A T T A C H M E N T A

Revise the Beaver Valley Power Station, Unit 2 Technical Specifications a follows:

Remove Page	Insert Page
---	3/4 9-14
---	3/4 9-15
B 3/4 9-3	B 3/4 9-3
5-6	5-6
5-7	5-7

LIMITING CONDITION FOR OPERATION

3.9.14 Fuel is to be stored in the spent fuel storage pool with:

a. The boron concentration in the spent fuel pool maintained greater than or equal to 1050 ppm when moving fuel in the spent fuel pool; and

b. Fuel assembly storage in Region 1 restricted to fuel with an enrichment less than or equal to 4.85 w/o stored in a 3 of 4 checkerboard configuration; and

c. Fuel assembly storage in Region 2 restricted to fuel which has been qualified in accordance with Table 3.9-1

APPLICABILITY: During storage of fuel in the spent fuel pool.

ACTION:

a. Suspend all actions involving movement of fuel in the spent fuel pool if it is determined a fuel assembly has been placed in the incorrect Region until such time as the correct storage location is determined. Move the assembly to its correct location before resumption of any other fuel movement.

b. Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined the pool boron concentration is less than 1050 ppm, until such time as the boron concentration is increased to 1050 ppm or greater.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.14.1 Prior to placing fuel or moving fuel in the spent fuel pool, verify through fuel receipt records for new fuel or by burnup analysis and comparison with Table 3.9-1 that fuel assemblies to be placed into or moved in the spent fuel pool are within the above enrichment limits.

4.9.14.2 Verify the spent fuel pool boron concentration is $\geq$ 1050 ppm:

a. Within 8 hours prior to and at least once per 24 hours during movement of fuel in the spent fuel pool, and

b. At least once per 31 days.

PROPOSED WORDING

Table 3.9-1

BEAVER VALLEY FUEL ASSEMBLY MINIMUM BURNUP VS. INITIAL U_{235} ENRICHMENT FOR STORAGE IN REGION 2 SPENT FUEL RACKS

Initial U_{235} Enrichment	Assembly Discharge Burnup (GWD/MTU)
3.6	0
4.0	2.6
4.4	5.3
4.85	8.2

NOTE: Linear interpolation yields conservative results.

PROPOSED WORDING

BASES

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.12 and 3/4.9.13 FUEL BUILDING VENTILATION SYSTEM

The limitations on the storage pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses. The spent fuel pool area ventilation system is non-safety related and only recirculates air through the fuel building. The fuel building portion of the SLCRS is safety related and continuously filters the fuel building exhaust air. This maintains a negative pressure in the fuel building.

3/4.9.14 FUEL STORAGE - SPENT FUEL STORAGE POOL

The requirements for fuel storage in the spent fuel pool ensure that: (1) the spent fuel pool will remain subcritical during fuel storage; and (2) a uniform boron concentration is maintained in the water volume in the spent fuel pool to provide negative reactivity for postulated accident conditions under the guidelines of ANSI 16.1-1975. The value of 0.95 or less for k_{eff} which includes all uncertainties at the 95/95 probability/confidence level is the acceptance criteria for fuel storage in the spent fuel pool.

The Action Statement applicable to fuel storage in the spent fuel pool ensures that: (1) the spent fuel pool is protected from distortion in the fuel storage pattern that could result in a critical array during the movement of fuel; and (2) the boron concentration is maintained at $\geq$ 1050 ppm (this includes a 50 ppm conservative allowance for uncertainties) during all actions involving movement of fuel in the spent fuel pool.

The Surveillance Requirements applicable to fuel storage in the spent fuel pool ensure that: (1) the fuel assemblies satisfy the analyzed U-235 enrichment limits or an analysis has been performed and it was determined that k_{eff} is $\leq$0.95; and (2) the boron concentration meets the 1050 ppm limit.

PROPOSED WORDING

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for maximum internal pressure of 45 psig and a temperature of 280.0°F.

PENETRATIONS

5.2.3 Penetrations through the reactor containment building are designed and shall be maintained in accordance with the original design provisions contained in Section 6.2.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with zircaloy-4. Each fuel rod shall have a nominal active fuel length of 144 inches. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of ~~3.3~~ 4.85 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is 9370 cubic feet at a nominal T_{avg} of 576°F.

PROPOSED WORDING

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

The fuel will be stored in accordance with the provisions described in FSAR Sections 4.3 and 9.1

CRITICALITY

5.6.1 The spent fuel storage racks are designed and shall be maintained with a minimum of 10.4375 inch center-to-center distance between fuel assemblies placed in the storage racks. to ensure a k_{eff} equivalent to ≤ 0.95 with the storage pool filled with unborated water. ~~The k_{eff} of ≤ 0.95 includes a conservative allowance of at least 1.4% $\Delta k/k$ for uncertainties.~~

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 751'-3".

CAPACITY

5.6.3 The fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1088 fuel assemblies.

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as Category I items in Section 3.7 of the FSAR shall be designed and maintained to the original design provisions with allowance for normal degradation pursuant to the applicant Surveillance Requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower shall be located as shown on Figure 5.1-1.

PROPOSED WORDING

A T T A C H M E N T B

Safety Analysis
Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Change No. 15

Description of amendment request: The proposed amendment would incorporate Section 3.9.14 and associated bases and revise Design Feature Sections 5.3.1 and 5.6.1 to set forth fuel assembly U-235 enrichment limitations on storage of fuel in the new and spent fuel storage racks. These changes are based on an evaluation performed by Westinghouse, "Criticality Analysis of Beaver Valley 2 Fuel Racks." The results of the evaluation provide justification for:

1. New fuel storage rack enrichment limit of 4.85 w/o,

2. Two spent fuel storage rack enrichment limits where Region 1 limits fuel enriched from 3.6 to 4.85 w/o to a three out of four cell checkerboard storage pattern, and fuel assemblies can be stored in all cells in Region 2 limited by the burnup dependent restrictions provided in Table 3.9-1.

Criticality of fuel assemblies in a fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies. The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (K_{eff}) of the fuel assembly array will be less than 0.95 as recommended in ANSI 57.2-1983 and ANSI 57.3-1983.

For accident conditions where reactivity is postulated to increase (i.e., misloading an assembly with a burnup and enrichment combination outside of the acceptable criteria provided in proposed Table 3.9-1, or dropping a fuel assembly between the rack and pool wall), the double contingency principle of ANSI 16.1-1975 is applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event. The presence of approximately 1000 ppm boron in the spent fuel pool will decrease reactivity by about 15 percent ΔK. Thus, for postulated accidents, should there be a reactivity increase, K_{eff} would be less than or equal to 0.95 due to the effect of the dissolved boron.

The maximum K_{eff} including uncertainties at the 95/95 probability/confidence level is presented for the limiting cases:

Case	K_{eff}
1. Spent Fuel Rack Region 1, 4.85 w/o 3 of 4 cell storage	.9417
2. Spent Fuel Rack Region 2, 3.6 w/o all cell storage	.9486
3. Fresh Fuel Rack, 4.85 w/o moderation - full density 1.0 gm/cm^3	.9264
4. Fresh Fuel Rack, 4.85 w/o moderation - optimum low density 0.076 gm/cm^3	.9398

The K_{eff} for each of the above limiting cases is less than 0.95 including uncertainties at the 95/95 probability/confidence level, therefore, the acceptance criteria for criticality is met under all conditions.

In accordance with the criticality analysis results, technical specification limitations on maximum enrichment are applicable for the spent fuel racks. Fuel assembly storage in Region 1 is limited to a maximum enrichment of 4.85 w/o in a 3 of 4 cell array. Fuel assemblies can be stored in all cells in Region 2, limited by the burnup dependent restrictions provided in Table 3.9-1. Both the spent and new fuel racks are analyzed for an enrichment limit of 4.85 w/o. Since technical specification limits have been placed on fuel assembly enrichment for storage in the spent fuel pool, no additional technical specification restrictions are required on the new fuel racks.

Design Feature section 5.3.1 has been revised to reflect the new fuel assembly enrichment limit of 4.85 w/o, and section 5.6.1 was revised to reference the applicable FSAR sections which describe the provisions for fuel storage. FSAR sections 4.3 and 9.1 are being revised to reflect the new criticality analysis which includes a description of the uncertainties applied. Therefore, the sentence describing the uncertainties is not required and has been deleted.

Storage of fuel in the new and spent fuel racks will be changed to reflect the "Criticality Analysis of Beaver Valley Unit 2 Fuel Racks". The criticality analysis supports the storage of fuel enriched up to 4.85 w/o U-235. This will facilitate longer fuel cycles, higher nuclear capacity factors and lower plant power generation costs.

Spent fuel pool Region 1 will provide for storage of fuel with enrichments up to 4.85 w/o in an administratively controlled 3 of 4 cell array. Region 2 will provide for storage of fuel assemblies with the burnup dependent enrichment limitations provided in Table 3.9-1. K_{eff} will be maintained less than 0.95 consistent with the current FSAR design basis. With the Region 1 checkerboard array, the segregation of fuel assemblies into Regions 1 and 2 and the proposed technical specification changes, no adverse safety considerations are introduced. The new criticality analysis satisfies the design basis for preventing criticality outside the reactor where, including uncertainties, there is a 95% probability at a 95% confidence level that K_{eff} of the fuel assembly array will be less than 0.95 in accordance with ANSI 57.2 - 1983 and ANSI 57.3 - 1983. Therefore, the proposed changes will not reduce the safety of the plant and are consistent with the current regulatory basis.

ATTACHMENT C

No Significant Hazards Evaluation
Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Change No. 15

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists in 10 CFR 50.92(c). A proposed amendment to an operating license for a facility involves no significant hazards if operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequence of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The proposed changes do not involve a significant hazards consideration because:

1. The criticality analysis acceptance criteria ($K_{eff} < 0.95$) is consistent with that stated in FSAR Sections 9.1.1 New Fuel Storage, 9.1.2 Spent Fuel Storage and 4.3.2.6 Criticality of the Reactor During Refueling and Criticality of Fuel Assemblies. Attachment D provides a revision to FSAR section 9.1.1 and 9.1.2 to describe the segregation of the spent fuel pool into regions 1 and 2 and how the Region 1 administrative controls ensure that the 4.85 w/o fuel and the 3 of 4 cell array is maintained. In addition to the administrative controls available to maintain the required checkerboard array in Region 1, the minimum boron concentration will provide an additional safety margin to ensure criticality will not be achieved. Even if new fuel assemblies were not stored in the specified checkerboard array, the dissolved boron would provide sufficient neutron absorption capability to preclude criticality.

   Attachment E provides a revision to FSAR Section 4.3.2.6 to incorporate changes to reflect the new criticality analysis. These FSAR changes are provided as background information for this technical specification change and will be included in a future FSAR update.

   Fuel assembly decay heat production is a function of core power level, and since the authorized core power level is not being changed, the decay heat load on the spent fuel pool cooling system will not be significantly impacted by the proposed enrichment limits.

   The proposed changes will not have a significant impact on the safety of the plant or on the operation of the spent fuel storage pool. The criteria setforth in Table 3.9-1 provide assurance that fuel assemblies are qualified for storage in Region 2 to ensure K_{eff} will be $\leq$ 0.95 at the 95/95 confidence level. Therefore, the proposed changes will not introduce any adverse safety considerations or involve a significant increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated.

2. The proposed changes are bounded by FSAR Section 15.7.4 Radiological Consequence of Fuel Handling Accidents and the activities in the fuel rod gap presented in Table 15.0-7 which use a conservative value of 650 days at a full power value of 2766 MWt to determine fission product inventories and calculate resultant doses. In accordance with the double contingency principle of ANSI N16.1-1975 it is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Therefore, the minimum boron concentration limits on the spent fuel pool ensure that even if new fuel assemblies were not spaced to maintain the checkerboard arrays or a fuel assembly was dropped on top of the rack (the rack structure pertinent for criticality is not excessively deformed and the dropped assembly has more than twelve inches of water separating it from the active fuel height of stored assemblies) that criticality would be precluded.

   The analysis of reactor core operation with up to 4.85 w/o reload fuel will be provided in the cycle-specific reload safety evaluations which are performed for each reload cycle (the standard reload design methods described in WCAP-9272 and 9273, "Westinghouse Reload Safety Evaluation Methodology", and/or other appropriate criteria to demonstrate that the core reload will not adversely affect the safety of the plant). Criticality accidents during fuel handling are precluded by stringent administrative procedures which require the qualification of fuel assemblies in accordance with Table 3.9-1 for fuel assembly storage in Region 2. Therefore, the probability for an accident or malfunction of a different type than previously evaluated will not be created.

3. Technical Specification 3.9.14 and associated bases provide the administrative controls required to assure that fuel assemblies with the potential to form a critical array are segregated such that the effective multiplication factor, K_{eff}, will be less than 0.95. Criticality will be prevented in Region 1 by limiting fuel assembly interaction by physical design of the fuel racks and maintaining a minimum soluble boron concentration in the pool water. Fuel assembly placement in Region 1 will be administratively controlled by storing fuel with an enrichment between 3.6 and 4.85 w/o in a 3 of 4 cell array. Where Region 1 is adjacent to Region 2, the arrangement will be maintained to limit fuel assembly interaction. This is consistent with the design basis criteria for preventing criticality outside the reactor where, including uncertainties, there is a 95% probability at a 95% confidence level that K_{eff} of the fuel assembly array will be less than 0.95 in accordance with ANSI 57.2 - 1983 and ANSI 57.3 - 1983. K_{eff} will be maintained less than 0.95 including uncertainties consistent with the current design basis. Therefore, the proposed changes will not involve a significant reduction in the margin of safety.

Therefore, based on considerations expressed above, it is proposed that this amendment application does not involve a significant hazards consideration.

A T T A C H M E N T   D

FSAR Changes

(provided for information only)

Ejected rod worths are given in Section 15.4.8 for several different conditions.

Allowable deviations due to misaligned control rods are discussed in the Technical Specifications.

A representative calculation for two banks of control rods simultaneously withdrawn (rod withdrawal accident) is given in Figure 4.3-36.

Calculation of control rod reactivity worth versus time following reactor trip involves control rod velocity and differential reactivity worth. The rod position versus time of travel after rod release normalized to "Distance to Top of Dashpot" and "Drop Time to Top of Dashpot" is given on Figure 4.3-37. For nulcear design purposes, the reactivity worth versus rod position is calculated by a series of steady-state calculations at various control rod positions, assuming all rods out of the core as the initial position in order to minimize the initial reactivity insertion rate. Also, to be conservative, the rod of highest worth is assumed stuck out of the core, and the flux distribution (and thus reactivity importance) is assumed to be skewed to the bottom of the core. The result of these calculations is shown on Figure 4.3-38.

The shutdown groups provide additional negative reactivity to assure an adequate shutdown margin. Shutdown margin is defined as the amount by which the core would be subcritical at hot shutdown if all rod cluster control assemblies are tripped, but assuming that the highest worth assembly remains fully withdrawn and no changes in xenon or boron take place. The loss of control rod worth due to the material irradiation is negligible, since only bank D may be in the core under normal operating conditions (near full power). The values given in Table 4.3-3 show that the available reactivity in withdrawn rod cluster control assemblies provides the design bases minimum shutdown margin, allowing for the highest worth cluster to be at its fully withdrawn position. An allowance for the uncertainty in the calculated worth of N-1 rods is made before determination of the shutdown margin.

4.3.2.6 Criticality of the Reactor During Refueling and Criticality of Fuel Assemblies

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer, shipping and storage facilities, and by administrative control procedures. The two principal methods of preventing criticality are limiting the fuel assembly array size and limiting assembly interaction by fixing the minimum separation between assemblies and/or inserting neutron poisons between assemblies.

The design basis for preventing criticality outside the reactor is that, considering possible variations, there is a 95 percent

ANSI 57.2-1983, ANSI 57.3-1983 and in NRC letter April 14, 1978

probability at a 95 percent confidence level that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than 0.95 as recommended in ~~ANSI N210-1976.~~ The following are the conditions that are assumed in meeting this design basis:

1. The fuel assembly contains the highest enrichment authorized without any control rods or any noncontained burnable poison and is at its most reactive point in life. ~~Refer to Section 9.1.2 for fuel properties used in criticality calculations.~~

2. For flooded conditions, the moderator is pure water at ~~the~~ a temperature of 68°F. ~~within the design limits which yields the largest reactivity.~~ A conservative value of 1.0 gm/cm³ is used for the density of water.

3. The array is either infinite in lateral extent or is surrounded by a conservatively chosen reflector, whichever is appropriate for the design.

4. Mechanical uncertainties are treated by either using "worst case" conditions or by performing sensitivity studies and obtaining appropriate uncertainties.

5. Credit is taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption.

6. Where borated water is present, credit for the dissolved boron is not taken, except under postulated accident conditions where the double contingency principle of ANSI N16.1-1975 is applied. This principle states that it shall require at least two unlikely, independent, and concurrent events to produce a criticality accident.

For fuel storage application, water is usually present. However, the design methodology also prevents accidental criticality when fuel assemblies are stored in the dry condition. For this case, possible sources of moderation, such as those that could arise during fire fighting operations, are included in the analysis. ~~The design basis k_{eff} is 0.98 as recommended in ANSI N210-1976.~~ The maximum rack K_{eff} under low density moderation conditions occurs at 0.076 gm/cm³ water density. The design method which ensures the criticality safety of fuel assemblies ~~outside the reactor~~ in the spent fuel storage rack uses the AMPX system of codes (Ford ~~et al. 1976~~ 1982 and Greene ~~et al~~ 1976) for cross-section generation and KENO IV (Petrie and Cross 1975) for reactivity determination.

The ~~218~~ 227 energy group cross-section library (Ford ~~et al 1976~~ 1982), that is the common starting point for all cross-sections, has been generated from ~~ENDF/B-IV~~ ENDF/B-V data. The NITAWL program (Greene ~~et al~~ 1976) includes in this library the self-shielded resonance cross-sections that are appropriate for ~~a~~ each particular geometry. The Nordheim Integral Treatment is used. Energy and spatial weighting of cross-sections is performed by the XSDRNPM program (Greene ~~et al~~ 1976), which is a one-dimensional S_N transport theory code. These multi-group cross-section sets are then used as input to KENO IV (Petrie and Cross 1975), which is a three-dimensional Monte Carlo theory program designed for reactivity calculations.

A set of ~~27~~ 33 critical experiments has been analyzed using the above method to demonstrate its applicability to criticality analysis and to establish the method bias and variability. The experiments range from water moderated oxide fuel arrays separated by various materials that simulate LWR fuel shipping and storage conditions ~~(Bierman et al 1977 and 1978)~~ (Baldwin 1979) to dry harder spectrum uranium metal cylinder arrays with various interspersed materials (Thomas 1973) that demonstrate the wide range of applicablity of the method.

Some descriptive facts about each of the ~~27~~ 33 benchmark critical experiments are given in Table 4.3-4. The average k_{eff} of the benchmarks is 0.992. ~~0.9998, which demonstrates that there is virtually no bias associated with the method. The standard deviation of the k_{eff} values is 0.0057Δk.~~ The 95/95 one sided tolerance limit factor for ~~27~~ 33 values is 2.19. ~~2.26.~~ There is thus a 95 percent probability with a 95 percent confidence level that the uncertainty in reactivity due to the method is not greater than ~~0.013~~ 0.0018 Δk.

~~The total uncertainty to be added to a criticality calculation is:~~

INSERT 1

$$\cancel{TU = \left[(ks)^2_{submethod} + (ks)^2_{KENO} + \Sigma (ks)^2_{mech} \right]^{\frac{1}{2}}}$$

~~where $(ks)_{submethod}$ is 0.013 as discussed above, $(ks)_{KENO}$ is the statistical uncertainty associated with the particular KENO calculation being used and the $(ks)_{mech}$ terms are a series of statistical uncertainties associated with mechanical tolerances such as thicknesses and spacings. If "worst case" assumptions are used for tolerances, this term will be zero.~~

~~The criticality design criteria are met when the calculated effective multiplication factor plus the total uncertainty (TU) is less than 0.95 or, in the special case defined above, 0.98.~~

These methods conform with ~~ANS N18.2-1973~~ ANSI 57.2-1983, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, Section 5.7, Fuel Handling System; ~~ANSI N210-1976,~~ Design Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations, Section ~~5.1.12~~ 6.4.2; ANSI N16.9-1975, Validation of Calculational Methods for Nuclear Criticality Safety; NRC Standard Review Plan, Section 9.1.2, Spent Fuel Storage; and the NRC guidance, NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications; ANSI 57.3-1983, Design Requirements for New Fuel Storage Facilities at Light Water Reactor Plants.

INSERT 1

The following equation is used to develop the maximum k_{eff}.

$$K_{eff} = K_{worst} + B_{method} + B_{part} + \sqrt{[(ks)^2_{worst} + (ks)^2_{method}]}$$

where:

K_{worst}	=	worst case KENO K_{eff} that includes centered fuel assembly positions, material tolerances, and mechanical tolerance which can result in spacing betw ien assemblies less than nominal
B_{method}	=	method bias determined from benchmark critical comparisons
B_{part}	=	bias to account for poison partical self-shielding
kS_{worst}	=	95/95 uncertainty in the worst case KENO K_{eff}
kS_{method}	=	95/95 uncertainty in the method bias

The criticality acceptance criteria is met when the effective multiplication factor (k_{eff}) including uncertainties at a 95/95 probability/confidence level is less than 0.95.

are obtained from the three dimensional TURTLE calculation from which constants are homogenized by flux-volume weighting.

Validation of the spatial codes for calculating power distributions involves the use of incore and excore detectors and is discussed in Section 4.3.2.2.7.

Based on comparison with measured data, it is estimated that the accuracy of current analytical methods is:

±0.1 percent $\Delta\rho$ for Doppler defect

±2 x 10^{-5}/°F for moderator coefficient

±50 ppm for critical boron concentration with depletion

±3 percent for power distributions

±0.2 percent $\Delta\rho$ for rod bank worth

±4 pcm/step for differential rod worth

±0.5 pcm/ppm for boron worth

±0.1 percent $\Delta\rho$ for moderator defect

4.3.4 Revisions

The design methods for the criticality of fuel assemblies outside the reactor now use the AMPX/KENO system of codes as described in Section 4.3.2.6.

The design methods for the nuclear analysis of the core now use both TURTLE (Barry and Altomare 1975) and PALADON (Camden et al 1978) for multi-dimensional analyses.

4.3.5 References for Section 4.3

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Replace with reference 5

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TABLE 4.3-4

BENCHMARK CRITICAL EXPERIMENTS

General Description	Enrichment wt% U235	Reflector	Separating Material	Characterizing Separation (cm)
1. UOd Rod lattice	2.35	Water	Water	11.92
2. UOd Rod lattice	2.35	Water	Water	8.39
3. UOd Rod lattice	2.35	Water	Water	6.39
4. UOd Rod lattice	2.35	Water	Water	4.46
5. UOd Rod lattice	2.35	Water	Stainless steel	10.44
6. UOd Rod lattice	2.35	Water	Stainless steel	11.47
7. UOd Rod lattice	2.35	Water	Stainless steel	7.76
8. UOd Rod lattice	2.35	Water	Stainless steel	7.42
9. UOd Rod lattice	2.35	Water	Boral	6.34
10. UOd Rod lattice	2.35	Water	Boral	9.03
11. UOd Rod lattice	2.35	Water	Boral	5.05
12. UOd Rod lattice	4.29	Water	Water	10.64
13. UOd Rod lattice	4.29	Water	Stainless steel	9.76
14. UOd Rod lattice	4.29	Water	Stainless steel	8.08
15. UOd Rod lattice	4.29	Water	Boral	6.72
16. U Metal cylinders	93.2	Bare	Air	15.43
17. U Metal cylinders	93.2	Paraffin	Air	23.84
18. U Metal cylinders	93.2	Bare	Air	19.97
19. U Metal cylinders	93.2	Paraffin	Air	36.47
20. U Metal cylinders	93.2	Bare	Air	13.74
21. U Metal cylinders	93.2	Paraffin	Air	23.48
22. U Metal cylinders	93.2	Bare	Plexiglass	15.74

TABLE 4.3-4 (Cont)

General Description	Enrichment wt% U235	Reflector	Separating Material	Characterizing Separation (cm)
23. U Metal cylinders	93.2	Paraffin	Plexiglass	25.43
24. U Metal cylinders	93.2	Bare	Plexiglass	21.74
25. U Metal cylinders	93.2	Paraffin	Plexiglass	27.94
26. U Metal cylinders	93.2	Bare	Steel	14.74
27. U Metal cylinders	93.2	Bare	Plexiglass, steel	16.67

Table 1. Benchmark Critical Experiments [5,6]

INSERT 2

	General Description	Enrichment w/o U235	Reflector	Separating Material	Soluble Boron ppm	k_{eff}
1.	UO_2 rod lattice	2.46	water	water	0	0.9857 ± .0028
2.	UO_2 rod lattice	2.46	water	water	1037	0.9906 ± .0018
3.	UO_2 rod lattice	2.46	water	water	764	0.9896 ± .0015
4.	UO_2 rod lattice	2.46	water	B4C pins	0	0.9914 ± .0025
5.	UO_2 rod lattice	2.46	water	B4C pins	0	0.9891 ± .0026
6.	UO_2 rod lattice	2.46	water	B4C pins	0	0.9955 ± .0020
7.	UO_2 rod lattice	2.46	water	B4C pins	0	0.9889 ± .0026
8.	UO_2 rod lattice	2.46	water	B4C pins	0	0.9983 ± .0075
9.	UO_2 rod lattice	2.46	water	water	0	0.9931 ± .0028
10.	UO_2 rod lattice	2.46	water	water	143	0.9928 ± .0025
11.	UO_2 rod lattice	2.46	water	stainless steel	514	0.9967 ± .0020
12.	UO_2 rod lattice	2.46	water	stainless steel	217	0.9943 ± .0019
13.	UO_2 rod lattice	2.46	water	borated aluminum	15	0.9892 ± .0023
14.	UO_2 rod lattice	2.46	water	borated aluminum	92	0.9884 ± .0023
15.	UO_2 rod lattice	2.[illegible]	water	borated aluminum	395	0.9832 ± .0021
16.	UO_2 rod lattice	2.46	water	borated aluminum	121	0.9848 ± .0024
17.	UO_2 rod lattice	2.46	water	borated aluminum	487	0.9895 ± .0026
18.	UO_2 rod lattice	2.46	water	borated aluminum	197	0.9885 ± .0022
19.	UO_2 rod lattice	2.46	water	borated aluminum	634	0.9921 ± .0019
20.	UO_2 rod lattice	2.46	water	borated aluminum	320	0.9920 ± .0020
21.	UO_2 rod lattice	2.46	water	borated aluminum	72	0.9939 ± .0020
22.	U metal cylinders	93.2	bare	air	0	0.9905 ± .0020
23.	U metal cylinders	93.2	bare	air	0	0.9976 ± .0020
24.	U metal cylinders	93.2	bare	air	0	0.9947 ± .0025
25.	U metal cylinders	93.2	bare	air	0	0.9928 ± .0019
26.	U metal cylinders	93.2	bare	air	0	0.9922 ± .0026
27.	U metal cylinders	93.2	bare	air	0	0.9950 ± .0027
28.	U metal cylinders	93.2	bare	plexiglass	0	0.9941 ± .0030
29.	U metal cylinders	93.2	paraffin	plexiglass	0	0.9928 ± .0041
30.	U metal cylinders	93.2	bare	plexiglass	0	0.9968 ± .0018
31.	U metal cylinders	93.2	paraffin	plexiglass	0	1.0042 ± .0019
32.	U metal cylinders	93.2	paraffin	plexiglass	0	0.9963 ± .0030
33.	U metal cylinders	93.2	paraffin	plexiglass	0	0.9919 ± .0032

CHAPTER 9

AUXILIARY SYSTEMS

9.1 FUEL STORAGE AND HANDLING

9.1.1 New Fuel Storage

The new fuel storage area is located in the fuel area shown on Figures 9.1-1, 9.1-2, and 9.1-3 and is designed to provide a safe, effective means for dry storage of the new fuel assemblies.

9.1.1.1 Design Bases

The new fuel storage area is designed in accordance with the following criteria:

1. General Design Criterion 2, as it relates to the ability of structures housing the facility components to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, and floods.

2. General Design Criterion 5, as it relates to shared structures, systems, and components important to safety being capable of performing required safety functions.

3. General Design Criterion 61, as it relates to the facility design for fuel storage.

4. General Design Criterion 62, as it relates to the prevention of criticality by physical systems or the process utilizing geometrically safe configurations.

5. Regulatory Guide 1.29, as it relates to the seismic design classification of facility components.

6. The new fuel storage facility is designed to store sufficient fuel for one refueling (one-third core) plus 17 spares, for a total of 70 assemblies, and maintain the fuel subcritical ($K_{eff} < 0.95$) when fully loaded and flooded with non-borated water. With fuel of the highest anticipated enrichment, assuming optimum moderation, the effective multiplication factor (K_{eff}) will not exceed ~~.98.~~ .95.

9.1.1.2 Facilities Description

The new fuel storage area is shown on Figures 9.1-1, 9.1-2, and 9.1-3. New fuel storage is provided for one-third core (53 fuel assemblies) plus 17 spare assemblies. New fuel assemblies are stored dry in a steel and concrete structure within the fuel building. The new fuel storage racks consist of a stainless steel support structure into which 70 stainless steel fuel guide assemblies are bolted in 14 parallel rows of five fuel guide assemblies each. It is not possible to insert a fuel assembly in other than a prescribed location due to the design of the fuel guide supporting structure.

The spacing of the new fuel assemblies, located in the new fuel guide assemblies, is a minimum of 21 inches center-to-center. Fuel assemblies are loaded into the fuel guide assemblies through the top. Adequate guidance is provided in each fuel guide assembly by means of a flared lead-in opening to preclude damage to the fuel assemblies during insertion or withdrawal. The accumulation of liquid in the new fuel storage area is prevented by a 4-inch floor drain located in the area.

9.1.1.3 Safey Evaluation

The new fuel storage area is located in the Seismic Category I fuel building. Handling of new fuel is done by a separate 10-ton hoist on the motor-driven platform crane (Section 9.1.4).

New fuel assemblies are stored vertically, with a minimum center to center spacing of 21 inches. This will maintain the fuel in a subcritical condition with the effective multiplication factor, K_{eff} less than 0.95~~, when the new fuel storage area is fully loaded and the storage area is flooded with non-borated water. With fuel of the maximum anticipated enrichment, assuming optimum moderation, K_{eff} will not exceed 0.98.~~ for both the full density (water at 68°F and 1 gm/cm^3) and low density (0.076 gm/cm^3) optimum moderation conditions.

The new fuel storage racks are designed to Seismic Category I requirements. A detailed analysis of the storage racks have been performed to verify the adequacy of the design to withstand the loadings encountered during normal operation, an operating basis earthquake (OBE), and the safe shutdown earthquake (SSE).

The motor-driven platform crane, which is used for transfer of fuel, is the only overhead crane which can pass over the new fuel.

Damage to the fuel assemblies and the new fuel racks by excessive uplift forces from the new fuel handling hoist are prevented by operating procedures and by a load cell attached to the crane. In addition, the new fuel storage area is protected from the effect of dropped heavy objects by interlocks on the fuel handling hoist, which limit the lifting capability of the crane to the weight of a fuel cell and its handling tool. Heavier loads will be handled by an administrative procedure, which will define the area over which these loads may be handled to prevent damage to the new fuel.

9.1.2 Spent Fuel Storage

The spent fuel storage area, located in the fuel building shown on Figures 9.1-1 and 9.1-2, is designed to provide a safe and effective means of storing spent fuel.

9.1.2.1 Design Bases

The spent fuel storage area is designed in accordance with the following criteria:

1. General Design Criterion 2, as it relates to structures housing the facility and the facility itself being capable of withstanding the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, and floods.

2. General Design Criterion 4, as it relates to structures housing the facility and the facility itself being capable of withstanding the effects of environmental conditions, external missiles, internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks, such that safety functions will not be precluded.

3. General Design Criterion 5, as it relates to shared structures, systems, and components important to safety being capable of performing required safety functions.

4. General Design Criterion 61, as it relates to the facility design for fuel storage and handling of radioactive materials.

5. General Design Criterion 62, as it relates to the prevention of criticality by physical systems or processes utilizing geometrically safe configurations.

6. General Design Criterion 63, as it relates to monitoring systems provided to detect conditions that could result in the loss of decay heat removal capabilities, to detect excessive radiation levels, and to initiate appropriate safety actions.

7. The spent fuel storage area is designed in accordance with the requirements of Regulatory Guides 1.13, 1.29, 1.115, and 1.117.

9.1.2.2 Facilities Description

The spent fuel storage area is divided into three areas separated by a stainless steel-lined concrete wall, with a removable gate provided between each area to allow movement of fuel elements between them. Each gate is equipped with an inflatable seal to prevent leakage from one area to another. The three areas are defined as the fuel cask area, the spent fuel pool, and the fuel transfer canal. Each area is lined with stainless steel and is normally filled with borated demineralized water.

The fuel cask area consists of two locations at different elevations, which allow for the safe movement of spent fuel into the shipping cask. The lower elevation provides a sufficient height of water above the fuel being transferred to allow for adequate shielding, while the upper elevation limits the potential spent fuel cask drop height and allows for preliminary decontamination using a floating spray ring.

The spent fuel pool houses the spent fuel storage racks, which provide sufficient space to store spent fuel from a total of 17 refuelings, plus the storage of one full core in the event the reactor must be emptied of fuel at any time during BVPS-2 life. The spent fuel racks consist of 17 rack assemblies, each having a storage capacity of 64 spent fuel elements. Total spent fuel pool storage capability is 1,088 spent fuel elements.

The spent fuel racks consist of two parts, a subbase beam system and the 17 individual rack assemblies. The system of interconnected base beams is provided to bridge the space between embedment pads so load transfer from the racks to the floor occurs only through the embedment pads. Each spent fuel rack, consisting of an 8x8 array of storage cells, is bolted to the base beams. Because the entire complement of base beams and 8x8 racks form a single structural unit, relative sliding between racks is eliminated. However, the base beam system is free to slide since it is not connected to the embedment plates. The storage racks are positioned such that adequate clearances are provided between the racks and pool walls to avoid impacting during seismic events.

ADD INSERT 3

The fuel transfer canal houses the fuel transfer system which provides for transfer of new and spent fuel elements between the fuel building and reactor containment during refueling. Spent fuel is transported between the fuel transfer canal, spent fuel pool, and the fuel cask area by the fuel building motor-driven platform crane. This platform incorporates separate 10-ton hoists for new fuel and spent fuel. A complete description of fuel handling and utilization of the movable platform with hoists is provided in Sections 9.1.4 and 9.1.5. Handling of the spent fuel casks utilizes the spent fuel cask trolley and is described in Section 9.1.5.

Normal makeup water for the spent fuel pool is provided by the primary grade water system. Borated makeup water may be supplied from the refueling water storage tank (RWST) through the fuel pool cleanup system, as described in Section 9.1.3. Boron concentration is normally maintained at 2,000 ppm and monitored by samples taken periodically.

INSERT 3

Fuel stored in the spent fuel pool is segregated into two areas (Region 1 and Region 2). Spent fuel pool Region 1 will provide for storage of fuel with enrichment between 3.6 and 4.85 w/o in a 3 of 4 cell array administratively controlled. The non-fueled cells will provide adequate spacing to prevent criticality. Criticality in Region 2 is prevented by limiting storage to fuel assemblies with burnup dependent enrichment limitations provided in the technical specifications. The soluble boron in the pool water provides available negative reactivity to maintain K_{eff} less than or equal to 0.95 for postulated accidents that would affect an increase in reactivity. These limitations satisfy the design basis for preventing criticality outside the reactor where, including uncertainties, there is a 95% probability at a 95% confidence level that the K_{eff} of the fuel assembly array will be less than 0.95.

Decay heat from fuel elements is removed by the fuel pool cooling system, as described in Section 9.1.3.

Ventilation in the fuel building is designed to maintain a negative pressure and is described in Section 9.4.2.

9.1.2.3 Safety Evaluation

In accordance with Regulatory Guide 1.13, the storage and handling of fuel in the fuel building is designed to protect the fuel, limit potential offsite exposures, and prevent loss of water from the fuel pool which may uncover the fuel.

The spent fuel pool, spent fuel pool liner, and all supporting structures are designed for SSE seismic loads as described in Sections 3.8.4 and 3.2.1.2. The BVPS-2 spent fuel pool structure and the spent fuel racks are classified, designed, and constructed as Seismic Category I items. The spent fuel pool liner and refueling cavity liner are classified, designed, and constructed as Seismic Category II items. The effects of tornadoes, hurricanes, and floods are described in Sections 3.3.1, 3.3.2, and 3.4.1. The capability of these components and structures to withstand the effects of external missiles, pipe whip, and jet impingement forces are described in Sections 3.5.1.1, 3.6.1, and 3.6.2.

The spent fuel pool is designed such that the water level in the pool cannot be decreased below the top of the fuel stored in the spent fuel racks. The fuel transfer gates do not extend below the top of the spent fuel assemblies, and all piping and piping penetrations of the spent fuel pool terminate no lower than 10 feet above the top of the fuel stored in the racks.

The fuel pool is lined with stainless steel and is equipped with a leak chase system and tell-tale drain connections which drain to a tell-tale drain tank located in the fuel building.

In the event of a loss of fuel pool cooling and normal makeup water supply, a supply of water is provided from the Seismic Category I service water system, as described in Section 9.1.3.

Radiation levels are kept at a minimum (Chapter 12) and optical clarity is maintained by the spent fuel pool cleanup system, as described in Section 9.1.3.

The release of radioactive material is prevented by the design of the fuel building ventilation system which maintains a negative pressure on the building and by the supplementary leak collection, as described in Sections 9.4.2 and 6.5.1.

The ASME III portions of the fuel pool cooling system and the ASME III portions of other systems important to safety of the spent fuel stored in the spent fuel storage facility undergo periodic inservice inspection and testing, as described in Sections 3.9.6 and 6.6.

Spent fuel assemblies are stored vertically in 17 free-standing high density storage racks. The racks utilize a neutron absorbing material (boron carbide in nonmetallic binders) in vented storage compartments to prevent the buildup of gases, and have a minimum center-to-center spacing of 10 7/16 inches to maintain the spent fuel in a subcritical condition. With spent fuel of a maximum enrichment of ~~3.6~~ 4.85 percent by weight UO_2, the fuel pool filled with pure water at

~~32~~ 68°F, the fuel stored in the worst feasible geometric configuration, and with the worst case seismic deflection, the effective multiplication factor, K_{eff}, will be less than 0.95. For the condition of a fuel assembly dropping on a storage rack and/or between the racks and the fuel pool liner, the K_{eff} is also verified as less than 0.95, due to the presence of approximately 1000 ppm dissolved boron.

The continued presence of neutron absorbing material is ensured by a poison surveillance program. This program provides samples which are exposed simultaneously to pool water and gamma radiation. Samples are exposed inside sample holders, which can be moved after each refueling to allow irradiation by fresh spent fuel.

Detailed criticality analyses are performed to demonstrate that the spent fuel racks are substantially subcritical (K_{eff} <0.95) for all credible combinations of the normal and abnormal fuel assembly/rack configurations. The criticality analyses are performed using the Monte Carlo Code, Keno IV.

The Monte Carlo 227 energy Code, Keno IV, is a multi-group neutron transport code utilizing ~~123~~ group crosssections which calculates K_{eff}, life-time and generation-time leakage fluxes, and fission densities. Extensive bench marking calculations are performed on criticality experiments involving storage of simulated, fresh pressurized water reactor fuel assemblies in poison storage cells.

REPLACE WITH INSERT 4 →

~~The normal and abnormal configurations considered in the analyses are:~~

~~Normal~~

~~1. Central positioning of fuel assemblies within storage cells of normal dimensions at normal temperature,~~

~~2. Eccentric positioning of adjacent fuel assemblies within the storage cells,~~

~~3. Variations in cell wall thickness, storage cell center-to-center pitch, and poison concentration and thickness, as permitted by fabrication tolerances, and~~

~~4. Variation in fuel parameters including enrichment and fuel rod pitch.~~

~~Abnormal~~

~~1. Bulk pool temperature variations from 32°F to 260°F with further reduction in water density to determine the effects of boiling.~~

INSERT 4

Configurations considered in the analyses are:

The maximum K_{eff} under normal conditions arises from consideration of mechanical and material thickness tolerances resulting from the manufacturing process in addition to asymmetric positioning of fuel assemblies within the storage cells. Studies of asymmetric positioning of fuel assemblies within the storage cells has shown that symmetrically placed fuel assemblies yield conservative results in rack K_{eff}. The sheet metal tolerances are considered along with construction tolerances related to the cell I.D., and cell center-to-center spacing. For the Region 1 racks this resulted in a reduction of the nominal 1.106" water gaps to their minimum values. Thus, the "worst case" KENO model of the Region 1 storage racks contains minimum water gaps of 1.007" with symmetrically placed fuel assemblies.

Most accident conditions will not result in an increase in K_{eff} of the rack. Examples are the loss of cooling systems (reactivity decreases with decreasing water density) and dropping a fuel assembly on top of the rack (the rack structure pertinent for criticality is not excessively deformed and the dropped assembly has more than twelve inches of water separating it from the active fuel height of stored assemblies which precludes interaction).

However, accidents can be postulated which would increase reactivity (i.e., or dropping a fuel assembly between the rack and pool wall). For these accident conditions, the double contingency principle of ANSI N16.1-1975 is applied. This states that one is not required to assume two unlikely, independent, concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition since not assuming its presence would be a second unlikely event.

The presence of approximately 1000 ppm boron in the pool water will decrease reactivity by about 15 percent ΔK. Thus, for postulated accidents, should there be a reactivity increase, K_{eff} would be less than or equal to 0.95 due to the effect of the dissolved boron.

2. Storage cells at minimum center-to-center spacing resulting from seismic vibration/displacement,

3. Fuel handling incident in which a fuel assembly is placed adjacent to a fully loaded rack.

4. Fuel handling incident in which a fuel assembly is dropped from a height of 2 feet above the top of the racks, and

5. Fuel handling incident in which a fuel assembly is lying across the top of the racks.

All analyses are performed assuming the fuel stored to be non-irradiated with 3.6 weight percent UO_2 enrichment in a pure water environment. An additional analysis is performed to determine the criticality effects of missing poison plates.

The high density spent fuel storage racks have been designed to meet the requirements for Seismic Category I structures. Detailed structural and seismic analyses of the storage racks have been performed to verify the adequacy of the design to withstand the loadings encountered during normal operation, OBE, and SSE.

As described in Sections 9.1.4 and 9.1.5, the moveable platform with hoists is the only crane operating over the spent fuel and is described in Section 9.1.4. The spent fuel cask trolley is described in Section 9.1.5, along with a description of the paths of travel and interlocks to preclude the dropping of heavy objects on stored spent fuel.

Cooling of spent fuel stored in the spent fuel storage racks is accomplished by the safety-related Seismic Category I fuel pool cooling system described in Section 9.1.3. The adequacy of natural circulation flow to cool the spent fuel assemblies was established by a thermal hydraulic analysis, which concluded that natural circulation in the spent fuel pool is adequate to prevent local boiling.

The design of the spent fuel racks is such that it is not possible to insert a spent fuel element in other than a design location, for example, between storage locations or between racks.

Damage to the spent fuel assemblies and the spent fuel racks by excessive uplift forces exerted by the spent fuel hoist during fuel handling are prevented by the hoist's load cell.

All materials used in construction are compatible with the spent fuel pool environment. All materials are corrosion resistant stainless steel, with the exceptions of the neutron absorbing material, gate seals, and fuel pool lights, and will not contaminate the fuel assemblies or pool environment.

A T T A C H M E N T E

Criticality Analysis

(provided for background information)