Components of the jib cranes are adequately proportioned to limit the overall deflections of the crane to safe limits under any position of the loaded trolley hoist. Maximum vertical static deflection of the boom with nameplate rated hook load is less than 0.6 in.; and

Attachment 2



WASHINGTON PUBLIC POWER
SUPPLY SYSTEM
NUCLEAR PLANT 2 FSAR

Simplified Section of Fuel Shipping Facilities (Section Z-Z)

Draw. No. 960690.75	Rev.	Figure 9.1-16

Form No. 960690

each damaged rod. Ten percent of the noble gases inventory (30% for ^{85}Kr) and 10% of the iodine inventory are assumed to be released to the fuel pool. The activity airborne in the secondary containment is presented in Table 15.7-3.

15.7.4.5.1.2 Fission Product Transport to the Environment. The transport pathway consists of mixing in the fuel pool, migration from the pool to the secondary containment atmosphere and release to the environment through the reactor building ventilation exhaust system (REA) and the SGTS. All of the noble gas and 1% (Reference 15.7-1) of the iodines in the fuel pool are assumed to become airborne in the secondary containment.

Release by means of the REA system will occur for a maximum of 6.5 sec. Continued airborne activity release to the environment over a 2-hr period will be after filtration by the SGTS (99% removal efficiency for iodine).

The release of activity to the environment is presented in Table 15.7-4.

15.7.4.5.1.3 Results. The calculated exposures for the design basis analysis are presented in Table 15.7-5 and are within the guidelines of 10 CFR 100.

15.7.5 SPENT FUEL CASK DROP ACCIDENT

ANSI N14.6 compliant

a lifting yoke

The spent fuel cask is equipped with ~~redundant sets of~~ lifting lugs and ~~yokes~~ compatible with the reactor building crane main hook. The reactor building crane is provided with sufficient redundancy such that no credible postulated failure of any crane component required to lift, hold, and move loads, will result in the dropping of the fuel cask. Therefore, an analysis of the spent fuel cask drop is not required.

15.7.6 REFERENCES

15.7-1 "Decontamination Factor For Fuel Handling Accident," NE-02-96-0011, August 1996.

15.7-2 ANSI N14.6-1993, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More," June 1993.

Evaluation of Significant Hazards Considerations

Summary of Proposed Change:

During readiness activities for dry cask spent fuel storage, preparation of a safety evaluation was begun to meet the requirements of 10 CFR 72.212(b)(4). 10 CFR 72.212(b)(4) states in part that: "Prior to use of the general license, determine whether activities related to storage of spent fuel under this general license involve any unreviewed facility safety question or change in the facility technical specifications, as provided under Part 50.59." A safety evaluation found that spent fuel cask handling activities as described in the Final Safety Analysis Report (FSAR) do not reflect actual plant design and represent an unreviewed safety question (USQ). This evaluation also found that the WNP-2 SER, NUREG-0892, "Safety Evaluation Report related to the operation of WPPSS Nuclear Project No. 2," did not evaluate the actual facility configuration or specific processes that will be used in support of spent fuel cask handling.

The original FSAR description of spent fuel departure from the spent fuel pool was based on using the GE IF-300 cask. New regulations and requirements for interim storage have changed the spent fuel handling process. At present we intend to use the 10 CFR 72.214 approved Holtec HI-STORM 100 spent fuel cask system. (The GE IF-300 cask is not included in the list of approved fuel storage casks under 10 CFR 72.214.) There are significant physical differences between the cask originally analyzed in the FSAR and the HI-STORM 100 spent fuel cask system that cause changes in procedures previously evaluated in the FSAR and approved in the SER. Also, the physical description of the facility, as discussed in the FSAR and the SER, does not accurately reflect the configuration as it exists today.

Changes are required to update the FSAR description of cask loading, handling, and spent fuel pool features of construction. The spent fuel cask loading area (e.g. wall, gate) descriptions also require revision in order to reflect the actual plant configuration. This new information changes the NRC stated bases for acceptability as noted in the SER. Energy Northwest has evaluated the revised facility and process descriptions and concluded that they continue to preserve the health and safety of the public. However, NRC review and approval of this new information is required in accordance with 10 CFR 50.59(c).

No significant Hazards Determination:

Energy Northwest has evaluated the proposed change using the criteria established in 10 CFR 50.92(c) and has determined that it does not represent a significant hazards consideration as described below.

The operation of WNP-2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated:

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences.

Accidents previously evaluated in the FSAR that could be influenced by these FSAR text changes regarding cask handling and spent fuel loading operations include the Spent Fuel Cask Drop Accident (FSAR 15.7.5) and the Fuel Handling Accident (FSAR 15.7.4).

Spent Fuel Cask Drop Accident: Sufficient redundancy is provided in the reactor building crane such that no credible postulated failure of any crane component will result in dropping of the fuel cask and rupturing the fuel storage pool. (Reference: WNP-2 FSAR Section 15.7.5, "Spent Fuel Cask Drop Accident"). The drop accident is not deemed credible and the revision of the FSAR description will continue to maintain the drop accident as incredible. Additionally, as a defense-in-depth measure, crane position interlocks prevent lifting a spent fuel cask over the spent fuel stored in the pool.

As the cask is moved in and out of the fuel pool, it passes over several cables and conduits supporting plant equipment. They include nonsafety-related cables such as those supplying the refueling bridge, and spent fuel pool temperature indicator FPC-TE-7. Additionally, a safety-related conduit for FPC-LE-5 is included in the cask load path. While a cask drop, which could damage or cut the cable to FPC-LE-5 is not credible, operator error in which the cable is damaged by the cask not clearing the conduit during cask movement may be credible. If the cable were damaged, it might inhibit one train of the automatic isolation signal for the fuel pool cooling system. The automatic isolation of interest occurs on low fuel pool water level, isolating the Seismic Category I cooling portion of the system from the Seismic Category II cleanup portion of the system. A fuel pool low water level coincident with a crane operator damaging the cable for FPC-LE-5 is an extremely low probability event. However, in the case of a damaged cable for FPC-LE-5, automatic isolation on low water level would still occur because a separate, redundant, logic train (from FPC-LE-4) would not be affected and would still be capable of accomplishing the isolation function described in FSAR Section 9.1.3.2.3. The cable for the redundant logic train is not in the cask load path. The cable for FPC-LE-5 also carries a signal for high/low spent fuel pool water level alarm, which has a redundant analogue signal (undamaged in this scenario) from FPC-LS-4.

Fuel Handling Accident: The fuel handling accident is analyzed in FSAR Section 15.7.4. In it, the assumption is made that a failure occurs in a fuel assembly lifting mechanism. The accident which produces the largest number of failed spent fuel rods is the drop of a spent fuel bundle into the reactor core when the reactor pressure vessel (RPV) head is off. The analysis assumes the accident occurs at the maximum height allowed by the fuel handling equipment above spent fuel (34 ft.). Since the same fuel handling mechanism is used in both the reactor (the analyzed accident location) and in the fuel pool, but at a considerably lower available drop height (approximately 3 ft.), the energy available to damage fuel rods is significantly less. As a result, the analyzed fuel handling accident consequences bound the consequences of a fuel assembly drop in the spent fuel pool.

Because fuel loaded in a cask will be within approximately 1 ft. of the elevation of a fuel pool rack, fuel handling for cask loading is essentially the same as other fuel handling within the pool and is also bounded by the FSAR analysis. Therefore the consequences of this accident evaluated previously in the FSAR will not be increased by the proposed change.

The proposed change does not entail any physical alteration to the present plant configuration. Therefore, individual precursors of an accident are unaffected and the probability of an accident previously evaluated is not expected to increase. In addition, since the functions and capabilities of systems designed to operate safely and/or mitigate the consequences of an accident have not changed, the consequences of an accident previously evaluated are not expected to increase.

The operation of WNP-2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated:

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration.

Information presented in the FSAR describing the spent fuel cask safe load path is revised by this amendment. To agree with the current plant configuration noted above, the FSAR will need to be changed to read, "At no time while being transported does the cask pass over any safe shutdown equipment." The objectives referenced in RG 1.13, Rev. 1, and the guidelines of NUREG-0612 (to prevent impact by heavy loads with safe shutdown equipment) will continue to be met. The proposed change does not entail any physical alteration to the present plant configuration. There are no new precursors of an accident created and no new or different kinds of accidents are created.

The operation of WNP-2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety for the following reasons:

There are no plant modifications required as a result of the proposed FSAR change. The proposed FSAR text changes correct inaccuracies partly resulting from incorrect original process descriptions. Since then, there have been significant changes to spent fuel cask handling and design requirements including the necessity for extended dry storage of spent fuel at independent spent fuel storage installations. With the proposed FSAR text changes incorporated, the FSAR will accurately describe actual plant configuration and processes related to spent fuel cask handling and the NRC certified Holtec HI-STORM 100 System.

The WNP-2 reactor building crane is single-failure-proof and therefore no credible postulated failure of any crane component will result in dropping of the fuel cask and rupturing the fuel storage pool. A single-failure-proof crane obviates the need for an isolated spent fuel cask transfer pool. In addition, safe load paths are defined that keep the spent fuel cask away from irradiated fuel and safe shutdown equipment. This is in accordance with defense-in-depth approach as described in NUREG-0612, Section 5.2, 'Bases for Guidelines'.

The proposed FSAR change contains information about WNP-2 spent fuel cask handling that has not been previously reviewed and approved by the NRC; however, there is no safety significance to this FSAR amendment request. The FSAR text corrections are in agreement with applicable regulations and no physical alteration to the plant configuration is required.

Therefore, this change will not involve a significant reduction in the margin of safety.

Environmental Assessment Applicability Review

Energy Northwest has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for categorical exclusion as provided for under 10 CFR 51.22(c)(9). This conclusion has been determined because the change requested does not pose any significant hazards considerations nor does it involve a significant increase in the amounts, or a significant change in the types of any effluent that may be released off-site. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.