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February 17, 1982

U.S. NUCLEAR REG. COMMISSION NMSS MAIL SECTION

G. CARL ANDOGNINI
VICE PRESIDENT
ELECTRIC OPERATIONS

Director of Nuclear Material Safety and Safeguards
US Nuclear Regulatory Commission
Washington, D. C. 20555

Attn: Mr. Norman Ketzlach

Subj: Special Nuclear Material License
Response to NRC Questions for
Palo Verde Nuclear Generating Station
Docket No. 70-2949 (SNM-1887)

Dear Mr. Ketzlach:

Attached is a copy of additional information requested by you in a letter dated January 8, 1982, to support the Arizona Public Service Company application for authorization to received, store, inspect and transport Special Nuclear Material and excore and incore detectors.

Should you have any questions or require additional information, please contact Mr. Steven R. Frost at (602) 271-3348.

Very truly yours,

G Carl Andognini

GCA/JRP/jc

Attachment

cc: Director, Region V, USNRC
NRC Resident Inspector - PVNGS
NRC Project Manager - M. Licitra

FEE EXEMPT

20290

20290

1. Page 19, Section 1.1.2

Describe the fuel assemblies in sufficient detail so that a nuclear criticality safety evaluation of the fuel handling and storage may be made (e.g., number and location of fuel rods and number of void positions/fuel assembly).

PVNGS Response

The PVNGS FSAR Section 4.2 refers to CESSAR Section 4.2 for a description of the fuel assemblies. Please find attached amended CESSAR Section 4.2.2.1 which provides a description of the fuel assemblies, including associated figures and including amended Table 4.2-1 showing the mechanical design parameters. These amended CESSAR Sections were forwarded to the NRC on CE Docket STN-50-470F in response to NRC Request for Additional Information, letter dated October 8, 1981. (LD81-069) Also attached is Figure 4.3-2 which provides the first cycle assembly fuel loadings and indicates the water hole and shim placement. (Reference Attachment 1) These changes will be included in a future CESSAR amendment.

2. Page 19, Section 1.1.3

Specify the maximum ^{235}U enrichment in a fuel assembly (only the "nominal" enrichment is stated).

PVNGS Response

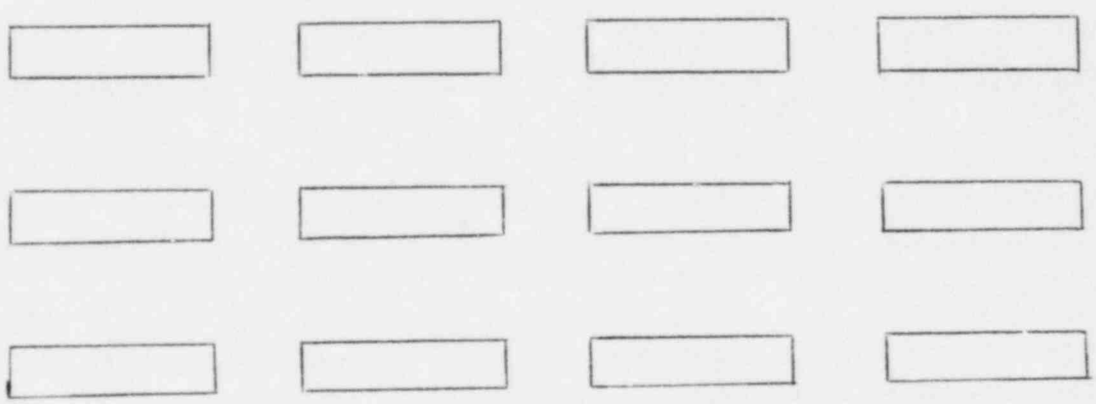
The maximum K_{eff} values have been calculated for enrichments up to 4.0 w/o U-235 for the new fuel storage, spent fuel storage, and the intermediate storage arrays. A description of the safety evaluation, including criticality safety for the storage arrays, are found in FSAR Sections 9.1.1.3.1, 9.1.2.3.1, and 9.1.2.4.2.1 (Attachment 2). Also, refer to response in Question No. 17.)

3. Page 20-21, Section 1.2.1

a. Describe the shipping container laydown array when the fuel assemblies are only in their inner container and in both their inner and outer containers and the maximum size of the array (total number of containers, number of containers in a stack).

PVNGS Response

Shipping containers are provided by Combustion Engineering (C-E) and are described in their application for SNM-1067 dated December 1980 (NRC Docket Number 70-1100). C-E shipping containers consist of two halves of a single barrier instead of an inner and outer container. Fuel assemblies will be stored in the shipping container laydown area in complete and closed shipping containers. No more than 20 shipping containers containing fuel assemblies will be allowed in the shipping container laydown area at any given time. No more than two containers will be stacked vertically, in an array of no more than 4 containers in the direction of the length of the container and 3 containers abreast.



3. Page 20-21, Section 1.2.1

   b. Provide the basis for the nuclear criticality safety of the laydown arrays.

PVNGS Response

C-E's fuel fabrication facility license (SNM-1067 Rev. 0, Docket Number 70-1100), on page II .8-50 provides an analysis of an array of containers stacked 3 high and of infinite extent in the horizontal plane, with no separation between containers, and concrete reflectors located above and below the array. The analysis concludes that the array is criticality safe under dry and flooded, with varying density water, conditions. This analysis envelops the proposed PVNGS shipping container array, and thus assures criticality safety for such an array.

3. Page 20-21, Section 1.2.1

c. Describe the maximum number of fuel assemblies in the inspection area (e.g., maximum number of assemblies at an inspection station, minimum edge-to-edge distance between individual assemblies and between the inspection area and the fuel storage area).

PVNGS Response

No more than one new fuel shipping container (two fuel assemblies) will be in the new fuel inspection station area at any time and no more than one fuel assembly will be out of a shipping container at a time in the new fuel inspection area. Procedures will require maintaining at least 6 inches edge-to-edge separation between fuel assemblies (the edge to edge distance provided in the C-E shipping container). The minimum possible edge-to-edge distance between a fuel assembly in the new fuel inspection area and one in the new fuel storage area is 48 inches.

3. Page 20-21, Section 1.2.1

   d. Specify the minimum center-to-center spacing between fuel assemblies in the new fuel storage racks and between fuel assembly storage locations in the spent fuel pool racks.

PVNGS Response

The minimum center-to-center spacing between fuel assemblies in the new fuel storage racks, containment fuel racks, and the spent fuel pool racks is as follows:

1) New Fuel Racks

   Minimum center-to-center

   18-1/8" length wise
   2' - 7-1/8" width wise

   (Refer to Joseph Oat Corp. Drawings 6676, 6675, 6681) (Attachment 3)

2) Containment (Intermediate) Fuel Racks

   Minimum center-to-center - 18-9/16"

   (Refer to Joseph Oat Corp. Drawings 6710, 6681) (Attachment 3)

3) Spent Fuel Pool Racks

   Minimum center-to-center

   13.287" (checkerboard arrangement)
   9.395" (high density arrangement)

   (Refer to CE Drawings E-4679-662-002, E-4679-667-006, E-4679-667-010) (Attachment 3)

3. Page 20-21, Section 1.2.1

   e. Correct the identification of the figure for the locations of the new fuel storage areas.

PVNGS Response

Figure 9.1-2, "Fuel building crane travel limits" shows the location of the new fuel storage areas in the fuel building. Figure 1.2-6, "General arrangement plans between El. 120'-0" & El. 140'-0", and Figure 1.2-7, "General arrangement plans between El. 140'-0" & El. 200'-0", also show the location of these areas. (Attachment 4)

3. Page 20-21, Section 1.2.1

f. Describe the blocking devices used to assure the fuel assemblies are stored in a checkerboard pattern in the spent fuel storage racks. Confirm they are always in their specified locations in the racks when used to store unirradiated fuel assemblies.

PVNGS Response

The cell blocking device is a springlike component that relies on the underload trip of the spent fuel handling machine (SFHM) to prevent inadvertent loading of a blocking cell. The device is expressly designed to enhance easy remote removal in support of installation of optional poison tubes. FSAR Figures 9.1-5 and 9.1-6 indicate the configuration of the blocking devices and their location. Attachment 5)

Administrative controls will be used to confirm that the blocking devices are in their specified locations in the spent fuel pool storage racks when storing fuel in the checkerboard array.

4. Page 21, Section 1.2.2

Confirm no construction activities will be performed in the fuel handling areas during fuel handling operations.

PVNGS Response

No construction activities will be permitted in, or immediately above areas of the fuel building where and when fuel is being handled.

5. Page 21, Section 1.2.3

a. Specify whether plastic sheeting is to be wrapped around the assemblies during storage. If so, confirm the sheeting is open at the bottom so that water would drain freely from the assemblies in the event of flooding and subsequent draining of the storage area.

PVNGS Response

If new fuel is stored in plastic wrap, the wrap will be perforated at the bottom to allow free draining of water in event of flooding and subsequent draining of the storage area.

5. Page 21, Section 1.2.3

   b. Specify whether neutron poisons are used to ensure the nuclear criticality safety of the spent fuel pool storage array. If so, specify the quality assurance program to confirm the poisons meet specifications and are properly installed in their design locations. Specify the requirements for periodically testing their continued presence in the design locations.

PVNGS Response

As stated in Section 1.2.1 of this license application, the checkerboard fuel storage mode will be the only storage array used for the term of this license. No neutron poisons are used in this application. Section 9.1.2 of the FSAR (Attachment 2) provides a discussion of the design basis and the facility description of the spent fuel pool, including the checkerboard fuel storage mode.

6. Page 25, Section 2.1.1

   a. Specify the training requirements for personnel who may be in areas where radioactive materials are handled and/or stored.

PVNGS Response

The training described in PVNGS FSAR Section 12.5.3.8 will be required for entry into the fuel building, which will be designated a controlled area (FSAR Section 12.5.3.2.B), prior to the receipt of new fuel. (Attachment 6)

6. Page 25, Section 2.1.1

b. Identify the position responsible for all aspects of radiation safety at the Palo Verde Nuclear Station. Confirm he is responsible to see that receiving, handling and storing of radioactive materials are in accordance with approved and documented procedures.

PVNGS Response

The Radiation Protection Supervisor is responsible to the Engineering and Technical Services Manager (who reports to the Manager of Nuclear Operations) for all aspects of radiation safety at PVNGS. This position is described in PVNGS FSAR Section 13.1.2.2.2.2. He is responsible to see that receiving, handling, and storing of radioactive materials are in accordance with approved and documented procedures. (Attachment 7)

7. Page 25, Section 2.1.2

   a. Confirm the monitoring program includes all operations involving radioactive materials, including routine surveys of fuel storage areas.

PVNGS Response

PVNGS FSAR Section 12.5.3.1 describes radiation and contamination survey procedures, and will be made effective for the fuel building prior to the receipt of new fuel. This section requires monitoring operations involving radioactive materials. Routine surveys of the fuel storage areas, commencing at new fuel receipt, will be conducted. (See Attachment 8)

7. Page 25, Section 2.1.2

   b. Confirm a commitment to Section 12.5.2 (Radiation Protection Procedures) of the referenced FSAR for the Part 70 license.

PVNGS Response

PVNGS-FSAR Section 12.5.3 describes radiation protection procedures that will be made effective prior to the receipt of material and used during the duration of the Part 70 license. (See Attachment 8)

8. Page 25, Section 2.1.3

State the minimum frequency and methods for calibration and testing of portable radiation survey instrumentation and laboratory radiation instrumentation.

PVNGS Response

Portable survey instruments are calibrated at least every six months to a source traceable to NBS and the Radiation Protection Section shall affix a sticker assuring the identification of those instruments in calibration. Response checks will be made at each use or daily (whichever is less frequent) and will be used to verify that the instruments are functioning properly between calibrations.

Calibrations of laboratory radiation instrumentation will be conducted prior to the initial use with sources traceable to NBS. These instruments will undergo calibration checks at each use or daily (whichever is less frequent). These checks will be more comprehensive than the response checks noted for portable instruments and will verify the initial calibrations or be the standard information used for necessary adjustments.

9. Page 25, Section 2.2.1

   a. Confirm the array in the Fuel Building shipping container laydown area is no larger than that in a single shipment or specify the criteria for larger storage arrays. If larger arrays are to be used, provide justirication for the larger storage arrays.

   b. Identify the shipping container used (e.g., NRC Certificate of Compliance No.) and specify the number of fuel assemblies per container.

PVNGS Response

a. See Response to Item 3a and 3b.

b. The shipping containers to be used under NRC Certificate of Compliance No. 6078, Revision No. 7, are Combustion Engineering Models Nos. 927A1 and 927C1 shipping packages.

The steel fuel bundle shipping containers consist of a strongback and fuel bundle clamping assembly, shock mounted to a steel outer container. A minimum 1/4" thick, 6" x 6" x 8" high steel separators are bolted between fuel bundles. The Model No. 927A1 container is approximately 43" in diameter by 189" long with an approximate gross weight of 6,200 lbs. The Model No. 927C1 container is approximately 43" in diameter by 216" long with an approximate gross weight of 7,000 lbs. Each shipping container carries two (2) fuel bundles, and the fuel is usually shipped in six (6) containers per shipload.

10. Page 25-26, Section 2.2.2

Provide the results of the nuclear criticality safety analysis for the fuel assembly arrays in both the new fuel and the spent fuel pool storage areas.

PVNGS Response

The results of the nuclear criticality safety analysis for the new fuel and spent fuel pool storage areas shows that the maximum K_{eff} values are less than 0.95. The calculation methodology is discussed in FSAR Section 9.1.1.3.1 and 9.1.2.3.1 (See Attachment 2). Details of these calculations are available at the Combustion Engineering offices for inspection.

11. Page 26, Section 2.2.3

Provide a demonstration that shows the effect of all degrees of moderation on the K_{eff} of the arrays. Describe the method(s) of analysis used and their validation.

PVNGS Response

See response to Question No. 10.

12. Page 26, Section 2.2.4

a. Specify the maximum number of fuel assemblies outside the authorized storage locations at any time. If more than a single assembly is at a particular location, provide justification.

PVNGS Response

Prior to initial fuel load for Unit One (duration of the Part 70 license), no more than one shipping container at a time will be outside of the authorized storage locations (while in transit). Following removal from a shipping container, no more than one fuel assembly will be outside the authorized storage locations at any time.

12. <u>Page 26, Section 2.2.4</u>

b. Specify the minimum edge-to-edge spacing between single fuel assemblies or single fuel assemblies and storage arrays. Provide justification for the distance between single assemblies if less than 36 inches, between single assemblies and fuel storage arrays or between array types if less than 12 feet.

<u>PVNGS Response</u>

Fuel assemblies will be removed from their shipping containers, inspected, and stored in the new fuel storage area or the spent fuel storage area, one at a time. Thus during handling, each fuel assembly will of necessity be at least the edge-to-edge distance from another fuel assembly or array assumed in the criticality safety studies discussed in C-E's fuel fabrication facility license (see response to question 3b, or in PVNGS FSAR Section 9.1.1.3 and 9.1.2.3). (Attachment 2)

13. Page 27, Section 2.2.5

Confirm you request an exemption from the provisions of 10 CFR 70.24 in accordance with 10 CFR 70.24(d).

PVNGS Response

PVNGS is requesting an exemption from the provisions of 10 CFR 70.24(b) in accordance with 10 CFR 70.24(d) as provided in 10 CFR 70.24(c).

14. Page 27, Section 2.3

Confirm no load heavier than a fuel assembly will be moved over the fuel assembly racks when containing assemblies, the racks will withstand accidental drops by heavier loads without compromising nuclear criticality safety, or describe the conditions under which the crane interlock override and administrative controls are used to allow heavier loads to be moved over the racks without compromising nuclear safety of the storage array.

PVNGS Response

PVNGS-FSAR Sections 9.1.1.3.1 and 9.1.2.3.1 describe the criticality safety analysis done for the new and spent fuel storage racks, and include the effect of dropping a fuel assembly from a height of 2 feet above the rack onto the rack with the assembly then falling horizontally across the top of the rack or falling into an area between the rack and cavity walls or between storage locations. No loads heavier than a fuel assembly will be moved over fuel assembly racks containing fuel assemblies. Crane and hoist limits are described in PVNGS FSAR Section 9.1.1.3.1 and 9.1.2.3.1. Additional administrative controls will be in effect for the new fuel handling crane that allow crane load movement over the new fuel storage racks only when the load is less than 2 feet above the top of the racks. (See Attachment 2)

15. Pages 28-29, Section 3.2.1

a. Identify the location for the storage of the Pu-Be startup sources. Provide the security controls to assure unauthorized personnel cannot handle the sources and no one can be exposed to the source outside their shielded shipping container.

PVNGS Response

The Pu-Be startup sources will be stored in their shielded shipping containers in a fenced and locked storage area that will be made a controlled area per PVNGS FSAR Section 12.5.3.2.B. Access to the storage area will be limited to persons having need for access and authorized by the Manager of Nuclear Operations, and will be controlled by the PVNGS Security Department. Unpacking and handling of these sources will be by procedure and require appropriate radiation protection measures under the supervision of the Radiation Protection Supervisor. (See Attachment 6)

15. Pages 28-29, Section 3.2.1

b. Confirm the storage area is a controlled area and is identified as a radiation area.

PVNGS Response

See response to 15a above. FSAR Section 12.5.3.2.B requires identification of radiation areas based on dose rate. (See Attachment 6)

15. Pages 28-29, Section 3.2.1

c. Provide the radiation protection controls to be exercised upon receipt of the sources, during their storage and after the installation of a support clamp at the upper end of the Pu-Be source body.

PVNGS Response

Radiation protection controls to be exercised upon receipt of Pu-Be sources will include:

1) Neutron and Gamma dose rate surveys of the shipping container exteriors and area near the containers per PVNGS FSAR Section 12.5.3.1. (See Attachment 6)

2) Personnel dosimetry (TLD's) for persons entering the controlled area per PVNGS FSAR Section 12.5.2.2.3. (See Attachment 6)

3) Smear surveys (leak tests) upon receipt and prior to moving shipping containers out of a controlled area.

4) Radiation protection controls for unpacking and handling of these sources will be specified by procedure as discussed in our response to question 15a.

15. Pages 28-29, Section 3.2.1

d. Confirm the Pu-Be source is placed back inside the shipping container after the support clamp is in position or prcvide the remaining source handling operations and related radiation protection controls.

PVNGS Response

PVNGS procedure for handling the Pu-Be sources will require that the support clamp is positioned prior to storage of Pu-Be sources in a shipping container.

16. Page 20, Section 3.2.2

a. Confirm both the excore and incore detectors are placed in secured storage when not in use by authorized personnel and shall be used only by or under the supervision of individuals designated by the Manager, Nuclear Operations or the Supervisor Radiation Protection.

PVNGS Response

Excore and incore detectors when not in use by authorized persons are stored in areas described in amendment 1 to the Physical Security Plan, Protection of Special Nuclear Materials low Strategic Significance. The qualification of persons permitted to perform work on or unpack these detectors is specified by work procedures approved by the Manager of Nuclear Operations or his designee. Radiological protection activities associated with such use is under the supervision of the Radiation Protection Supervisor.

16. Page 29, Section 3.2.2

b. Identify the ANSI standard by title and the date of most recent issue (e.g., ANSI/ASME N45.22-1978).

PVNGS Response

Section 3.2.2 should identify ANSI N45.2.2-1972. Our commitment to this standard is in accordance with the regulatory position of Regulatory Guide 1.38 (Revision 2, May 1977) as discussed in PVNGS-FSAR Section 1.8.

17. Page 3i

Confirm 3.30% ^{235}U is the maximum fuel enrichment (including analytical and sampling errors) to be in the fuel assemblies.

PVNGS Response

Arizona Public Service Company has placed the order for the initial core with the U.S. Department of Energy for fuel enrichments of a maximum of 3.30% ^{235}U. The DOE contract requires that enrichments are delivered in the range of $\pm$ 0.05% ^{235}U from nominal values including analytical and sampling errors.

ATTACHMENT 1

ATTACHMENT 1

31

4.2.2 DESCRIPTION AND DESIGN DRAWINGS

This subsection summarizes the mechanical design characteristics of the fuel system and discusses the design parameters which are of significance to the performance of the reactor. A summary of mechanical design parameters is presented in Table 4.2-1. These data are intended to be descriptive of the design: limiting values of these and other parameters will be discussed in the appropriate sections.

4.2.2.1 Fuel Assembly

The fuel assembly (Figure 4.2-6) consists of 236 fuel and poison rods, 5 guide tubes, 11 fuel rod spacer grids, upper and lower end fittings, and a holdcown device. The outer guide tubes, spacer grids, and end fittings form the structural frame of the assembly.

The fuel spacer grids (Figure 4.2-7) maintain the fuel rod array by providing positive lateral restraint to the fuel rod but only frictional restraint to axial fuel rod motion. The grids are fabricated from preformed Zircaloy or Inconel strips (the bottom spacer grid material is Inconel) interlocked in an egg crate fashion and welded together. Each cell of the spacer grid contains two leaf springs and four arches. The leaf springs press the rod against the arches to restrict relative motion between the grids and the fuel rods. The perimeter strips contain features designed to prevent hangup of grids during a refueling operation.

The TEN Zircaloy-4 spacer grids are fastened to the Zircaloy-4 guide tubes by welding, and each grid is welded to each guide tube at eight locations, four on the upper face of the grid and four on the lower face of the grid, where the spacer strips contact the guide tube surface. The lowest spacer grid (Inconel) is not welded to the guide tubes due to material differences. It is supported by an Inconel 625 skirt which is welded to the spacer grid and to the perimeter of the lower end fitting. TYPICAL AXIAL SPACING BETWEEN THE SPACER GRIDS IS 15 23/32 INCHES.

The upper end fitting is an assembly consisting of ~~two~~ FOUR cast 304 stainless steel plates, five machined posts and helical Inconel X-750 springs. THE END FITTING attaches to the guide tubes to serve as an alignment and locating device for each fuel assembly and has features to permit lifting of the fuel assembly. The lower cast plate locates the top ends of the guide tubes and is designed to prevent excessive axial motion of the fuel rods.

The upper cast plate of the assembly, called the holddown plate, together with the helical compression springs, comprise the holddown device. The holddown plate is movable, acts on the underside of the tube sheet tubes and is loaded by the compression springs. Since the springs are located at the upper end of the assembly, the spring load combines with the fuel assembly weight to counteract upward hydraulic forces. The determination of upward hydraulic forces includes factors accounting for flow maldistribution, fuel assembly component tolerances, crud buildup, drag coefficient, and bypass flow. The springs are sized and the spring preload selected such that a net downward force will be maintained for all normal and anticipated transiet flow and temperature conditions. The design criteria limit the maximum stress under the most adverse tolerance conditions to below yield strength of the spring material. The maximum stress occurs during cold conditions and decreases as the reactor heats up. The reduction in stress is due to a decrease in spring deflection resulting from differential thermal expansion between the Zircaloy fuel bundles and the stainless steel internals.

During normal operation, a spring will never be compressed to its solid height. However, if the fuel assembly were loaded in an abnormal manner such that a spring were compressed to its solid height, the spring would continue to serve its function when the loading condition returned to normal.

The lower end fitting ASSEMBLY is a simple stainless steel casting consisting of a plate with flow holes and a support leg at each corner (total of four legs) that aligns the lower end of the fuel assembly with the core support structures alignment pins. Each alignment pin is required to position the corners of UP TO four lower end fittings. A CENTER POST IS THREADED INTO THE CENTRAL PORTION OF THE FLOW PLATE AND TACK WELDED INTO POSITION.

The four outer guide tubes have a widened region at the upper end which contains an internal thread. Connection with the upper end fitting is made by passing the externally threaded end of the guide posts through holes in the lower cast flow plate and into the guide tubes. When assembled, the flow plate is secured between flanges on the guide tubes and on the guide posts. The connection with the upper end fitting is locked with a mechanical crimp. Each outer guide tube has, at its lower end, a welded Zircaloy-4 fitting. This fitting has a threaded portion which passes through a hole in the fuel assembly lower end fitting and is secured by a Zircaloy-4 nut. This joint is secured with a stainless steel locking ring tack welded to the lower end fitting in four places.

The central instrumentation guide tube inserts into sockets in the upper and lower end fittings and is thus retained laterally by the relatively small clearance at these locations. The upper end fitting socket is created by the center guide tube post which is threaded into the lower cast flow plate and tack welded in two places. The lower end fitting socket is AN EXTENSION FROM THE CENTER POSITION of the lower end fitting ASSEMBLY. There is no positive axial connection between the central guide tube and the end fittings.

The five guide tubes have the effect of ensuring that bowing or excessive swelling of the adjacent fuel rods cannot result in obstruction of the control element pathway. This is so because:

A. There is sufficient clearance between the fuel rods and the guide tube surface to allow an adjacent fuel rod to reach rupture strain without contacting the guide tube surface.

B. The guide tube, having considerably greater diameter and wall thickness (and also, being at a lower temperature) than the fuel rod, is considerably stiffer than the fuel rods and would, therefore, remain straight, rather than be deflected by contact with the surface of an adjacent fuel rod.

Therefore, the bowing or swelling of fuel rods would not result in obstruction of the control element channels such as could hinder CEA movement.

The fuel assembly design enable reconstitution, i.e., removal and replacement of fuel and poison rods, of an irradiated fuel assembly. The fuel and poison rod lower end caps are conically shaped to ensure proper insertion within the fuel assembly grid cage structure; the upper end caps are designed to enable grappling of the fuel and poison rod for purposes of removal and handling. Threaded joints which mechanically attach the upper end fitting to the control element guide tubes will be properly torqued and locked during service, but may be removed to provide access to the fuel and poison rods.

Loading and movement of the fuel assemblies is conducted in accordance with strictly monitored administrative procedures and, at the completion of fuel loading, an independent check as to the location and orientation of each fuel assembly in the core is required.

The serial number provided on the fuel assembly upper end fitting enables verification of fuel enrichment and orientation of the fuel assembly. The serial number is also provided on the lower end fitting to ensure preservation of fuel assembly identity in the event of upper end fitting removal. Additional markings are provided on the fuel rod upper end caps as a secondary check to distinguish between fuel enrichments and burnable poison rods, if present.

During the manufacturing process, the lower end cap of each rod is marked to provide a means of identifying the pellet enrichment, pellet lot and fuel stack weight. In addition, a quality control program specification requires that measures be established for the identification and control of materials, components, and partially fabricated subassemblies. These means provide assurance that only acceptable items are used and also provide a method of relating an item or assembly from initial receipt through fabrication, installation, repair, or modification to an applicable drawing, specification, or other pertinent technical document.

4.2.2.2 Fuel Rod

The fuel rods consist of slightly-enriched UO_2 cylindrical ceramic pellets, a round wire Type 302 stainless steel compression spring, and an alumina spacer disc located at each end of the fuel column, all encapsulated within a Zircaloy-4 tube seal welded with Zircaloy-4 end caps. The fuel rods are internally pressurized with helium during assembly. Figure 4.2-8 depicts the fuel rod design.

Each fuel rod assembly includes both a serial number and a visual identification mark. The serial number ensures traceability of the fabrication history of each fuel rod component. The identification mark provides a visual check on pellet enrichment batch during fuel bundle fabrication.

The fuel cladding is cold worked and stress relief annealed Zircaloy-4 tubing 0.025 inches thick. The actual tube forming process consists of a series of cold working and annealing operations, the details of which are selected to provide the properties discussed in Section

The UO_2 pellets are dished at both ends in order to better accommodate thermal expansion and fuel swelling. The density of the UO_2 in the pellets is 10.38 g/cm^3, which corresponds to 94.75% of the 10.96 g/cm^3 theoretical density (TD) of UO_2. However, because the pellet dishes and chamfers constitute about 3% of the volume of the pellet stack, the average density of the pellet stack is reduced to 10.06 g/cm^3. This number is referred to as the "stack density".

The compression spring located at the top of the fuel pellet column maintains the column in its proper position during handling and shipping. The alumina spacer disc at the lower end of the fuel rod reduces the lower end cap temperature, while the upper spacer disc prevents UO_2 chips, if present, from entering the plenum region. The fuel rod plenum which is located above the pellet column, provides space for axial thermal differential expansion of the fuel column and accommodates the initial helium loading and evolved fission gases.

4.2.2.3 Burnable Poison Rod

Fixed burnable neutron absorber (poison) rods, Figure 4.2-9, will be included in selected fuel assemblies to reduce the beginning-of-life moderator coefficient. They will replace fuel rods at selected locations. The poison rods will be mechanically similar to fuel rods, but will contain a column of burnable poison pellets instead of fuel pellets. The poison material will

TABLE 4.2-1

MECHANICAL DESIGN PARAMETERS

(Sheet 1 of 4)

Core Arrangement

Number of fuel assemblies in core, total	241
Number of CEAs	89
Number of fuel rod locations	56,876
Spacing between fuel assemblies, fuel rod surface to surface, inches	0.208
Spacing, outer fuel rod surface to core shroud, inches	0.214
Hydraulic diameter, nominal channel, feet	0.0393
Total flow area (excluding guide tubes), ft^2	60.9
Total core area, ft^2	112.3
Core equivalent diameter, inches	143.6
Core circumscribed diameter, inches	152.46
Total fuel loading, Kg U (ASSUMING ALL ROD LOCATIONS ARE FUEL RODS)	102.7×10^3
Total fuel weight, lb UO_2 (ASSUMING ALL ROD LOCATIONS ARE FUEL RODS)	257.1×10^3
Total weight of Zircaloy, lb	71,758
Fuel volume (including dishes), ft^3	409.6

Fuel Assemblies

Batch	No. of Assemblies	Enrichment (wt%) U-235	No. of Poison Rods per Assembly
A_0	69	1.92	0
B_1	44	12 RODS WITH 1.92 208 RODS WITH 2.78	16
B_2	64	12 RODS WITH 1.92 208 RODS WITH 2.78	16
C_0	40	12 RODS WITH 2.78 224 RODS WITH 3.30	0
C_1	24	12 RODS WITH 2.78 208 RODS WITH 3.30	16
	241		

Fuel Rod array square 16 x 16
Fuel Rod Pitch, inches 0.506

Spacer Grid

Type	Leaf spring
Material	Zircaloy-4
Number per assembly	10
Weight each, lb	1.8

TABLE 4.2-1 (Cont'd.)

MECHANICAL DESIGN PARAMETERS
(Sheet 2 of 4)

Fuel Assemblies (Cont'd.)	
Bottom Spacer Grid	
Type	Leaf spring
Material	Inconel 625
Number per assembly	1
Weight each, lb WITH SKIRT	3.2
Weight of fuel assembly, lb	1436
Outside dimensions	
Fuel rod to fuel rod, inches	7.972 x 7.972
Fuel Rod	
Fuel rod material (sintered pellet)	UO_2
Pellet diameter, inches	0.325
Pellet length, inches	0.390
Pellet density, g/cm^3	10.38
Pellet theoretical density, g/cm^3	10.96
Pellet density (% theoretical)	94.75
Stack height density, g/cm^3	10.061
Clad material	Zircaloy-4
Clad ID, inches	0.332
Clad OD, (nominal), inches	0.382
Clad thickness, (nominal), inches	0.025
Diametral grap, (cold, nominal), inches	0.007
Active length, inches	150
Plenum length, inches	9.677

TABLE 4.2-1 (Cont'd.)

MECHANICAL DESIGN PARAMETERS

(Sheet 3 of 4)

Control Element Assemblies (CEA)	Full Length	Part Length
Number	76	13
Absorber elements, No. per assy.	12 & 4	4
Type	Cylindrical rods	Cylindrical rods
Clad material	Inconel 625	Inconel 625
Clad thickness, inches	0.035	0.035
Clad OD, inches	0.816	0.816
Diametral gap, inches	0.009	0.009
Elements		
Poison material	B_4C/Felt metal and reduced dia. B_4C	Inconel/B_4C
Poison length, inches	135 1/2 / 12 1/2	75/16
B_4C Pellet		
Diameter, inches	0.737/0.674	0.737
Density, % of theoretical density of 2.52 g/cm^3	73	73
Weight % boron, minimum	77.5	77.5

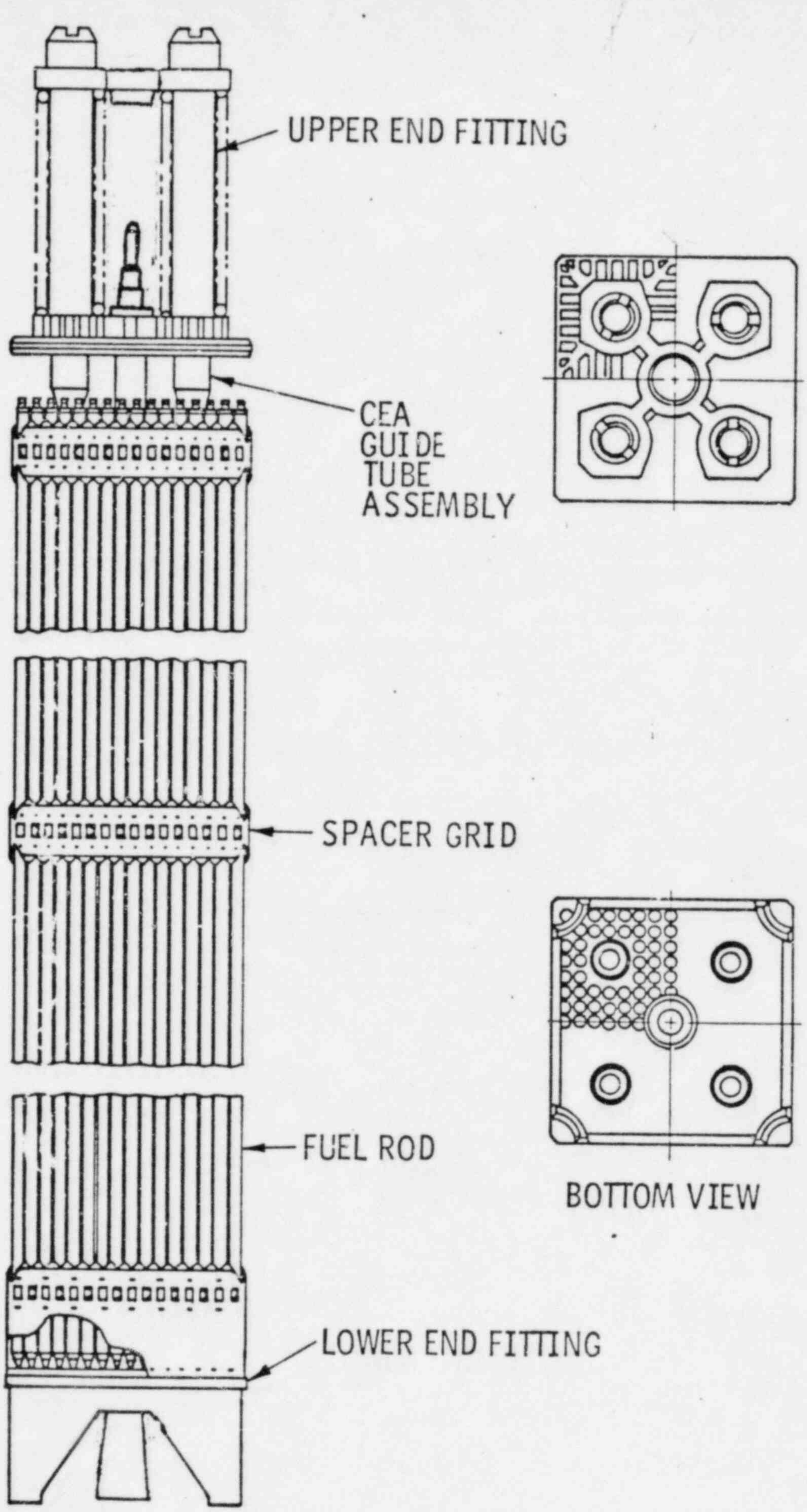
Burnable Poison Rod

Absorber material	Al_2O_3-B_4C
Pellet diameter, inches	0.307
Pellet length, inches, min	0.500
Pellet density (% theoretical), min	93
Theoretical density, Al_2O_3, g/cm^3	3.94

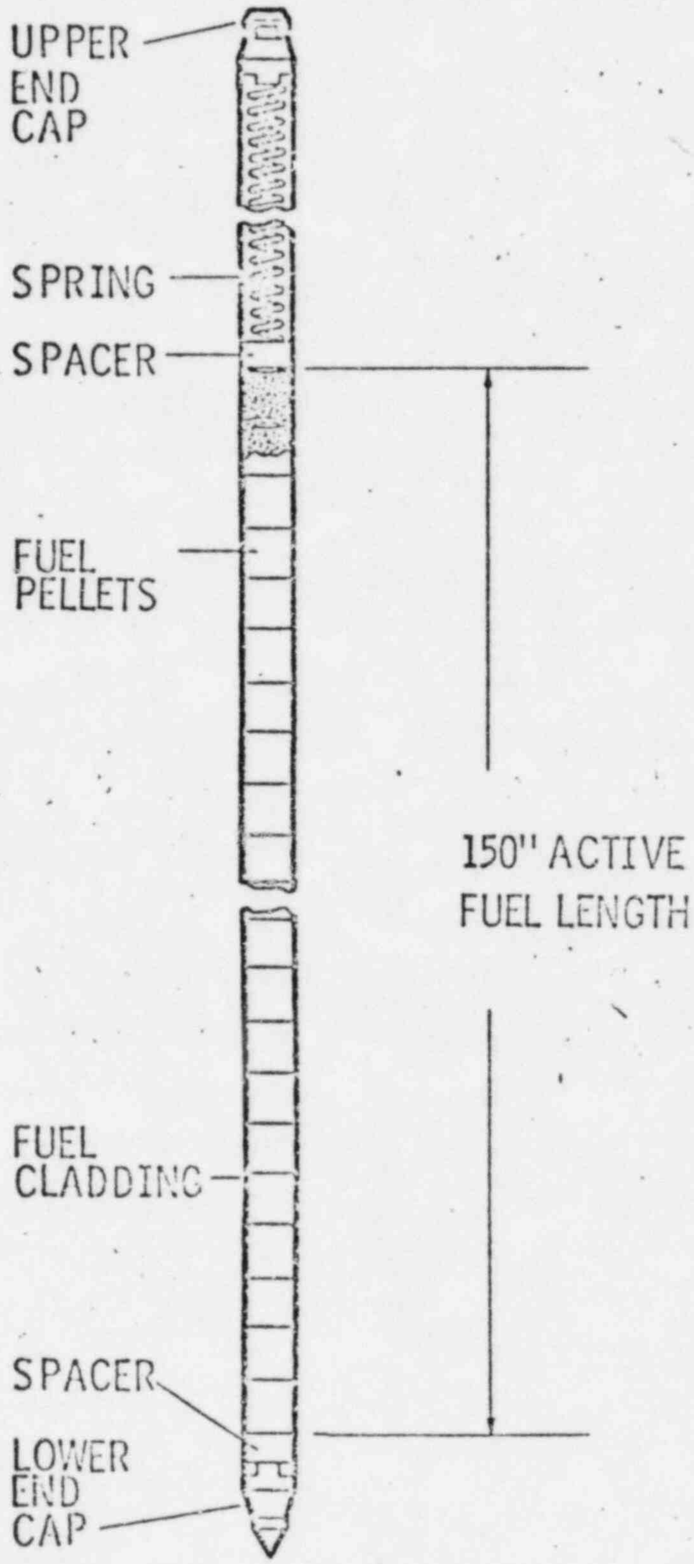
TABLE 4.2-1 (Cont'd.)

MECHANICAL DESIGN PARAMETERS
(Sheet 4 of 4)

Burnable Poison Rod (Cont'd.)	
Theoretical density, B_4C, g/cm^3	2.52
Clad material	Zircaloy-4
Clad ID, inches	0.332
Clad OD, inches	0.382
Clad thickness, (nominal), inches	0.025
Diametral gap, (cold, nominal), inches	0.025
Active length, inches	136.0
Plenum length, inches	11.090



C-E SYSTEM 80	FUEL ASSEMBLY	Figure 4.2-6



C - E SYSTEM 80	FUEL ROD	Figure 4.2-8

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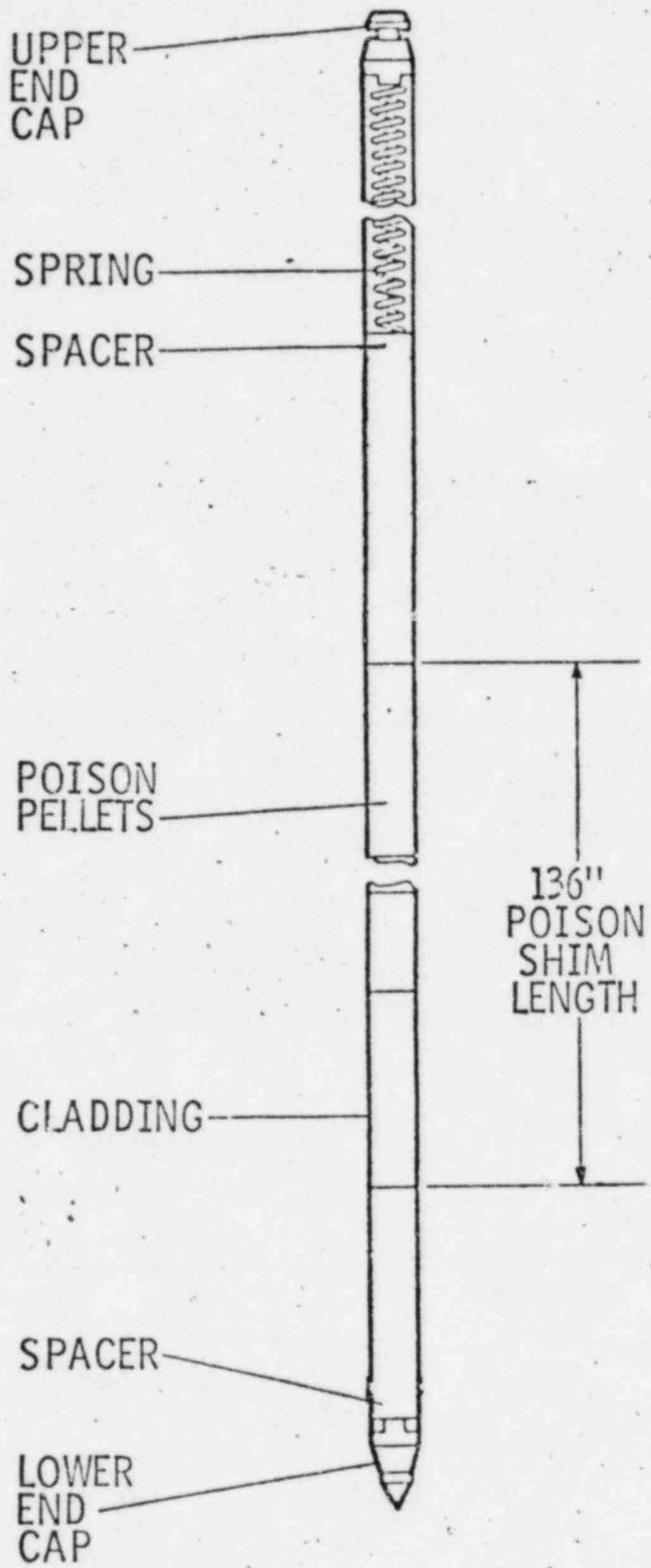
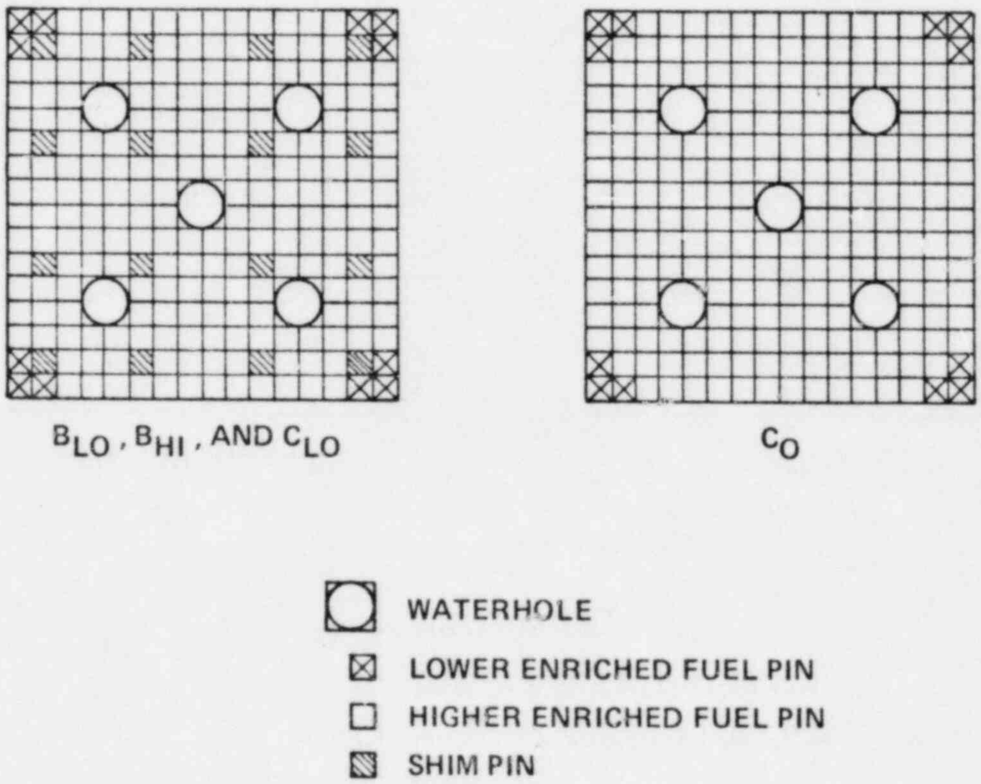


C-E SYSTEM 80

BURNABLE POISON ROD

Figure 4.2-9

ASSEMBLY TYPE	NUMBER OF ASSEMBLIES	FUEL ENRICHMENT W/T % U235	No. OF FUEL RODS PER ASSEMBLY	No. OF SHIM RODS/ ASSEMBLY	gm B^{10} /IN.
A	69	1.92	236	0	–
B_{LO}	44	1.92 2.78	12 208	16	0.01842
B_{HI}	64	1.92 2.78	12 208	16	0.02532
C_O	40	2.78 3.30	12 224	0	–
C_{LO}	24	2.78 3.30	12 208	16	0.01151



Amendment No. 3
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C-E SYSTEM 80

FIRST CYCLE ASSEMBLY FUEL LOADINGS
WATERHOLE AND SHIM PLACEMENT

Figure 4.3-2

ATTACHMENT 2

9. AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

9.1.1 NEW FUEL STORAGE

9.1.1.1 Design Bases

The following design bases are imposed on the storage of new fuel:

A. Accidental criticality shall be prevented for the most reactive arrangement of new fuel stored, with optimum moderation, by assuring that K_{eff} is less than 0.98. This design basis shall be met under any normal or accident conditions.

B. The requirements of Regulatory Guide 1.13 shall be met.

C. The storage racks and facilities shall be qualified as Seismic Category I.

D. Storage shall be provided for at least one-third core of new fuel.

9.1.1.2 Facilities Description

The rack assemblies are made up of individual racks similar to those shown in figure 9.1-1. A minimum edge-to-edge spacing between fuel assemblies, as required by section 9.1.1.3.1, is maintained between assemblies in adjacent rows. These spacings are the minimum values after allowances are made for rack fabrication tolerances and the predicted deflections resulting from postulated accident conditions, discussed in section 9.1.1.3.1.

The specific location of the new fuel racks in the fuel building is shown in figures 1.2-6, 1.2-7, 1.2-12, and 9.1-2.

The stainless steel construction of the storage racks is compatible with water and zirconium clad fuel.

The top structure of the racks is designed such that there is no opening between adjacent fuel cavities that is as large as the cross-section of the fuel bundle. In addition, the outer structure of the racks precludes the inadvertent placement of a bundle against the rack closer than the prescribed edge-to-edge spacing.

9.1.1.3 Safety Evaluation

The new fuel storage rack design and location, discussed in section 9.1.1.2, ensures that the design bases of section 9.1.1.1 are met. The capability of PVNGS new fuel storage is described below.

9.1.1.3.1 Criticality Safety

The following postulated accidents were considered in the design of the new fuel storage racks.

A. Flooding; complete immersion of the entire storage array in pure, unborated, room temperature water.

B. Envelopment of the entire array in a uniform density aqueous foam or mist of optimum density that maximizes the reactivity of the array. It is postulated that these conditions could be present as a result of fire fighting.

C. A fuel assembly dropped from a height of 2 feet onto the rack which then falls horizontally across the top of the rack.

D. Tensile load on the rack of 5000 pounds (limited by adjustment of the motor stall torque or load limiting device of the crane used to load fuel into the racks.)

Although the above accident conditions have been postulated, the fuel handling equipment, new fuel racks, and the building arrangement are designed to minimize the possibility of these accidents or the effects resulting from these accidents by:

A. Providing positive hoist travel limits and interlocks to ensure proper equipment operation and sequencing.

B. Limiting the crane loads when installing fuel into or removing fuel from the fuel rack.

C. Designing the new fuel racks for SSE conditions and dropped fuel bundle conditions.

D. Maintaining K_{eff} less than 0.95 in the event the fuel area becomes flooded.

E. Designing the new fuel handling crane to preclude the new fuel handling crane, or any part thereof, from falling into the new fuel handling area.

The following assumptions are made in evaluating criticality safety.

A. Under postulated conditions of complete flooding by unborated room temperature water, the storage array is treated as an infinite array of assemblies having an infinite fueled length.

B. Under postulated conditions of envelopment by aqueous foam or mist, a range of foam or mist densities is examined to ensure that the maximum reactivity of the array is established. The foam or mist is assumed to be pure water.

C. For the analyses presented here, the poisoning effects of rack structure have been neglected. Prior calculations have shown this to be a conservative assumption, where the degree of conservatism depends on the exact rack structure design. It is also assumed that no supplemental fixed poisons are utilized in the storage array.

D. The array is assumed to be surrounded on all six faces by a 2-foot thick, close-fitting reflector of concrete. This assumption is conservative since the concrete walls are several inches away from the outer rows of fuel assemblies, the floor is several inches below the bottom of the active fuel, and the materials above the active fuel provide a substantially poorer reflector than the assumed thick concrete reflectors. Calculations indicate that the assumption of concrete reflectors is conservative relative to the assumption of thick water reflectors.

E. The rack is assumed to be filled to capacity with fuel assemblies.

F. No burnable poison shims or other supplemental neutron poisons (e.g., CEAs) are assumed to be present in the fuel assemblies.

Criticality safety margins are maintained by:

A. Limiting the size of the array to 90 assemblies.

B. Defining an overall array configuration as shown in figure 9.1-1.

C. Providing adequate mechanical separation of fuel assemblies in the array, even under postulated accident conditions.

The mechanical separation provided is discussed in section 9.1.1.2.

In evaluating criticality safety, neutron cross section data for representative fuel rod cells, and material between and around assemblies (e.g., water at various densities and concrete), were generated with CEPAK (see CESSAR Section 4.3.3). Spatial calculations are performed using the two-dimensional transport code DOT-2W[1] for a typical repeating lattice unit of the fuel rack for a selection of uniform water densities covering the ranges in which reactivity peaks occur. For foam or mist conditions, the neutron effects of the leakage are assessed using the three-dimensional code KENO IV[2] based on the conservative assumption that the storage array is reflected on all faces by thick reflectors of concrete.

Maximum K_{eff} values have been calculated for this storage array and enrichments higher than U-235 expected loadings. 7
The results are:

- Flooded condition, 4.0 wt% U-235 - k_{eff} <0.91
- Uniform aqueous foam, 4.0 wt% U-235 - k_{eff} <0.95 7

These k_{eff} values are substantially below the limiting values allowed by ANSI Standard N18.2 and provide adequate margin for calculation uncertainty.

If enrichments higher than 4.0 wt% are planned for subsequent 7
fuel loadings, new criticality safety analyses and/or specific storage arrangements will be provided as needed.

The rack structure provides at least 10 inches between the top of the active fuel and the top of the rack to preclude criticality in the event a fuel assembly is dropped into a horizontal position on the top of the rack.

The new fuel storage area is protected from the effects of missiles or natural phenomena as discussed in section 3.5.

9.1.1.3.2 Compliance with Regulatory Guide 1.13

New fuel storage complies with Regulatory Guide 1.13

9.1.1.3.3 Seismic Classification

New fuel storage racks and facilities are qualified as Seismic Category I.

9.1.1.3.4 Storage Capacity

Storage is provided for at least one-third of a core of new fuel.

9.1.2 SPENT FUEL STORAGE

9.1.2.1 Design Bases

The following design bases are imposed on the storage of fuel within the spent fuel pool:

A. Accident criticality shall be prevented for the most reactive arrangement of fuel stored with optimum moderation by avoiding a K_{eff} greater than 0.95. This design basis shall be met under any normal or accident conditions.

B. The requirements of Regulatory Guide 1.13 shall be met.

C. The storage racks and facilities shall be Seismic Category I.

D. Storage shall be provided for up to 1329 fuel assemblies.

E. The storage racks and spent fuel pool facilities shall prevent extensive bulk boiling in the fuel racks and prevent fuel assembly peak clad temperatures from exceeding 650F.

F. Shielding of spent fuel shall be adequate to ensure that the radiation zone criteria of section 12.3 are met.

9.1.2.2 Facility Description

9.1.2.2.1 Spent Fuel Pool

The spent fuel pool is a stainless steel lined, concrete walled pool that is an integral part of the fuel building, as shown in figures 1.2-5 through 1.2-7, 1.2-9, and 1.2-12.

9.1.2.2.2 Spent Fuel Pool Storage Racks

The spent fuel pool storage racks are made up of individual modules. A module is an array of fuel storage cells similar to that shown in figure 9.1-3. The storage racks are comprised of 17 modules: twelve 8 by 9 and four 8 by 12 arrays, and one 9 by 9 array. The storage racks are stainless steel honeycomb structures with rectangular fuel storage cells. The stainless steel construction of the racks is compatible with fuel assembly materials and the spent fuel borated water environment. These racks have the capability to store new or spent fuel in three modes. Each mode is described below. 6

9.1.2.2.2.1 Checkerboard Fuel Storage Mode. A minimum edge-to-edge spacing between fuel assemblies is maintained by storing fuel in a checkerboard pattern while using inserts to properly locate a fuel assembly within a storage cavity (see figure 9.1-4). In addition, fuel assembly blocking devices are used in rack cells that are not within the acceptable checkerboard storage pattern (see figures 9.1-5 and 9.1-6). This prevents accidental insertion of a fuel assembly into an interstitial position so as to preclude criticality. The fuel assembly spacings are minimum values after allowances are made for rack fabrication tolerances and predicted deflections resulting from a safe shutdown earthquake (SSE). Storage for up to 665 new or spent fuel assemblies can be provided in this mode.

9.1.2.2.2.2 <u>Borated Fuel Storage Mode.</u> A minimum edge to edge spacing between fuel assemblies is maintained by storing fuel in rack locations that have been lined with a neutron
7 poison. This arrangement and the cell details of the neutron poison tube are shown in figure 9.1-7. The fuel assembly spacings are minimum values after allowances are made for rack fabrication tolerances and predicted deflections resulting from an SSE. Storage for up to 1329 new or spent fuel assemblies can be provided in this mode.

9.1.2.2.2.3 <u>Mixed Mode Fuel Storage</u>. This mode uses variable numbers of L-inserts and cell blocking devices in conjunction
7 with corresponding numbers of neutron poison inserts to provide a segregated mix of checkerboard storage and full neutron
7 poison inserts to provide storage for an appropriate number of fuel assemblies. In this mode, individual regions of the racks are either in the checkerboard mode or the borated mode. All
6 cavities in a checkerboard region immediately adjacent to a borated region must be empty and have cell blocking devices in place. This requirement assures that the criticality criterion for the racks will be met.

9.1.2.3 <u>Safety Evaluation</u>

The spent fuel pool storage rack design and location, discussed in section 9.1.2.2, provides assurance that design bases of section 9.1.2.1 are met as noted in the following sections.

9.1.2.3.1 Criticality Safety

The following postulated accidents were considered in the design of the spent fuel pool storage racks.

A. A fuel assembly dropped from a height of 2 feet above the rack onto the rack with the assembly then falling horizontally across the top of the rack or falling between the rack and surrounding spent fuel pool walls or falling into a blocked-off fuel storage cavity.

B. Tensile load on the rack of 5000 pounds (limited by adjustment of the motor stall torque or load limiting device of the crane used to load fuel into the racks).

Although the above accident conditions have been postulated, the fuel handling equipment, fuel racks, and the building arrangement are designed to minimize the possibility of these accidents or the effects resulting from these accidents by:

A. Providing positive mechanical travel hoist limits and interlocks to ensure proper equipment operation and sequence.

B. Limiting the crane loads when installing fuel into or removing fuel from the fuel rack.

C. Designing the fuel racks for SSE conditions and dropped fuel bundle conditions.

D. Designing the fuel handling machine as Seismic Category I to preclude the fuel handling machine, or any part thereof, from falling into the spent fuel pool.

The following assumptions are made in evaluating criticality safety:

A. No control element assemblies (CEA) are assumed to be present in the fuel assemblies.

B. The rack is assumed to be filled to capacity with fuel assemblies of the type whose criticality safety is being evaluated.

C. For normal operation, no credit is assumed for the boron normally found in the spent fuel pool water. Also, an optimum temperature is assumed for the water
5 moderator. In evaluating the criticality limits of a dropped bundle accident, it is assumed that boron concentration in the pool water is at least 2000 ppm.

D. An infinite fuel assembly array is assumed.

E. Only one fuel assembly is assumed to be dropped in a fuel handling accident.

Criticality safety margins are assured by:

A. Neglecting the neutron absorption effects associated with the boron normally in the spent fuel pool water, during normal operations and assuming that pool boron
5 concentration is less than one-half of normal during a bundle drop accident.

B. When fuel is stored in the borated or mixed modes, credit is taken for the neutron absorption affects
7 associated with the neutron poison inserts.

C. No credit is taken for the lesser reactive nature of spent fuel.

In evaluating criticality safety, neutron cross-section data for representative fuel rod cells, and material between and around assemblies (e.g., water at various densities and concrete), were generated with CEPAK (see CESSAR Section 4.3.3). Spatial calculations are performed using the two-dimensional transport code DOT-2W[1] for a typical repeating lattice unit of the fuel rack for a selection of uniform water densities covering the ranges in which reactivity peaks occur.

7 Maximum k_{eff} values have been calculated for enrichments up to 4.0 w/o U-235.

For all conditions, i.e. normal and accident, K_{eff} is less than 0.95. The K_{eff} values are substantially below the limiting values allowed by ANSI Standard N18.2 and provide adequate margin for calculation uncertainty.

The spent fuel storage area is protected from the effects of missiles or natural phenomena as discussed in section 3.5.

9.1.2.3.2 Compliance with Regulatory Guide 1.13

The PVNGS spent fuel storage facility complies with Regulatory Guide 1.13.

9.1.2.3.3 Seismic Classification

The PVNGS spent fuel pool storage racks and facilities are Seismic Category I.

9.1.2.3.4 Storage Capacity

Storage is provided for up to 1329 fuel assemblies.

9.1.2.3.5 Fuel Assembly Cooling

The PVNGS spent fuel pool storage racks are designed to prevent extensive bulk boiling in the racks as well as maintain fuel cladding temperatures well below 650F for the following collective conditions:

A. Natural convection water circulation within the spent fuel pool.

B. Maximum pool water temperature of 145F at the fuel rack inlet flow passages.

C. Maximum fuel pool heat load as described in section 9.1.3.

9.1.2.3.6 Shielding

Concrete and water shielding are provided as shown in figures 1.2-5 through 1.2-7, 1.2-9 and 1.2-12. This shielding attenuates radiation from the maximum design loading of stored fuel assemblies such that the radiation zone criteria of section 12.3 are met.

9.1.2.4 <u>Design Bases For Containment Fuel Storage Racks</u>

The following design bases are imposed on the storage of fuel within the containment fuel rack:

A. Accident criticality shall be prevented for the most reactive arrangement of fuel stored in unborated water by designing to a k_{eff} less than 0.95. This design basis shall be met under any normal or accident conditions. 7

cavities. The rack is located adjacent to the core support barrel laydown area, which provides access to the refueling machine for insertion and removal of fuel bundles.

9.1.2.4.2 Safety Evaluation

The containment fuel storage rack design and location, discussed in section 9.1.2.4.1 provides assurance that design bases of section 9.1.2.4 are met as noted in the following sections.

9.1.2.4.2.1 Criticality Safety. The following assumptions are made in evaluating criticality safety:

A. No control element assemblies (CEA) are assumed to be present in the fuel assemblies.

B. The rack is assumed to be filled to capacity with fuel assemblies of the type whose criticality safety is being evaluated. 7

C. For normal operation, no credit is assumed for the boron normally found in the spent fuel pool water.

D. An infinitely long fuel assembly is assumed.

Criticality safety margins are assured by:

A. Neglecting the neutron absorption effects associated with the boron normally in the spent fuel pool water during refueling operations.

B. When fuel is stored in the rack, no credit is taken for the neutron absorption affects of the rack structure.

C. No credit is taken for burnable shims or other supplemental neutron poisons (e.g., CEAs).

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In evaluating criticality safety, neutron cross-section data for representative fuel rod cells, and material between and around assemblies (e.g., water at various densities and concrete), were generated with CEPAK (see CESSAR Section 4.3.3). Spatial calculations were performed using the two-dimensional
7 transport code DOT-2W[1].

Maximum k_{eff} values have been calculated for enrichments up to 4.0 w/o U-235. For all conditions, i.e. normal and accident, k_{eff} is less than 0.95. The k_{eff} values are substantially below the limiting values allowed by ANSI Standard N18.2 and provide adequate margin for calculational uncertainty.

9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Bases

9.1.3.1.1 Fuel Pool Cooling Design Bases

The following design bases are imposed on fuel pool cooling:

A. Two independent, 100% capacity cooling systems must be provided to cool the spent fuel pool and prevent thermal damage to spent fuel elements stored therein under normal and accident conditions.

ATTACHMENT 6

12.5.2.2.3 Personnel Monitoring Instruments

Personnel monitoring instrumentation is provided to determine external and internal contamination levels and radiation doses received by personnel.

The criteria for selection of dose measuring devices were to have devices that could be quickly and accurately evaluated by station personnel (thermoluminescent dosimeters) and that could be easily read by the individual (self-reading dosimeters). The criteria for selection of external contamination measuring equipment were to have devices available at checkpoints and other areas that could be used to determine the location of contamination (friskers) and at the normal exit from the controlled area that do not require any action by personnel being checked (portal monitors). The principal criterion for selection of the whole-body counting system was to have a system readily available to supply information concerning internal exposure levels.

The friskers, portal monitors, and thermoluminescent dosimeter (TLD) readers are calibrated electronically and/or with a source at least semiannually. A calibration check on the friskers and portal monitors is performed monthly and on the TLD readers daily. A calibration check of the self-reading
5 dosimeters is performed at least semi-annually and complies
with the regulatory position of Regulatory Guide 8.4. The whole-body counting system is calibrated at least semiannually using a phantom containing various radionuclides. A check of the calibration of the whole-body system is performed by computer each time an analysis is performed.

5 Quantities of each type of device will be obtained to permit
calibration and repair without diminishing the radiation pro-
5 tection supplied. The devices and minimum numbers of each
include:

A. Count rate meters that are used as friskers to detect beta-gamma external contamination. They are normally
5 used with G-M detectors. (At least 30).

B. One portal monitor used to check for beta-gamma external contamination at the exit of the controlled area of each unit and two in the guard house. The monitor consists of an instrument console and a portal. The portal contains several liquid scintillation detector channels to provide head-to-foot detection capability. A sensor activates the counting circuit when a person steps into the portal. Visual and audible alarms are provided. (A back up set is available to substitute for an inoperative portal monitor). 5

C. Self-indicating dosimeters of various ranges. These dosimeters may include some electronic alarming and/or rate dependent types for use when deemed necessary by the Radiation Protection Supervisor. Dosimeters shall be tested and used in accordance with the recommendations of the regulatory position of Regulatory Guide 8.4. (At least 300 dosimeters). 5

D. One automatic and three manual TLD reading systems. These systems are used in the in-plant determination of whole-body exposure and job-related exposure.

E. One whole-body counter, located onsite. The multichannel analyzer and computer are programmed to analyze the data and report the radionuclides detected together with the percent body (or thyroid) burden. 5

F. Containers for collection of urine samples (normally used for tritium) and for fecal samples (possibly used under accident conditions) will be available. These samples are sent to a vendor for analysis. (at least 600 containers). 5

12.5.2.2.4 Area Radiation Monitoring (ARM) System

The ARM system provides readout and alarms locally and in the control room as described in section 11.5.

are performed in the same machine which eliminates the pos-
5 sibility of any accidental contamination during transfer
between washers and dryers. Radioactive waste is trapped by
6 filters. Aqueous or hydrocarbon soluble contamination is
5 separated during solvent distillation and re-condensation.

The liquid waste from the laundry passes to the Unit 1 Radwaste
Building sump by way of a floor drain. From the sump the waste
is pumped to the LRS holdup tanks for processing through the
LR and SR systems. The dry, solid waste from the laundry is
manually bagged and carried to the radwaste baler in the Unit 1
6 Radwaste Building. This waste is compacted into 55 gallon
drums in preparation for offsite disposal.

The liquid wastes from the decontamination facility are piped to a line common to the laundry which goes to the Radwaste Building sump. The solid waste generated from the decontamination facility is handled in a manner similar to that discussed above.

12.5.3 PROCEDURES

Radiation protection procedures are established to keep personnel radiation exposures ALARA and within the limits of 10CFR20. These procedures are discussed in section 13.5.2. Policy and operational considerations for maintaining personnel radiation exposures are discussed in sections 12.1.1 and 12.1.3.

12.5.3.1 <u>Radiation and Contamination Surveys</u>

Radiation protection personnel normally perform routine radiation and contamination surveys of accessible areas of the units. These surveys consist of radiation dose rate measurements and/or contamination smears as appropriate for the specific area. Air samples are routinely taken in accessible portions of controlled areas. Surveys related to specific activities may be performed if necessary prior to, during, or after activities that would be expected to produce additional significant radiation exposure to individuals. Survey procedures and routine survey schedules are provided in the Station Manual.

12.5.3.2 Procedures and Methods to Maintain Exposures ALARA

Operating, maintenance, and radiation protection procedures are reviewed, as discussed in section 12.1.1, to identify situations in which potential exposures could be reduced. Such ALARA considerations include:

A. Restricted Areas

Restricted Areas as defined in 10CFR20.3 (14) are established at the protected area fence and access is controlled at that point for the purpose of protecting individuals from exposure to radiation and radioactive materials.

B. Controlled Areas

Procedures establish permanent and temporary controlled areas within the restricted areas where access is further administratively controlled for the purpose of limiting exposure of individuals to radiation and radioactive materials. Within controlled areas, radiation and high radiation areas are identified and segregated in accordance with 10CFR20.202 and 10CFR20.203, utilizing access control and posting of radiation and high radiation areas. High radiation areas having dose rates greater than 100 mrem/h, but less than 1000 mrem/h will be barricaded with a rope, gate, or other suitable unlocked barrier. Access to high radiation areas having dose rates greater than 1000 mrem/h will meet the requirements of 10CFR20.203(c).

C. Radiation Exposure Permits

Procedures require Radiation Exposure Permits for entry into areas where the dose rate exceeds 15 mrem/h, or which contain removable contamination levels in excess and Schedule B of 10CFR30 are subject to material controls for radiological protection. Those controls include:

A. Monitoring of packages containing radioactive materials for external dose rate and removable contamination upon receipt at the station and prior to shipment away from the station.

B. Monitoring of sources for removable surface contamination (leakage testing) at 6-month intervals. Excluded are sealed sources of 100 μCi or less of beta and/or gamma emitting materials and 10 μCi or less of alpha emitting materials.

C. Labeling of licensed sources with the radiation symbol, stating the activity, radionuclide, and source identification number.

D. Storage in a locked area of sources which are not installed in an instrument or other piece of equipment.

E. Inventory of sources every year.

12.5.3.8 Radiation Protection Training

As part of the general employee training, members of the permanent operating organization whose duties entail entering controlled areas, or directing the activities of others who enter controlled areas, are instructed in the fundamentals of radiation protection. They must pass an examination to be allowed to enter controlled areas unescorted. These personnel are also required to attend a retraining program in radiation protection held at intervals not to exceed 2 years.

General employee training is discussed in section 13.2. The training program includes instruction in applicable provisions of the NRC regulations for the protection of personnel from radiation and radioactive material (10CFR20) and instruction to women concerning prenatal radiation exposure.

Additional training and testing is given to candidates for reactor operator or senior reactor operator licenses to meet the requirements of 10CFR55. Their training is discussed in section 13.2.

In addition to general employee training, radiation protection personnel also receive training in areas which apply to their specific job functions such as radiation and contamination surveys, air sampling techniques, use of portable and laboratory instrumentation, release limits, and safe handling of sources.

ATTACHMENT 7

13.1.2.2.2.2 Radiation Protection Supervisor. The Radiation Protection Supervisor is responsible to the Engineering and 5
Technical Services Manager for the preparation, coordination, and conduct of the station chemistry and radiological programs, including delineating the operating philosophy and procedures for maintaining occupational radiation exposures as low as is reasonably achievable. His position corresponds to "Radiation Protection Manager" as discussed in Regulatory Guide 1.8.

Reporting to the Radiation Protection Supervisor are the 5
Supervising Radiation Physicists for each unit. The Radiation Protection Supervisor is responsible for control of radiation exposures to personnel, maintenance of related records, conduct of surveillance, and approval of radioactive waste disposal activities. A Supervising Radiation Physicist is designated as backup to provide coverage in event of absence of the Radiation Protection Supervisor. The Supervising Radiation Physicist 5
shall have a minimum of five years experience, of which two years shall be at a professional level.

13.1.2.2.2.3 Operations Engineering Supervisor. The Operations Engineering Supervisor is responsible to the Engineering and Technical Services Manager for mechanical and electrical 5
engineering support, including monitoring station performance and the inservice inspection program.

13.1.2.2.2.4 Computer Supervisor. The Computer Supervisor is responsible to the Engineering and Technical Services Manager 5
for coordinating station computer activities, including hardware and software.

13.1.2.2.2.5 Chemistry Supervisor. The Chemistry Supervisor is responsible to the Engineering and Technical Services Manager for the conduct of the water chemistry program and coordinates 5
with the Radiation Protection Supervisor on radiation exposures and contamination problems associated with the chemistry program.

ATTACHMENT 8

are performed in the same machine which eliminates the pos-
5 sibility of any accidental contamination during transfer between washers and dryers. Radioactive waste is trapped by
6 filters. Aqueous or hydrocarbon soluble contamination is
5 separated during solvent distillation and re-condensation.

The liquid waste from the laundry passes to the Unit 1 Radwaste Building sump by way of a floor drain. From the sump the waste is pumped to the LRS holdup tanks for processing through the LR and SR systems. The dry, solid waste from the laundry is manually bagged and carried to the radwaste baler in the Unit 1
6 Radwaste Building. This waste is compacted into 55 gallon drums in preparation for offsite disposal.

The liquid wastes from the decontamination facility are piped to a line common to the laundry which goes to the Radwaste Building sump. The solid waste generated from the decontamination facility is handled in a manner similar to that discussed above.

12.5.3 PROCEDURES

Radiation protection procedures are established to keep personnel radiation exposures ALARA and within the limits of 10CFR20. These procedures are discussed in section 13.5.2. Policy and operational considerations for maintaining personnel radiation exposures are discussed in sections 12.1.1 and 12.1.3.

12.5.3.1 Radiation and Contamination Surveys

Radiation protection personnel normally perform routine radiation and contamination surveys of accessible areas of the units. These surveys consist of radiation dose rate measurements and/or contamination smears as appropriate for the specific area. Air samples are routinely taken in accessible portions of controlled areas. Surveys related to specific activities may be performed if necessary prior to, during, or after activities that would be expected to produce additional significant radiation exposure to individuals. Survey procedures and routine survey schedules are provided in the Station Manual.

12.5.3.2 Proceo ~es and Methods to Maintain Exposures ALARA

Operating, maintenance, and radiation protection procedures are reviewed, as discussed in section 12.1.1, to identify situations in which potential exposures could be reduced. Such ALARA considerations include:

A. Restricted Areas

Restricted Areas as defined in 10CFR20.3 (14) are established at the protected area fence and access is controlled at that point for the purpose of protecting individuals from exposure to radiation and radioactive materials.

B. Controlled Areas

Procedures establish permanent and temporary controlled areas within the restricted areas where access is further administratively controlled for the purpose of limiting exposure of individuals to radiation and radioactive materials. Within controlled areas, radiation and high radiation areas are identified and segregated in accordance with 10CFR20.202 and 10CFR20.203, utilizing access control and posting of radiation and high radiation areas. High radiation areas having dose rates greater than 100 mrem/h, but less than 1000 mrem/h will be barricaded with a rope, gate, or other suitable unlocked barrier. Access to high radiation areas having dose rates greater than 1000 mrem/h will meet the requirements of 10CFR20.203(c).

C. Radiation Exposure Permits

Procedures require Radiation Exposure Permits for entry into areas where the dose rate exceeds 15 mrem/h, or which contain removable contamination levels in excess of 100,000 dpm/100 cm^2. These permits are a principal administrative means of managing personnel radiation exposure and describe the radiological controls required to perform the activity and maintain personnel radiation exposure ALARA. The permit contains information pertinent to the activity such as radiation and/or contamination levels in the area, allowable stay times, protective clothing requirements, respiratory protection equipment requirements, special personnel monitoring requirements, temporary shielding requirements, and personnel authorized to receive radiation exposure while performing the activity for which the Radiation Exposure Permit was issued. Radiation Exposure Permits require the approval of radiation protection supervision or designated alternates.

D. Selected Operating and Maintenance Activities

Operating and maintenance activities that can result in significant individual exposures are controlled by written procedures. Procedures controlling refueling, radwaste handling, spent fuel handling, radiochemical sampling, loading and shipping of radioactive materials, and procedures controlling inservice inspections, normal operation, routine maintenance, and calibrations that are expected to require issuance of a Radiation Exposure Permit, are reviewed for ALARA considerations, as discussed in section 12.1.1.

12.5.3.3 Control of Access and Stay Time in Radiation Areas

Physical and administrative controls assure that the philosophy of maintaining personnel exposures ALARA is implemented, as specified in sections 12.1 and 12.5.3.2.

Authorized personnel who enter station restricted areas are issued appropriate dosimetry devices in accordance with Station Manual procedures.

12.5.3.4 Contamination Control

Contamination limits for personnel, equipment, and areas are listed in the Radiation Protection Division of the Station Manual. Surveys are performed as discussed in section 12.5.3.1, to determine contamination levels. Areas found contaminated beyond specified limits are roped off or otherwise delineated with a physical barrier, posted appropriately, and decontaminated as soon as practical. A stepoff pad or other appropriate means may be used to prevent the spread of contamination.

Station Manual procedures incorporate those recommendations of Regulatory Guide 1.39 which are considered applicable for housekeeping activities occurring during the operations phase that are comparable to those occurring during the construction phase (refer to section 1.8).

Tools and equipment used in contaminated areas are monitored prior to removal from the controlled area. If they are to be released to an uncontrolled area, they must meet uncontrolled (clean) area limits, and must be decontaminated or packaged and labelled as necessary to meet these limits. Decontamination facilities are discussed in section 12.5.2. If the tools and equipment are to be transferred to another controlled area through an uncontrolled area, they may be bagged, wrapped, or similarly enclosed to prevent the spread of contamination while being transferred.

Control of personnel contamination (external and internal) is provided by use of protective clothing and respiratory protection equipment as discussed in section 12.5.2. Each individual is responsible for monitoring himself and his clothing when he crosses a local control point or the main access control point, as discussed in section 12.5.2. If contamination above allowable limits is found, the individual is decontaminated using facilities previously described in section 12.5.2.

12.5.3.5 Airborne Activity Exposure Control

When airborne radioactivity is detected in excess of the limits of 10CFR20.203(d), the area is posted as an airborne radioactivity area, and access is controlled in accordance with section 12.5.3.2.

Occupancy is restricted or respiratory protection equipment is provided to maintain exposures within the limits of Appendix B, Table 1, Column 1 of 10CFR20, if personnel entry is required into areas where the source of airborne radioactivity cannot be removed or controlled. An air sampling program is used to ensure that appropriate respiratory protective equipment is specified on the Radiation Exposure Permit. The respiratory protection program is organized to conform to the applicable portions of ANSI Z88.2-1969. Effectiveness of the respiratory protection program is evaluated by various types of bioassay analyses or nasal smears, or respirator facepiece interior smears.

Respiratory equipment discussed in section 12.5.2 is available near the main access control point to permanent controlled areas. Supplementary emergency respiratory equipment is available in the control room and in emergency kits.

The following controls are incorporated in the program:

A. Each respirator user is advised that he may leave an airborne radioactivity area for psychological or physical relief from respirator use. Each user shall leave the area in the case of respirator malfunction or any other condition that might cause reduction in the protection afforded the user.

B. Air samples and surveys are made to identify the presence of airborne radioactivity and to estimate individual exposures so that selection of appropriate respiratory equipment can be made.

C. Procedures are established to ensure correct fitting, use, maintenance, and cleaning of respirator equipment. Each individual qualified to use respiratory protection equipment receives a quantitative fit test annually and performs a qualitative fit test prior to use of respiratory protection equipment.

12.5.3.6 Personnel Monitoring

Station employees, contractor personnel, support personnel, and visitors are required to wear thermoluminescent dosimeters (TLD) or self-reading dosimeters when in a controlled area. In addition, job dosimeters are issued to individuals working under a Radiation Exposure Permit. The exposure readings of these job dosimeters are used for specific ALARA job exposure evaluation as well as to indicate current individual exposure
5 status. Use of neutron dosimeters complies with the recommendations of Regulatory Guide 8.14.

The Station Manual requires bioassays, including whole body counting, consistent with the recommendations of Regulatory
5 Guides 8.9 and 8.26. The type of determination and the frequency of determination depends upon the work environment of the individual and the work situation.

Job dosimeters are used for specific ALARA job exposure evaluation as well as to indicate current individual exposure status. Personal dosimeters are evaluated on a monthly basis and are used as the dosimetry of record for the individual unless evaluation determines an alternate exposure evaluation is more representative of the dose received.

Exposure data of personnel issued personal dosimeters in accordance with 10CFR20.202 is maintained on NRC Form 5, or equivalent. Occupational exposures incurred by individuals prior to working at PVNGS are summarized on NRC Form 4, Occupational External Radiation Exposure History, or the equivalent. These records are maintained at PVNGS and will be preserved for the lifetime of the plant or until the NRC authorizes their disposal. Reports of overexposure to radiation workers are made to the NRC and the individual involved pursuant to 10CFR19 and 10CFR20.

12.5.3.7 Handling of Radioactive Material

Licensed sources used for calibration are used by or under the direction of personnel who have received training in the safe use and handling of sources. A Radiation Exposure Permit will be required for this use if such use could result in significant personnel exposure.

Suitable methods for the safe handling of radioactive materials are implemented to maintain external and internal doses at levels that are ALARA. External doses are minimized by a combination of time, distance, and shielding considerations. Internal doses are minimized by the measurement and control of loose contamination and airborne radioactivity. Station Manual procedures provide instructions for handling radioactive sources.

Sealed radionuclide sources have activities greater than the quantities of radionuclides defined in Appendix C to 10CFR20 and Schedule B of 10CFR30 are subject to material controls for radiological protection. Those controls include:

A. Monitoring of packages containing radioactive materials for external dose rate and removable contamination upon receipt at the station and prior to shipment away from the station.

B. Monitoring of sources for removable surface contamination (leakage testing) at 6-month intervals. Excluded are sealed sources of 100 μCi or less of beta and/or gamma emitting materials and 10 μCi or less of alpha emitting materials.

C. Labeling of licensed sources with the radiation symbol, stating the activity, radionuclide, and source identification number.

D. Storage in a locked area of sources which are not installed in an instrument or other piece of equipment.

E. Inventory of sources every year.

12.5.3.8 Radiation Protection Training

As part of the general employee training, members of the permanent operating organization whose duties entail entering controlled areas, or directing the activities of others who enter controlled areas, are instructed in the fundamentals of radiation protection. They must pass an examination to be allowed to enter controlled areas unescorted. These personnel are also required to attend a retraining program in radiation protection held at intervals not to exceed 2 years.

General employee training is discussed in section 13.2. The training program includes instruction in applicable provisions of the NRC regulations for the protection of personnel from radiation and radioactive material (10CFR20) and instruction to women concerning prenatal radiation exposure.

Additional training and testing is given to candidates for reactor operator or senior reactor operator licenses to meet the requirements of 10CFR55. Their training is discussed in section 13.2.

In addition to general employee training, radiation protection personnel also receive training in areas which apply to their specific job functions such as radiation and contamination surveys, air sampling techniques, use of portable and laboratory instrumentation, release limits, and safe handling of sources.

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