Revision 0
09/28/81

APPLICATION

FOR

SPECIAL NUCLEAR MATERIAL LICENSE

TO

RECEIVE, POSSESS AND STORE

UNIRRADIATED FUEL ASSEMBLIES

FOR

WPPSS NUCLEAR PROJECT NO. 2

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

TABLE OF CONTENTS

1.0 GENERAL INFORMATION

Washington Public Power Supply System (hereinafter "the applicant"), a municipal corporation and a joint operating agency of the State of Washington, with its main office at 3000 George Washington Way, Richland, Washington hereby applies for a license for the receipt, possession, and storage of unirradiated nuclear fuel assemblies for Nuclear Project No. 2 formally known as Hanford No. 2 (hereinafter "the Plant"), which is presently under construction as authorized by the U. S. Nuclear Regulatory Commission (hereinafter "NRC") under Construction Permit CPPR-93.

The Applicant requests that the license applied for herein be in effect until the full operating license for the Plant is issued pursuant to 10CFR Part 50.

1.1 REACTOR AND FUEL

1.1.1 Identification of Reactor, Geographic Location and Docket Number

The Plant and its geographic location are identified in subsections 1.1 (Reference 1) and 2.1.1.1 (Reference 2) of the Final Safety Analysis Report (FSAR) submitted on NRC Docket 50-397.

1.1.2 Description of Fuel Bundles

Each fuel bundle contains 62 fuel rods and two nonfueled rods called water rods. The 64 rods are spaced and supported in a square (8 x 8) array by seven spacers and a lower and upper tie plate. The lower tie plate has a nose piece that fits into the fuel support piece which distributes coolant flow to the fuel rods. The upper tie plate has a handle for lifting the fuel bundle. Both tie plates are made of stainless steel. Three types of fuel rods are used in the bundle; tie rods, water rods, and standard rods.

The third and sixth fuel rods along each outer edge of the bundle are tie rods. The eight tie rods in each bundle have threaded end plugs which screw into the lower tie plate casting and extend through the upper tie plate casting. A stainless steel hexagonal nut and locking tab are installed on the upper end plug of each tie rod to hold the bundle together. The water rods are hollow water tubes which are used to improve the nuclear performance of the fuel. One water rod is used to axially position the Zircaloy fuel rod spacers vertically in the bundle. The fuel rod spacers are equipped with Inconel springs to maintain rod-to-rod spacing, and to accommodate differential axial thermal expansion. The remaining fuel rods in a bundle are standard rods each of which has a single tube of fuel pellets. The standard rods are the same length as the tie rods. The end plugs of the water and standard rods have pins which fit into holes in the tie plates. An Inconel expansion spring, located over the top end plug of each rod, keeps the rods seated in the lower tie plate while allowing them to expand axially by sliding within the holes in the upper tie plate. The sliding is allowed to accommodate differential axial thermal expansion.

There are three types of bundles used within a core. The Type 1 fuel bundles (Natural U) are located on the periphery of the core while Type 2 and 3 bundles (1.76% and 2.19%) are in the interior of the core. Bundle Types 2 and 3 both contain a burnable poison, gadolinia, for supplementary control. These bundles differ in the amount and placement of the gadolinia.

1.1.3 Quantity of Special Nuclear Material

The total weight of uranium to be received and stored by the Applicant under the requested license will be approximately 140,069 kilograms contained in 764 fuel bundles. The average U-235 enrichments of the three types of fuel are .711%, 1.76%, and 2.19% with a full core average enrichment of 1.87% with no thorium present.

1.2 STORAGE CONDITIONS

1.2.1 Description of Storage Areas

A. Primary Storage Area - New and Spent Fuel Storage Area

Up to 240 fuel assemblies may be stored in the new fuel storage vault. Assemblies not stored in the new fuel storage vault may be stored in the spent fuel pool which will be dry during storage of the unirradiated fuel assemblies. A detailed description of the fuel storage areas is given in Section 9.1 of the FSAR (Reference 3). No activities other than those involving fuel handling, inspection, and storage normally takes place in any fuel storage area. No construction work which could cause physical damage to the fuel will be allowed in the fuel storage areas during fuel handling inspection or storage.

When fuel assemblies are stored in the new or spent fuel storage areas, access to the areas will be restricted with only authorized personnel being allowed to enter. The only means of access to the new or spent fuel storage areas will be through doors that are controlled by the electronic access control system, or by key locks with a security watchperson present whenever the door(s) are unlocked.

The new fuel will be moved into the inspection and storage locations by the Reactor Building auxiliary hoist and the refueling jib cranes. Descriptions, design bases, and safety evaluations for the Fuel Handling Systems are provided in subsection 9.1.1, 9.1.2, and 9.1.4 of the FSAR.

B. Alternate Storage Area

As an alternative, an exterior protected area would be used as a temporary storage location. This area will be protected by an eight foot high security fence of at least eleven AWG fence fabric with a barbed wire topping. A fence sensor intrusion detection system will be employed and the area will be lighted at night. A security watchperson will be posted at the facility 24 hours per day to control access. Two storage configurations may be utilized in the alternate storage area. They are:

1) Configuration 1. The fuel assemblies would be stored in the original General Electric Model RA series container (two fuel assemblies per container). This container consists of a wooden outer shipping box and an inner metal box. Containers would be stacked in six wide by five high arrays. Containers within each array would be separated using wooden 4" x 4" (nominal) vertical spacers. Arrays would be spaced approximately five feet by forty feet to allow manipulation of equipment necessary to move the containers from truck trailers to storage area and later to the Plant refueling floor.

2) Configuration 2. This configuration would be similar to Configuration 1 except that the metal boxes would be removed from the wooden boxes prior to stacking the metal boxes in six wide by four high arrays. Containers within each array would be separated using wooden 4" x 4" (nominal) horizontal and vertical spacers.

New fuel would be transported to the Reactor Building and moved into the inspection and storage locations by the Reactor Building auxiliary hoist and the refueling jib cranes. Each stack would be covered by a waterproof and fire retardant material for added protection. For drawings of the alternate storage area, see Reference 6.

1.2.1.1 Fuel Handling System

All fuel handling equipment will be pre-operationally tested for safe operation prior to its use for fuel handling activities. The fuel handling equipment and fuel bundles and assemblies are designed for the fuel handling activities in this application.

A more complete description of the Fuel Handling System is contained in subsection 9.1.4 of the applicant's FSAR.

1.2.1.2 Fuel Handling Activities

Upon arrival of a shipment of fuel, the following will normally take place:

1) The shipment is directed from the gate to the railroad bay in the Reactor Building at elevation 441' or the alternate storage area.

2) The tarps and chains will be removed from the truck.

3) The shipment and shipping containers will be verified to comply with shipping papers presented by the carrier. The wooden shipping crates will be inspected for damage. The Plant Technical Department and Fuel Supply Department are responsible for evaluation and resolution of discrepancies.

4) Upon proper acceptance of shipping papers, the truck may be unloaded.

5) Health Physics will survey the fuel boxes.

6) Health Physics will perform a survey on the truck.

7) It will be verified that the tamper-safe seals on the outer container have not been broken.

8) The metal container number and the two bundle serial numbers will be verified to agree with the General Electric shipping document. If the fuel is stored in the alternate storage area, this inspection will not occur until final transfer to new fuel storage vault and/or spent fuel pool.

The metal shipping containers will be removed from the wooden shipping crates and then will be hoisted to the fuel handling area, 606' elevation, of the Reactor Building. The fuel may now be readied for inspection, channeling and storage or inspection and storage. All personnel involved in the inspection operation will be familiar with and adhere to all criticality rules. Inspection, channeling and storage will proceed in accordance with written operational procedures. Storage shall be in the new fuel storage vault and spent fuel pool.

The Plant Manager has overall responsibility for special nuclear materials on the plant site.

1.2.2 Fire Protection

A. New and Spent Fuel Storage Areas

The materials used in the construction of the fuel storage areas are concrete and steel. The fuel bundles themselves and the fuel racks which hold the fuel are also constructed of noncombustible materials. To cover and protect the fuel bundles while being handled and while in storage, they are normally wrapped in polyethylene sheeting. Fire in the storage or adjacent areas is highly unlikely. Fire hose racks connected to the Plant fire main system are located on the refueling floor level with access to the new fuel storage vault and spent fuel area.

B. Alternate Storage Area

Materials used in construction of the alternate storage areas would consist primarily of concrete and steel or other noncombustible or fire retardant materials. Fire hydrants or hose racks connected to the existing fire protection system would be installed nearby or within the storage location.

1.3 PHYSICAL PROTECTION

The fuel assemblies subject to this license contain no material enriched in U-235 to greater than 3.0 average enrichment by weight, also the assemblies contain no U-233 or plutonium. Thus, the protective requirements of 10CFR Part 73 do not apply. Physical protection is, however, addressed in the Physical Security Plan for protection of the received, unirradiated fuel bundles. The Physical Security Plan is considered security proprietary. This plan is submitted under separate cover.

1.4 TRANSFER OF SPECIAL NUCLEAR MATERIAL

The General Electric Company, fabricator of the nuclear fuel bundles is responsible for shipment of the fuel to the Plant site. To include physical protection in accordance with 10CFR70.67(g)(3), all fuel assemblies are to be delivered to the site in accordance with shipping procedures and arrangements of the General Electric Company, for container Model RA series, authorized for use by that company under separate special nuclear material license SNM-1097. Applicant will be responsible for packaging, and shipment of, any fuel which is required to be returned to the General Electric Wilmington facility.

1.5 FINANCIAL PROTECTION AND INDEMNITY

Prior to the shipment of the special nuclear materials delineated in this application, the Applicant shall furnish proof of financial protection with the NRC pursuant to the manner specified in 10CFR 140.15. The amount of financial protection to be in force at this time will be $1,000,000 as defined in 10CFR 140.13.

2.0 HEALTH AND SAFETY

2.1 RADIATION CONTROL

2.1.1 Training and Experience

The educational, experience and technical qualifications for the responsible onsite plant personnel are detailed in Reference 5.

Qualifications of the Radiation Protection personnel are as set forth in Regulatory Guide 1.8, Revision 1. The Supply System is committed to ensure that, in most cases, the minimum qualifications for individual function levels will be exceeded by those persons assigned to the discipline. If an individual is placed in a discipline who does not meet the minimum qualification criteria, it will be specifically pointed out and justification or explanation provided.

2.1.2 Contamination Monitoring

Administrative controls will be covered under the sections of the Plant Procedures Manual which govern the Plant Health Physics Program. These procedures include receipt of radioactive material, storage of radioactive material, inventory control of radioactive material, personnel monitoring, establishing and posting controlled areas, operation of portable survey instruments, radiation work permits, and others.

Contamination controls, as described previously, will be provided by issuing radiation work permits governing work and access to the fuel handling and storage areas.

The Supply System is committed, consistent with recommendations of Regulatory Guide 8.8, to establish a Health Physics Program to maintain occupational and general public exposure to radiation As Low As Reasonably Achievable (ALARA). The Health Physics Program objectives are discussed in section 12.5 (Reference 4) of the FSAR. The Plant Health Physics Supervisor is responsible for establishing administrative and technical controls for radiation protection, radiation exposure reduction, and for the subsequent administration of this program.

Procedures for Radiation Protection are discussed in section 12.5.3 of the FSAR. Health Physics equipment is described in section 12.5.2.3 of the FSAR.

Radiation survey inspections of the loaded fuel shipping crates will be performed upon receipt.

Portable survey instrumentation will be calibrated at six month intervals using either approved plant procedures and National Bureau of Standards (NBS) traceable calibration sources or a contracted calibration service which has been evaluated and placed on the Qualified Suppliers List for safety related services.

Laboratory instrumentation will be calibrated at twelve month intervals using NBS traceable calibration sources. Functional checks will be performed daily or prior to use to ensure that the instrument is operating properly and remains in calibration.

Additional detail concerning the frequencies and methods for calibration of instruments is discussed in sections 12.5.2.2 and 12.5.2.4 of the FSAR.

2.2 NUCLEAR CRITICALITY SAFETY

The applicant requests that it be exempted from the requirements of 10CFR70.24 insofar as they apply to the storage of nuclear fuel assemblies at the plant. This request for exemption is based on the fact that the storage facilities described in this application provide assurance that inadvertent criticality cannot occur. The fuel would remain in the metal containers and/or wooden shipping crates at all times when in an outdoor temporary fuel storage location. The General Electric Model RA containers (outer plus inner combination) have been demonstrated to be criticality safe in an infinite geometry in the undamaged condition and loaded with fuel enriched to 3.2% U235. Without the wooden containers, critical safety has been demonstrated in closed pack conditions of up to 260 containers in worst credible accident conditions. Fuel in the alternate storage area will be kept in groups of much less than 260 containers. The General Electric analysis of criticality safety for RA containers (SNM 1097) verifies the above criticality statements.

2.3 ACCIDENT ANALYSES

Accident analyses for fuel handling equipment and storage areas are provided throughout sections 9.1.1, 9.1.2 and 9.1.4 of the FSAR. The potential for accidents affecting the safety of fuel in the storage areas is limited to the dropping of fuel assemblies over the storage area. No overhead load greater than one fuel assembly will be allowed in or over the fuel storage array. This requirement shall be enforced by operating procedures controlling crane operations. The seismic design of the Reactor Building, and of cranes, racks, and pools precludes the credibility of more severe accidents. In the unlikely event of a dropped fuel assembly in the storage areas, the consequences affecting safety would be minimal. Due to the spacing of the storage arrays, a criticality condition would not be possible under these accident conditions. The consequences of the accident would be limited to the minimal effect of possible rupture of fuel rods and subsequent release of unirradiated uranium dioxide fuel.

REFERENCE 1

WNP-2 FSAR 1.1

Page 1.1-1 and Figure 1.1-1

CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

1.1.1 TYPE OF LICENSE REQUIRED

This Final Safety Analysis Report (FSAR) is submitted in support of an application of the Washington Public Power Supply System (WPPSS) for a Class 103 operating license for a single unit nuclear power plant. The facility is known as the WPPSS Nuclear Project No. 2 (WNP-2) and was formerly known as the Hanford No. 2 Plant.

1.1.2 IDENTIFICATION OF APPLICANT

Washington Public Power Supply System (also referred to as WPPSS or the Supply System) is the applicant for the operating license for WNP-2. The plant has been designed, constructed, and will be operated under the responsibility of WPPSS.

1.1.3 NUMBER OF PLANT UNITS

WNP-2 is a single unit plant.

1.1.4 DESCRIPTION OF LOCATION

WNP-2 is located within the Hanford Reservation of the Department of Energy (DOE), Benton County, Washington, approximately 12 miles north of the City of Richland. The site is approximately 3 miles west of the Columbia River at River Mile 352.

1.1.5 TYPE OF NUCLEAR STEAM SUPPLY

This plant will have a boiling water reactor nuclear steam supply system as designed and supplied by the General Electric Company and designated as BWR 5.

1.1.6 TYPE OF CONTAINMENT

The containment system, designed by Burns and Roe, Inc. (B&R) with specific vessel design by Pittsburgh Des Moines Steel Co., limits the release of radioactive materials to the environs subsequent to the postulated occurrence of a loss-of-coolant accident so that the offsite doses are below the "reference values" stated in 10 CFR Part 100. The design employs the drywell/pressure-suppression feataures of the BWR/Mark II containment concept. The containment provides dual barriers consisting of the primary containment and the secondary containment. The primary containment is a steel pressure-suppression system of the over-and-under configuration. The secondary containment is the reactor building which encloses the reactor and its primary containment.

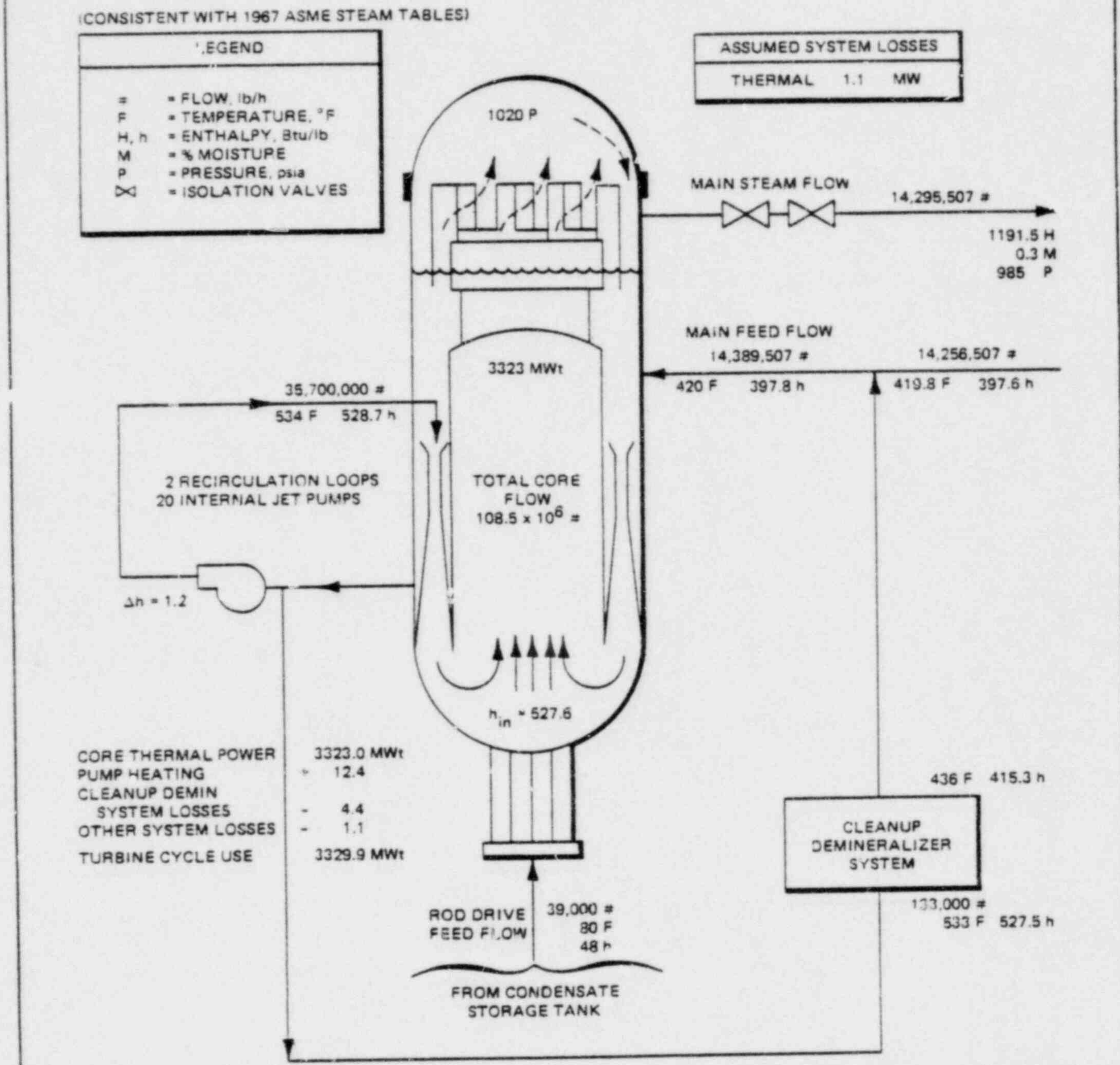
1.1.7 CORE THERMAL POWER LEVELS

The information presented in this FSAR pertains to WNP-2, with rated power level of 3323 MWt and design power level of 3468 MWt. The station utilizes a single-cycle forced circulation boiling water reactor (BWR) provided by General Electric (GE). The heat balance for rated power is shown in Figure 1.1-1. The station is designed to operate at a gross electrical power output of approximately 1205 MWe and a net electrical power output of approximately 1145 MWe.

1.1.8 TYPE OF CONDENSER COOLING

Condenser cooling is provided by water circulated through mechanical induced draft cooling towers.

1.1.9 DELETED



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	HEAT BALANCE AT RATED POWER.	FIGURE 1.1-1

REFERENCE 2

WNP-2 FSAR 2.1

Page 2.1-1 and Figures 2.1-1, 2.1-2

CHAPTER 2

SITE CHARACTERISTICS

2.1 GEOGRPAHY AND DEMOGRAPHY

2.1.1 SITE LOCATION AND DESCRIPTION

2.1.1.1 Specification of Location

WNP-2 is located in the southeast area of the U.S. Department of Energy's (DOE) Hanford Site in Benton County, Washington. The site is approximately 3 miles west of the Columbia River at River Mile 352, approximately 8 miles north of North Richland, 18 miles northwest of Pasco and 21 miles northwest of Kennewick (Figures 2.1-1 and 2.1-2).

The reactor is located at 46° 28' 18" north latitude and 119° 19' 58" west longitude.

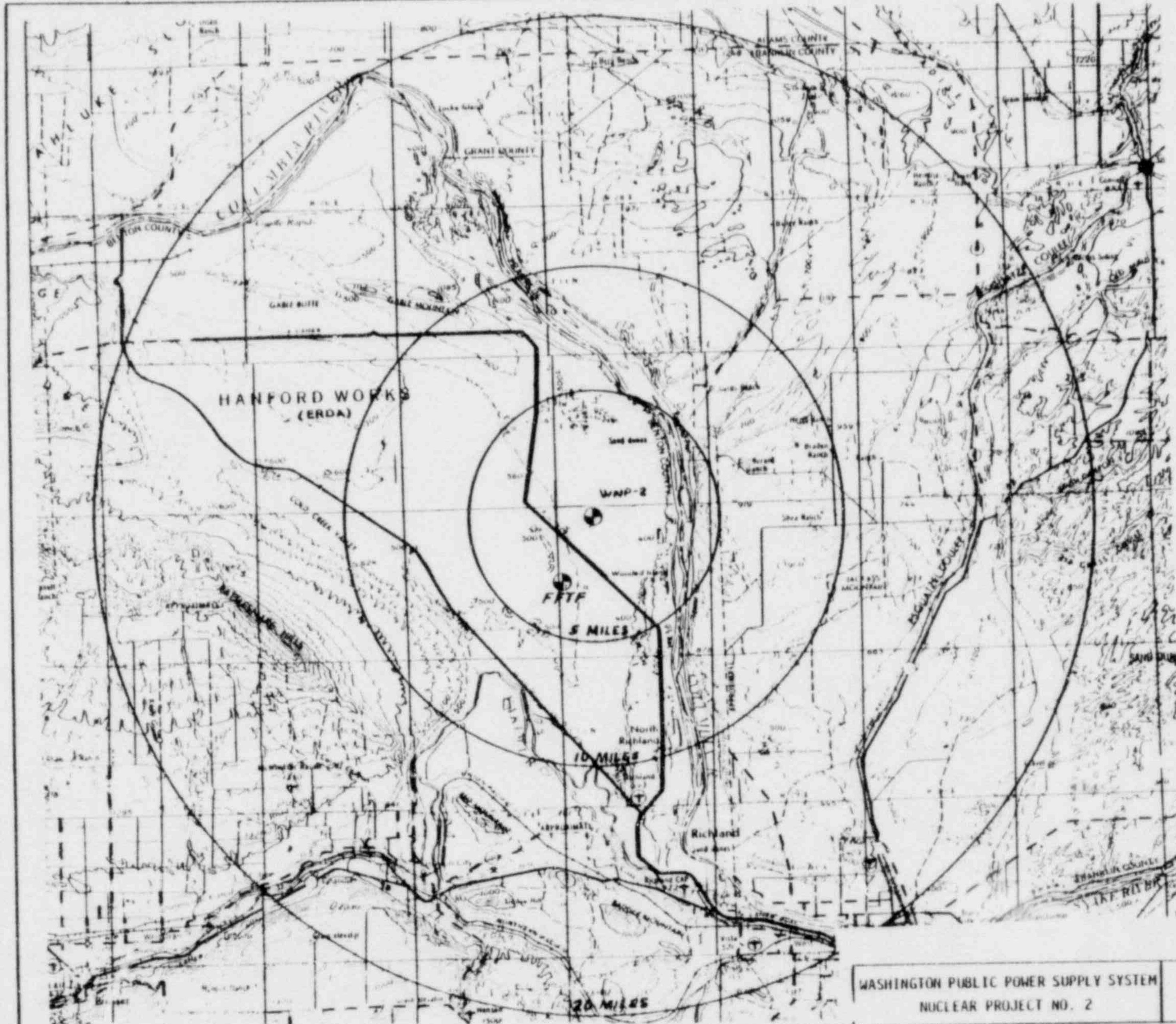
The approximate Universal Transverse Mercator coordinates are 5,148,840 meters north and 320,930 meters east.

2.1.1.2 Site Area Map

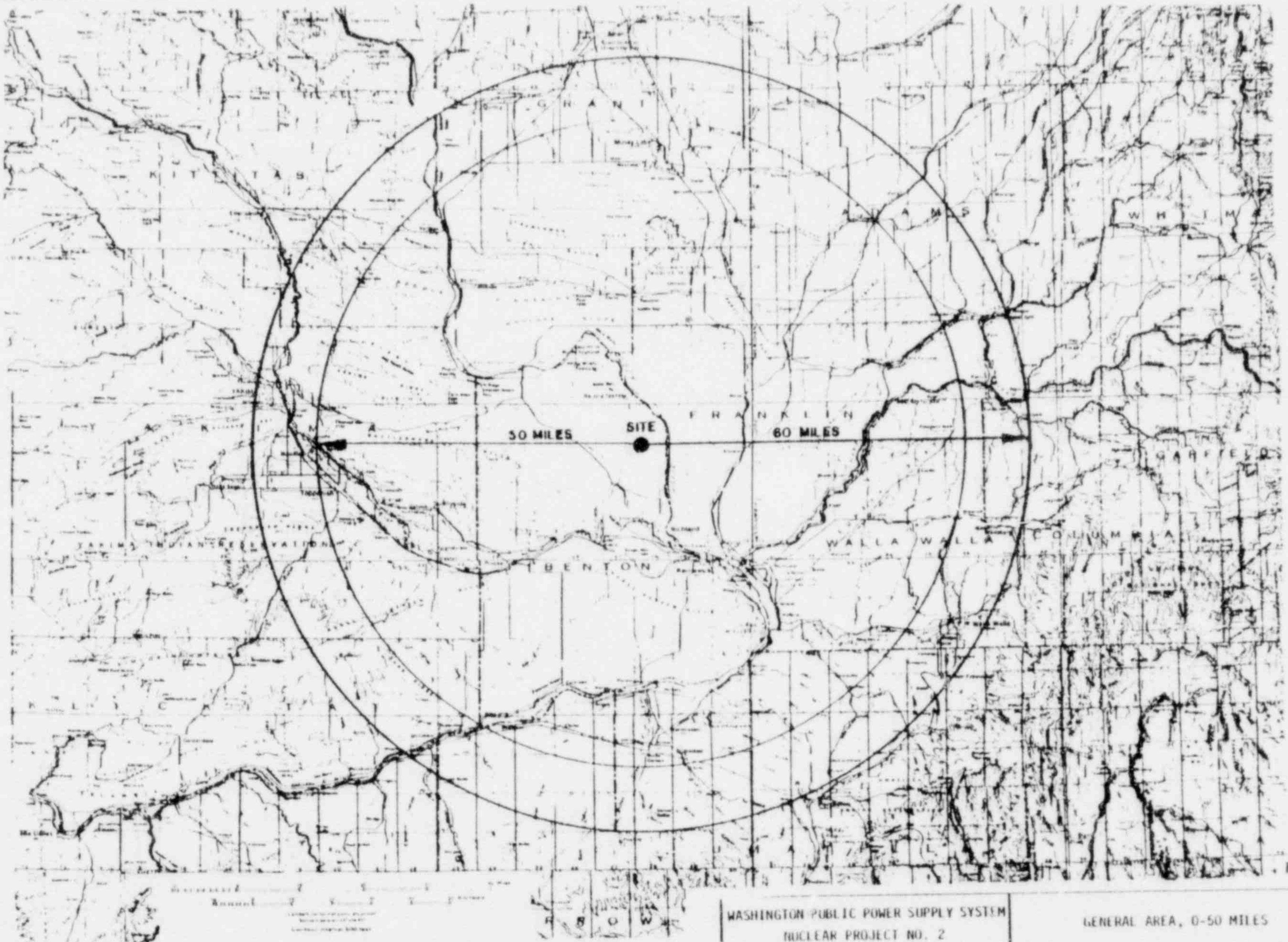
The WNP-2 site area is that real estate over which WPPSS has the legal right to control access of individuals to and is described by the area enclosed by the exclusion zone plus the plant property lines as shown in Figure 2.1-3. The plant site grade level is 441 feet above Mean Sea Level (MSL). The relative locations of the plant buildings, the circulating water system and the railroad spur linking the site with the Burlington Northern Railroad at Richland are shown in Figure 2.1-4.

The boundary of the exclusion area is a circle with its center at the reactor and a radius of 1950 meters. Industrial facilities located in the site area are the H.J. Ashe Substation and the Supply System's Nuclear Projects Nos. 1 and 4. Highway and railway facilities located within the site area are shown in Figure 2.1-3. The exclusion zone boundary shown in Figure 2.1-3 is the joint boundary for the three reactors.

Ownership and control of the land outside the WNP-2 property line but within the site exclusion area are discussed in 2.1.2. The site is situated near the middle of the relatively flat, essentially featureless plain which is best described as a shrub steppe with sage brush interspersed with perennial native and introduced annual grasses extending in a northerly, westerly and southerly direction for several miles. The plain is characterized by slight topographic relief with a maximum across the plant site of approximately ten feet.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	GENERAL AREA, 0-20 MILES	FIGURE 2.1-1



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	GENERAL AREA, 0-50 MILES	FIGURE 2.1-2

REFERENCE 3

WNP-2 FSAR 9.1

Pages: 9.1-1 thru 9.1-22
9.1-37 thru 9.1-71
and
Figures 9.1-1 thru 9.1-17

CHAPTER 9.0

AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

9.1.1 NEW FUEL STORAGE

9.1.1.1 Design Bases

9.1.1.1.1 Safety Design Bases

9.1.1.1.1.1 Safety Design Bases - Structural

a) The new fuel storage racks containing a full complement of fuel assemblies are designed to withstand all credible static and dynamic loadings, to prevent damage to the structure of the racks and the contained fuel, and to minimize distortion of the rack arrangement. (See Table 3.9-2(s)).

b) The racks are designed to protect the fuel assemblies from excessive physical damage so as to prevent the release of radioactive materials in excess of 10 CFR 20 limits under normal or abnormal conditions.

c) The racks are constructed in accordance with the Quality Assurance Requirements of 10 CFR Part 50, Appendix B.

d) The new fuel storage racks are categorized as Safety Class 2 and Seismic Category I.

e) The building containing the new fuel storage vault (new fuel storage facility) is designed to comply with the objectives set forth in Regulatory Guide 1.13, and General Design Criteria 2, 3, 4, 5, 61, 62 and 63 of 10 CFR Part 50, Appendix A.

9.1.1.1.1.2 Safety Design Bases - Nuclear

a) The new fuel storage racks are designed and maintained with sufficient spacing between the new fuel assemblies to assure that the array, when racks are fully loaded, shall be subcritical by at least 5% Δk, including allowance for calculation biases and uncertainties. In the calculations performed to assure that $k_{eff} \leq 0.95$, the standard lattice methods (Reference 9.1-1) used at General Electric are employed. Under conditions where diffusion theory is valid, the reference 9.1-1 method is used in calculations (i.e., conditions where fuel is flooded with water at a density of between 0.7 and 1.0 g/cc).

It is assumed that the storage array is infinite in all directions. Since no credit is taken for leakage, the values reported as effective neutron multiplication factors are in reality infinite neutron multiplication factors.

The biases between the calculated results and experimental results, as well as the uncertainty involved in the calculations, are taken into account as part of the calculational procedure to assure that the specified k_{eff} limits are met.

9.1.1.1.2 Power Generation Design Bases

a) New fuel storage racks provide for approximately 30% of the full core fuel load.

b) New fuel storage racks are designed and arranged so that the fuel assemblies can be handled efficiently during refueling operations.

9.1.1.2 Facilities Description

The new fuel storage vault containing the new fuel storage racks is a concrete structure adjacent to the spent fuel pool at the refueling floor level of the reactor building (see Figure 1.2-6). The reactor building is built to Seismic Category I requirements and is further discussed in 3.2. The new fuel rack design features are as follows:

a) The new fuel storage vault contains 24 sets of castings which may contain up to 10 fuel assemblies; therefore a maximum of 240 fuel assemblies may be stored in the fuel vault.

b) There are three tiers of castings which are positioned by fixed box beams. This holds the fuel assemblies in a vertical position, the fuel assemblies are supported at the lower and upper tie plate, with additional lateral support at the center of gravity of the fuel assembly.

c) The lower casting supports the weight of the fuel assembly and restricts the lateral movement; the center and top casting restricts only lateral movement of the fuel assembly.

d) The new fuel storage racks are made from aluminum. Materials used for construction are specified in accordance with the latest issue of applicable ASTM specifications. The material choice is based on a consideration of the susceptibility of various metal combinations to electrochemical reaction. When considering the susceptibility of metals to galvanic corrosion, aluminum and stainless steel are similar insofar as their coupled potential is concerned. The use of stainless steel fasteners in aluminum to avoid detrimental galvanic corrosion is a recommended practice and has been used successfully for many years by the aluminum industry.

e) The minimum center-to-center spacing for the fuel assembly between rows is 11.875 inches. The minimum center-to-center spacing within the rows is 6.535 inches. Fuel assembly placement between rows is not possible.

f) Lead-in and lead-out of the casting provides guidance of the fuel assembly during insertion or withdrawal.

Fuel spacing (7 inches minimum center-to-center within a rack, 12 inches minimum center-to-center between adjacent racks) within the rack and from rack-to-rack will limit the effective multiplication factor of the array (k_{eff}) to not more than 0.95. The fuel assemblies are loaded into the rack through the top. Each hole for a fuel assembly has adequate clearance for inserting or withdrawing the assembly channeled or unchanneled. Sufficient guidance is provided to preclude damage to the fuel assemblies. The upper tie plate of the fuel element rests against the rack to provide lateral support. The design of the racks prevents accidental insertion of the fuel assembly into a position not intended for the fuel. This is achieved by abutting the sides of each casting to the adjacently installed casting. In this way, the only spaces in the assembly are those into which it is intended to insert fuel. The weight of the fuel assembly is supported by the lower tie plate which is seated in a chamfered hole in the base casting.

The floor of the new fuel storage vault is sloped towards a drain located at the low point. This removes any water that may be accidentally and unknowingly introduced into the vault. the drain is part of the floor drain subsystem of the liquid radwaste system.

The radiation monitoring equipment for the new fuel storage area is described in 12.3.4.

9.1.1.3 Safety Evaluation

9.1.1.3.1 Criticality Control

The calculations of k_{eff} are based upon the geometrical arrangements of the fuel array, and that subcriticality does not depend upon the presence of neutron absorbing materials. To meet the requirements of General Design Criterion 62, geometrically-safe configurations of fuel stored in the new fuel array are employed to assure that k_{eff} will not exceed 0.95 if fuel is stored in the dry condition or if the abnormal condition of flooding (with water with a density of 1 g/cc) occurs. In the dry condition, k_{eff} is maintained $\leq$0.95 due to under-moderation. In the flooded condition, the geometry of the fuel storage array assurs the k_{eff} will remain $\leq$0.95 due to over-moderation.

The fuel storage rack is designed using non-combustible materials. Plant procedures and inspections assure that combustible materials are restricted from this area. The primary approach to fire prevention is the elimination of combustible materials. The fire suppressant is water which will not inhibit or negate criticality control. Refer to 9.5.1 for discussion of WNP-2 fire protection systems.

9.1.1.3.2 New Fuel Rack Structural Design

The new fuel storage racks are designed to meet Seismic Category I requirements.

The maximum stress in the fully load rack in a faulted condition is 16.5 Kip. (See Table 3.9-2(s)). This is significantly lower than the allowable stress.

The storage rack is designed to withstand horizontal combined loads up to 222,000 lbs., well in excess of expected loads.

The storage rack is designed to withstand the pull-up force of 4000 lbs. and a horizontal force of 1000 lbs. There are no readily available forces in excess of 1000 lbs. The racks are designed with lead outs to prevent sticking. However, in the event of a stuck fuel bundle, the lifting bail will yield at a pull up force less than 1000 lbs.

The storage rack structure (Figure 9.1-1) is designed to withstand the impact resulting from a falling weight. Tests using a simulated fuel bundle have been conducted to verify that the rack casting can withstand the impact from a bundle dropped from above the array. During testing the lowest drop to cause the rack casting to exceed ultimate stress was a drop of 6.17 feet (4314 foot pounds) made at mid span. Therefore, procedural requirements dictate that no more than one bundle at a time can be handled over the storage array. These requirements assure that the racks cannot be displaced in a manner causing critical spacing as a result of impact from a dropped fuel assembly. Since the 125 ton reactor building crane can traverse the full length of the refueling floor, administrative controls will prohibit carrying loads over the new fuel.

9.1.1.3.3 New Fuel Handling

New fuel is carried to the new fuel vault and placed in the storage rack with either the 15 ton reactor building auxiliary hoist or by use of jib cranes MT-CRA-9A or 9B. To handle the new fuel, a general purpose grapple is used which interfaces with the auxiliary hoist at the upper end and the fuel bundle bail at the lower end.

During the positioning of a new fuel assembly into the new fuel rack, the grapple is always above the upper fuel rack casting. The grapple interfaces only with the fuel bundle bail, thus precluding engagement of the fuel rack. The transfer devices used for new fuel handling to the new fuel vault, therefore, cannot impose uplift loads on the rack castings. See 9.1.4.2.10.1 for further discussion of new fuel handling.

9.1.1.3.4 Other New Fuel Storage Design Factors

The new fuel racks are designed to be restrained by holddown bolts to assure that rack spacing does not vary during the SSE.

The storage rack structure is so designed that the height of the rack is less than the length of the fuel bundle. Therefore, the upper tie plate of the bundle cannot pass below the top cross member of the rack. Also, the fuel bundle insertion spaces in the top casting of the rack have a lead in chamber on the upper and lower surfaces. These design features prevent any tendency of the fuel bundle to jam during insertion into or removal from the rack.

The new fuel storage racks are made from aluminum and are secured by stainless steel fasteners. Materials used for construction are specified in accordance with ASTM Specifications B108, B179, B209, B211, and B221 dated January 1971.

The new fuel storage racks do not require any periodic special testing or inspection for nuclear safety purposes.

9.1.2 SPENT FUEL STORAGE

9.1.2.1 Design Bases

9.1.2.1.1 Safety Design Bases

9.1.2.1.1.1 Safety Design Bases - Structural

a. The spent fuel storage racks are designed to withstand the affects of the Safe Shutdown Earthquake and remain functional and maintain subcriticality. The racks are also designed to withstand the impact of a dropped fuel assembly or the upward force of a stuck assembly without loss of function.

b. The spent fuel storage facility is located so that no missiles can enter the fuel pool with the necessary energy to cause any damage to the fuel.

c. The reactor building containing the spent fuel storage facility provides the capability for limiting the potential off-site exposures in accordance with 10 CFR Part 100 in the event of significant release of radioactivity from the stored fuel.

9.1.2.1.1.2 Safety Design Bases - Nuclear

a. The center-to-center spacing between stored fuel assemblies in a fully loaded rack is sufficient to maintain a k_{eff} less than about 0.95. This design basis is met with fresh fuel of up to 3.25 weight percent enrichment, a conservative water temperature (68°F) and no credit for fixed poison in the fuel assembly. Credit is taken for the fixed poison in the fuel racks.

b. The spent fuel storage rack design precludes storage of a fuel assembly other than where intended.

c. The spent fuel storage racks are designed to allow adequate cooling of the stored spent fuel assemblies.

d. Shielding for the spent fuel storage arrangement is sufficient to protect plant personnel from exposure to radiation in excess of 10 CFR Part 20 limits. Since provisions for portable shielding are not provided in the drywell, administrative control is used during refueling operations to avoid overexposure of personnel as the result of a postulated fuel drop accident such as a drop occurring on the reactor seal plate.

9.1.2.1.2 Power Generation Design Bases

a. Spent fuel storage space in the fuel storage pool is for 2658 fuel assemblies.

b. Spent fuel storage racks are designed and arranged so that fuel assemblies can be handled efficiently during refueling operations.

9.1.2.2 Facilities Description

9.1.2.2.1 Spent Fuel Storage Racks

Spent fuel storage racks provide a place in the fuel pool for storing the spent fuel discharged from the reactor vessel. They are top entry racks, designed to maintain the spent fuel in a space geometry that precludes the possibility of criticality under both normal and abnormal conditions. This is accomplished with the aid of neutron absorbing plates. The location of the spent fuel pool within the plant is shown in Figure 1.2-6.

The spent fuel storage rack design, shown in Figure 9.1-2, consists of fuel storage cells which are square stainless steel tubes with neutron absorbing B_4C plates between them. A stainless steel plate grid at the top and the bottom of the tubes, to which the tubes are welded, form the tubes into racks and maintain center-to-center spacing between the tubes at 6.5 inches. The racks are welded together into modules which are held firmly in place by seismic restraints attached between the rack modules and the pool wall. The storage racks are made of stainless steel. The square tube storage cells are 1/8 inch thick.

The neutron absorber plates have nominal dimensions of 19 inches long, 5.88 inches wide, and 0.2 inches thick. They are composed of B_4C granular material bonded together to form a plate of uniform properties. They have a nominal B^{10} loading of 0.0959 grams per square centimeter of plate and a plate density of

0.05 lbs/in^3. The plate has been shown by tests to have negligible corrosion in water and thermally stable over the range of pool water temperatures that can occur. The plates are seal welded in a stainless steel cavity to prevent water intrusion.

There are no load bearing requirements for the plates. Based on the results of the Modulus of Rupture tests, the plates will withstand approximately two times the calculated stresses caused by a postulated seismic event. Plate integrity and mechanical properties have been verified by comprehensive tests. These tests included Modulus of Rupture and Modulus of Elasticity tests. The Modulus of Rupture testing was performed using a three point support method and was done on specimens at temperatures varying from ambient to 300°F, specimens soaked in water, and irradiated specimens. The Modulus of Elasticity was performed using a resonance procedure and was done at varying temperatures and after the plate had been immersed in water. The tests showed no swelling, cracking or dimensional changes and provided verification of the plate mechanical properties required for the rack design.

In addition to the mechanical tests, extensive irradiation induced offgassing tests have been performed using gamma sources. These test results clearly indicate that the amount of offgassing is negligible and will not cause rack distortion.

Different rack sizes are used (12 x 16, 12 x 13, 8 x 13, 7 x 18 and 11 x 16 arrays) to take full advantage of the fuel storage space in the pool (see Figure 9.1-3). The upper rack structures are welded to an elevated base plate which, in turn, is supported by a system of welded beams and stiffeners. The base serves to support the weight of the fuel assemblies and to distribute the load on the pool floor. The base plate contains an opening at each fuel assembly storage location which accommodates the fuel assembly lower nozzle. Natural circulation of pool water flows upward through the lower nozzle and the fuel assembly to remove decay heat. The storage cells are designed to provide lateral support for the stored assemblies.

The seismic restraints are stainless steel turnbuckles located between the pool walls and the racks around the periphery of the pool (Figure 9.1-3). They are located at both the top and bottom of the rack and, once adjusted will transmit the seismic forces of the OBE and the SSE between the racks and the walls and remain functional. The turnbuckles are connected at the wall to stainless steel bands which are embedded in the concrete wall and seal welded to the pool liner.

9.1.2.2.2 Spent Fuel Storage Pool

The spent fuel storage pool is designed to withstand earthquake loadings as a Seismic Category I structure. It is a reinforced concrete structure completely lined with stainless steel, which provides a leakproof membrane that is resistant to abrasion and damage during normal and refueling operations. The stainless steel liner plates are seamwelded

BLANK

together and are anchored to the surrounding concrete by concrete anchors in the walls and structural members in the floor welded to the liner plates. Each liner weld seam is backed up by a drainage monitoring channel. These channels form a series of interconnecting systems designed to provide the following:

a. Detection, measurement, and location of any liner leakage;

b. Prevention of pressure buildup behind the liner plates due to leakage; and

c. Prevention of uncontrolled loss of contaminated pool water to other relatively clean locations within the secondary containment.

This network of drainage monitoring channels is embedded in the concrete behind the liners and is designed to permit free gravity drainage to the radioactive floor drain system, the flow of which is monitored.

The refueling canal connecting the spent fuel storage pool to the reactor well is provided with two gates in series, with a monitor drain between them. This arrangement permits monitoring for any leaks and facilitates repair of a gate or seal, if required.

A gamma scan collimator port is located 15'-11" from the bottom of the fuel pool and extends through the side of the pool. Through this port, gamma scanning of radioactive reactor components and spent fuel assemblies can be performed. This non-destructive method of analysis by the measurement of the gamma radiation being emitted by a material can indicate fuel enrichment, reactor power distribution, or fission product content of the component being analyzed.

The water supply to the spent fuel pool is provided by the condensate supply system (See 9.2.6) with emergency makeup water from the standby service water system (See 9.2.7). Surge tank water level is monitored to maintain a pool water level above the fuel sufficient to provide shielding for normal building occupancy (See 9.1.3).

9.1.2.3 Safety Evaluation

9.1.2.3.1 Criticality Control

The design of the spent fuel storage racks provides for a subcritical multiplication factor (k_{eff}) of equal to or less than 0.95 for both normal and abnormal storage conditions. Normal conditions exist when the fuel storage racks are covered with a normal depth of water (about 25 feet above the stored fuel) for radiation shielding, and with the maximum number of fuel assemblies or bundles in their design storage position. An abnormal condition may result from damage caused by accidental drop of a fuel assembly or stuck fuel assembly during attempted withdrawal.

The criticality analyses of the normal condition included several conservative assumptions as well as the effect of uncertainties in calculation method and geometric and material variations of the fuel storage rack. The following conservative assumptions were used in the calculation:

a. Fresh fuel of 3.25 weight per cent U-235 enrichment.

   Initially the maximum enrichment will be much lower than this, but could approach this value if an 18-month fuel cycle is used. The enrichment selected is higher than the average enrichment of any fuel expected to be stored in the spent fuel pool. It was chosen because the fully loaded rack of fuel with this enrichment gives a more reactive condition than any presently foreseen.

b. Uniform planar array of 3.25 weight per cent enrichment fuel

   Calculations have shown that this is conservative compared to the realistic, planar distributed enrichments within an assembly.

c. Spent Fuel Pool Bulk Water Temperature 68°F

   This is considerably lower than expected. Nevertheless, a calculation was done to determine the increase in reactivity due to a decrease in pool temperature to 32°F. The results showed the effect to be negligible.

d. Fuel racks are infinite in three dimensions.

e. Fixed neutron poisons in the fuel assembly are neglected.

The majority of the calculations were performed with methods commonly used in light water reactor design; i.e., 4-group diffusion theory cell calculations using PDQ-7 (Reference 9.1-2). Cross sections for these calculations are generated with NUMICE-2, (Reference 9.1 3) the NUS Corporation version of the Westinghouse LEOPARD code. This code uses the same cross section library tape and calculational techniques as LEOPARD. Selected cases were checked and the final design multiplication factors were verified with Monte Carlo calculations using KENO-IV (Reference 9.1-4), with a 123-group cross section library generated from a basic GAM-THERMOS library using two subroutines, NITAWL and XSDRNPM, in the AMPX1 (Reference 9.1-5) code package. Both the PDQ-7 and the KENO-IV calculation methods, as described above, have been benchmarked. These calculation methods, as described, were used for the WNP-2 calculation and do not contain any significant modifications.

Under normal conditions, for a center-to-center spacing of 6.5 inches between fuel assemblies with B_4C plates surrounding each stored fuel assembly, the k_{eff}, as determined using KENO, is 0.851. With the void space between the B_4C plates and the stainless steel box flooded with water, the KENO calculation yielded a lower k_{eff}. Calculation uncertainties were determined from comparison between calculation and experiments using KENO and a statistical evaluation of Monte Carlo runs. The results indicated a calculational uncertainty for the former of 0.013 Δk and for the latter 0.010 Δk at a 95% confidence level; this represents a total calculation uncertainty of 0.023 Δk. Mechanical spacing and tolerances acting in a direction close to the water gaps between adjacent racks result in a slight reactivity increase of 0.002 Δk. Production tolerances of B_4C plates result in a reactivity increase of 0.003 Δk.

To determine the effect of reduced or missing neutron absorbing material, it was assumed that one out of every twenty-five neutron absorber plates was missing. This case is extremely unlikely but shows poison variation sensitivity. The results of the calculation was an increase in reactivity of 0.015 Δk. A temperature decrease in pool water temperature to 32°F was included, i.e., 0.004 Δk. Adding the total calculational uncertainties of 0.023 Δk and the total geometric and material uncertainties of 0.024 Δk to the nominal k_{eff} results in a k_{eff} of 0.897 with a confidence level of 95%. This is well below the design basis of equal to or less than 0.95 for the normal wet condition.

Two abnormal conditions were also considered. They are (a) a dropped assembly assumed to lay across the top of a fuel rack and (b) a fuel assembly in transport in a vertical position, accidentally dropped into the water channels between racks. Of these two, the second condition is more severe. For the first condition, the end fittings on the top of the BWR assembly prohibits a spacing between the dropped assembly and the active fuel in the storage racks of less than 11.6 inches. Using the same techniques, assumptions, and uncertainties previously discussed, the second case resulted in a k_{eff} of 0.903. This is only slightly different from the conservative normal condition and within the design basis k_{eff} of equal to or less than 0.95.

9.1.2.3.2 Spent Fuel Storage Rack Structural Design

In accordance with Regulatory Guide 1.13, Revision 1, the spent fuel storage racks are designed to Seismic Category I requirements. Structural integrity of the racks when subjected to normal and abnormal loads, as well as seismic loads, is demonstrated in accordance with the load requirements and acceptance criteria of NRC Standard Review Plan 3.84. The loads considered are:

a. Dead loads which are the dead weight of the rack and fuel assemblies, and hydrostatic loads.

b. Live loads, i.e., the effect of lifting empty racks during installation.

c. Thermal loads, which include the uniform thermal expansion of the racks due to increases in average pool temperature, and a thermal gradient between adjacent storage locations.

d. The seismic forces of the OBE and SSE.

e. Accidental drop of a fuel assembly from the maximum possible height consistent with fuel handling operations, which is four feet above the top of fuel rack.

f. Postulated stuck fuel assembly causing an upward force of 1200 lbs., equal to the fuel grapple load limit to be exerted on the assembly upon attempted withdrawal.

The spent fuel storage racks were analyzed for six combinations of these loads using elastic working stress design methods. The combinations are:

a. Dead loads plus live loads,

b. Dead loads plus OBE,

c. Dead loads plus thermal loads plus OBE,

d. Dead loads plus thermal loads plus SSE,

e. Dead loads plus thermal loads plus fuel assembly drop,

f. Dead loads plus thermal loads plus stuck fuel assembly.

Live loads are not included in load combinations b. through f. The only live load on the rack is that due to lifting of the racks which is performed with the racks empty. In all cases the loads were below the strength limits, which were determined from Part I of the AISC "Specification for the Design, Fabrication and Reaction of Structural Steel for Buildings," February 12, 1969 and Supplements 1, 2 and 3. (Supplement 3 was effective June 12, 1974.)

Individual fuel racks are welded into either 1 x 2 or 2 x 2 rack arrays or "super modules". These super modules are then attached to the fuel pool walls at two elevations by adjustable seismic restraints which are essentially large turnbuckles. These restraints are designed and positioned to eliminate any significant thermal growth loads on the walls.

The fundamental frequency of lateral vibration of the welded rack array yielding the lowest frequency was determined using the STARDYNE3 (Reference 9.1-6) computer program. The model, consisting of beam and plate elements and lumped masses, represents a three-dimensional 2x2 rack array. The model has an array of beams, representing fuel storage cans, connecting the upper and lower grids, and resting on a base grid which is held off the pool floor by support feet. The rack is restrained laterally by two levels of springs, representing the seismic restraints. The stiffnesses of the fuel assemblies and B_4C neutron absorber plates conservatively were neglected. However, the mass of the fuel assemblies, as well as the mass of the B_4C plates and an effective mass of water was considered to be uniformly distributed over the height of the fuel boxes. The results of the STARDYNE analysis showed a lateral natural frequency of 14.7 Hz. A dynamic analysis was performed using the horizontal floor response spectra (damping 1/2% of critical).

The fundamental frequency of vertical vibration of the rack was also determined using the STARDYNE computer program. The same model replacing lateral mass with vertical mass was utilized. In this case, since the fuel rests on the base frame, the entire mass of the fuel was lumped at the base grid. Since the calculated frequency was 50.9 Hz and the vertical floor response spectra (damping 1/2% of critical) showed constant acceleration at frequencies in excess of 18 Hz, the effects of the vertical accelerations were considered using the zero-period acceleration in a static analysis. The lateral and vertical loads were considered to be acting simultaneously.

In the general seismic/structural analysis of the fuel racks, the mass of a fuel assembly is assumed to be uniformly distributed along the length of each of the fuel storage cans. Since a maximum gap on the order of 3/8" exists between the side of a fuel assembly and the can (when the fuel is not encased in a channel), the fuel will actually move within the can during a seismic event and cause impact loads to be transmitted to the fuel rack restraints. The effects of this fuel can interaction are determined using a simplified finite element model of the rack and fuel. A nonlinear dynamic analysis is performed utilizing the ANSYS computer program. Details of this analysis are given in NUS Corporation Technical Report #2060, entitled "Fuel-Can Interaction Analysis," October, 1977.

Using the given loads, load combinations and analytical methods, stresses were calculated at critical sections of the rack and compared to the structural acceptance criteria. In all cases, the calculated stress did not exceed the allowable stress.

To assure the integrity of the spent fuel storage racks, specially designed control samples, consisting of B_4C plates sealed in storage tube stainless steel material and fabricated using the same procedures employed for the production of the fuel racks, will be placed in a readily accessible position in the spent fuel pool. These samples are subjected to periodic visual examination and neutron attenuation tests, if visual examination indicates evidence of corrosion.

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 9.1-17. 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 four feet above the top of the fuel rack). See 9.1.2.3.2.e.

Transfer of fuel assemblies between the reactor well and the spent fuel pool is performed with the refueling platform (see 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 (lbs)	Auxiliary Fuel Hoist (lbs)
Load limiting switch	1200	1000
Load sensing switch	485	485
Stall torque or 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 accidently 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. The pool cask storage area is separated from the spent fuel pool by a wall over which the spent fuel is transferred.

For removal of spent fuel from the plant, a shipping cask is lowered into the cask area. Transfer of fuel to the cask is made over the wall between the spent fuel pool 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 9.1-17. 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 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 assures 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 backflow. 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 9.1.3 for additional evaluation of continuous cooling capabilities of the spent fuel pool cooling and cleanup system.

9.1.2.3.4 Spent Fuel Rack Design Features

The rack, rack modules and restraints are all stainless steel, as is the spent fuel pool liner, to minimize the potential for galvanic corrosion. Stainless steel has also been shown to be compatible with spent fuel pool water and the stored assemblies.

The fuel rack base is elevated above the floor to assure adequate flow under the rack in each fuel assembly. Analyses have been performed and show that sufficient flow is induced by natural convection to preclude local boiling in the hottest storage location.

The analyses were based on the following assumptions:

a. The fuel element inlet temperature is the mixed hot design temperature of the pool.

b. A hot assembly peaking factor of 1.74 is applied to the core average assembly energy release rate of 5.3×10^4 Btu/hr.

c. The maximum local peaking factor is 2.49, giving a maximum local heat flux of 1334 Btu/hr-ft^2.

d. A film coefficient of 31 Btu/hr-ft^2-^oF is based on pure conduction through a stagnant boundary layer at the fuel rod surface.

e. The downcomer region feeds 12 assemblies in a row, each assumed to be generating the maximum heat rate defined in Assumption b.

f. One dimensional fluid flow analysis applies.

During full core offload with the bulk pool temperature at a design value of 150^oF, the mixed temperature of the water exiting from the hottest storage location is less than 181^oF. This is 58^oF below the local saturation temperature of 239^oF, indicating that adequate margin to bulk boiling exists. Under normal operating conditions, the fuel rod surface temperature calculated on the basis of the heat flux and film coefficient defined above is more than 14^oF below the local saturation temperature. Local boiling is thus precluded.

The fuel racks are designed, constructed and fabricated with a high degree of reliability and integrity. A list of the industry design codes and standards used for the spent fuel storage racks is given below.

Design Codes

a. AISC Manual of Steel Construction, 7th Edition, 1970.

b. ASME Boiler and Pressure Vessel Code Section III-1971, Nuclear Power Plant Components. (Tables I-2.2, I-5.0, I-6.0 are used for yield strength values, coefficients of thermal expansion, and moduli of elasticity.)

c. AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, February 12, 1969 and Supplements 1, 2 and 3. (Supplement 3 effective 6/12/74.)

d. ASTM Specification A 240-75a, Specification for Heat-Resisting Chromium and Chromium-Nickle Stainless Steel Plate Sheet and Strip for Fusion-Welded Unfired Pressure Vessels.

e. ASTM Specification A 320-74, Specification for Alloy Steel Bolting Materials for Low-Temperature Service.

f. AWS A 5.9, Corrosion Resisting Chromium and Chromium Nickel Steel Welding Rods and Bare Electrodes.

g. ASME Boiler and Pressure Vessel Code Section IX-1974, Welding and Brazing Qualifications.

9.1.2.3.5 Spent Fuel Storage Facilities Design

The spent fuel storage pool is designed to Seismic Category I requirements to prevent earthquake damage to the stored fuel. The exterior wall and roof of the reactor building is designed as a low-leakage barrier to confine potential airborne radiation contamination within the reactor building and the exhaust air-treatment systems.

Release of radioactive products from damaged or failed fuel in the spent fuel pool will be detected by a high level gamma radiation monitor located in the fuel pool vicinity. Backup detection is provided by high radioactivity monitors in the reactor building ventilation system (See 9.4.2). These also will initiate reactor building ventilation system isolation and operation of the standby gas treatment system (See 6.5.1) to block potential leakage of contaminated air to the environment. This instrument has a range of 100 to 10^6 mrem/hr with remote readout and alarm in the main control room. In the event of high levels of radioactive releases from the fuel pool area, this monitor will send a signal to initiate automatic isolation of the reactor building ventilation system (See 9.4.2) and operation of the standby gas treatment system (See 6.5.1). (See Chapter 15 for radiological considerations.)

The spent fuel storage pool contains water to a total depth of 37 feet 9 inches. A low water level alarm is provided in the control room in the event of loss of pool water. As a backup, flow alarms are provided in the drain lines to detect leakage in the reactor vessel to drywell seal, drywell to concrete seal, and fuel pool gate. An adequate fuel pool water level can be maintained, even in the unlikely event of a pipe break between the skimmer surge tanks and the fuel pool cooling and cleanup system pumps, since fuel pool discharge to the skimmer surge tanks is by overflow only. A pipe break would drain the skimmer surge tank but would not reduce the spent fuel storage level. A check valve in each supply pipe from the fuel pool cooling and cleanup system prevents siphon backflow to the cleanup system. Provision is also made so that standby service water can be used as backup for fuel pool makeup upon failure of the normal makeup system. This connection is capable of supplying enough water to prevent the uncovering of the spent fuel. By use of the standby service water as makeup, the fuel pool will be cooled by evaporation of the pool water. The RHR system operating in parallel with the fuel pool cooling system is available during shutdown conditions to remove abnormal heat loads in the fuel pool.

The reactor building below the refueling floor is designed to be tornado proof. Tornado missiles below this elevation cannot impair the structural integrity of the pool. Missiles which could reach the pool from above are small enough that they could not impair the structural integrity of the pool. Metal siding above the refueling floor blows out during a tornado. All large objects on the refueling floor are secured so that they cannot be carried into the fuel pool. (See 3.3 and GE Topical Report, APED-5696, Tornado Protection for the Spent Fuel Storage Pool.)

Leak protection channels are provided on the concrete sides of the spent fuel storage pool liner. Surveillance of flow indications from these leak channels permits early determination and localization of any leakage.

9.1.2.3.6 Radiological Considerations

9.1.2.3.6.1 Normal Operation

Three sources of exposure to personnel in the area of the spent fuel storage pool are considered:

a. Direct dose from the stored fuel

b. Dose from the radionuclides in the spent fuel pool water

c. Dose from airborne tritium

Direct dose from the stored fuel is negligible due to the height of water above the storage positions. Calculations indicate a direct dose from stored fuel of less than 1×10^{-5} millirem per hour.

Most of the personnel exposure in the area of the spent fuel storage pool comes from the radionuclide inventory in the water. Estimates of the dose from this source were calculated to be 3.5 millirem per hour during the first refueling, increasing to 6.9 millirem per hour during the eleventh refueling. These calculated values are given in Table 9.1-8.

The dose from airborne tritium is limited by spent fuel storage pool water temperature limitations as detailed in 9.1.3 and by the fuel pool exhaust ventilation described in 9.4.2.

9.1.2.3.6.2 Radiological Consequences of Accidents

The accidents considered are a fuel-handling accident in the spent fuel storage pool, impact of heavy objects and seismic events. Refer to Chapter 15 for discussion.

The fuel-handling accident involves the drop of a fuel assembly from the maximum height probable onto the fuel storage racks. As discussed in 9.1.2.3.2 the storage racks are designed to withstand such an impact and protect the stored fuel.

The storage racks, storage pool and associated equipment, and the reactor building in which the pool is located, are designed to remain functional through and after a Safe Shutdown Earthquake.

The impact of heavy objects on the racks has been considered in the design. The reactor building below the refueling floor is tornado proof and tornado missiles that could reach the pool from above that level are small enough that the structural integrity of the pool (and racks therein) could not be impaired. The drop of a spent fuel shipping cask is precluded by redundancy features of the crane and other design features discussed in 9.1.2.3.3.

9.1.2.3.7 Conclusions

From the foregoing analyses, it is concluded that the spent fuel storage arrangement and design comply with Revision 1, the objectives set forth in Regulatory Guide 1.13.

The channel gauging fixture consists basically of a frame, gauging plate, and gauging block. The gauging plate is shimmed to correspond to the outside dimension of a usable fuel channel. The gauging block conforms to the inside dimension of the lower end of a usable fuel channel.

The channel gauging fixture is installed in the vertical position, between the two fuel preparation machines and hangs from the fuel storage pool curb.

9.1.4.2.3.8 General Purpose Grapple

The general purpose grapple, Figure 9.1-11, is a handling tool used generally with the fuel. The grapple can be attached to the reactor building auxiliary hoist, jib crane, and the auxiliary hoists on the refueling platforms. The general purpose grapple is used to remove new fuel from the vault, place it in the inspection stand, and transfer it to the fuel pool. It can be used to handle fuel during channeling.

9.1.4.2.4 Servicing Aids

General area underwater lights are provided with a suitable reflector for illumination. Suitable light support brackets are furnished to support the lights in the reactor vessel and to allow the light to be positioned over the area being serviced independent of the platform. Local area underwater lights are small diameter lights for additional illumination. Drop lights are used for illumination where needed.

A radiation hardened designed portable underwater closed circuit television camera is provided. The camera may be lowered into the reactor vessel and/or fuel storage pool to assist in the inspection and/or maintenance of these areas. The camera is also equipped with a right angle lens to allow viewing at 90°.

A general purpose, plastic viewing aid is provided to float on the water surface to permit better visibility. The sides of the viewing aid are brightly colored to allow the operator to observe it in the event of filling with water and sinking. A portable, submersible type, underwater vacuum cleaner is provided to assist in removing crud and miscellaneous particulate matter from the pool floors, or the reactor vessel. The pump and the filter unit are completely submersible for extended periods. The filter "package" is capable of being remotely changed, and the filters will fit into a standard shipping container for off-site burial. Fuel pool tool accessories are also provided to meet servicing requirements.

A fuel sipper is provided. This is to be used to detect defective fuel assemblies during open vessel periods while the fuel is in the core. The fuel sipper head isolates individual fuel assemblies by sealing the top of the fuel channel and pumping water from the bottom of the fuel assembly through the fuel channel, to a sampling station, and return to the primary coolant system. After a "soaking" period a water sample is obtained and is radio-chemically analyzed.

9.1.4.2.5 Reactor Vessel Servicing Equipment

The essentiality and safety classifications, the quality group, and the seismic category for this equipment is listed in Table 9.1-6. Following is a description of the equipment designs in reference to that table.

9.1.4.2.5.1 Reactor Vessel Service Tools

These tools are used when the reactor is shut down and the reactor vessel head is being removed or reinstalled. Tools in this group are:

Stud Handling Tool

Stud Wrench

Nut Runner

Stud Thread Protector

Thread Protector Mandrel

Bushing Wrench

Seal Surface Protector

Stud Elongation Measuring Rod

Dial Indicator Elongation Measuring Device

Heat Guide Cap

These tools are designed for a 40 year life in the specified environment. Lifting tools are designed for a safety factor of 5 or better with respect to the ultimate strength of the material used. When carbon steel is used, it is either hard chrome plated, parkerized, or coated with an acceptable paint.

9.1.4.2.5.2 Steam Line Plug

The steam line plugs are used during reactor refueling or servicing; they are inserted in the steam outlet nozzles from inside of the reactor vessel to prevent a flow of water from the reactor well into the main steam lines during servicing of safety relief valves, main steam isolation valves, or other components of the main steam lines. (The reactor water level is raised to the refueling level during servicing.)

The steam line plug design provides two seals of different types. Each one is independently capable of holding full head pressure. The equipment is constructed of non-corrosive materials. All calculated safety factors are 5 or greater. The plug body is designed in accordance with the "Aluminum Construction Manual" by the Aluminum Association.

9.1.4.2.5.3 Shroud Head Bolt Wrench

This is a hand held tool for operation of shroud head bolts. It is designed for a 40 year life, and it is made of aluminum to be easy to handle and to resist corrosion. Testing has been performed to confirm the design.

9.1.4.2.5.4 Head Holding Pedestal

Three pedestals are provided for mounting the reactor vessel head on the refueling floor. The pedestals have studs which engage three evenly spaced stud holes in the head flange. The flange surface rests on replaceable wear pads made of aluminum. When resting on the pedestals, the head flange is approximately 3 feet above the floor to allow access to the seal surface for inspection and O-ring replacement.

The pedestal structure is a carbon steel weldment, coated with an approved paint. It has a base with bolt holes for mounting it to the concrete floor. The structure is designed in accordance with "The Manual of Steel Construction" by AISC.

9.1.4.2.5.5 Head Nut and Washer Rack

The RPV head nut and washer rack is used for transporting and storing up to 6 nuts and washers. The rack is a box shaped aluminum structure with dividers to provide individual compartments for each nut and washer. Each corner has a lug and shackle for attaching a 4-leg lifting sling.

The rack is designed in accordance with the "Aluminum Construction Manual" by the Aluminum Association, and for a safety factor of 5.

9.1.4.2.5.6 Head Stud Rack

The head stud rack is used for transporting and storage of 8 reactor pressure vessel studs. It is suspended from the auxiliary building crane hook when lifting studs from the reactor well to the operating floor.

The rack is made of aluminum to resist corrosion.

9.1.4.2.5.7 Dryer and Separator Sling

The dryer and separator sling is a lifting device used for transporting the steam dryer or the shroud head with the steam separators between the reactor vessel and the storage pools. The sling consists of a cruciform shaped structure which is suspended from a hook box with four wire ropes and turnbuckles. The hook box, with two hook pins, engages the reactor service crane sister hook. On the end of each arm of the cruciform is a socket with a pneumatically operated pin for engaging the four lift eyes on the steam dryer or shroud head.

The sling has been designed such that one hook pin and one main beam of the cruciform is capable of carrying the total load and so that no single component failure will cause the load to drop or swing uncontrollably out of an essentially level attitude.

The safety factor of 11 lifting members is 5 or better in reference to the ultimate breaking strength of the material. The structure is designed in accordance with "The Manual of Steel Construction" by AISC. The completed assembly is proof tested at 125 percent or greater of rated load and all structural welds are magnetic particle inspected after load test.

9.1.4.2.5.8 Head Strongback

The RPV head strongback is used for lifting both the pressure vessel head and the drywell head. It is a cruciform shape with four equally spaced lifting points on the ends of the arms. In the center it has a hook box which engages with two pins to the reactor service crane sister hook.

The strongback is designed such that one leg of the cruciform will support the rated load and such that no single component failure will cause the load to drop or swing uncontrollably out of an essentially level attitude. The structure is designed in accordance with "The Manual of Steel Construction" by AISC. All welding is in accordance with the ASME Boiler and Pressure Vessel Code Section IX. A safety factor of 5 or greater in reference to the ultimate material strength is used for the design. The completed assembly is proof tested at 125 percent rated load. After the load test, all structural welds are magnetic particle inspected.

9.1.4.2.5.9 Service Platform

The service platform is provided to facilitate maintenance work on reactor internals. It provides a working platform for people and hand guided tools, and it also has provision for supporting a jib crane. The service platform is supported by four wheels which run on a circular track resting on the vessel flange and confined by the vessel closure studs.

The service platform is Seismic Class II equipment, and it has been designed for 0.75 g horizontal and 0.00 g vertical. The physical size of the device is such that it cannot enter the reactor pressure vessel.

The structure design is in accordance with "The Manual of Steel Construction" by AISC. Materials are in accordance with ASTM Standards. Welding is in accordance with ASME Boiler and Pressure Vessel Code Section IX or AWS D1.1 structural welding.

The electrical system is in accordance with ANSI-ANS C1, National Electrical Code, and NEMA Publications No. IC1 and MG1.

Painting and surface preparation is in conformance with SSPC and in compliance with Regulatory Guide 1.54.

9.1.4.2.5.10 Service Platform Support

The service platform support serves as a sealing surface protector for the reactor vessel flange, and as a track for the service platform. It has continuous vertical support on the vessel flange, and horizontally it is confined by the vessel studs by strapping to the outer edge of the flange. The service platform support is made from aluminum and all welding is done in accordance with AWS Code D1.0.

9.1.4.2.5.11 Steam Line Plug Installation Tool

The steam line plug installation tool is suspended from the building crane auxiliary hook for transporting and installing the steam line plugs in the steam line nozzles of the reactor vessel. This tool is made of aluminum; it is designed for a safety factor of 5, and is in accordance with "Aluminum Construction Manual" by the Aluminum Association.

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 tube into the vessel. The incore dry tube is then decoupled from the strongback and is guided into place while being supported by the instrument handling tool. Final incore insertion is accomplished from below the reactor vessel. The instrument handling tool is attached to the refueling platform auxiliary hoist and is used for removing and installing fixed incore dry tubes as well as handling neutrons source holders and the source range monitor/intermediate range monitor dry tubes.

Each incore instrumentation guide tube is sealed by an O-ring on the flange and in the vent that the seal needs replacing, an incore guide tube sealing 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 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 which 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 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 verify hook engagement and grapple load, prevent unsafe operation over the vessel during control rod movements, and limit vertical travel of the grapple. Two 1000 pound capacity auxiliary hoists, one main trolley mounted and one auxiliary trolley mounted, are provided for servicing such as LPRM replacement, fuel support replacement, jet pump servicing, and control rod replacement. The grapple in its fully retracted position provides 8 feet 6 inches minimum water shielding over the active fuel during transit.

9.1.4.2.8 Storage Equipment

Specific storage equipment is used for new, used, or spent fuel, for control rods and for other core components.

Specially designed fuel storage racks are provided in the spent fuel pool and in the new fuel vault. Additional storage equipment is listed on Table 9.1-4. For fuel storage racks description and fuel arrangement, see 9.1.1 and 9.1.2.

Fuel sipping heads, panels and containers are separate pieces of equipment used for out-of-core wet sipping at any time. They are used to isolate a fuel bundle while circulating water through the fuel bundle in a closed system. The containers cannot be used for transporting a fuel bundle. The bail on the container head is designed not to fit into any of the grapples.

9.1.4.2.9 Under Reactor Vessel Servicing Equipment

The primary function of the under reactor vessel servicing equipment is to: (1) remove and install control rod drives, (2) service thermal sleeve and control rod guide tube, and (3) install and remove the neutron detectors. Table 9.1-4 lists the equipment and tools required for servicing. Of the equipment listed, the equipment handling platform and the control rod drive handling equipment are powered electrically.

The control rod drive handling equipment is used for the removal and installation of the control rod drives from their housings. This equipment is designed in accordance with the requirements of National Electrical Manufacturers Association (NEMA, MG1: Motor and Generator Standards), American National Standards Institute Standards (ANSI C1, National Electric Code), Occupational Safety and Health Administration (OSHA, 1910.179), American Instutute of Steel Construction (AICS, Manual of Steel Construction). All lifting components are equipped with adequate brakes or gearing to prevent uncontrolled movement upon loss of power or component failure.

The equipment handling platform is also powered electrically and provides a working surface for equipment and personnel performing work in the under vessel area. It is a polar platform capable of 360° rotation. This equipment is designed in accordance with the applicable requirements of OSHA (Vol. 37, No. 202, Part 191 ON), AISC, and ANSI C1, (National Electric Code).

The spring reel is used to pull the incore guide tube seal or incore detector into the incore tube during incore servicing.

The thermal sleeve installation tool locks, unlocks, and lowers the thermal sleeve from the control rod drive guide tube.

Incore flange seal test plug is used to determine the pressure integrity of the incore flange O-ring seal. It is constructed of non-corrosive material. The key bender is designed to install and remove the anti-rotation key that is used on the thermal sleeve.

9.1.4.2.10 Description of Fuel Transfer

The fuel handling system provides a safe and effective means for transporting and handling fuel from the time it reaches the plant until it leaves the plant after post-irradiation cooling. The previous subsection has described the equipment and methods utilized in fuel handling. The following paragraphs describe the integrated fuel handling system which ensures that the design bases of the fuel handling system is satisfied.

9.1.4.2.10.1 Arrival of Fuel on Site

The new fuel arrives at the site in special shipping containers. Each shipping container has space for two fuel bundles. The shipping container consists of a metal inner container positioned by means of cushioning materials within a wooden outer container. These containers with their contents are designated "RA Series" packages, and meet all NRC requirements. Both inner and outer containers are reusable.

After arrival of the fuel shipment, the fuel containers are off-loaded from the carrier at a pre-designated fuel receiving area.

The outer wooden containers will be inspected for evidence of damage during shipping. If there are indications of rough handling or damages, further investigation, for possible damage to fuel bundles, will be performed.

With the use of appropriate lifting equipment and rigging, the metal inner container is removed and placed in a horizontal position. The metal container is first vented and then the cover is removed to expose the fuel bundles. The fuel bundles are then visually inspected for any obvious damage that may have occurred during shipment, and a radiation survey for contamination is made.

Special bundle holddown bars will be installed on the metal container in preparation for raising it to the vertical position.

Using the reactor building auxiliary hoist with a spreader bar sling which is attached to existing lifting lugs at the top of container, the container is raised to the vertical position and secured. The fuel shipping containers are then hoisted to the refueling floor.

All incoming fuel shall be inspected prior to being placed in the reactor vessel. Preferably, the inspection will be done before the fuel is stored. However, depending on specific plans for the initial fueling and/or subsequent refueling, fuel may be stored until a more desirable time for inspection.

Inspections are performed with the fuel bundles in the vertical position, securely seated in the fuel inspection stand. Inspections are performed, by qualified inspectors, in accordance with detailed procedures and inspection check-off sheets.

After satisfactory inspection, the fuel bundle held by the general purpose grapple is released from the fuel inspection stand. It is then transferred to a storage rack in the new fuel vault or to a storage rack in the spent fuel storage pool by the reactor building crane auxiliary hoist or jib crane. The fuel bundle may or may not be channeled prior to storage.

If a fuel bundle fails inspection, it is tagged and discrepancies are noted on the inspection check-off sheet. Action will be taken to repair the fuel bundle or it will be set aside for disposition by the fuel manufacturer.

9.1.4.2.10.2 Refueling Procedure

Fuel handling procedures are described below and shown visually in Figure 9.1-13 through Figure 9.1-16.

The refueling floor layout is shown in Figure 9.1-17 and component drawings of the principal fuel handling equipment are shown in Figures 9.1-5 through 9.1-12.

The fuel handling process takes place primarily on the refueling floor above the reactor. The principal locations and equipment are shown on Figure 9.1-13. The reactor and fuel pool are connected to each other by slot (A) which is open during reactor refueling. At other times the slot is closed by means of blocks and gates, which make watertight barriers.

New fuel, in shipping crates, is brought up to the refueling floor through the hatches; and spent fuel, in a shipping cask, is lowered through the hatches to a truck or rail car near grade level.

The handling of new fuel on the refueling floor is illustrated in Figure 9.1-14. The transfer of the bundles between the crate (C) and the new fuel inspection stand (D) and/or the new fuel storage vault (E) is accomplished using the 15-ton auxiliary hoist of the reactor building crane equipped with a general-purpose grapple. The fuel bundle cannot be handled horizontally without support, so the crate is placed in an almost vertical position before being opened. The top and front of the crate are opened and the bundles removed in a vertical position.

Either the auxiliary hoist or the jib crane is also used with a general-purpose grapple to transfer new fuel from the new fuel vault or inspection stand to a storage rack position in the fuel pool. From this point on, the fuel is handled by the telescoping grapple on the refueling platform.

The storage racks in both the vault and the fuel pool hold the fuel bundles or assemblies vertical, in an array which is subcritical under all possible conditions.

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 general-purpose grapples and the fuel grapple of the refueling platform have redundant hooks and an indicator which confirms positive grapple engagement.

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 fully retracted state, maintains adequate water shielding over the fuel being handled.

The reactor refueling procedure is shown schematically in Figure 9.1-15. The refueling platform (G) 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. Here it is placed in one of the fuel prep machines (K).

An operator, using a long-handled wrench, removes the screws and springs from the top of the channel. The channel is then held, while a carriage lowers the fuel bundle out of the channel. The channel is then moved aside, and the refueling platform grapple carries the bundle and places it in a storage rack. The channel handling boom hoist (L) moves the channel to storage, if appropriate.

In actual practice, channeling and dechanneling may be performed in many sequences, depending on whether a new channel is to be used, or a used channel is to be installed on a new bundle and returned to the core. The channel racks are conveniently located near to the fuel prep machines, for temporary storage of channels which are to be reused.

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.

When spent fuel is to be shipped, 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 cask area of the pool, and lowers it into the cask (M). When the cask is loaded, the building crane sets the cask cover (N) on the cask, 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 at near grade level.

Additional detailed information is provided below.

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 will be delivered to a receiving station within the reactor building. The crates are unloaded from the transport vehicle and examined for damage during shipment. The crate dimensions are approximately 32" x 32" x 18 feet long. Each crate contains two fuel bundles supported by an inner metal container. Shipping weight of each unit is approximately 3000 pounds. The receiving station will include a separate area where the crate covers can be removed and the inner metal container can be removed from the crate. Both inner and outer shipping containers are reuseable. Handling during uncrating is to be accomplished by traveling crane, MT-CRA-13.

9.1.4.2.10.2.1.2 Channeling New Fuel

New fuel is unloaded from the new fuel vault and transported to the fuel racks in the fuel pool. Usually channeling new fuel is done concurrently with dechanneling spent fuel. Two fuel preparation machines are located in the fuel pool, one used for dechanneling spent fuel and the other to channel new fuel. The procedure is as follows: Using a jib crane and the general purpose grapple a spent fuel bundle is transported to the fuel prep machine. The channel is unbolted from the bundle using the channel bolt wrench. The channel handling tool is fastened to the top of the channel and the fuel prep machine carriage is lowered removing the fuel from the channel. The channel is then positioned over a new fuel bundle located in the fuel prep machine #2 and the process reversed. The channeled new fuel is stored in the pool storage racks ready for insertion into the reactor.

9.1.4.2.10.2.1.3 Equipment Preparation

Prior to the plant shutdown for refueling, all equipment will be placed in readiness. All tools, grapples, slings, strongbacks, stud tensioners, etc. will be given a thorough check and any defective (or well worn) parts will be replaced. Air hoses on grapples will be routinely leak tested. Crane cables will be routinely inspected. All necessary maintenance and interlock checks will be performed to assure no extended outage due to equipment failure.

The incore flux monitors, in their shipping container, will be on the refueling floor. The channeled new fuel and the replacement control rods will be ready in the storage pool.

9.1.4.2.10.2.2 Reactor Shutdown

The reactor is shut down according to a prescribed procedure. During cooldown the reactor pressure vessel is vented and filled to above flange level to equalize cooling. The reactor well shield plugs are removed using the reactor building crane and the supplied slings.

This operation can be immediately followed by removal of the canal plugs and the slot plugs. The outer fuel pool gate is also removed at this time. The gate sling is attached to the gate lifting lugs, and the reactor building crane lifts the gate and places it on the fuel pool storage lugs.

9.1.4.2.10.2.2.1 Drywell Head Removal

Immediately after removal of the reactor well shield plugs, the work to unbolt the drywell head can begin. The drywell head will be attached by removable bolts. The bolts are unscrewed and together with their nuts are removed and stored.

The head strongback is attached to the unbolted drywell head and lifted by the overhead reactor building crane to its appointed storage space on the refueling floor. The drywell seal surface protector is installed before any other activity proceeds in the reactor well area.

9.1.4.2.10.2.2.2 Reactor Well Servicing

When the drywell head has been removed, an array of piping is exposed that must be serviced. Various vent piping penetrations through the reactor well must be removed and the penetrations made water tight. Vessel head piping and head insulation must be removed and transported to storage on the refueling floor.

Water level in the vessel is now brought to flange level in preparation for head removal.

9.1.4.2.10.2.3 Reactor Vessel Opening

9.1.4.2.10.2.3.1 Vessel Head Removal

The stud tensioner is transported by the reactor building crane and positioned on the reactor vessel head. Each stud is tensioned and its nut loosened in a series of 2-3 passes. When the nuts are loose, they are backed off using a nut runner until only a few threads remain engaged. The vessel nut handling tool is engaged in the upper part of the nut and the nut is rotated free from the stud. The nuts and washers are placed in the racks provided for them and transported to the refueling floor for storage. With the nuts and washers removed, the vessel stud protectors and vessel head guide caps are installed.

The head strongback, transported by the reactor building crane, is attached to the vessel head and the head transported to the head holding pedestals on the refueling floor. The head holding pedestals keep the vessel head elevated to facilitate inspection and "O" ring replacement.

The six studs in line with the fuel transfer canal are removed from the vessel flange and placed in the rack provided. The loaded rack is transported to the refueling floor for storage.

9.1.4.2.10.2.3.2 Dryer Removal

The dryer-separator sling is lowered by the reactor building crane and attached to the dryer lifting lugs. The dryer is lifted from the reactor vessel and transported to its storage location in the dryer-separator storage pool adjacent to the reactor well. The dryer is transported in air. However, if the dryer should become highly contaminated, the reactor well and storage pool can be flooded and a wet transfer effected.

9.1.4.2.10.2.3.3 Separator Removal

In preparation for separator removal, the service platform and service platform support are installed on the vessel flange. From the service platform work area, the four main steam lines are plugged from inside the vessel using the furnished plugs for this duty. Servicing of the safety and relief valves can thus be accomplished without adding to the critical refueling path time. Working from the service platform, the separator is unbolted using the shroud head bolt wrenches furnished.

When the unbolting is accomplished, the service platform is removed and stored on the refueling floor. The service platform support remains on the vessel flange during the remainder of the refueling outage and acts as the flange seal surface protector.

The dryer-separator sling is lowered into the vessel and attached to the separator lifting lugs. The water in the reactor well and in the dryer-separator storage is raised to fuel pool water level, and the separator is transferred underwater to its allotted storage place in the adjacent pool.

9.1.4.2.10.2.3.4 Fuel Bundle Sampling

During reactor operation, the core off-gas radiation level is monitored. If a rise in off-gas activity has been noted, the reactor core will be sampled during shutdown to locate any leaking fuel assemblies. The fuel sampler or sipper rests on the channels of a four bundle array in the core. An air bubble is pumped into the top of the 4 fuel bundles and allowed to stay about 10 minutes. This stops water circulation through the bundles and allows fission products to concentrate if a bundle is defective. After 10 minutes, a water sample is taken for fission product analysis. If a defective bundle is found, it is taken to the fuel pool and if required, may be stored in a special defective fuel storage container to prevent the spread of contamination in the pool.

9.1.4.2.10.2.4 Refueling and Reactor Servicing

The remaining gate isolating the fuel pool from the reactor well is now removed thereby interconnecting the fuel pool, the reactor well, and the dryer-separator storage pool. The actual refueling of the reactor can now begin.

9.1.4.2.10.2.4.1 Refueling

The refueling pattern will depend on various factors such as fuel performance, plant loading, and fuel design. Immediately prior to a refueling outage, detailed procedures concerning fuel management will be written and approved for that particular outage.

Detailed procedures will also be developed for various refueling tasks such as fuel receipt, fuel inspection, fuel transfer from vault to pool, fuel movements within the pool, fuel movements within the reactor, or between the spent fuel pool and the reactor, as well as procedures for handling of other core components.

During a normal equilibrium outage, approximately 25% of the fuel is removed from the reactor vessel, 25% of the fuel is shuffled in the core (generally from peripherial to center locations) and 25% new fuel is installed. The actual fuel handling is done with the fuel grapple which is an integral part of the refueling platform. The platform runs on rails over the fuel pool and the reactor well. In addition to the fuel grapple, the refueling platform is equipped with 2 auxiliary hoists which can be used with various grapples to service other reactor internals.

To move fuel, the fuel grapple is aligned over the fuel assembly, lowered and attached to the fuel bundle bail. The fuel bundle is raised out of the core, moved through the refueling slot to the fuel pool, positioned over the storage rack and lowered to storage. Fuel is shuffled and new fuel is moved from the storage pool to the reactor vessel in the same manner.

9.1.4.2.10.2.5 Vessel Closure

The following steps, when performed, will return the reactor to operating condition. The procedures are the reverse of those described in the proceeding sections. Many steps are performed in parallel and not in the sequences listed.

a. Install inner fuel pool gate.

b. Core verification. The core position of each fuel assembly must be verified to assure the desired core configuration has been attained.

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

d. Replace separator.

e. Drain dryer-separator storage pool and reactor well.

f. Decontaminate reactor well.

g. Install service platform, bolt separator, and remove the 4 steam line plugs. Return the service platform and platform support to storage on refueling floor.

h. Remove drywell seal surface covering.

i. Open drywell vents, install vent piping.

j. Replace fuel pool outer gate.

k. Replace steam dryer.

l. Decontaminate dryer-separator storage pool.

m. Replace vessel studs.

n. Replace slot plugs.

o. Install reactor vessel head.

p. Install vessel head piping and insulation.

q. Replace dryer-separator canal plugs.

r. Hydro-test vessel, if necessary.

s. Install drywell [illegible]d.

t. Install reactor well shield plugs.

u. 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 incore flux monitors are calibrated.

9.1.4.2.10.3 Departure of Spent Fuel From Site

Spent fuel is shipped from the plant in an NRC licensed shielded cask.

The reactor building and cask handling facilities (See 9.1.4.2.2 for description of the crane) are designed to handle the casks the size of the G.E. IF-300 and larger casks of the future.

The following description of the spent fuel departure is based on the licensed G.E. 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 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 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 off-site transportation requirements. The remaining cask closure nuts are replaced and tightened.

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.

9.1.4.3 Safety Evaluation, Fuel Handling

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 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 is controlled by travel path (Figure 9.1-17) and interlocks that prohibit the crane from traveling over the storage area. At no time while being transported does the cask pass over any other safety related 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. 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 9.1.4.2.2.2 prevents the cask from physically contacting the floor. See 9.1.4.2.2.2 for safety features which assure crane stability during tornado and seismic excitation.

Jib cranes MT-CRA-9A, 9B, and 13 are designed and equipped for Class A1 (standby) service in accordance with 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 non-rotating 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 ½".

e. Structural members/shapes not covered by the above safety factors are designed in accordance with CMAA Specification #70, and if not covered by CMAA Specification #70, in accordance with the AISC Specification for the design, fabrication and erection of structural steel for buildings except that the allowable stress shall not exceed 80% of the AISC allowable design stress.

Jib cranes MT-CRA-9A, 9B and 13 are designed to Seismic Category I requirements (with lifted load).

The cranes are designed so as to be capable of operating within the following tolerances:

a. With all brakes adjusted for normal operation, it is possible to control the vertical movement to within ¼ inch under all conditions of loading.

b. Cranes operate through full hook lift without noticeable rotation of load.

c. With hook carrying 100 percent of rated load and lowering at full speed, the motor does not exceed 125 percent of synchronous speed.

Since jib cranes MT-CRA-9A and 9B are to be used on the reactor vessel service platform and around the fuel pool for handling new and spent fuel and other components in the work area, the following safety features are provided:

To preclude the possibility of raising radioactive materials out of the water, the cable on the jib crane hoist incorporates an adjustable removable stop which will jam the hoist cable, thereby preventing drum rotation when the end of the cable is at a preset distance below water level.

The hoist is motorized with a motorized boom and jib. The unit is equipped with two full capacity brakes, as well as adjustable up-travel limit switches. On hoisting, the first two independently adjustable switches shall automatically stop the hoist approximately 8 ft. below floor level. Continued hoisting shall be possible by operating a key lock contact up-travel override push button on the control pendant together with the normal hoisting push button. Two additional independent switches shall automatically cut hoist power at the maximum safe up-travel limit.

A mechanical force gage is supplied to automatically stop the hoist on an overload signal of 1000±50 lbs. Two additional microswitches are wired in parallel and the three leads are brought out with the power leads for connection to a platform receptacle. The two additional switches are adjusted to open at 400±50 lbs.

Safety aspects (evaluation) of the fuel servicing equipment are discussed in 9.1.4.2.3 and safety aspects of the refueling equipment are discussed throughout 9.1.4.2.7. A description of fuel transfer, including appropriate safety features, is provided in 9.1.4.2.10. In addition, the following summary safety evaluation of the fuel handling system is provided below.

The fuel prep machine removes and installs channels with all parts remaining under water. Mechanical stops prevent the carriage from lifting the fuel bundle or assembly to a height where water shielding is less than 8 feet. Irradiated channels, as well as small parts such as bolts and springs, are stored underwater. The spaces in the channel storage rack have center posts which prevent the loading of fuel bundles into this rack.

There are no nuclear safety problems associated with the handling of new fuel bundles, singly or in pairs. Equipment and procedures prevent an accumulation of more than two bundles in any location.

The refueling platform is designed to prevent it from toppling into the pools during a SSE. Redundant safety interlocks are provided as well as limit switches to prevent accidentally running the grapple into the pool walls. The grapple utilized for fuel movement is on the end of a telescoping mast. At full retraction of the mast, the grapple is eight feet below water surface, so there is no chance of raising a fuel assembly to the point where it is inadequately shielded by water. The grapple is hoisted by redundant cables inside of the mast and is lowered by gravity. A digital readout is displayed to the operator, showing him the exact coordinates of the grapple over the core.

The mast is suspended and gimbaled from the trolley, near its top, so that the mast can be swung about the axis of platform travel, in order to remove the grapple from the water for servicing and for storage.

The grapple has two independent hooks, each operated by an air cylinder. Engagement is indicated to the operator. Interlocks prevent grapple disengagement until a "slack cable" signal from the lifting cables indicates that the fuel assembly is seated.

In addition to the main hoist on the trolley, there is an auxiliary hoist on the trolley, and another hoist on its own monorail. These three hoists are precluded from operating simultaneously, because control power is available to only one of them at a time. The two auxiliary hoists have load cells with interlocks which prevent the hoists from moving anything as heavy as a fuel bundle.

The two auxiliary hoists have electrical interlocks which prevent the lifting of their loads higher than 8 feet under water. Adjustable mechanical jam-stops on the cables back up these interlocks.

In summary, the fuel handling system complies with Regulatory Guide 1.13, Revision 1, General Design Criteria 2, 3, 4, 5, 61, 62, and 63, and applicable portions of 10 CFR Part 50.

A system-level, qualitative-type failure mode and effects analysis relative to this system is discussed in 15.A.6.5.

The safety evaluation of the new and spent fuel storage is presented in 9.1.1.3 and 9.1.2.3.

9.1.4.4 Testing and Inspection Requirements

9.1.4.4.1 Testing and Inspection of Cranes

The main and auxiliary hooks of the reactor building crane are proof tested in the vertical direction with a total uniformly applied load equal to 150% of their rated capacity. Tests on the main hook are made with a load suspended. After the load tests, the hooks are checked by magnetic particle inspection and for any dimensional change. When completely assembled, the crane components (except for wire ropes), are operated to assure the accuracy of fabrication and the quality of workmanship.

After erection in the reactor building, the crane is statically tested to 125% of rated capacity (156.25 tons for the main hoist and 18.75 tons for the auxiliary hoist). The tests are performed in accordance with written procedures which include movement in all positions of hoisting, lowering and trolley and bridge travel. After completion of the static test, a full performance test with 100% of the design rated load is performed.

Operational tests and visual inspections are to be made at periodic intervals during the life of the crane to demonstrate its ability to safely perform its functions. The crane hooks are to be inspected by the dye-penetrant method. These inspections and tests will be scheduled to precede major fuel handling activities.

Jib Crane hooks (9A, 9B, and 13) shall be either magnetic particle inspected in accordance with ASTM E-109 (July, 1975) or liquid penetrant inspected in accordance with ASTM E-165 (July, 1975). Standard shop tests shall be performed on all equipment. Field tests shall be performed upon each piece of equipment, after completion of erection and installation, in order to verify the overall performance of the equipment against requirements and to check the mechanical performance of the equipment with regard to applicable specification requirements.

Periodic operational tests and inspection will be performed on the jib cranes to demonstrate their ability to perform their safety function.

9.1.4.4.2 Inspection and Testing Requirements of Refueling and Servicing Equipment

9.1.4.4.2.1 Inspection

Refueling and servicing equipment is subject to the strict controls of quality assurance, incorporating the requirements of 10 CFR Part 50, Appendix B. Components defined as essential to safety, such as the fuel storage racks and refueling platform, have an additional set of engineering specified "quality requirements" that identify safety-related features which require specific QA verification of compliance to drawing requirements.

For components classified as American Society of Mechanical Engineers (ASME) Section III, the shop operation must secure and maintain an ASME "N" stamp, which requires the submittal of an acceptable ASME quality plan and a corresponding procedural manual.

Additionally, the shop operation must submit to frequent ASME audits and component inspections by resident state code inspectors.

Prior to shipment, every component inspection item is reviewed by QA supervisory personnel and combined into a summary product quality checklist (PQL). By issuance of the PQL, verification is made that all quality requirements have been confirmed and are on record in the product's historical file.

9.1.4.4.2.2 Testing

Qualification testing is performed on refueling and servicing equipment prior to multi-unit production. Test specifications are defined by the responsible design engineer and may include; sequence of operations, load capacity and life cycles tests. These test activities are performed by an independent test engineering group and, in many cases, a full design review of the product is conducted before and after the qualification testing cycle. Any design changes affecting function, that are made after the completion of qualification testing, are re-qualified by test or calculation.

Functional tests are performed in the shop prior to the shipment of production units and include electrical tests, leak tests and sequence of operations tests.

When the unit is received at the site, it is inspected to insure no damage has occurred, during transit or storage. Prior to use and at periodic intervals, each piece of equipment is again tested to ensure the electrical and/or mechanical functions are operational.

Passive units, such as the fuel storage racks are visually inspected prior to use.

There is an operation and maintenance instruction manual for each tool that additionally requires a series of functional checks each time the unit is operated for reactor refueling or servicing.

Fuel handling and vessel servicing equipment preoperational tests are described in 14.2.12.1.12.

9.1.4.5 Instrumentation Requirements

9.1.4.5.1 Instrument Requirements - Cranes

Operation of the reactor building crane is from the cab or from the floor by radio control. Control at any one time is from one point only. Sufficient electrical circuitry in the system is provided such that no single credible electrical component failure will cause the load to drop. The reactor building crane and its safety features are described in 9.1.4.2.2.

The refueling platform contains a position-indicating system that indicates the position of the full grapple over the core. Interlocks on both the fuel grapple hoist and the auxiliary hoists are provided for safety purposes. The refueling interlocks are described and evaluated in 7.7.1.13.

For a description of jib crane safety-related interlocks, see 9.1.4.3. Limit switches are provided for all electrically operated hoists to prevent overhoisting. Pushbutton pendant controls are provided with stepped controls for multispeed motions.

9.1.4.5.2 Instrumentation Requirements - Refueling and Servicing Equipment

The majority of the refueling and servicing equipment is manually operated and controlled by the operator's visual observations. This type of operation does not necessitate the need for a dynamic instrumentation system.

However, there are several components that are essential to prudent operation that do have instrumentation and control systems.

9.1.4.5.2.1 Refueling Platform

The refueling platform has a non-safety-related X-Y-Z position indicator system that informs the operator which core fuel cell and fuel grapple is accessing. Interlocks and control room monitor are provided to prevent the fuel grapple from operating in a fuel cell where the control rod is not in the proper orientation for refueling. Refer to 7.7.1.13 for discussion of refueling interlocks.

Additionally, there are a series of mechanically activated switches and relays that provide monitor indications on the operator's console for grapple limits, hoist and cable load conditions, and confirmation that the grapple hook is either engaged or released.

A series of load cells are installed to provide automatic shutdown whenever threshold limits are exceeded on either the fuel grapple or the auxiliary hoist units.

9.1.4.5.3 Fuel Support Grapple

Although the fuel support grapple is not essential to safety, it has an instrumentation system consisting of mechanical switches and indicator lights. This system provides the operator with a positive indication that the grapple is properly aligned and oriented and that the grappling mechanism is either extended or retracted.

9.1.4.5.4 Other

Refer to Table 9.1-4 for additional refueling and servicing equipment not requiring instrumentation.

9.1.4.5.5 Radiation Monitoring

The radiation monitoring equipment for the refueling and servicing area is discussed in 11.5.2.1.2 and 12.3.4.

9.1.5 REFERENCES

9.1-1 C.L. Martin, "Lattice Physics Methods", General Electric Company, NEDO-20913.

9.1-2 WAPD-TM-678 PDQ-7 Reference Manual by W.R. Caldwell, Bettis Atomic Power Laboratory, January 1967.

9.1-3 NUS-894 NUMICE-2, A Spectrum Dependent Non-spatial Cell Depletion Code by Y.S. Kim, NUS Corporation, March 1972. Numice is NUS' version of LEOPARD.

9.1-4 ORNL-4938 KENO-IV An Improved Monte Carlo Criticality Program by L.M. Petrie and N.F. Cross, ORNL, November 1975.

9.1-5 ORNL-TM-3706 AMPX-L Modular Code System for Generating Coupled Multi-group Neutron Gamma Libraries from ENDF/B by N.M. Greene, et al., ORNL, November 1974.

9.1-6 MRI/STARDYNE3 (Version 3), developed by Mechanics Research, Inc., Los Angeles, California.

BLANK

TABLE 9.1-1

FUEL POOL COOLING AND CLEANUP SYSTEM PERFORMANCE DATA

Design Heat Load Btu/hr	4.0×10^6
Maximum Heat Load Btu/hr	8.0×10^6
System Design Pressure, psig	150
System Design Temperature, ^oF	150
Fuel Pool Water Volume, gal	350,000
Dryer-Separator Pool Water Volume, gal	293,600
Reactor Well Water Volume, gal	210,000

TABLE 9.1-2

FUEL POOL COOLING AND CLEANUP SYSTEM EQUIPMENT DATA

Fuel Pool Heat Exchangers

Number	2
Type	Tube and Shell
Material Tube/Shell	SS/CS
Capacity, Btu/hr/heat exchanger	4.0×10^6
Cooling Water Flow, gpm/heat exchanger	575
Code and Standards	ASME/III-Class 3 and TEMA-Class R
Seismic Category	II

Fuel Pool Circulation Pumps

Number	2
Type	Horizontal, centrifugal
Material	SS
Flow, gpm	575
Head, Ft of H_2O	160
Motor Size hp	40
Seismic Category	II
Code	ASME/III-Class 3

Fuel Pool Filter Demineralizer

Number	2
Design Flow Rate, gpm	1000 Maximum, 575 Normal
Design Pressure, psig	150
Design Temperature, °F	150
Material	CS-Plastic Lined
Code	ASME/III-Class 3
Seismic Class	II

Piping and Valves

Design pressure, psig	150/300
Design Temperature, °F	220
Material	CS/SS
Code	ASME/III-Class 3

TABLE 9.1-3

ESTIMATED REFUELING CYCLE DATA***

Cycle* Number	Number of Fuel Assemblies Discharged	Equivalent Irradiation Time (Days)	Decay Time (Days)
1	204	333.718	4035
2	200	662.325	3670
3	168	914.470	3305
4	152	998.287	2940
5	160	1043.492	2575
6	160	1095.003	2210
7	160	1139.089	1845
8	160	1143.125	1480
9	176	1275.495	1115
10**	176	1271.148	750
11**	176	1257.620	385
12**	176	985.186	20
Total	2068		

* The word "cycle" is used here to describe the spent fuel leaving the reactor at the end of the irradiation cycle having the same number.

** Uranium fuel with plutonium added.

*** Normal ("average") conditions except last four years are low stream flow (high plant capacity factor).

TABLE 9.1-4
TOOLS AND SERVICING EQUIPMENT

Fuel Servicing Equipment
Fuel Preparation Machines
New Fuel Inspection Stand
Channel Bolt Wrenches
Channel Handling Tool
Fuel Pool Sipper
Channel Gauging Fixture
General Purpose Grapples
Fuel Bundle Inspection Fixture

Servicing Aids
Pool Tool Accessories
Actuating Poles
General Area Underwater Lights
Local Area Underwater Lights
Drop Lights
Underwater TV Monitoring System
Underwater Vacuum Cleaner
Viewing Aids
Light Support Brackets
In-Core Detector Cutter
In-Core Manipulator

Reactor Vessel Servicing Equipment
Reactor Vessel Servicing Tools
Steam Line Plugs
Shroud Head Bolt Wrenches
Head Holding Pedestals
Head Stud Rack
Dryer-Separator Sling
Head Strongback
Service Platform
Service Platform Support
Steam Line Plug/Installation Tool

In-Vessel Servicing Equipment
Instrument Strongback
Control Rod Grapple
Control Rod Guide Tube Grapple
Fuel Support Grapple
Grid Guide
Control Rod Latch Tool
Instrument Handling Tool
Control Rod Guide Tube Seal
In-Core Guide Tube Seals
Blade Guides
Fuel Bundle Sampler
Peripheral Orifice Grapple
Orifice Holder
Peripheral Fuel Support Plug
Fuel Bail Cleaner
Dummy Fuel Assembly

Refueling Equipment
Refueling Equipment Servicing Tools
Refueling Platform

Storage Equipment
Spent Fuel Storage Racks
Channel Storage Racks
In-Vessel Racks
New Fuel Storage Rack

Under Reactor Vessel Servicing Equipment

Control Rod Drive Servicing Tools
CRD Hydraulic System Tools
Spring Reels
Control Rod Drive Handling Equipment
Equipment Handling Platform
Thermal Sleeve Installation Tool
In-Core Flange Seal Test Plug
Key Bender

TABLE 9.1-5

FUEL SERVICING EQUIPMENT

COMPONENT NO.	IDENTIFICATION	ESSENTIAL CLASSIFICATION (a)	SAFETY CLASSIFICATION (b)	QUALITY GROUP (c)	SEISMIC CATEGORY (d)
1	Fuel Prep Machine	PE	2	E	I
2	New Fuel Inspection Stand	NE	0	E	NA
3	Channel Bolt Wrench	NE	0	E	NA
4	Channel Handling Tool	NE	0	E	NA
5	Fuel Pool Sipper	NE	0	E	NA
6	Fuel Inspection Fixture	NE	0	E	NA
7	Channel Gauging Fixture	NE	0	E	NA
8	General Purpose Grapple	PE	2	E	I

Table Notes

(a) NE - Nonessential
PE - Passive Essential

(b) 0 - Other

(c) B - ASME Code Section III Class-2
D - ANSI B31.1
E - Industrial Code Applies
I - Electrical Codes Apply

(d) NA - No Seismic Requirements

TABLE 9.1-6

REACTOR VESSEL SERVICING EQUIPMENT

COMPONENT NO.	IDENTIFICATION	ESSENTIAL CLASSIFICATION (a)	SAFETY CLASSIFICATION (b)	QUALITY GROUP (c)	SEISMIC CATEGORY (d)
1	Reactor Vessel Serv. Tools	NE	O	E	NA
2	Steam Line Plug	PE	O	E	NA
3	Shroud Head Bolt Wrench	NE	O	E	NA
4	Vessel Nut Handling Tool	NE	O	E	NA
5	Head Holding Pedestal	NE	O	E	NA
6	Head Nut and Washer Rack	NE	O	E	NA
7	Head Stud Rack	NE	O	E	NA
8	Dryer and Separator Sling	PE	O	E	NA*
9	Head Strongback	PE	O	E	NA*
10	Service Platform	NE	O	E	NA
11	Service Platform Support	NE	O	E	NA
12	Steam Line Plug Inst. Tool	NE	O	E	NA

Table Notes

(a) NE - Nonessential
PE - Passive Essential

(b) O - Other

(c) B - ASME Code Section III Class-2
D - ANSI B31.1
E - Industrial Code Applies
I - Electrical Codes Apply

(d) NA - No Seismic Requirements

*Dynamic analysis methods for seismic loading are not applicable, as this equipment is supported by the reactor service crane. Lifting devices have been designed with a minimum safety factor of 5 and undergo proof testing.

TABLE 9.1-7

UNDER-REACTOR VESSEL SERVICING EQUIPMENT AND TOOLS

Equipment/ Tool	Classification	Safety Class	Seismic Category
1. CRD Handling Equipment	Non-Essential	Other	NA
2. Equipment Handling Platform	Non-Essential	Other	NA
3. Spring Reel	Non-Essential	Other	NA
4. Thermal Sleeve Removal Tool	Non-Essential	Other	NA
5. In-Core Flange Seal Test Plug	Non-Essential	Other	NA
6. Key Bender	Non-Essential	Other	NA

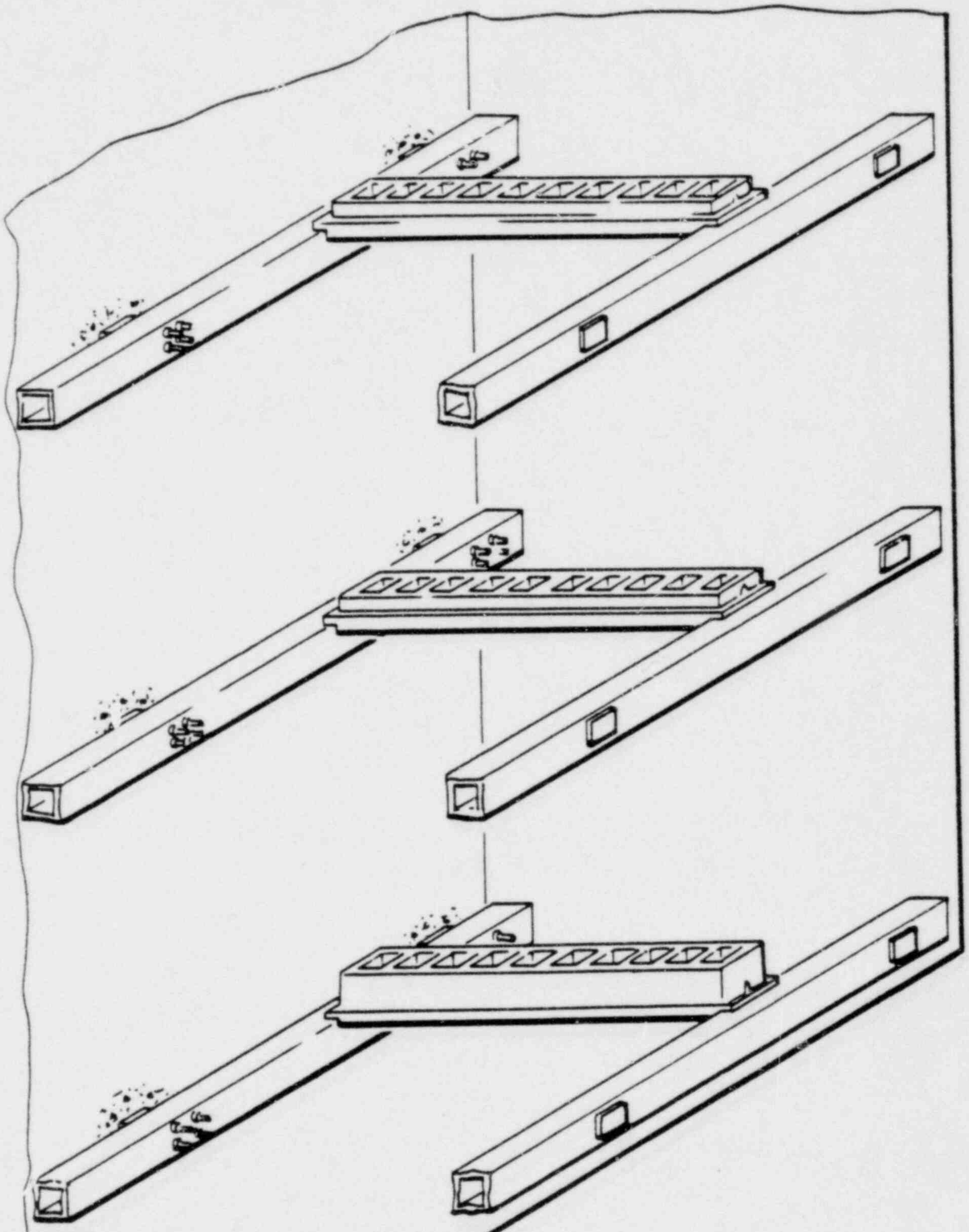
Table Notes

NA - No Seismic Requirements

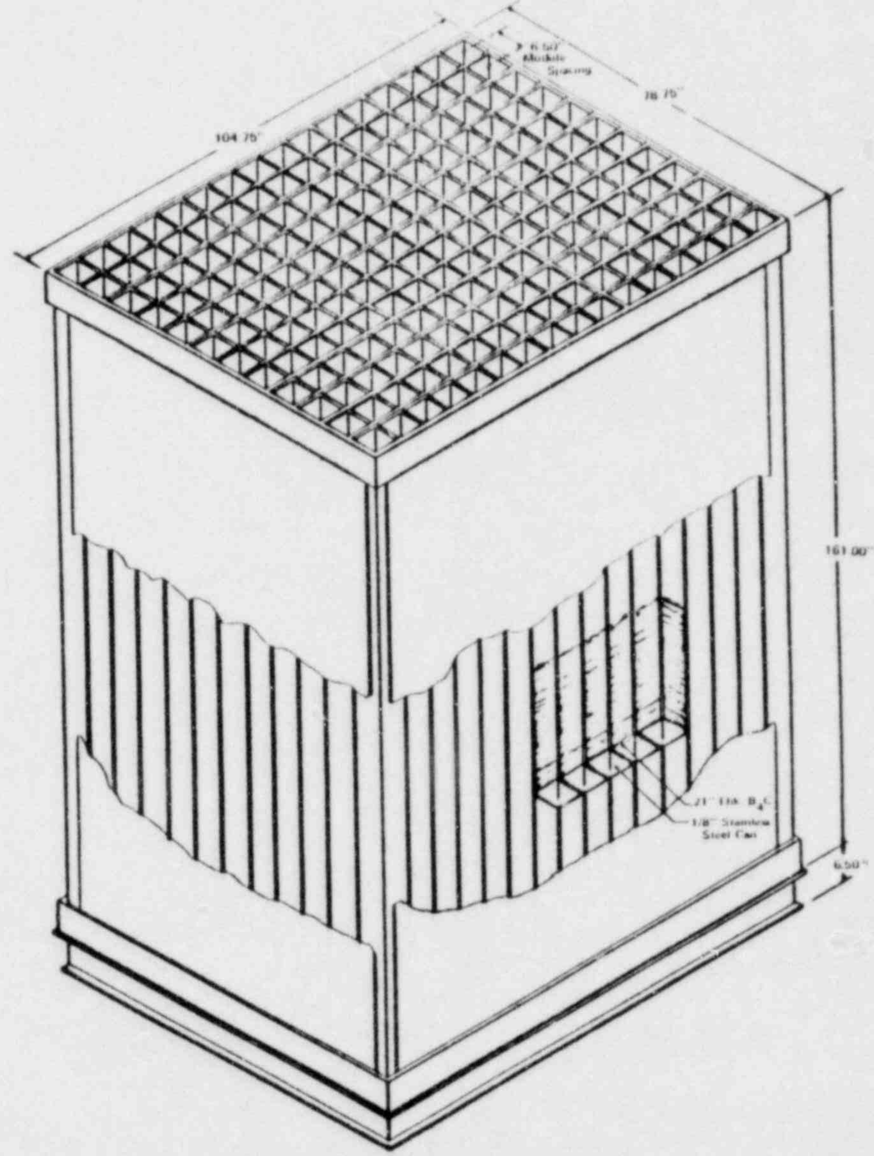
TABLE 9.1-8

DOSE RATE AT 5 FEET ABOVE FUEL POOL WATER LEVEL DURING REFUELING

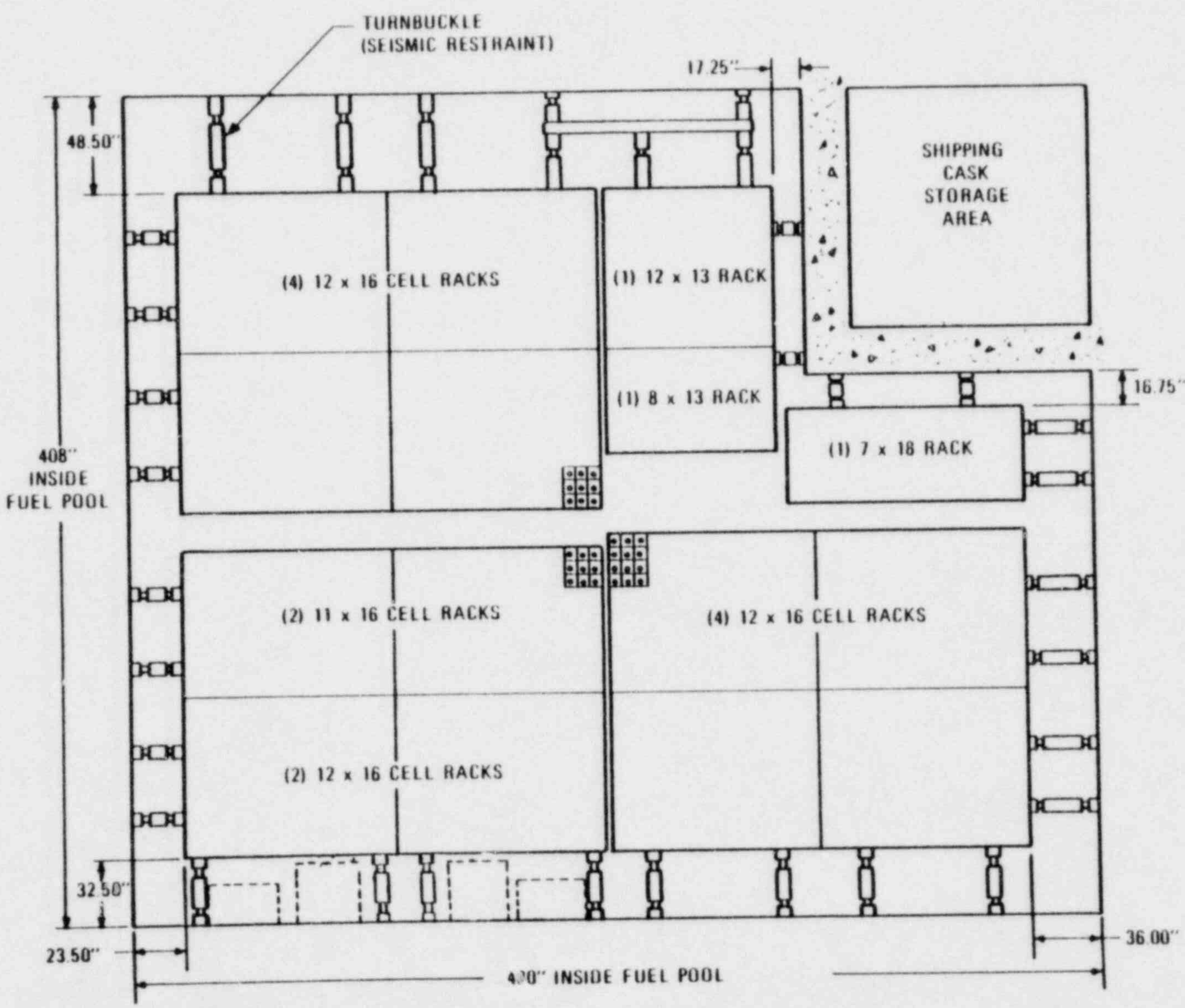
Refueling No.	Calculated Dose Rate (mrem/hr)
1	3.5
2	4.3
3	4.8
4	5.3
5	5.7
6	6.0
7	6.2
8	6.4
9	6.6
10	6.8
11	6.9



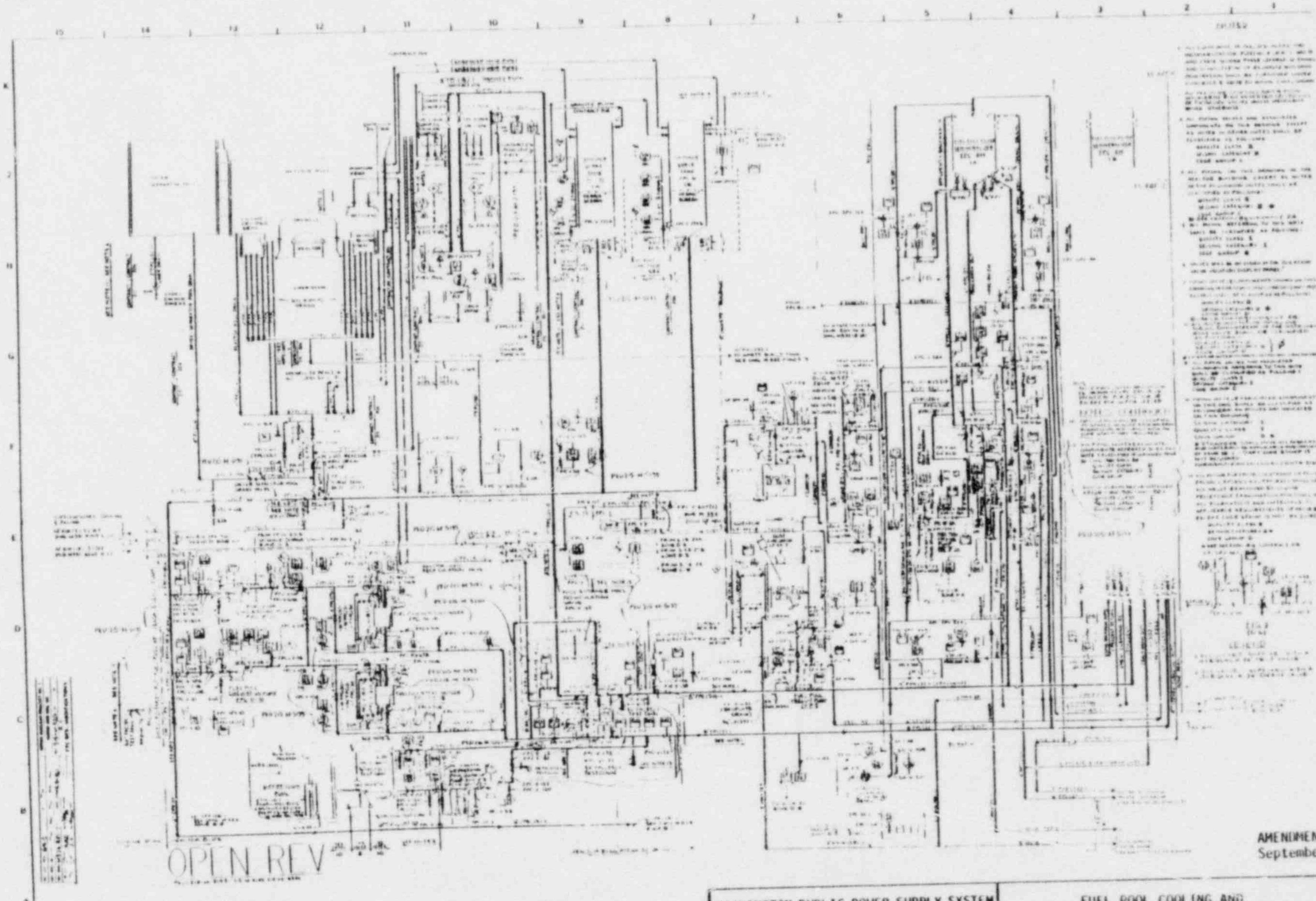
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEW FUEL STORAGE	FIGURE 9.1-1



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	TYPICAL SPENT FUEL RACK 12 x 16 ARRAY	FIGURE 9.1-2

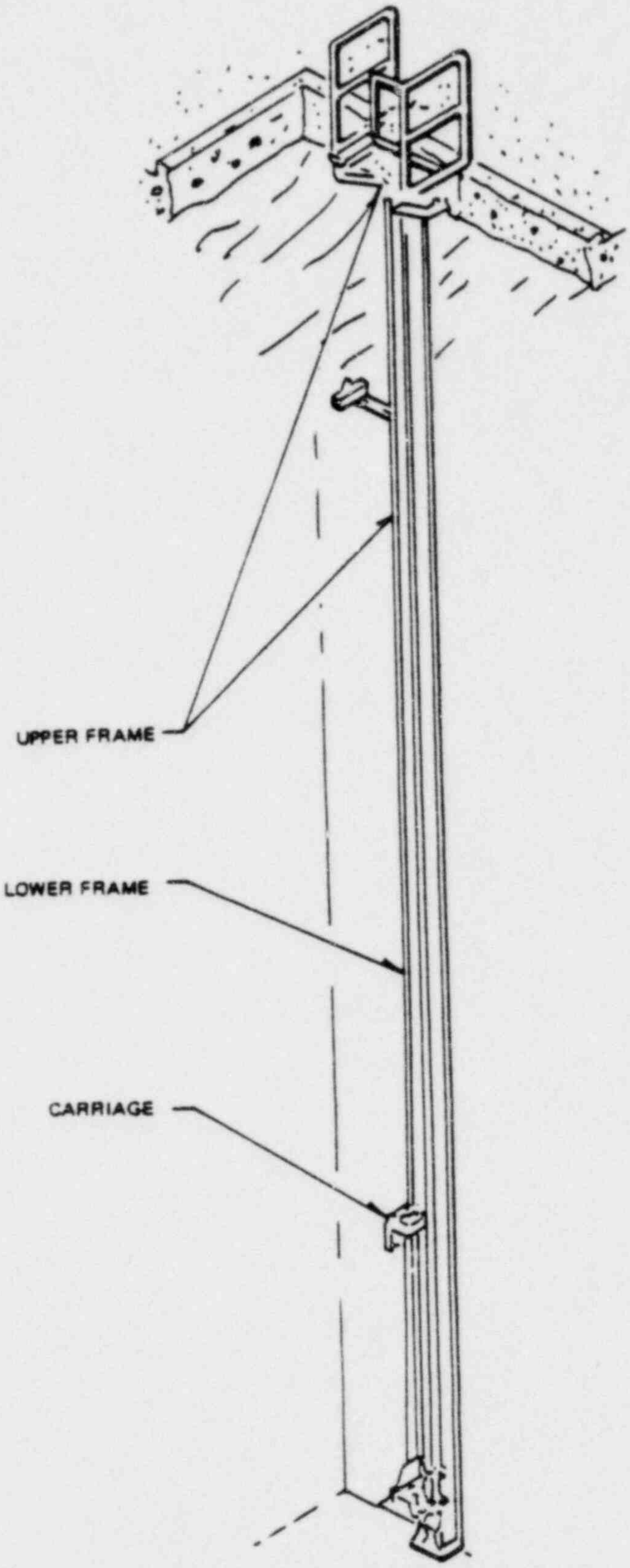


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SPENT FUEL POOL ARRANGEMENT	FIGURE 9.1-3

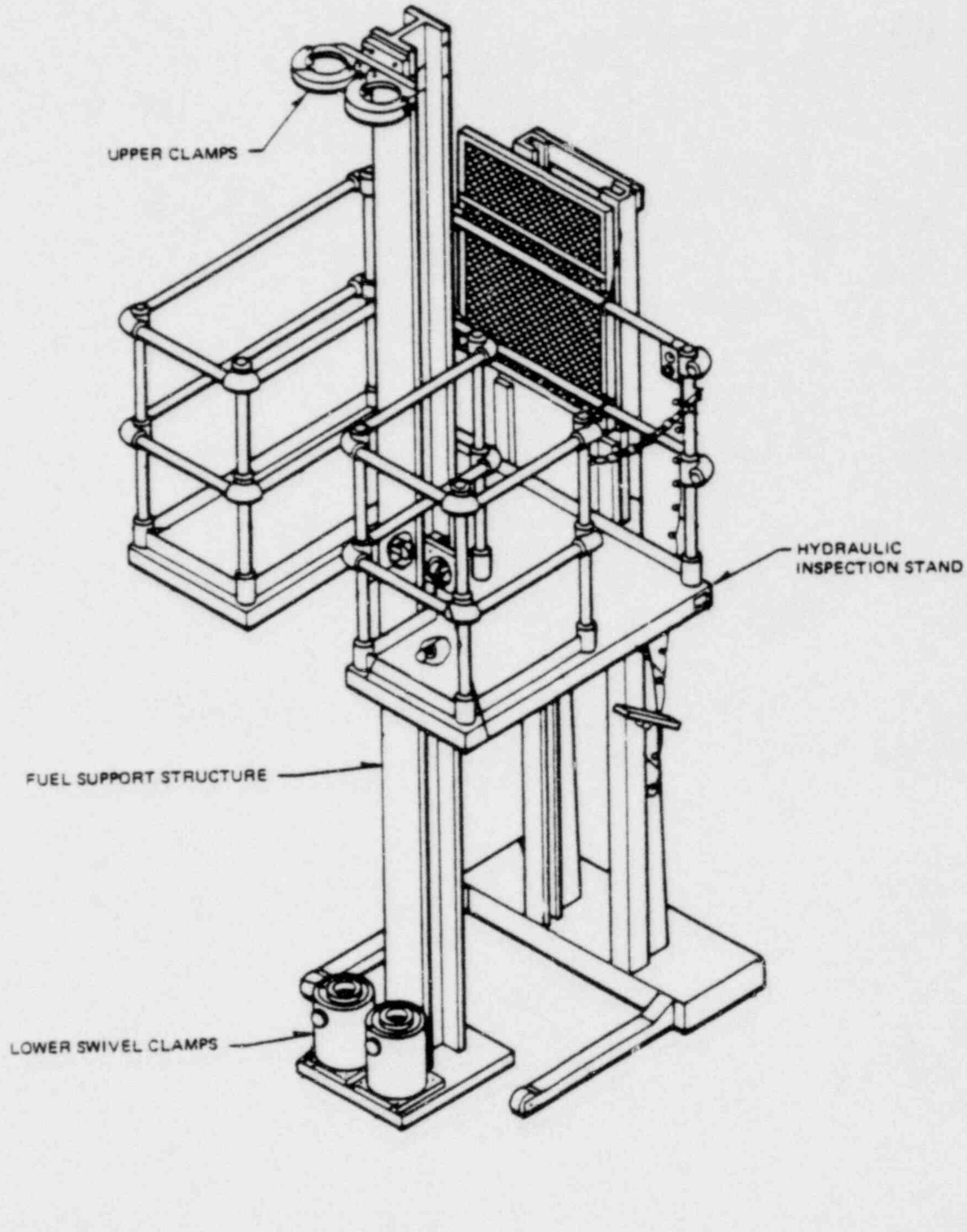


AMENDMENT NO. 11
September 1980

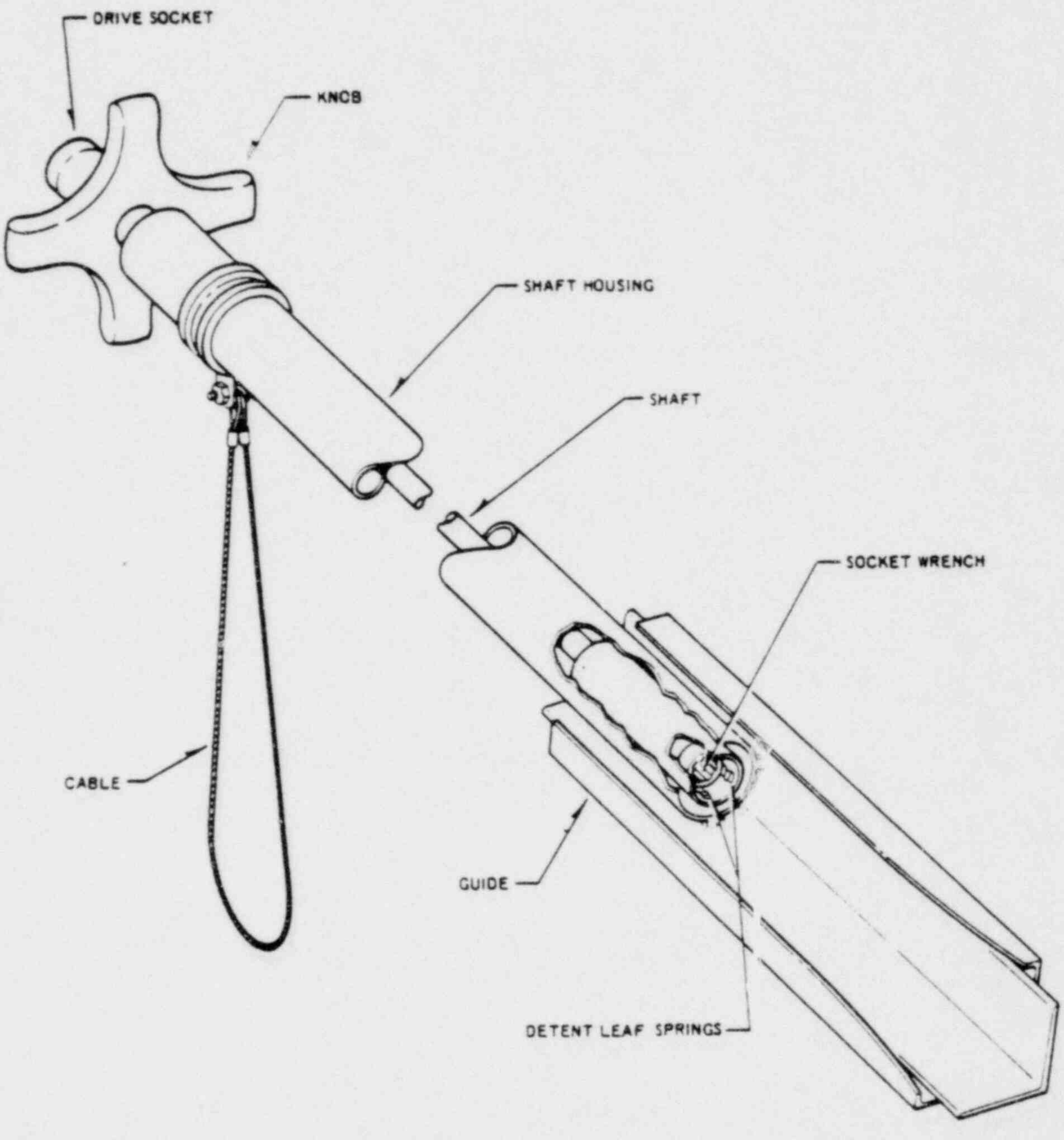
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FUEL POOL COOLING AND CLEANUP SYSTEM	FIGURE 9.1-4



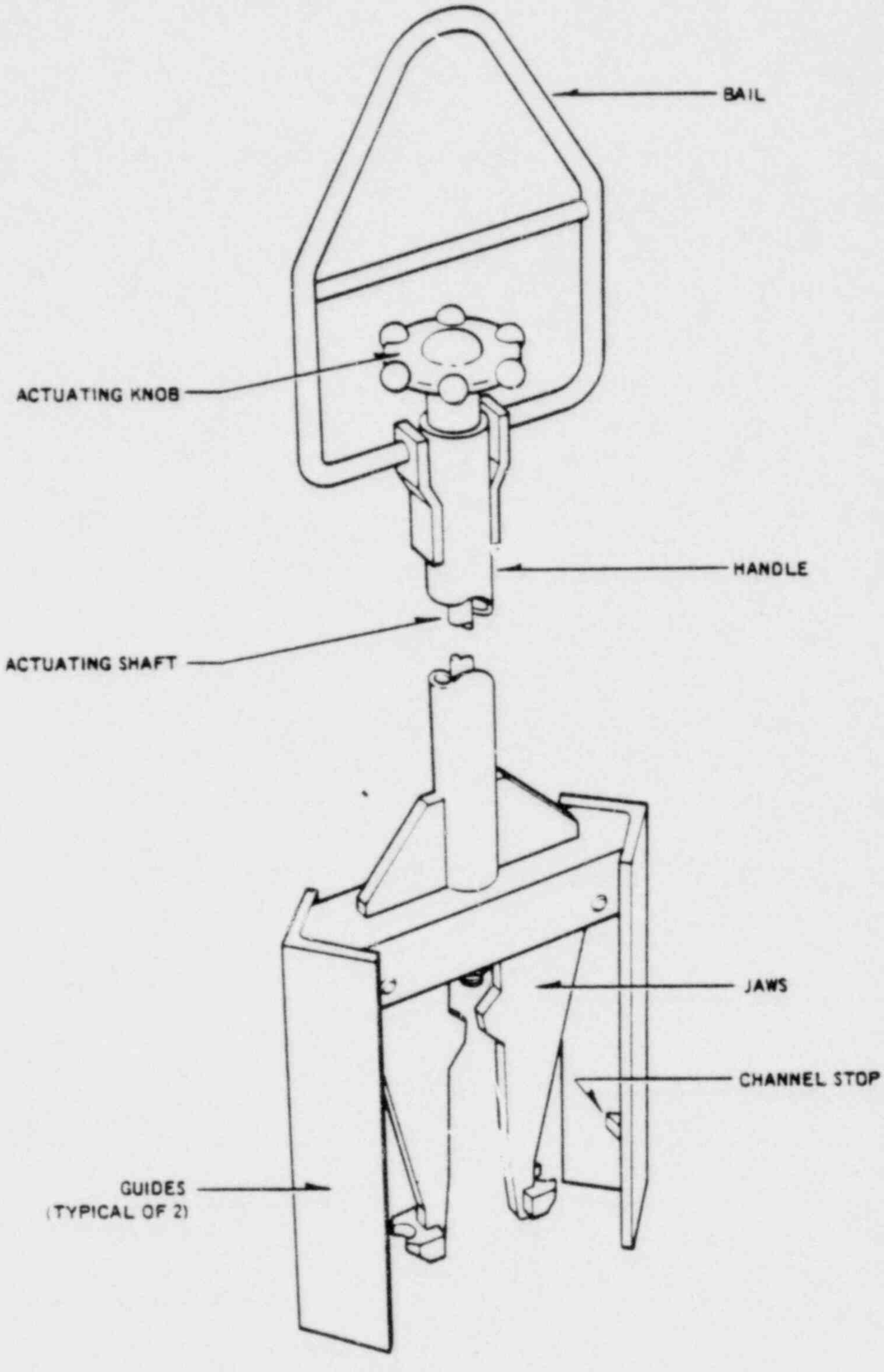
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FUEL PREPARATION MACHINE SHOWN INSTALLED IN FACSIMILE FUEL POOL	FIGURE 9.1-5



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NEW FUEL INSPECTION STAND	FIGURE 9.1-6

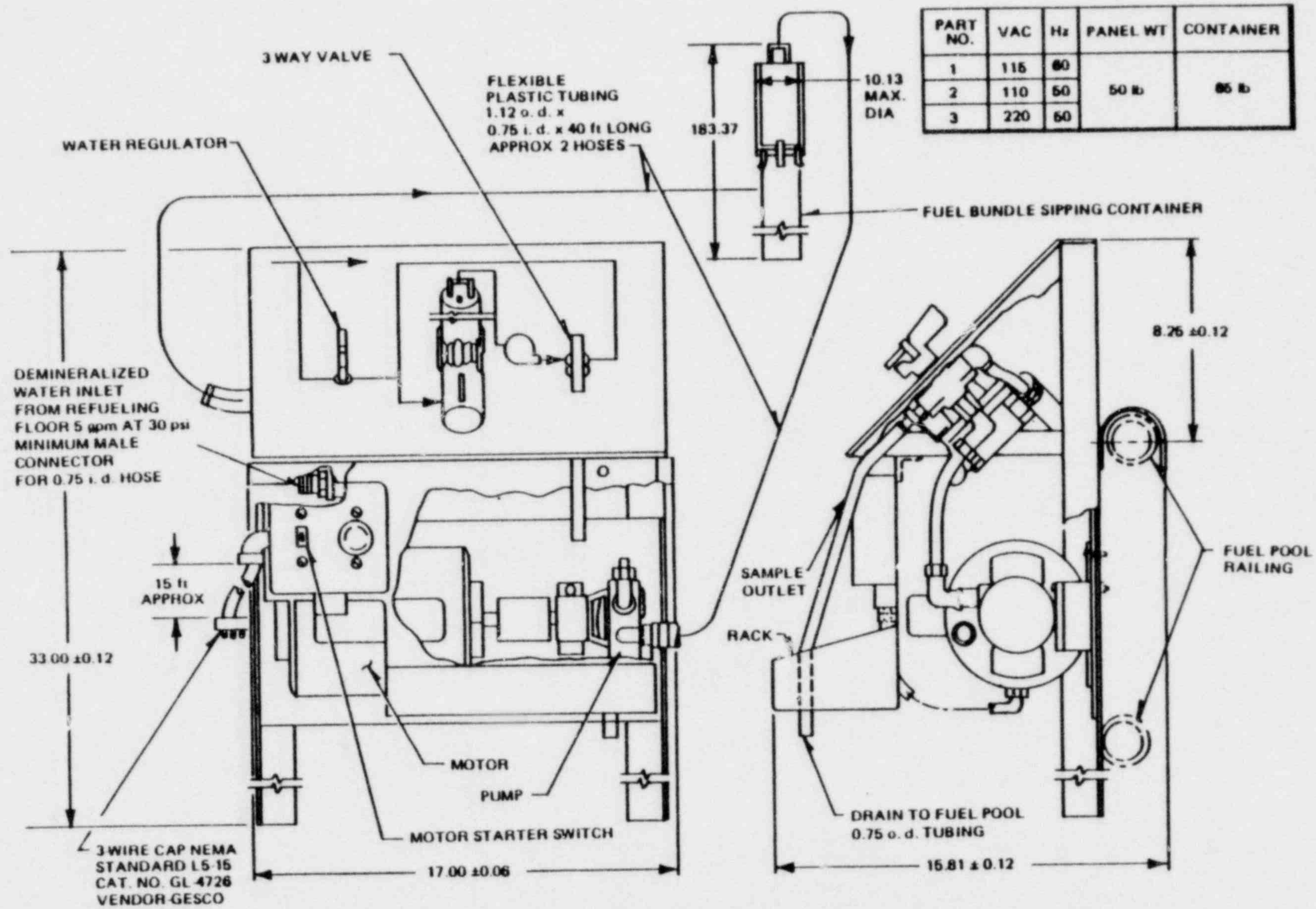


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CHANNEL BOLT WRENCH	FIGURE 9.1-7



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CHANNEL HANDLING TOOL	FIGURE 9.1-8

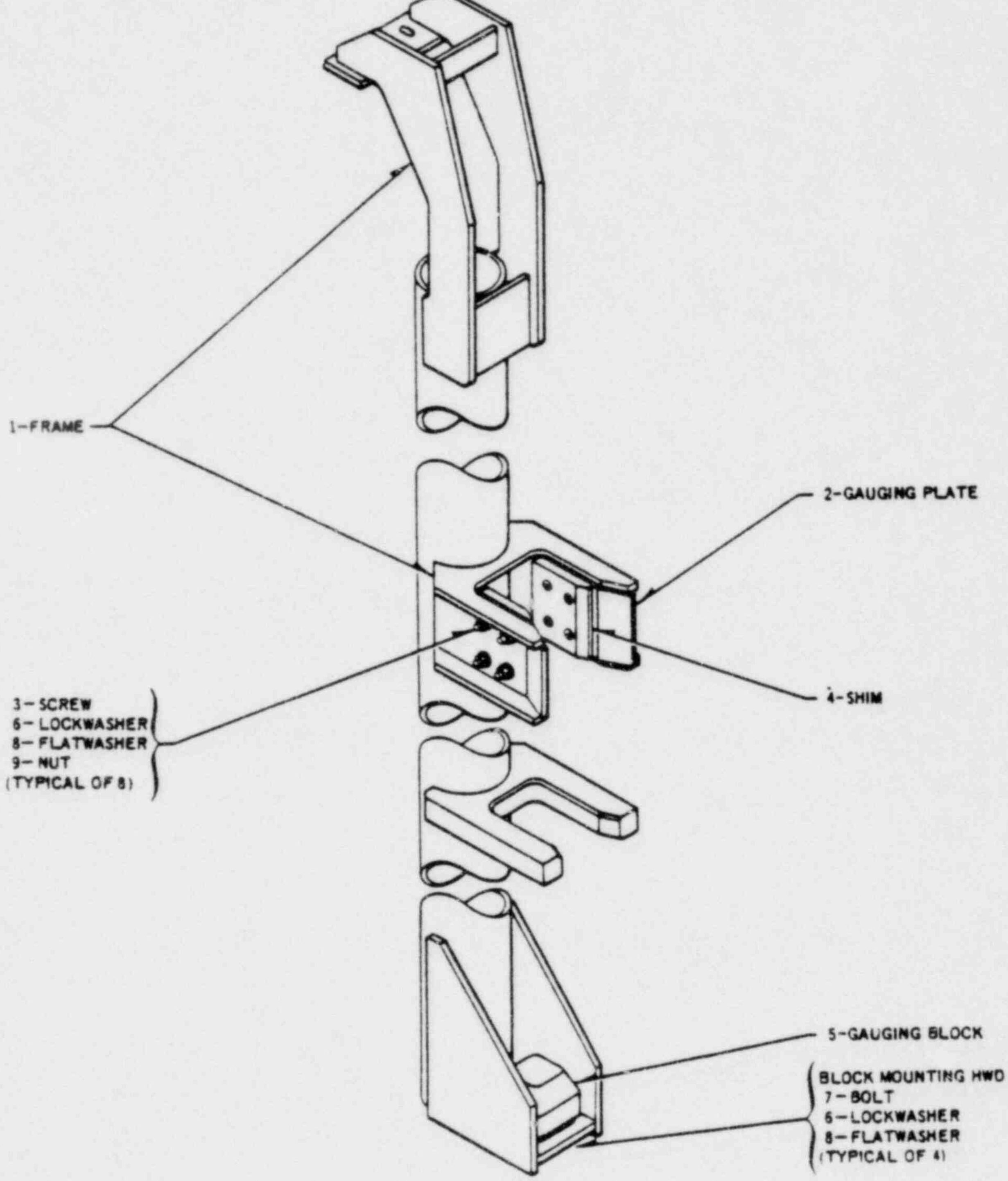
PART NO.	VAC	Hz	PANEL WT	CONTAINER
1	115	60	50 lb	85 lb
2	110	50		
3	220	50		



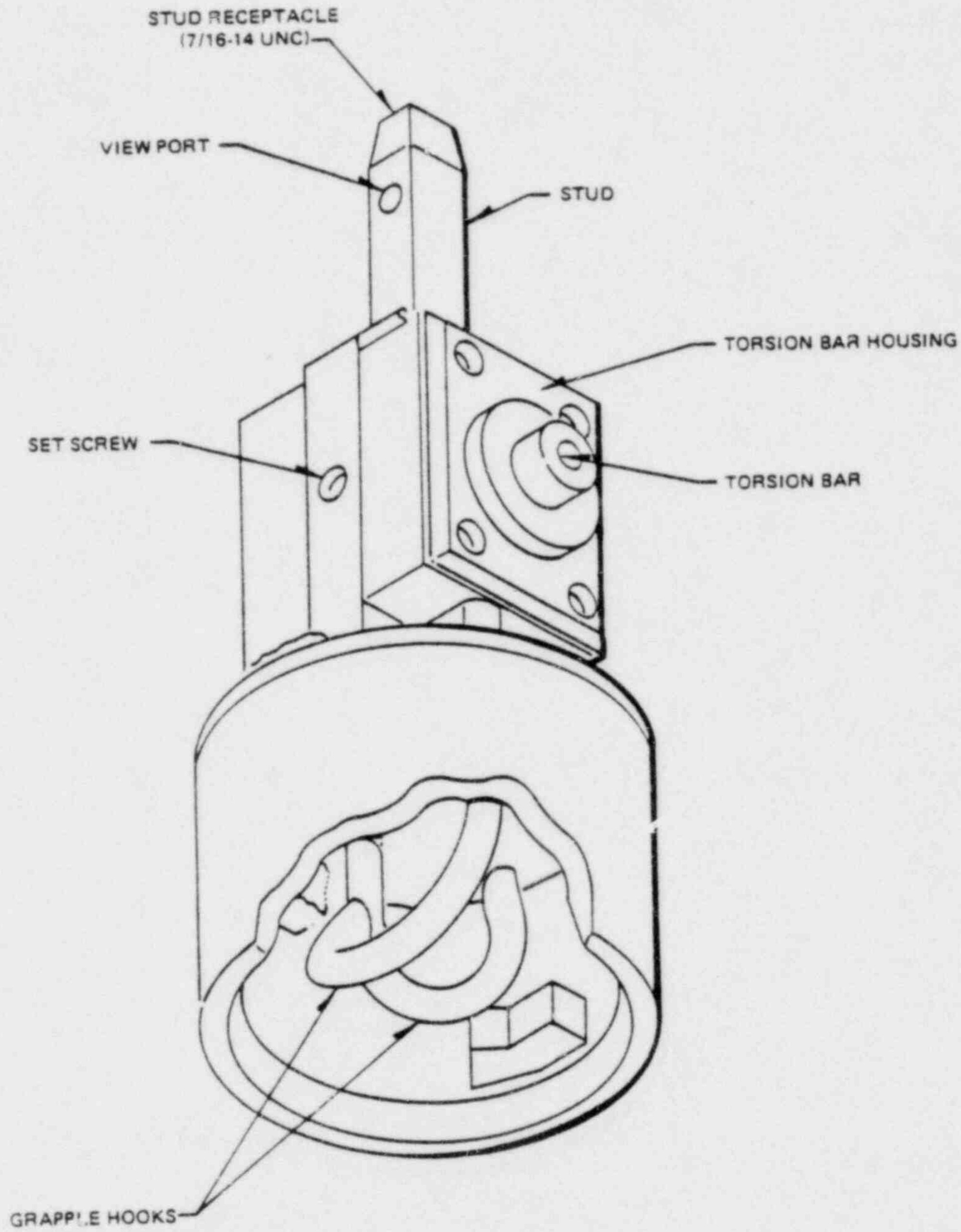
WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

FUEL POOL SIPPER

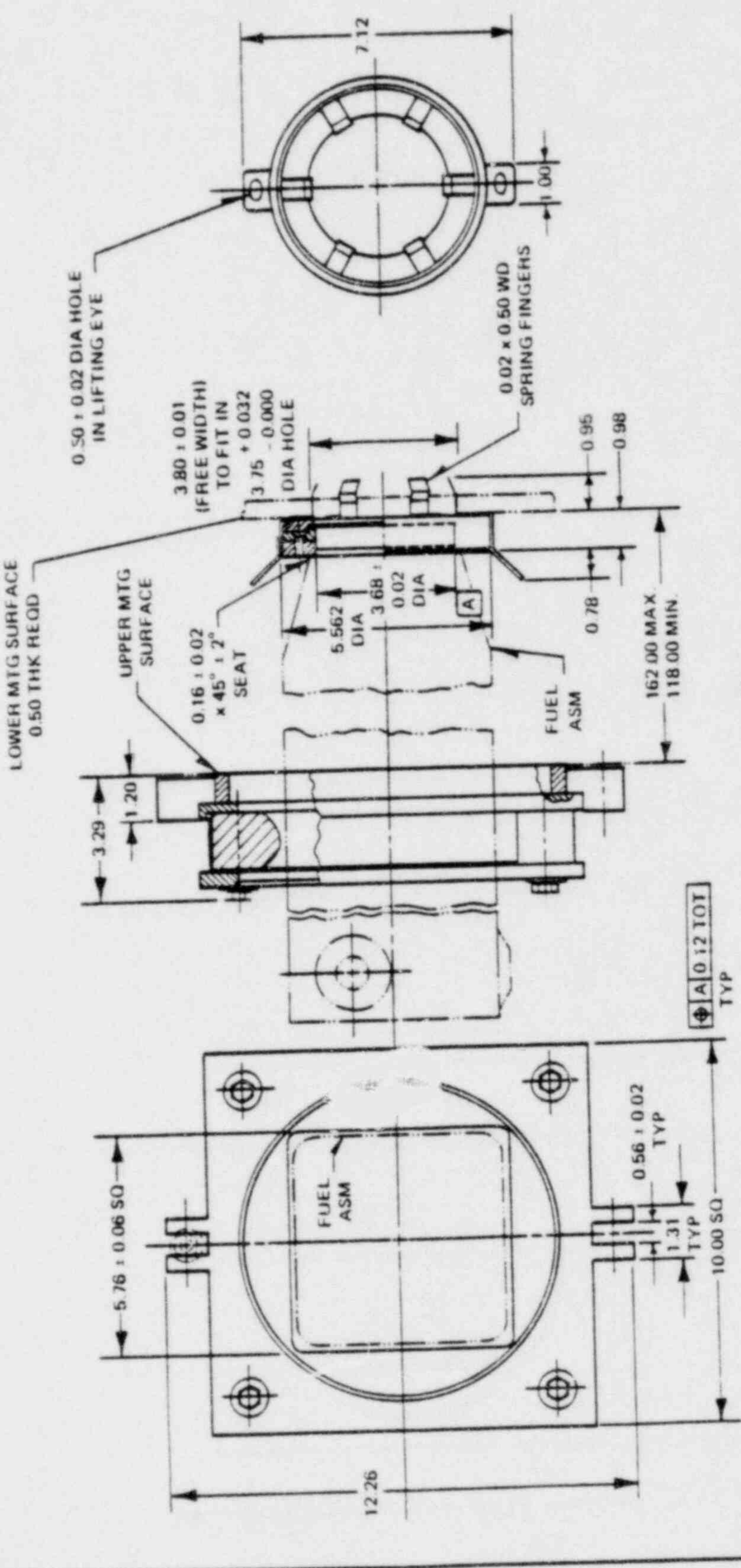
FIGURE 9.1-9



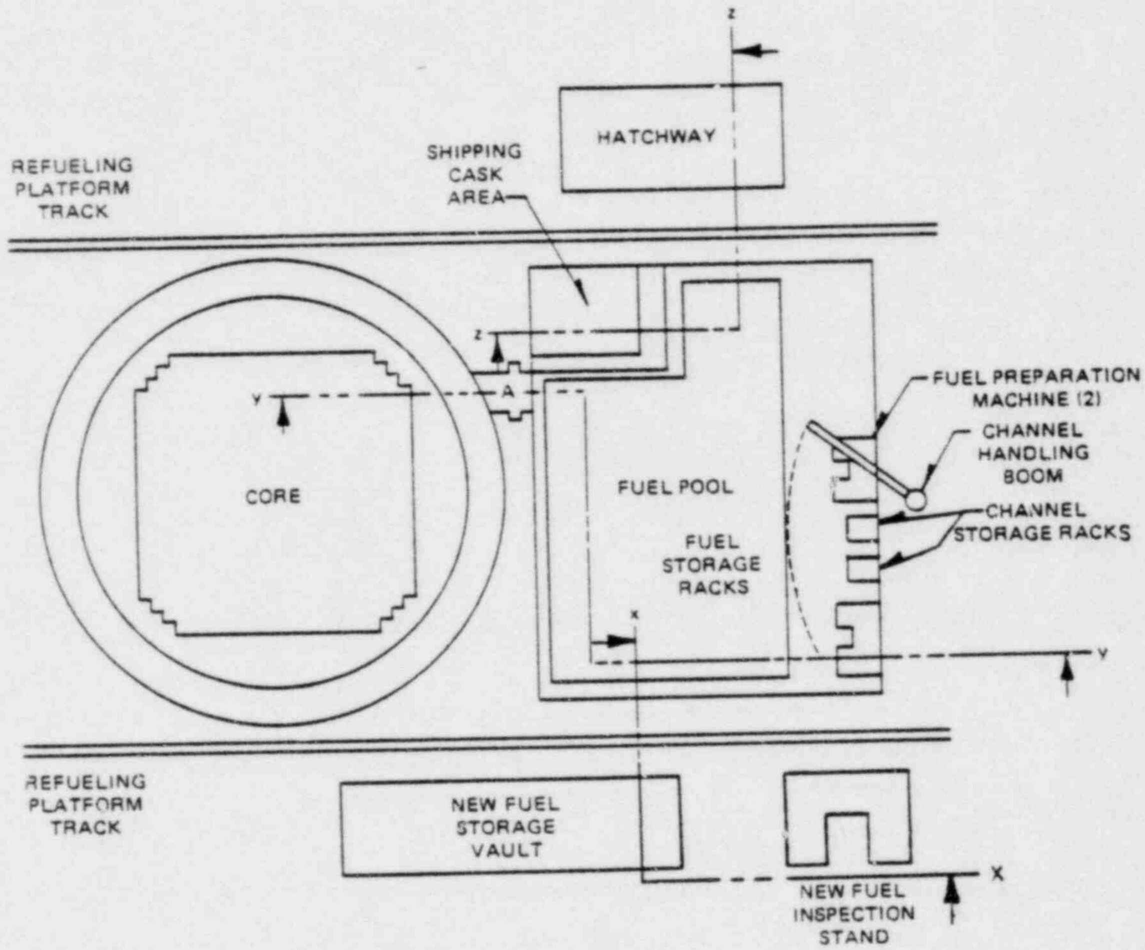
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	CHANNEL GAUGING FIXTURE	FIGURE 9.1-10



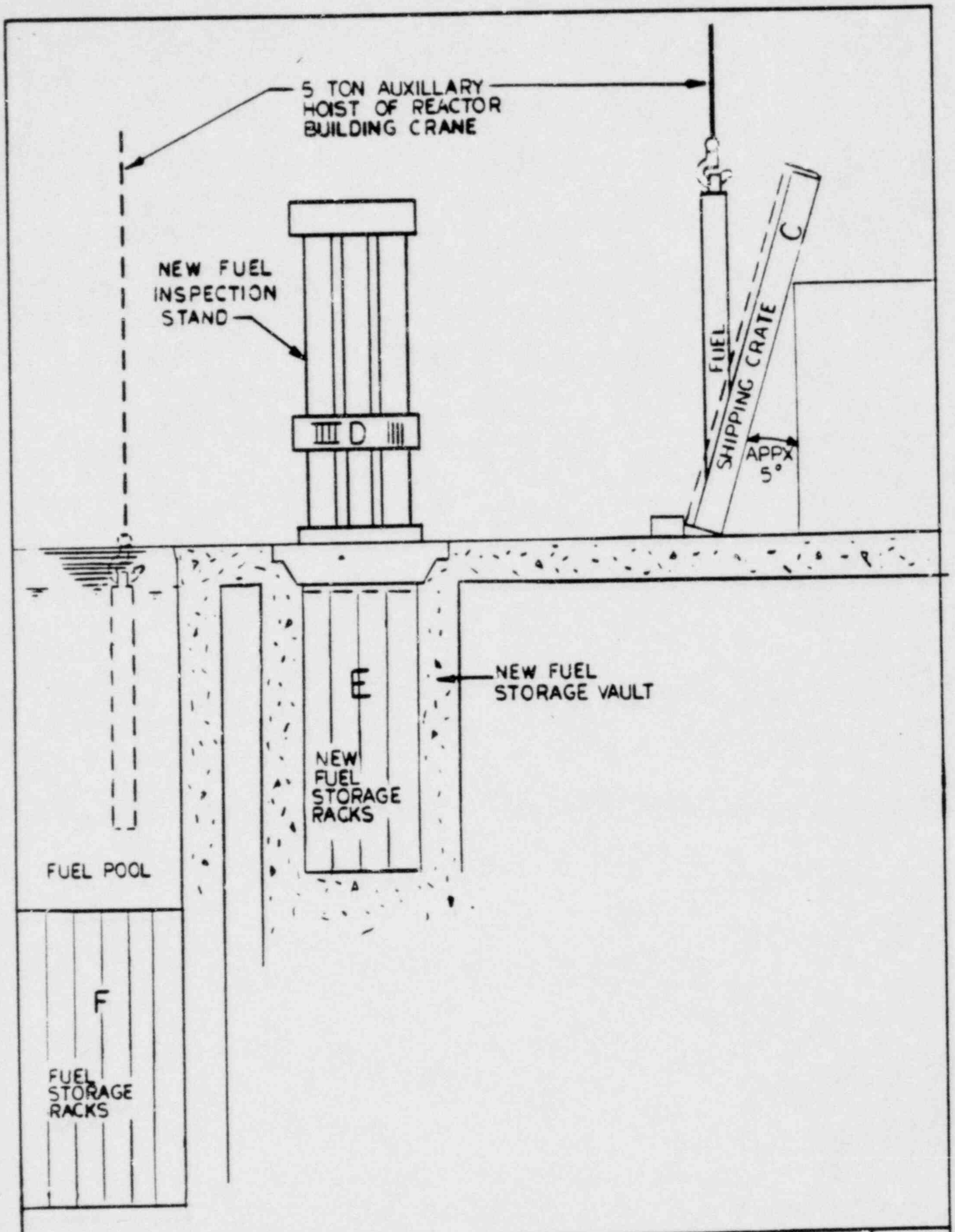
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	GENERAL PURPOSE GRAPPLE	FIGURE 9.1-11



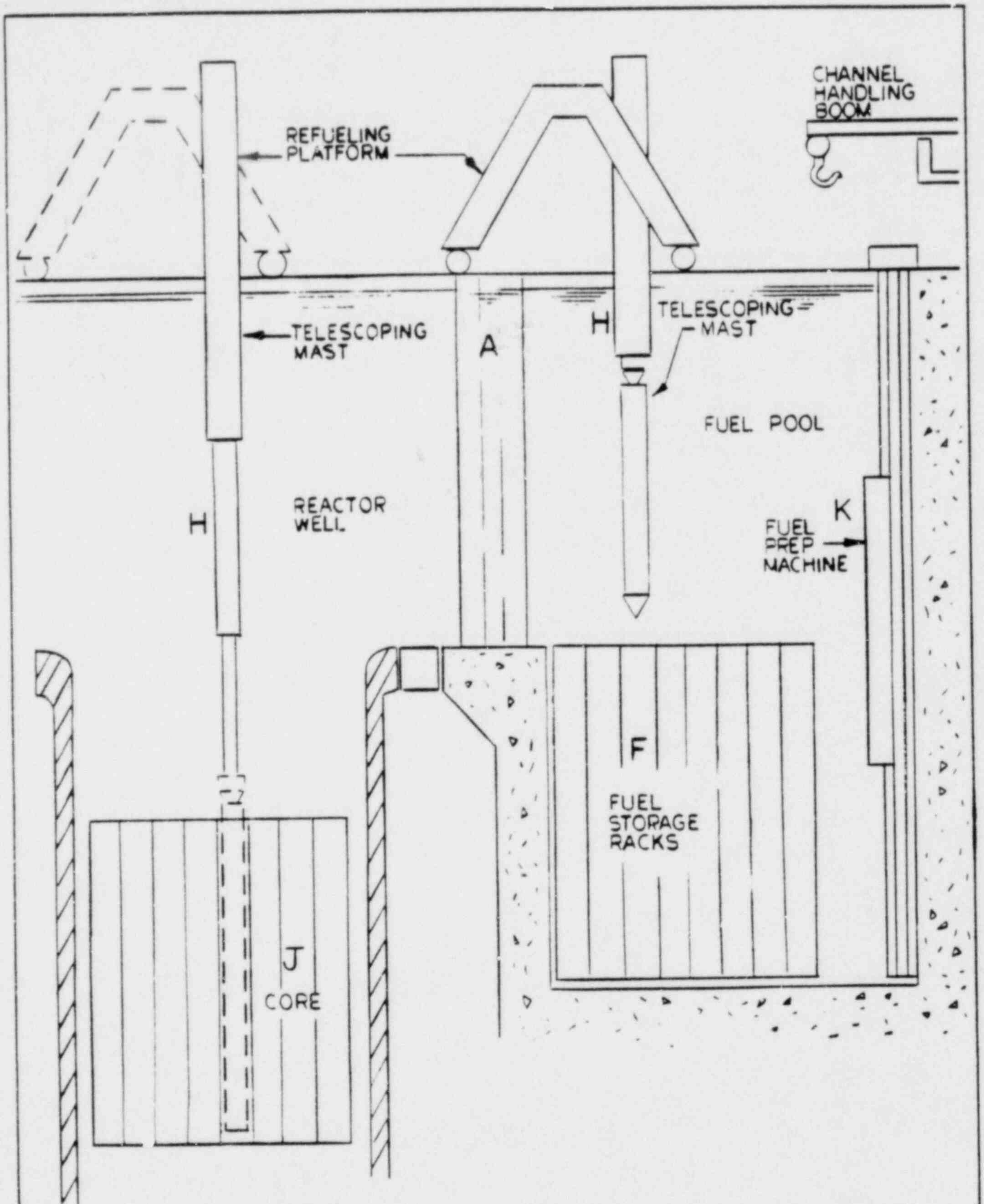
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FUEL INSPECTION FIXTURE	FIGURE 9.1-12



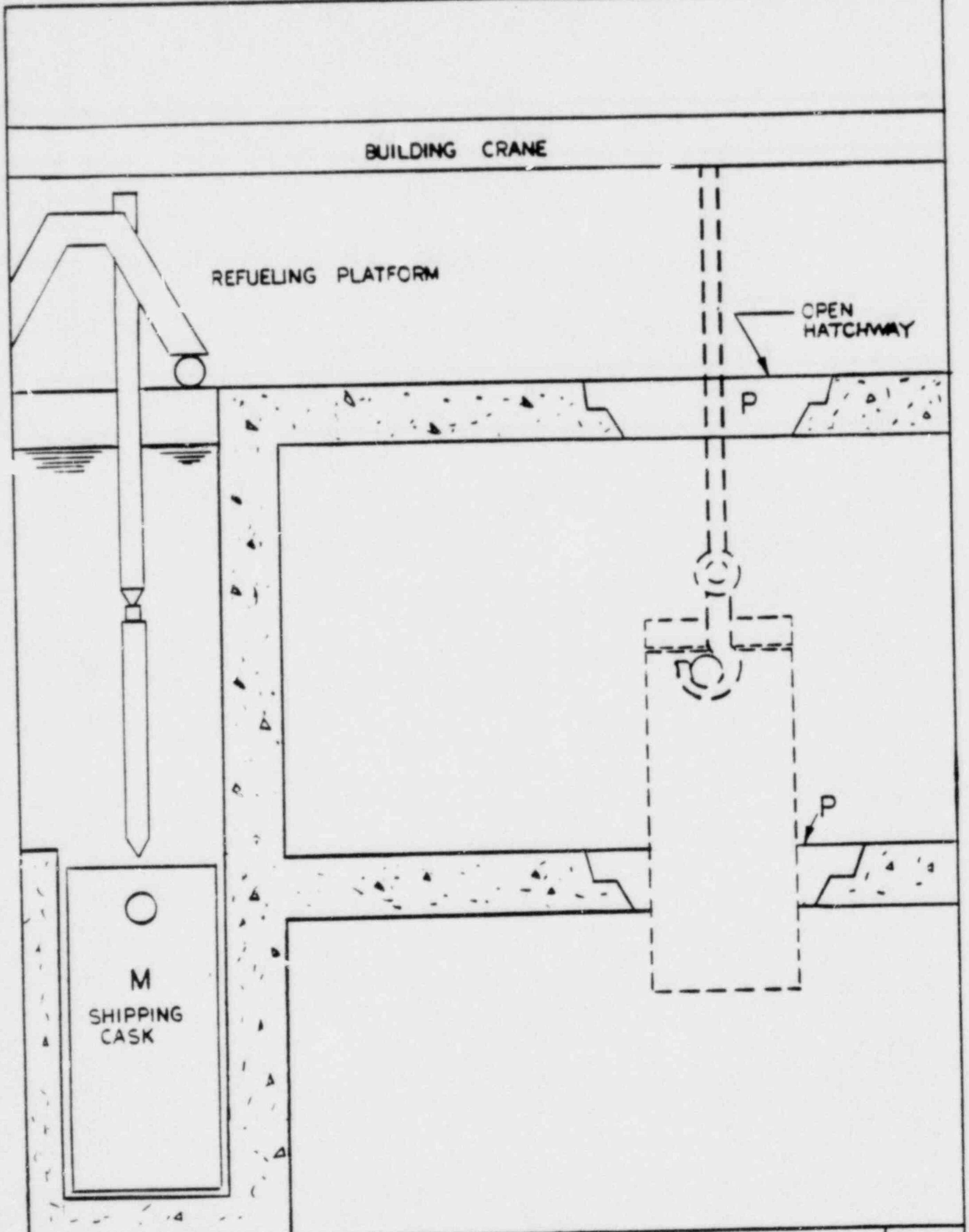
WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PLANT VIEW OF REFUELING FACILITIES	FIGURE 9.1-13



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SIMPLIFIED SECTION OF NEW FUEL HANDLING FACILITIES (SECTION X-X, FIGURE 9.1-13)	FIGURE 9.1-14

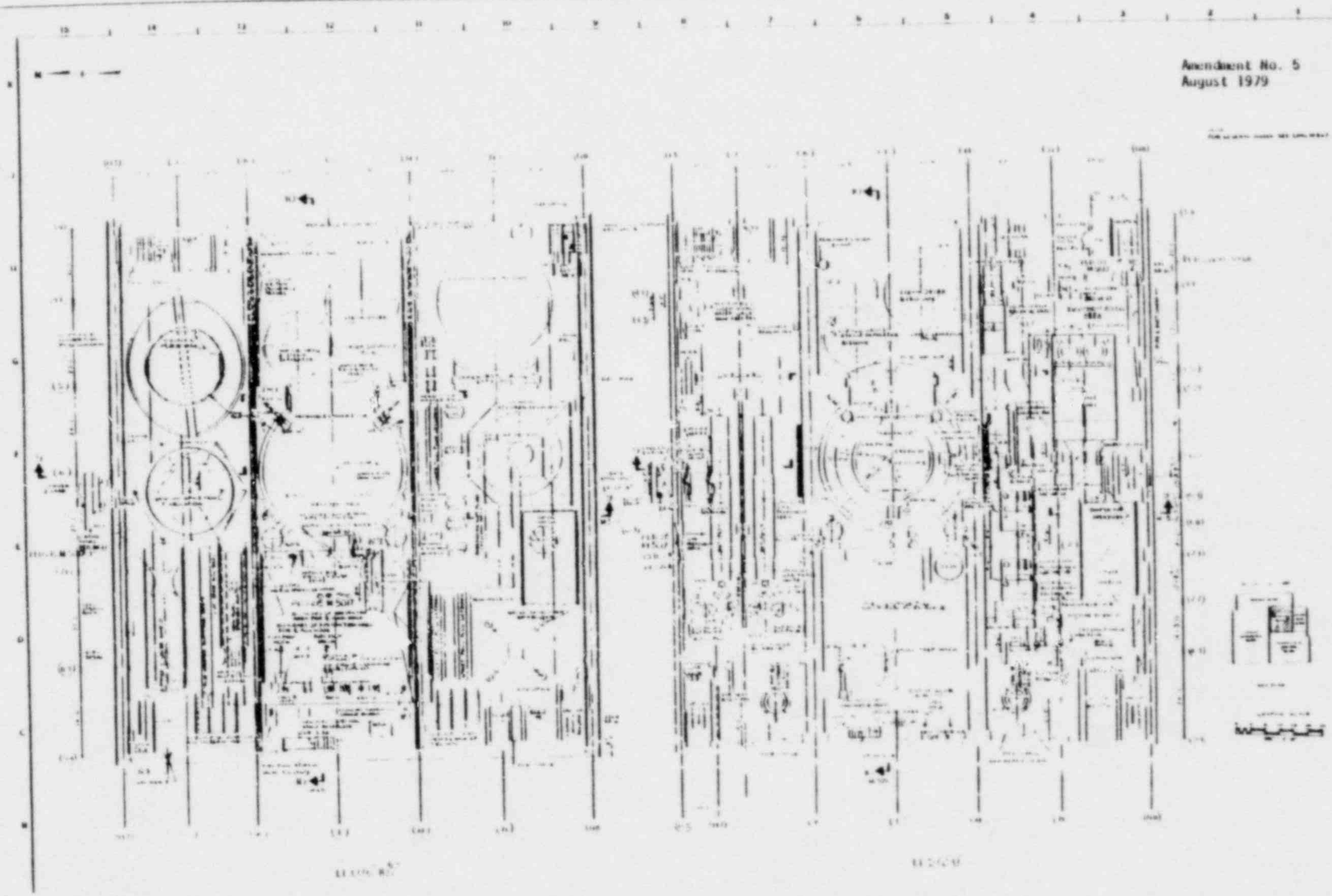


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SIMPLIFIED SECTION OF REFUELING FACILITIES (SECTION Y-Y, FIGURE 9.1-13)	FIGURE 9.1-15



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SIMPLIFIED SECTION OF FUEL SHIPPING FACILITIES (SECTION ZZ)	FIGURE 9.1-16

Amendment No. 5
August 1979



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	GENERAL ARRANGEMENT PLAN EL. 572' AND EL. 606'- 10 1/2" REACTOR BUILDING	FIGURE 9.1-17

REFERENCE 4

WNP-2 FSAR 12.5

Pages: 12.5-1 thru 12.5-23

12.5 HEALTH PHYSICS PROGRAM

12.5.1 ORGANIZATION

The administrative organization of the WNP-2 plant radiation safety program is described below. The plant Health Physics/ Chemistry section organization is supported by the Radiological Programs Department described in 12.1.1. The health physics organization is also audited for compliance to regulations and to ensure that exposures are ALARA by personnel outside of the operations division. Regulatory Guide 1.8 has been followed in the selection of personnel for the health physics organization and in developing a training program for all plant workers. WPPSS pre-employment practices include screening to determine that plant employees are responsible and conscientious and qualified to perform their duties safely. The experience and qualifications of the personnel responsible for the health physics program and for handling and monitoring radioactive materials including special nuclear source and byproduct materials, are described in Chapter 13. Chapter 13 also describes the minimum qualification requirements for specific plant personnel, using the criteria outlined in Regulatory Guide 1.8.

The WNP-2 Plant Manager is responsible for the radiological safety of all plant personnel. In turn, all employees share this responsibility and are required to follow the rules and procedures established for radiological safety. WPPSS employees are also responsible for the radiological safety of visitors and contractor personnel under their direction at the WNP-2 site.

The plant Health Physics Supervisor is responsible to the WNP-2 Health Physics/Chemistry Supervisor for establishing administrative and technical controls for radiation protection, radiation exposure reduction, and for the subsequent administration of this program. With the assistance of the Health Physics/Chemistry Technicians, he maintains radiological surveillance, trains personnel in radiation safety, locates and evaluates radiological problems, maintains personnel radiation exposure records, and make recommendations for the control and/or elimination of the radiological conditions that increase personnel exposure or the release of radioactivity. He reviews plant operations and maintenance to ensure that the radiological program objectives are accomplished. The main objectives of the health physics program are:

a. To minimize the exposure to ionizing radiation of all persons who enter the restricted areas (as defined in 10CFR20)

b. To prevent, contain, and minimize the spread of radioactive contamination.

c. To prevent radioactive materials or devices (other than those shipped under license) from leaving the restricted area.

d. To provide all regularly assigned plant personnel a continuing progressive educational program in radiation safety concepts and practices in order that there will be an intelligent and willing conformance with established health physics practices and regulations, and to instruct all personnel assigned to work in the restricted areas in the rules of conduct for radiation work. This training program is designed to meet the requirements of 10CFR19 and to provide for ALARA indoctrination.

e. Maintain surveillance of the location and intensity of radiation and radioactive contamination within the facility.

f. Measure and record radiation exposures received by personnel within the plant.

g. Preplan work within the plant to reduce the radiation exposures to as low as reasonably achievable.

h. Provide records of all surveillance activities that will point out problem areas and assist in job planning.

12.5.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

This section describes the equipment, instrumentation, and facilities available for implementation of the radiological safety program and the criteria used for selection of the instrumentation and equipment. The guidance provided by Regulatory Guides 8.3, 8.4,8.8 and 8.9 has been followed with the exception of the two specific items listed below:

a. Regulatory Guide 8.3, "Film Badge Performance Criteria" will be followed if film badges are used in the plant program, however, thermoluminescent dosimeters are used as the primary method for compliance with 10CFR20.202(a) and 20.401. Specific performance criteria are applied to the TLD program which meet or exceeds the intent of Regulatory Guide 8.3.

b. Regulatory Guide 8.4 is implemented except for C.2.b, which states, "The calibration/response test result should not exceed ±10% of an exposure from a source traceable to the National Bureau of Standards". This is accepted on the minus side, but is considered excessively stringent on the positive side. Since the error on the positive side results in exposure conservatism to the worker, +20% is a more reasonable limit for rejection of a pencil dosimeter.

12.5.2.1 Criteria for Selection

a. Radiation and Contamination Survey Instrumentation - This equipment was selected to cover the wide range requirements extending from picocurie quantity measurements in the laboratory to the thousand R/hour ranges necessary for emergency dose rate determinations. The laboratory instrumentation was chosen to provide capability for the quantitative and qualitative analyses required to identify and measure the radionuclides encountered in a power reactor. The portable instrumentation includes low level detection capabilities for alpha, beta, and gamma contamination and wide ranges of dose rate measuring instruments for beta, gamma, and neutron radiation. The criteria for quantity selection were to provide adequate available counting time for anticipated demand in the laboratory and sufficient portable instruments to cover normal operational and emergency requirements in all areas of the WNP-2 facility.

b. Airborne Radioactivity Monitoring - Basic criteria for selection of this equipment was to provide a means for determining radioactive airborne effluents released from the plant, and to effectively monitor airborne radioactivity levels within the plant environs. Provisions have been made for continuing response monitoring of noble gases discharged from gaseous release points from the reactor, radwaste and turbine building, and for continuous sampling of radioiodines and particulates at these same locations. Internal plant air monitoring instrumentation is used within these buildings with readout locally and in the control room. Selected devices, capable of airborne particulate monitoring, are included in all the instruments and for radiohalogens and gases where appropriate.

c. Area Radiation Monitoring - This system was designed to provide continuous surveillance of radiation levels throughout the plant with local alarm at predetermined levels, local indication and control room annunciation and recording. Functions of the system include warning of excessive gamma radiation levels in fuel storage and handling areas, detection of unauthorized or inadvertent movement of radioactive materials in the plant, local alarms to warn personnel in an area of a substantial increase in radiation levels, provision for supervisory information in the control room so that correct decisions may be made in the event of a radiation incident, backup to other systems for detection of abnormal migrations of radioactive materials in or from the process streams, and providing a permanent record of gamma radiation levels at selected locations within the various plant buildings.

d. Personnel Monitoring - Personnel dosimetery devices were chosen to provide a record of exposure received by all workers and visitors entering the controlled areas of the plant under normal or accidental conditions.

Personnel dosimeter badges containing thermoluminescent dosimeters, film, or other acceptable dosimeter provide the primary legal record of exposure incurred by personnel. Each person entering plant controlled areas is assigned a badge, which is recorded with the wearer's identification. Results of the badge and the period of exposure are recorded on a document kept as a legal WPPSS record. Badges used will be capable of recording exposure over a range of at least 10 mrem to greater than 1,000 rem.

Neutron exposures are assigned by determining the dose rate versus the time spent in any areas where significant neutron exposure is present, or by use of state of the art neutron recording badges. The total neutron dose for each exposed individual will be added to his permanent record at least quarterly.

In addition to the dosimeter badge, persons entering the restricted area may be required to wear other dosimetry assigned by the radiation protection staff such as direct reading pocket ion chambers, integrating dose meters, personnel alarm devices, extremity badges, and finger rings.

12.5.2.2 Facilities

Health physics facilities at the WNP-2 plant include the following:

a. Two locker-change rooms are located in the service and radwaste buildings respectively. These rooms are provided with lockers for personnel clothing storage, clean protective clothing supplies and respiratory protective equipment. Personnel decontamination showers and sinks are located in the radwaste building building (487' level) locker change room. Temporary change areas are set up as necessary in other areas of the plant to localize and prevent the spread of contamination while performing maintenance activities. Smaller inventories of protective clothing and respiratory protective equipment are also stored in the emergency relocation centers, operation and radwaste control rooms and strategic locations throughout the plant.

b. Personnel and equipment monitoring stations are provided at the radiological access control area and various posted areas within the radiological access control area to survey for radioactive contamination prior to exiting. Monitoring equipment include low level smear counters, hand and foot monitors, and/or portable personnel monitoring equipment (friskers).

c. Medical first aid facilities are equipped to provide care for injuries, including those with radioactive contamination involved.

d. Facilities for equipment and tool decontamination exist in the Radwaste Building, Turbine Building and Reactor Building. The locations and facilities are:

(1) Radwaste Building

The general decontamination area is shown in Figure 12.3-1, approximate column location Q.1-13.2 at the 437'-0" level. Facilities include curbing, sink, monorail hoist and drains. At the 487'-0" level, Figure 12.3-5, column coordinates R.2-14.0, tools and small equipment can be decontaminated. Facilities include an ultrasonic cleaner, sink, bench space and drains.

(2) Turbine Building

Figure 12.3-1, columns H-9.5, elevation 441'-0" identifies the Turbine Building decontamination area. Facilities include a monorail, curb, sink, shower and drains.

(3) Reactor Building

The head washdown area is shown in Figure 12.3-4 at column coordinates N-5.8 at the 606'-10]" level and contains a curb and drain.

The CRD room area, Figure 12.3-3, columns M-3.4, 501'-0" elevation contains curbing, sink, monorail, bench, storage racks and tables. An additional small decontamination area exists at the same elevation, same Figure at K-8.3 and has curbing, sink bench space and shower.

e. The Health Physics/Chemistry Supervisor's office is located in the Service Building. The Health Physics Supervisor and Health Physics/Chemistry Technicians are located in the Radwaste and Service Buildings adjacent to the locker change rooms. These locations provide for ready access to the radiation protection staff by other plant workers and an area to generate and store all records.

f. A hot machine shop nd a hot instrument shop are provided in the radwaste building for work on taminated equipment under controlled conditions.

g. A laboratory complex is provided in the radwaste building consisting of a sample room, hot radiochemistry laboratory and a counting room where radioactive samples will be qualitatively and/or quantitatively analyzed.

h. Facilities for calibration of all plant health physics instrumentation are provided either at the plant or they are performed by a qualified contractor. Calibration equipment includes pulse generators, sources, and electronic test equipment with traceability to the National Bureau of Standards. Periodic maintenance and checks are also performed on all test equipment. Calibration records are maintained by computer to ensure recalibration at specified intervals. Instrument maintenance facilities include the normal maintenance instrument shop and a hot instrument shop, both located in the radwaste building. Storage of portable instruments is provided at selected points throughout the facility to make the proper instrument readily available for use. Storage areas include the two health physics areas in the radwaste and service building, cabinets inside the various buildings, the control room, and in the emergency relocation centers.

Portable monitoring instrument calibration is performed in the calibration facility located at the south end of the water treatment laboratory, Figure 12.3-6, Column E-4. The facility contains a multiple source gamma calibrator, condenser R meters, benches, tables, instrument storage cabinets and portable sampling device flow calibration equipment. Sources used for calibration are traceable to NBS standards. Survey instruments will be calibrated every six (6) months. Instruments used to monitor radiographic operations or for alpaha (α) surveys will be calibrated quarterly.

12.5.2.3 Equipment

Health physics equipment, other than instrumentation, is described below:

a. Protective clothing and accessories are provided for personnel required to work on contaminated areas. Clothing requirements for a particular task or area are prescribed by the radiation protection staff based upon the actual or potential conditions. Clothing available includes:

   (1) Coveralls and laboratory coats

   (2) Gloves - rubber and/or cotton

   (3) Head covers

   (4) Foot protection

   (5) Plastic suits - with or without supplied air.

b. Respiratory protection equipment is provided and required for personnel when levels of airborne radioactive materials approach or exceed applic able limits or when a potential for this con dition exists. The respiratory protection pro gram is conducted within the requirements of 10CFR20.103 and exposure is limited to average concentrations less than the valu. specified in Appendix B, Table 1, column 1, of 10CFR20. Allowance is made for use of respiratory protec tive equipment, as prescribed in 20.103, in determining an individual's inhalation of air- borne radioactive materials. The following types of equipment are used:

(1) Full face air purifying respirators

(2) Airline supplied full face masks. (Pressure demand regulated)

(3) Self contained breathing apparatus. (Pressure demand regulated)

c. Air sampling equipment, in addition to the continuous air monitors, includes high and low volume portable air samplers, low volume constant air samplers, and air samplers with a self-contained power source. Collection media (filters) employed are capable of collecting particulate and radioiodine samples.

d. Emergency supplies are maintained in the designated emergency relocation centers and some supplied in the control room. Detailed inventories are incorporated into the WNP-2 Emergency Plan, and include:

(1) Radiation monitoring instrumentation - both GM survey and wide range dose rate types

(2) Air sampling and counting equipment

(3) Portable electric power supplies

(4) Protective clothing

(5) Respiratory protective equipment

(6) Personnel monitoring devices

(7) Comminications equipment

(8) Medical first aid equipment

(9) Personnel decontamination supplies

Other than emergency supplies, the primary storage areas for radiation protection equipment are the two health physics control areas located in the service and radwaste buildings. Temporary storage facilities are set up in localized areas as required.

12.5.2.4 Instrumentation

The plant portable and laboratory health physics instrumentation is described in Table 12.5-1. All of this instrumentation is calibrated at least semiannually or more frequently. Checks and tests are performed at the minimum frequency required by the Technical Specifications. Electronic calibrations of instrument components are performed using test equipment traceable to the National Bureau of Standares. Overall calibration measuring instruments is performed using radioactive standards traceable to a recognized source in a known, reproducible geometry. Calibration of low level radiation detection instruments is done with a pulse generator.

12.5.3 PROCEDURES

Section 12.1.3 described a process that was incorporated into the preparation and revision of all plant procedures which provided a positive method of assuring health physics input and ALARA consideration into all radiation exposure related activities. The intent of this process is to incorporate the general guidance of appropriate Regulatory Guides plus the previous experience of power reactor radiation protection work into all applicable WNP-2 plant procedures.

12.5.3.1 Personnel Control Procedures

The WPPSS Program and Procedure Manual contains the administrative procedures for control of access to radiation and high radiation areas, and for control of time spent within these areas by all plant workers. Basically, the procedures limit entry to these areas to time required for necessary operational maintenance, and surveillance activities only. The primary tool used to ensure this control at WNP-2 is the Radiation Work Permit Procedure. Two types of Radiation Work Permits are used within this program and these are described below:

a. Regular Work Permit - This type of permit is issued for a particular task or function, and is required before entering radiation or high radiation areas when a significant amount of time will be spent in that area. This permit provides current data on radiation levels within the area of interest, any restrictions on allowable work time, protective clothing and respiratory protective requirements, information on special tools or equipment needed, special radiation safety and personnel monitoring requirements, and any other special instructions necessary. A section of the permit is used to incorporate the criteria given in Regulatory Guides 8.8, 8.10, 8.2 into each individual task, even though it has already been included in job procedures through the system for ALARA consideration described in Section 12.1.3. The Regular Work Permit requires approval from both the Operations and Health Physics groups prior to starting work and reading and initialing by personnel involved to ensure understanding.

b. Extended Work Permit - This type of permit is issued to cover entry into radiation or high radiation areas for certain routine functions, for limited time periods, and where the radiological conditions are known to be stable. Provision is also made on this type of permit for special instructions and requirements which ensure that personnel using this type of permit do not use it to complete tasks requiring a Regular Work Permit.

In addition to the administrative controls used at WNP-2, certain physical controls are established which restrict entry to radiation and high radiation areas to personnel authorized to enter them. Radiation areas are posted as required by 10CFR20, and high radiation areas are locked or otherwise controlled as specified by this same regulation.

The plant security control system complements both the administrative and physical restraints described by allowing access to specific plant areas only to personnel required to be in them.

12.5.3.2 As Low As Reasonably Achievable (ALARA) Procedures

The procedures described here for assuring ALARA occupational radiation exposure at the WNP-2 plant are in addition to the ones described in 12.1.3 and above in 12.5.3.1 and include:

a. Post Operation Review - This is conducted by the Health Physics Supervisor or his appointed alternate for tasks determined to be major dose contributors in the plant. The review is made either with members of the department involved, or in cases involving large numbers of personnel, with the Plant Operations Committee. Purpose of the review is to point out the sources of exposure and jointly provide methods of reducing this exposure.

b. Pre-Operation Instruction - This type of instruction is provided in general terms in the training courses described separately, and is provided for specific tasks by inclusion on the work permit, by direct verbal communication, and in the case of large projects involving considerable exposure, by group sessions for that purpose. This type of instruction may vary from simple voice communications to the use of "dry runs" where it is considered feasible.

c. Remote Handling Procedures - Use of this technique and special tools and equipment are factored into work wherever practicable.

d. Recordkeeping Procedures - These records are maintained in a fashion to provide data that will point out problem areas, activities that contribute significantly to exposure, personnel or groups of personnel receiving high exposures, and will otherwise assist in conducting the ALARA program.

All plant procedures are reviewed by the Health Physics/ Chemistry section for ALARA considerations. Health physics requirements, prerequisites and precautions are incorporated into these procedures during the procedure acceptance review stage. In addition, a Radiation Work Permit is issued for those tasks having radiological implications. Special procedures for such tasks as ISI work, outage work and refueling will be reviewed for ALARA considerations in the same way as the plant operations and maintenance procedures. These special procedures may, in addition, require Radiation Work Permits where the specific radiological controls and conditions are specified.

BLANK

The above methods of assuring as low as reasonably achievable occupational radiation exposure describes the WNP-2 philosophy only in general terms, since all health physics procedures are written to reduce radiation exposure either directly or indirectly.

12.5.3.3 Radiation Survey Procedures

A radiation survey is here defined as an evaluation of the direct radiation present. A routine radiation survey at WNP-2 consists of measuring the exposure rates throughout the plant, from all types of radiation present, in order to detect changes in these rates and keep an up to date posting of radiation levels present. Periodic general radiation surveys are made by the Radiation Protection Group in all plant areas. The frequency of these routine surveys in any given area depends on the type of area, its use, its potential hazard, or the uncertainity of radiation conditions in that area. Specific radiation surveys are performed on a request or need basis. For example, a specific survey is usually conducted prior to authorizing work in radiation and high radiation areas to determine protection requirements and plan exposure reduction methods. Continuing radiation surveys are made of occupied areas if there is potential for significant change of radiation level while these areas are occupied.

The frequency of direct radiation surveys in any area is determined by the conditions listed in the preceding paragraph. The following are considered minimum frequencies.

<u>Daily</u> - Check the area radiation monitoring system for changes in levels throughout the plant. Investigate and determine source of any significant increases. Check all personnel monitors, air monitors, and other continuously operating radiation monitors for changes in background and investigate any unexplained increases.

<u>Weekly</u> - Perform a survey of all occupied controlled areas and update survey records as required.

<u>Annually</u> - Perform a survey of all buildings within the restricted area.

<u>Non-Routine</u> - As requested and when known changes in facility, design or operations parameters exist.

A survey of all normally inaccessible areas is performed after each shutdown or prior to the first entry into an area. Areas are posted or de-posted as conditions dictate and survey record sheets are updated.

Procedures for conducting radiation surveys are given in the WPPSS Health Physics Program and Procedures manual.

12.5.3.4 Procedures for Radioactive Contamination Control

This section describes the bases and methods used for monitoring and control of radioactive contamination on personnel, equipment and plant surfaces. The following limits are used for release of materials and equipment from the plant controlled areas:

a. "Loose", "Smearable", or "Transferable" Contamination - Is kept at a minimum and maintained below 1,000 dpm/100cm^2 β, γ and 100 dpm/100cm^2 α to be considered clean. α surveys are not routinely made unless plant conditions indicate α may be present.

b. "Fixed" Contamination - Is measured by direct survey, and a surface is considered radioactively contaminated if it exceeds 100 CPM above background per GM probe area, with the background less than 500 CPM. Measurements are made at an average distance of 1 cm, through a maximum absorber thickness of 7 mg/cm^2. Fixed contamination is further defined as that radioactive material remaining after successive attempts at removal by approved procedures which are no longer effective.

Procedures for personnel entering and leaving contaminated areas and the associated survey requirements are given in the WPPSS Health Physics Manual.

12.5.3.5 Procedures for Control of Airborne Radioactivity

Evaluation of airborne radioactivity concentrations is done procedurally by several methods. Routine airborne surveys consist of observing the continuous air monitors located in various areas of the plant and also the effluent monitors. These observations are supplemented by grab samples taken on a routine basis and by laboratory analysis of selected particulate and charcoal filters used on the continuous monitors. Special airborne surveys are made with portable samplers when a constant air monitor indicates increases in airborne radioactivity, or to evaluate conditions in a specific area or on a specific job.

The portable air sampling equipment consists of both high and low volume collectors with appropriate media for collecting particulates and radioiodines. These samplers are used for both spot evaluations by collection of grab samples and longer term evaluations by use of low volume samplers to collect over the period of a specific job or activity. Laboratory analysis is made of air samples for gross radioactivity and, where warranted, for specific isotope identification and quantification to determine and record airborne concentrations.

Selected numbers of the routine air samples collected are analyzed for specific isotope content to ensure that MPC levels are not being approached. Special samples are taken for this purpose whenever unexplained increases occur on constant air monitors or when gross activity levels indicate there is a potential for having 25% of the value specified in 10CFR20, Appendix B, Table 1, Column 1 of any isotope present in the mixture.

Airborne radioactive iodine monitoring includes integrated sample collection and laboratory analysis plus portable sampling and analysis. Portable sampling encompasses iodine collection on charcoal cartridges of nominal dimension of 2" disc diameter by 1" thickness at calibrated flow rates up to 2.5 scfm. Duration of sampling is determined by the anticipated ambient concentration levels whereas a nominal sampling period in excess of five minutes is selected to minimize sampling errors. Where gross noble gas concentrations exist, the sample cartridge may be purged in the laboratory with clean filtered air to minimize noble gas interferences. The cartridge will be sealed in a clean plastic bag and taken to the analytical laboratory counting room for analysis.

Areas are barricaded and posted as airborne radioactivity areas whenever average concentrations in that area exceed 25% of the values specified in 10CFR20, Appendix B, Table 1, Column 1. Entry into airborne radioactivity areas without approved respiratory protective equipment is not permitted until it is determined by isotope measurement that exposure during a 40 hour week will be limited to average concentrations less than the values specified in Appendix B, Table 1, Column 1 of 10CFR20. Respiratory protective equipment is also required when a significant potential for an airborne hazard exists, or when entering an area of unmonitored, unknown airborne contamination.

Various procedures for control and reduction of airborne activity are incorporated in the WPPSS Health Physics Manual, and include proper use of the ventilation system, use of specially designed equipment to collect radioactive airborne contaminants, methods for reducing and containing contamination to prevent it becoming airborne, and procedures for cleanup of primary water prior to opening this system.

The respiratory protective equipment program is designed to meet the regulation specified in 10CFR20.103 and the guidelines established for compliance. Procedures for fitting, training, maintenance, and testing of the respiratory protective equipment are included. All equipment is required to have appropriate NIOSH/Bureau of Mines approval if available. Unless the requirements are met, the protection factors are not used. Unapproved equipment may be used in some instances where reduction of intake of radioactive material will result, but no protection factor is taken for its use. An example of this is use of charcoal cartridges in atmospheres where radioiodines are present to reduce the inhalation of these materials.

12.5.3.6 Procedures for Handling and Storage of Special Nuclear Materials (SNM)

WNP-2 has implemented a program to assure the safe storage, handling, and use of sealed and unsealed nuclear, source, and byproduct materials. Included in the program are procedures for the following:

a. Receiving and opening shipments as required by 10CFR20.205.

b. Storage of licensed material as required by 10CFR20.207.

c. Control-sealed and unsealed sources are shielded as necessary.

d. Posting - all radioactive materials are posted and labeled in accordance with 10CFR20 requirements.

e. Leak tests - sources are checked for leakage or loss of material at least semi-annually.

f. Disposal - all licensed material disposals are made in accordance with 10CFR20 requirements. or by transfer to an authorized recipient as provided in 10CFR30, 40, or 70.

Procedures for handling unsealed sources, such as the liquid standard solutions used for calibration of plant instrumentation are provided. Included are provisions for ventilation control, shielding, waste collection, contamination control, and monitoring.

Procedures for handling high level sealed sources incorporate remote handling devices, shielding, and personnel monitoring.

The WPPSS Health Physics Programs and Procedures Manual includes radiation safety procedures and instructions to personnel involved in handling and assigns responsibilities for control and monitoring of these materials. The WNP-2 Health Physics/Chemistry Supervisor, Health Physics Supervisor, and Chemistry Supervisor are responsible for control and monitoring of sealed and unsealed source and byproduct materials. The WNP-2 Nuclear Engineer is accountable for SNM. Monitoring during handling of these materials is provided by the health physics group.

Required byproduct, source, or special nuclear materials in the form of reactor fuel; sealed neutron sources for reactor startup; sealed sources for reactor instrument and radiation monitoring equipment calibration; or as fission detectors; will be limited to the amounts required for reactor operation.

12.5.3.7 Personnel Dosimetry Procedures

12.5.2 describes the monitoring devices used to provide the primary legal records of exposure incurred by personnel and additional equipment used to backup and supplement this data. Records of exposure are maintained for each individual for whom personnel monitoring is required by 10CFR20.202. Such record is kept on a form containing all the information required by NRC Form 5. If individuals are permitted to receive exposure to radiation in excess of the limits specified in paragraph (a) of 10CFR20.101, a Form NRC-4 is prepared prior to the exposure, in accordance with the requirements of 10CFR20.102. Reports to individuals of records required by 10CFR20.401 are issued as required by 10CFR19.13. Reports of exposure to the Commission are issued as required by 10CFR20.405 and 10CFR20.408.

Procedurally, personnel dosimeter badges are normally processed monthly for radiation workers, but may receive interim processing if an abnormal exposure is suspected. The pocket dosimeters, and other auxiliary monitoring devices, are used to maintain an estimate of an individual's dose during the interim period between processing of dosimeter badges. The use of dosimeters as a permanent record of an individual's dose is restricted to times when badges are lost or damaged, or give a false result. When a large discrepancy exists between the two devices, it is established that the TLD is in error before the dosimeter result is assigned as the permanent record.

Plant supervisors are notified of their assigned worker's exposure status, and are responsible for maintaining these and their own exposure to as low as reasonably achievable and within specified limits.

A documented quality assurance program is required of any contractor providing the plant badge service. The contractor's service is audited and cross checked to verify the quality of the program.

Internal dose assessment is primarily determined by whole body counting. Annual whole body counts are normally performed on WNP-2 personnel who work routinely in the restricted areas. Representative personnel are given a thyroid count at this time. Personnel are also given a whole body count if it is suspected or known that they were exposed to an amount of airborne radioactivity in excess of those allowed by 10CFR20.

Whole body counting is performed at WPPSS facilities or by a commercial contractor. A dose assessment is made when the measured deposition is significant (i.e. 10% MPBB).

Since ^{90}Sr and ^{3}H are not measurable by whole body counting urinalysis will be made when the plant radiation surveillance program indicates a potential need. This is based on in-plant airborne analysis indicating that an individual could inhale the equivalent of 10% of the MPBB of either of these radionuclides.

All results obtained from whole body counting, urinalysis, and other methods of internal assessment become part of the individual's record.

12.5.3.8 Radiation Protection Surveillance Program

Previous sections have described the procedures, methods, and equipment used in the WNP-2 radiation protection program. This section describes the practices incorporated into the overall structure to ensure that the health physics program is maintained at a high level and upgraded to meet new requirements and problems.

a. 12.1.1 describes the organization structured to provide assurance that the ALARA policy is effective. It is also pointed out in this section that the plant radiation protection program has several levels of review from a performance standpoint.

b. 12.1.3 describes the process for review of all plant procedures for ALARA consideration. This also includes all procedures revised and new procedures developed, and so ensures health physics consideration continuously in all plant operations.

c. The Special Work Permit Program and other records previously described provide a valuable source of information and are used to determine where the occupational radiation exposures are occurring and as a means of review for possible methods of exposure reduction.

d. The Health Physics Supervisor and his staff work on an individual and group basis with other plant organizations to determine what their principal sources of exposure are and to look for methods of reducing these exposures.

e. Plant administrative practices provide for regular review and updating of all procedures.

f. Procedures provide for routine maintenance, calibration, and testing of all radiation instrumentation and equipment. New equipment will be added as necessary for replacement and to supplement that existing. Written procedures are provided for use of equipment where required.

g. Plant facilities are routinely reviewed for possible improvements from a radiation protection standpoint. 12.1.3 describes several changes that have been incorporated into plant design for this purpose. Other considerations are additional shielding where practicable, improved ventilation control, additional equipment, and increased physical restriction.

h. The routine and special surveys previously described point out levels of radioactive contamination in plant areas. The WNP-2 staff is committed to maintaining a clean plant and considers it routine procedure to reduce levels of contamination whenever such action will not result in an increase of occupational radiation exposure to personnel.

i. One aspect that is considered important and used in implementing the radiation protection program is the incorporation of previous reactor and power reactor experience in this area. Previously successful methods, procedures, and equipment are used whenever possible.

j. Training of all personnel who work in the plant in radiation safety practices is mandatory and given a high priority by Supply System and WNP-2 Management. The System Training Manager is supplemented by a full time Training Coordinator in each plant and is responsible for development of all training programs, including radiation safety indoctrination.

Radiological Programs personnel and Plant Radiation Safety Staff personnel assist in this training by providing instructors for some phases. The degree of training provided each plant worker is dependent on his function and degree of responsibility, however, all radiation workers in the plant are provided training considered necessary or required for their position. The training programs provided are designed to meet the requirements of 10CFR19.12 and the guidance of Regulatory Guides 8.8, 8.10, and 1.8.

TABLE 12.5-1

WNP-2 PLANT HEALTH PHYSICS INSTRUMENTATION

Number	Type	Minimum Accuracy	Minimum Radiation
1	Multichannel γ Energy Analytical System with GE-Li Detector	Not Applicable	Not Applicable
1	Multichannel γ Energy Analytical System with NaI Detector	Not Applicable	Not Applicable
3	β - γ Proportional Counters	Not Applicable	Not Applicable
1	Single Channel Analyzer - Well Detection Systems	Not Applicable	Not Applicable
1	Liquid Scintillation Counter	Not Applicable	Not Applicable
4	Scaler - GM Detector - Shield Units	Not Applicable	Not Applicable
14	Ion Chamber Dose Rate Instruments	±10% of Full Scale	0-5R/hr
4	Telescoping Dose Rate Instruments	Full Scale ±10%	0.01 R/hr-999.9 R/hr
2	Neutron Dose Rate Instruments*	Not Specified	0 mrem/hr - 5 rem/hr
14	β - γ Survey Meters with Standard, End Window, or Pancake Type Probe	±10% of Full Scale	0-70,000 CPM
2	Alpha Counters - Gas Proportional Portable Survey Instruments	±10% of Full Scale	0-500,000 CPM
10	Frisker Type Personnel Contamination Monitors	±10% of Full Scale	0-500,000 CPM
5	Hand and Foot Counters	±10% of Full Scale	0-50,000 CPM
1	Portal Monitor	Not Specified	Not Specified
1	Condenser R-Meter	±3% of Full Scale	0-250R
3	Portable Area Monitors with Adjustable Alarm	±15% of Actual Intensity	1-1000 mr/hr
6	Constant Air Monitors - Moving Filter β Detector	±10% of Actual Intensity	50-50,000 CPM
1	Portable Constant Air Monitors	±10% of Full Scale	0-50,000 CPM
900	Personnel Dosimeters - Direct Reading	+20% to -10%	0-500 mR
100	Personnel Dosimeters - Direct Reading High Ranges	+20% to -10%	Various Ranges to 100 R
6	Personnel [illegible] Dose Integr[illegible]	Not Specified	Not Specified

* Provides d[illegible] mrem/hr for neutrons with energies between 0.[illegible] [illegible]KeV, directional response ±10%

REFERENCE 5

EDUCATIONAL, EXPERIENCE AND TECHNICAL QUALIFICATIONS FOR ONSITE RADIATION CONTROL PERSONNEL

TRAINING AND EXPERIENCE

RG GRAYBEAL - MANAGER, HEALTH PHYSICS/CHEMISTRY

TYPE OF TRAINING	WHERE TRAINED	DURATION OF TRAINING	ON THE JOB	FORMAL COURSE
a. Principles and Practices of Radiation Protection	General Electric Company Hanford Atomic Products Operation, Richland, WA	7.0 years	Yes	Yes
	Dairyland Power Coop LACBWR, LaCrosse, WI	3.5 years	Yes	Yes
	State Hygienic Laboratory University of Iowa Iowa City, IA	3.75 years	Yes	Yes
	Iowa Electric Light & Power Company Cedar Rapids, IA	3.75 years	Yes	Yes
	Washington Public Power Supply System Richland, WA	6.0 years	Yes	No
b. Radioactivity Measurement Standardization and Monitoring Techniques and Instruments	Same as a.			
c. Mathematics and Calculations Basic to the Use and Measurement of Radioactivity	Same as a.			
d. Biological Effects of Radiation	Same as a.			

EXPERIENCE WITH RADIATION (Actual Use of Radioisotopes or Equivalent Experience)

ISOTOPE	MAXIMUM AMOUNT	WHERE EXPERIENCE WAS GAINED	DURATION OF EXPERIENCE	TYPE OF USE
Mixed Fission & Corrosion Products	Curie Quantities	Same as a.	25 years	Reactor Operation Waste Disposal Instrument Calibration Analytical Procedures

TRAINING AND EXPERIENCE

LG BERRY - SUPERVISOR, HEALTH PHYSICS

	TYPE OF TRAINING	WHERE TRAINED	DURATION OF TRAINING	ON THE JOB	FORMAL COURSE
a.	Principles and Practices of Radiation Protection	U.S. Navy Nuclear Propulsion Power School Mare Island, California Idaho Falls, Idaho	6.0 years	Yes	Yes
		General Electric Company Vallecitos Nuclear Research Pleasanton, California	2.75 years	Yes	Yes
		Lawrence Livermore Lab. University of California/DOE Livermore, California	1.0 year	Yes	Yes
		Washington Public Power Supply System Richland, Washington	1.5 years	Yes	Yes
b.	Radioactivity Measurement Standardization and Monitoring Techniques and Instruments	Same as a.			
c.	Mathematics and Calculations Basic to the Use and Measurement of Radioactivity	Same as a.			
d.	Biological Effects of Radiation	Same as a.			

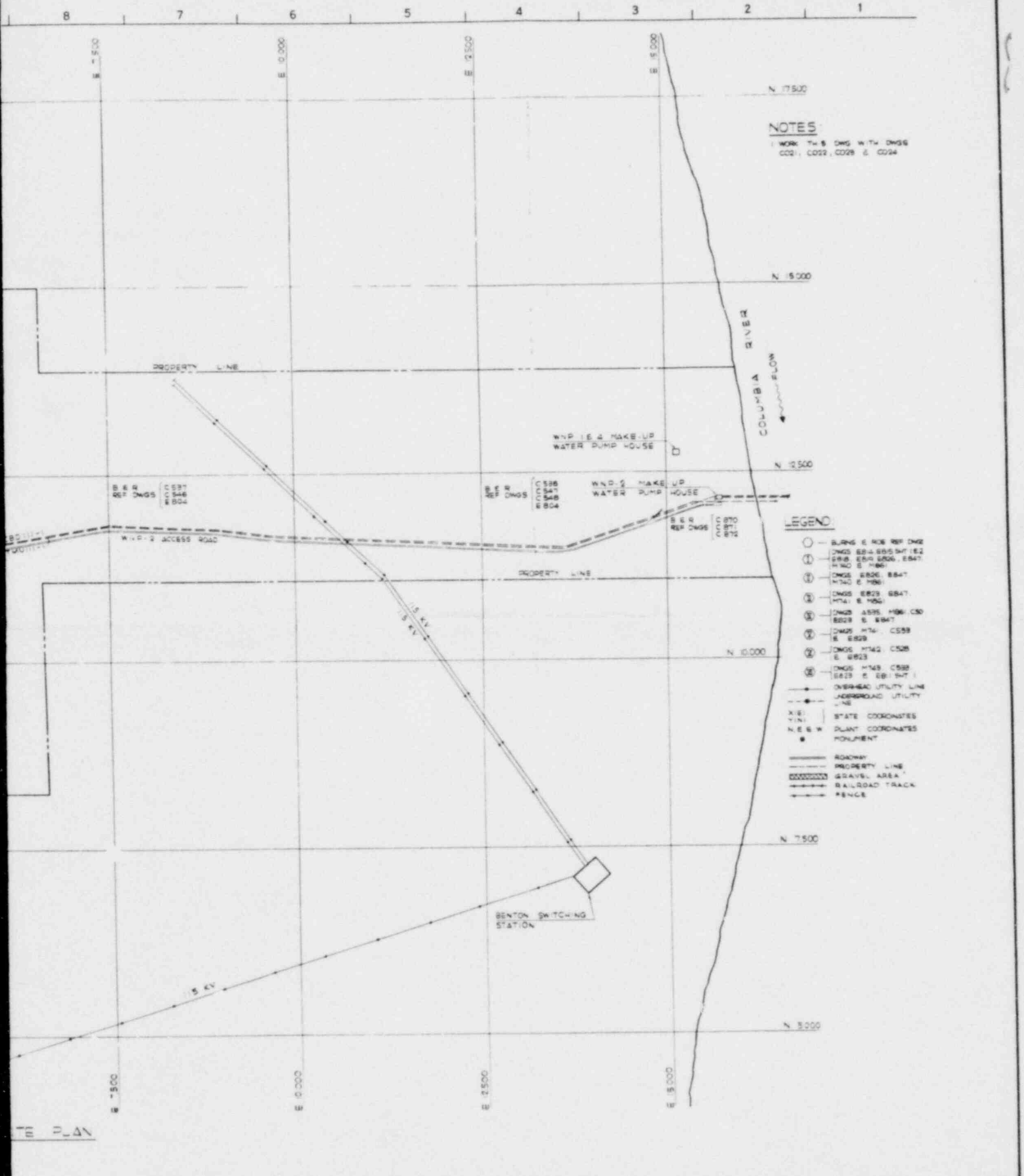
EXPERIENCE WITH RADIATION (Actual Use of Radioisotopes or Equivalent Experience)

ISOTOPE	MAXIMUM AMOUNT	WHERE EXPERIENCE WAS GAINED	DURATION OF EXPERIENCE	TYPE OF USE
Mixed Fission & Corrosion Products	Curie Quantities	Same as a.	12 years	Reactor Operation Waste Disposal Instrument Calibration Analytical Procedures

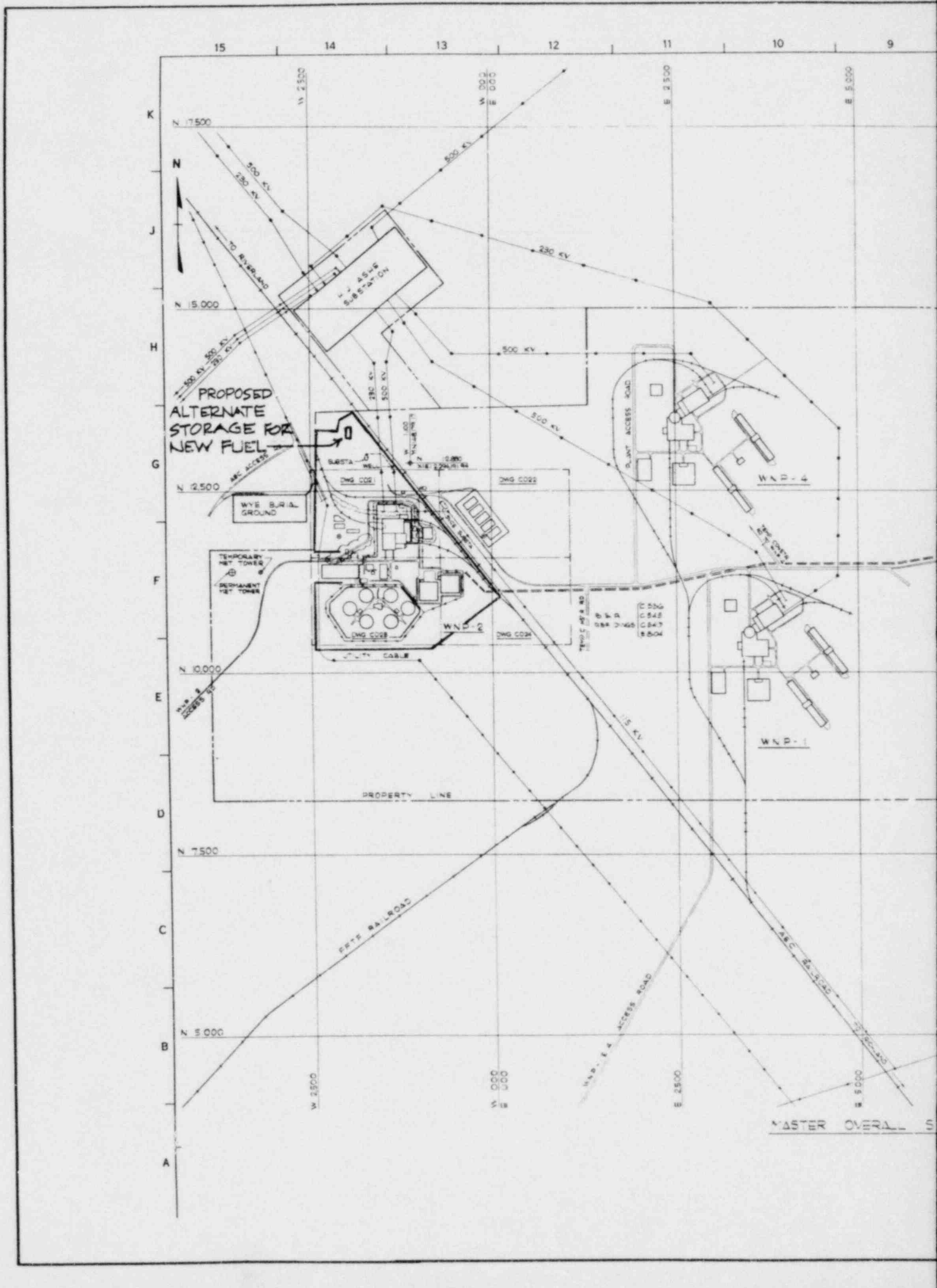
REFERENCE 6

ALTERNATE STORAGE AREA DRAWINGS

1. Alternate Storage - New Fuel
2. Alternate New Fuel Storage Facility - Plan, Sections, and Details
3. Alternate New Fuel Storage Facility - Misc. Sections and Details



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ALTERNATE STORAGE NEW FUEL



PROPOSED ALTERNATE STORAGE FOR NEW FUEL
WNP-2
WNP-4
WNP-1
WYE BURIAL GROUND
TEMPORARY MET TOWER
PERMANENT MET TOWER
PROPERTY LINE
UTILITY CABLE
PLANT ACCESS ROAD
FFTF RAILROAD
AEC RAILROAD
WNP-1 & 4 ACCESS ROAD
AEC ACCESS
TO RIVERLAND
115 KV
230 KV
500 KV
SUBSTA.
N 17,500
N 15,000
N 12,500
N 10,000
N 7,500
N 5,000
W 2500
W 0,000
E 2500
E 5,000
MASTER OVERALL S