APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NO. NPF-3

FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NO. 1

Attached are the requested changes to the Davis-Besse Nuclear Power Station, Unit No. 1 Facility Operating License No. NPF-3. Also included are the Safety Evaluation and Significant Hazards Consideration.

The proposed changes (submitted under cover letter Serial No. 1479) concern:

Section 3/4.9, Refueling Operations, Specification 3.9.13, Spent Fuel Pool Fuel Assembly Storage;
Section 3/4.9, Refueling Operations, Specification 4.9.13.1 and 4.9.13.2., Spent Fuel Pool Fuel Assembly Storage;
Bases, Section 3/4.9.13, Spent Fuel Pool Fuel Assembly Storage;
Section 5.3, Reactor Core, Specification 5.3.1, Fuel Assemblies; and
Section 5.6, Fuel Storage, Specification 5.6.1.2.a,b,c, Criticality.

By: _____
D. C. Shelton, Vice President, Nuclear

Sworn and subscribed before me this 11th day of April, 1988.

_____
Notary Public, State of Ohio

My commission expires 5/15/91

LAURIE A. HINKLE
Notary Public, State of Ohio
My Commission Expires May 15, 1991

The following information is provided to support issuance of the requested changes (Attachment 3) to the Davis-Besse Nuclear Power Station, Unit No. 1 Operating License No. NPF-3, Appendix A, Technical Specification Sections 3.9.13, 4.9.13.1, 4.9.13.2, 5.3.1 and 5.6.1.2.a,b,c and Bases Section 3/4.9.13:

A. Time Required to Implement: This change will be implemented by the licensee within 45 days following NRC issuance of the License Amendment. The change is requested to be issued by October 15, 1988 to facilitate enrichment contracting efforts for fuel to be manufactured.

B. Reason for Change (Facility Change Request (FCR) 87-0134): Revise the Technical Specifications to increase the maximum allowable fuel enrichment from the present 3.30 wt% uranium-235 to 3.80 wt% uranium-235.

C. Safety Evaluation: See attached Safety Evaluation (Attachment No. 1).

D. Significant Hazards Consideration: See attached Significant Hazards Consideration (Attachment No. 2).

SAFETY EVALUATION

DESCRIPTION OF PROPOSED ACTIONS

The purpose of this safety evaluation is to review proposed changes to Technical Specification 5.3.1 to allow an increase in the maximum uranium-235 enrichment in the nuclear fuel from 3.3 wt% U-235 to 3.8 wt% U-235. Since the maximum allowable enrichment for the fuel is determined by and limited by criticality considerations in the Spent Fuel Storage Racks and New Fuel Storage Racks, Technical Specification 5.6.1.2 must also be amended to include the new criticality criterion for optimum moderation or "mist" conditions in the New Fuel Storage Racks, and a new Limiting Condition for Operation, 3.9.13, along with its associated Applicability, Actions, Surveillance Requirements, and Bases, will be added to address the restrictions on fuel assembly location in the Spent Fuel Storage Racks.

Increasing the maximum allowable U-235 enrichment in the fuel will permit longer fuel cycles with smaller fuel assembly feed batch sizes. This results in more efficient uranium utilization and fewer spent fuel assemblies to be stored on site. The latter is especially important considering the limited capacity of the Spent Fuel Storage Racks and the current state of the high level nuclear waste disposal program. It is anticipated that enrichments above the current limit of 3.3 wt% U-235 will be required at Davis-Besse for Cycle 7 (fuel batch 9).

SYSTEMS AND COMPONENTS AFFECTED

Reactor Core
Nuclear Fuel
Spent Fuel Storage Racks (SFSR)
New Fuel Storage Racks (NFSR)

DOCUMENTS AFFECTED

USAR Section 1.2.7, Fuel Handling

USAR Section 9.1.1, New Fuel Storage

USAR Section 9.1.2, Spent Fuel Storage

USAR Section 9.1.4, Fuel Handling System

USAR Section 15.4.7, Fuel-Handling Accident

Technical Specification 5.3.1, Fuel Assemblies

Technical Specification 5.6.1, Criticality

SAFETY FUNCTIONS OF THE AFFECTED SYSTEMS AND COMPONENTS

The nuclear fuel in the reactor core produces heat through the fissioning of uranium and plutonium. This heat is ultimately used to produce steam which drives the turbine to produce electricity. The safety functions performed by the reactor core and the nuclear fuel assemblies are to retain the fuel in an appropriate geometry for heat removal, and to prevent the migration of radioactive fission products away from the uranium dioxide fuel pellets by encapsulating the fuel pellets in Zircaloy cladding.

The NFSR are designed to store new, non-irradiated nuclear fuel assemblies in a dry, vertical configuration. The safety functions of the NFSR are to prevent damage to the fuel assemblies through seismic or other similar events, and to maintain the fuel assemblies in a non-critical configuration. The NFSR are Seismic Class I structures. These racks were designed with a nominal 21 inch center-to-center spacing between fuel assemblies which maintains a k-effective of less than 0.95 even if the entire rack and assembly configuration is flooded with non-borated water [4].

The SFSR are designed to store either new, non-irradiated fuel or burned, irradiated fuel in a vertical configuration under water. The safety functions of the SFSR are to prevent damage to the fuel assemblies during seismic and other similar events, and to maintain the fuel assemblies in a non-critical configuration. The SFSR are Seismic Class I structures. These spent fuel racks are designed with a 12 31/32 inch nominal center-to-center fuel assembly spacing in one direction, and a 13 3/16 inch nominal spacing in the other orthogonal direction. This spacing, along with the "flux trap" construction of the racks, is designed to maintain a k-effective of less than 0.95 when immersed in non-borated water [4].

EFFECTS ON SAFETY

Increasing the uranium-235 enrichment of the fuel to 3.8 wt% U-235 will in no way adversely affect the safety functions performed by the reactor core and the nuclear fuel assemblies. Increasing the fuel enrichment does not increase the mass of the fuel assembly and, therefore, has no effect on the ability of the structural materials in the core to maintain the fuel assemblies in the proper configuration. Any changes in the nuclear properties of the reactor core as a result of increasing the fuel enrichment would be analyzed in the corresponding reload analysis for that core. Also, increasing the fuel enrichment would have little, if any, effect on fission product inventories in the core or in individual fuel assemblies since fission product inventories are mainly a function of fuel assembly power and burnup and not initial fuel enrichment. Any changes in fission product inventories that might result from the higher burnups achievable using higher enriched fuel in the reactor core would also be analyzed as part of the reload analysis for the core.

As was stated above, increasing the U-235 enrichment of the fuel does not increase the mass of the fuel. Therefore, increasing the enrichment of the fuel will have no effect on the ability of the NFSR or the SFSR to maintain the fuel in a proper configuration during normal and off-normal events. The amount of decay heat produced by the fission products in the spent fuel will be within the capability of the spent fuel pool cooling systems.

The only major effect that increasing the fuel enrichment will have is on criticality in the NFSR and the SFSR. The criticality analysis for both storage facilities has been re-evaluated [1] and is discussed below.

CRITICALITY ANALYSIS

1. Background

Criticality defines the condition of a fissile system when the number of nuclear fissions produced by one generation of neutrons equals the number of fissions produced by the next generation of neutrons, which indicates that a fission chain reaction is occurring. This condition is highly undesirable in a fuel storage facility since the chain reaction will produce large amounts of heat and neutron and gamma radiation. Such storage facilities are generally designed such that, under worst case conditions, the system is always at least 5 percent subcritical (k-effective < 0.95) when flooded with non-borated water, and at least 2 percent subcritical (k-effective < 0.98) when immersed in a hydrogenous "mist" that provides optimum moderation [6,7]. The latter criterion applies only to dry storage facilities, such as the NFSR [11].

2. Methodology

The primary computer code used for performing this analysis was the KENO-IV Monte Carlo code. KENO-IV allows the modeling of relatively complex geometries with a simple input description and calculates a k-effective and associated uncertainty for the fissile system. The KENO-IV code was developed at Oak Ridge National Laboratory and is widely used throughout the nuclear industry for criticality analyses.

Cross sections for KENO-IV were obtained from either the standard 123 energy group XSDRN cross section library or from the NULIF/NUTAN 111 energy group library. It was necessary to use the NULIF/NUTAN library to model irradiated fuel since the 123 group library did not contain adequate fission product cross section data for this purpose. The NULIF/NUTAN codes were developed by Babcock & Wilcox and have been used to process cross sections for past analyses on Davis-Besse cores. The NITAWL code was used to generate resonance self-shielding corrections for the 123 group cross sections. The XSDRNPM code was used to generate a representative flux spectrum for spatially weighting the cross sections from the 123 group library. The XSDRN library, as well as the NITWAL and XSDRNPM codes, was developed at Oak Ridge.

3. Biases And Uncertainties

Aside from the statistical uncertainty associated with each KENO-IV calculation, the primary uncertainty in this analysis was the accuracy of KENO-IV when compared against actual critical experiments. B&W has benchmarked KENO-IV against several critical experiments conducted at its Lynchburg Research Center [13]. When using the standard 123 group cross section library, it was determined that KENO-IV exhibited a non-conservative bias in k-effective that was a function of the spacing between fuel assemblies. When adjusted for the actual spacing between assemblies in the Davis-Besse spent fuel pool, this bias, combined with its associated uncertainty, had a value of 0.023 delta k. This bias was determined to be bounding for all fuel assembly spacings and water densities that could be present in either the SFSR or the NFSR. Therefore, this bias of 0.023 delta k was added to all results generated by KENO-IV.

When using the NULIF/NUTAN cross section libraries, NUTAN collapsed the cross sections from 111 groups to 13 groups. When the 13 group model was benchmarked against the 123 group model, it was discovered that the 13 group NULIF/NUTAN cross sections produced an overly conservative k-effective. Therefore, a reactivity credit was defined and used in cases where the 13 group NULIF/NUTAN library was employed in KENO-IV. This credit had a value of 0.01583 delta k and had an associated 1 sigma uncertainty of 0.00555 delta k. This uncertainty was statistically combined with all other uncertainties when calculating the final k-effective for a particular case. The 13 group NULIF/NUTAN cross section library was used only in cases where it was necessary to model irradiated fuel, and, therefore, this reactivity credit was applied only in those cases. An additional set of benchmarks was performed by comparing pin cell KENO-IV cases run with the NULIF code itself. It was shown that KENO-IV slightly underpredicted the k-effective calculated by NULIF, and, consequently, an additional reactivity penalty of 0.002 delta k with a 1 sigma uncertainty of 0.00145 delta k was applied to all KENO-IV cases that used the 13 group NULIF/NUTAN cross section library.

When burnable poison rod assemblies (BPRAs) are inserted into fuel assemblies, the neutron spectrum seen in these fuel assemblies is hardened, thus leading to less U-235 utilization and more plutonium production. Once the BPRA is removed from the fuel assembly, that fuel assembly will have a higher reactivity (k-effective) than an assembly with an identical fuel burnup but with no BPRA history. Since the 13 group NULIF/NUTAN cross section library generated for the KENO-IV cases assumed that no BPRAs were present in the fuel, a reactivity penalty had to be defined to conservatively account for this effect. A special NULIF depletion to 18,577 MWD/MTU was performed assuming the fuel contained a BPRA with 1.6 wt% boron carbide and the reactivity penalty was determined to be 0.006 delta k. This value is conservative for two reasons: one, BPRAs with greater than 1.4 wt% boron carbide are not anticipated to be

used at Davis-Besse and the effect increases with the boron carbide enrichment; and, two, fuel assemblies with BPRA loadings this large do not reach 18,577 MWD/MTU of burnup in one eighteen month cycle, and the effect increases with the amount of burnup incurred while the BPRA was present. This reactivity penalty was applied only to those KENO-IV cases that modeled irradiated fuel.

4. Assumptions

The analysis was done to determine whether an increase in fuel enrichment to 3.8 wt% U-235 could be tolerated. This value was chosen because it bounded any enrichment that would be used in an 18 month fuel cycle, even if the reload feed batch size was reduced to 60 assemblies. Further, in order to address the potential uncertainty in the actual enrichment of the fuel pellets, a one percent tolerance (per B&W fuel pellet specification [15]) was added to the enrichment. Therefore, as is required per ANSI/ANS-57.2-1983 [9] and ANSI/ANS-57.3-1983 [10], an enrichment of at least 3.838 wt% U-235 was used for the "most reactive fuel assembly". Throughout this text, nominal values for enrichment are quoted even though the one percent tolerance was included in the actual calculations.

For the NFSR, the following assumptions were used:

a) The center-to-center spacing between fuel assemblies is the minimum possible based upon dimensional tolerances for the NFSR and for the fuel assemblies themselves. This minimum spacing, which is 20.57 inches, will conservatively bound, with respect to k-effective, any other possible spacing.

b) No burnable poison, control rod components, rack structural materials, assembly spacer grids, upper or lower assembly end fittings, or other structural neutron absorbers were modeled in the criticality calculations.

c) Criticality was examined for both flooded conditions (water density of 1.00 grams per cubic centimeter) and for optimum moderation or "mist" conditions. The "mist", as described in ANSI/ANS-57.9-1984 [11] could be created as a result of a water or steam pipe break or as a result of using a foam-type fire retardent in a fire-fighting situation in the vicinity of the NFSR. Analysis for the type of criticality phenomenon is required per USNRC Standard Review Plan Section 9.1.1 [6]. For the "mist" condition, sufficient cases were run to determine the moderator density which provided the highest value of k-effective.

d) Due to the design of the NFSR, it was not necessary to examine the effects of dropping or misplacing a fuel assembly into the racks in such a manner as to compromise the minimum assembly spacing discussed above.

For the SFSR, the following assumptions were used:

a) No structural materials in the SFSR were considered with the exception of the 0.125 inch thick 304 stainless steel can surrounding each fuel assembly.

b) No credit was taken for soluble boron in the water in the spent fuel pool. This is required per ANSI/ANS-57.2-1983 [9] and per USNRC Standard Review Plan Section 9.1.2 [7].

c) To conservatively account for the effects of fuel densification, the fuel pellets were assumed to be of nominal diameter and length, but of maximum density.

d) The SFSR were considered to be infinite in the horizontal X-Y directions, and to have a twelve inch water reflector immediately above and below the active fuel length. This conservatively ignores neutron leakage in the horizontal directions and provides sufficient distance for thermal spectrum decoupling in the vertical directions.

e) No burnable poisons, control rod components, assembly spacer grids, or upper and lower assembly end fittings were modeled in the criticality calculations.

f) For irradiated fuel, it was assumed that all xenon had decayed away before storage in the spent fuel racks and that samarium had been allowed to accumulate for seven days before storage. The latter assumption is slightly non-conservative since current Technical Specifications allow fuel movement within three days of shutdown. Subsequent calculations indicated that the increase in reactivity between seven days of decay and three days of decay was very small and could be adequately accounted for in the extra conservatism incorporated in the burnable poison reactivity penalty defined above for irradiated fuel.

g) To account for dimensional tolerances in the construction of the SFSR and in the fuel assemblies themselves, the center-to-center assembly spacings were assumed to be 13-1/8 inches in one direction and 12-29/32 inches in the other orthogonal direction.

h) The temperature of the water in the spent fuel pool was assumed to be that which produced the highest value of k-effective. Sufficient cases were examined to determine this optimum water temperature.

5. New Fuel Storage Rack Criticality

When the NFSR were modeled as being completely filled (80 assemblies) with 3.8 wt% fuel and in a flooded condition, KENO-IV yielded a k-effective of 0.9018 plus or minus 0.0042. When combined with the bias of 0.023 defined above and the statistical uncertainty, the final k-effective was 0.9332. This is well under the acceptance criterion of 0.95, as is mandated by USNRC Standard Review Plan 9.1.1 [6], ANSI/ANS-57.3-1983 [10], and Technical Specification 5.6.1.2 [5]. However, it was determined that 80 fuel assemblies of this enrichment in the NFSR would, under "mist" conditions, become critical. Therefore, it was necessary to reduce the number of fuel assemblies that could be stored in the NFSR so as to meet the "mist" criticality criterion of k-effective less than 0.98 (USNRC Standard Review Plan 9.1.1 [6]).

Various geometries were examined and it was determined that 64 fuel assemblies of 3.8 wt% U-235 or less could be stored in the new fuel storage racks if no fuel assemblies were loaded into rows 4 or 7 (see Figure 1). Administrative controls will be implemented to ensure that fuel assemblies will not be placed in any locations in rows 4 and 7 of the NFSR. Using this scheme, several KENO-IV cases were run to determine the optimum moderator density, which was found to be 0.07 grams per cubic centimeter. From a KENO-IV run for the stated conditions and geometry, a k-effective of 0.9352 plus or minus 0.0041 was obtained. Statistically combining the KENO-IV bias and the case uncertainty yielded a maximum k-effective of 0.9663. This value is also well below the acceptance criterion of 0.98, as defined above.

A worst-case scenario was examined by assuming 64 3.8 wt% assemblies in the above geometry under "mist" conditions with an additional dropped fuel assembly of 3.8 wt% lying horizontally on the new fuel storage rack cover above rows 5 and 6. The KENO-IV run for this scenario produced a k-effective that was statistically identical to the case with no dropped fuel assembly, and, therefore, it was not necessary to define a reactivity penalty for this type of accident.

6. Spent Fuel Storage Rack Criticality

In order to perform the spent fuel storage rack criticality analysis, it was necessary to define several penalties to account for various off-normal and accident conditions that could occur in the spent fuel pool.

First, the optimum temperature of the spent fuel pool water with respect to the system k-effective was determined using KENO-IV and this analysis assumed the racks to be fully loaded with 3.9 wt% fuel. The optimum water temperature was determined to be 90 degrees Fahrenheit and all final analysis cases were run at that temperature.

Since the stainless steel cans in the SFSR are slightly larger than the fuel assemblies, it was necessary to examine the effects of a worst-case off-center loading of fuel. Two KENO-IV cases were run, one with the fuel assemblies centered in the cans (see Figure 2) and a second with the fuel assemblies positioned as shown in Figure 3. The fuel assemblies all contained 3.9 wt% U-235 and the following table shows the results:

Geometry	K-effective	Uncertainty
Off-Center	0.94126	0.00421
Normal	0.93627	0.00428

Penalty = 0.00499 delta k plus or minus 0.00600

The penalty is obtained by taking the difference in the two k-effective values. The 1 sigma penalty uncertainty is calculated by statistically combining the uncertainties of the two cases. As was calculated here, this penalty was bounding for all off-center configurations and was added to all final k-effective values for the SFSR.

It was necessary to define a bounding criticality accident for the SFSR and to define the reactivity penalty associated with that bounding accident. Due to the construction of the SFSR, it is physically impossible to load a fuel assembly anywhere that was not meant to have fuel assemblies [2]. Also, since the SFSR are Seismic Class I structures, there was no possibility of the racks themselves physically moving. These design features precluded all but the dropped assembly or "T-bone" accident, in which a fuel assembly is dropped from some fuel transfer mechanism and lies horizontally on top of the racks. This accident configuration was analyzed with KENO-IV using the geometry shown in Figure 4. This analysis was conservative in that the dropped fuel assembly was modeled as lying directly on top of the active fuel in the vertical assemblies in the racks as opposed to approximately 8 inches above the active fuel in reality. Also, the geometry used in the model assumes that every seventh row in the spent fuel racks has a dropped assembly on top of it, instead of just one dropped assembly in the entire pool, and that each dropped fuel assembly is infinite in length. All of the fuel assemblies in this analysis contained 3.9 wt% U-235 fuel. The accident case was compared with the "normal" case described above, and the results are shown in the following table:

Geometry	K-effective	Uncertainty
"T-bone"	0.94946	0.00413
Normal	0.93627	0.00428

Penalty = 0.01319 delta k plus or minus 0.00595

This penalty was bounding for all accident conditions and was applied to all final results for the SFSR.

The first portion of the criticality analysis was to determine if 3.8 wt% unburned fuel could be loaded into all locations in the spent fuel storage racks without exceeding a k-effective of 0.95. This would be the most ideal approach since it would not require any administrative controls on fuel assembly location in the spent fuel racks. An initial KENO-IV case using 3.5 wt% fuel and a pool temperature of 90 degrees Fahrenheit yielded a k-effective of 0.91836 plus or minus 0.00775. When combined with the above penalties and uncertainties and the KENO-IV bias, a final k-effective of 0.98247 was obtained. The following equation demonstrates how the final k-effective was calculated:

$$\text{Final k-effective} = 0.91836 + 0.023 + 0.00499 + 0.01319 + 2[(0.00775)^2 + (0.00600)^2 + (0.00595)^2]^{0.5} = 0.98247$$

It should be noted that by statistically combining the various calculational uncertainties to obtain an overall 2 sigma uncertainty and then adding this 2 sigma uncertainty to the final result, a 95/95 confidence level is achieved, which is required for criticality analyses [12].

The k-effective calculated above was clearly unacceptable and indicated that it would not be possible to store 3.8 wt% fuel in the SFSR without some kind of administrative control. It was determined that the highest enrichment that could be stored in the SFSR without administrative controls would be approximately 3.36 wt% U-235, which was very close to the value (3.30 wt%) determined in the previous safety evaluation for these racks [2].

The administrative control decided upon was to use a checkerboard pattern which interspersed the fresh 3.8 wt% U-235 fuel assemblies with fuel assemblies of a lower reactivity, i.e., lower initial enrichment and/or a specified amount of burnup. The type of checkerboard pattern used is shown in Figure 5. The fuel assemblies with combinations of initial enrichment and assembly burnup that could be stored adjacent to the fresh 3.8 wt% fuel assemblies while maintaining a final k-effective of less than 0.95 were then determined.

The geometry for the KENO-IV cases for this type of analysis is shown in Figure 6. The KENO-IV cases defined a curve that began at a low enrichment with no burnup extending to 3.8 wt% enrichment with a large burnup. The following table shows that results of these cases:

Checkerboard Enrichments (B/A)	Checkerboard Burnups (B/A)	K-effective	Uncertainty	Final K-effective
3.8/3.8	0/34,114	0.88949	0.00566	0.94619
3.8/3.1	0/25,136	0.89603	0.00565	0.95273
3.8/3.1	0/26,932	0.88599	0.00477	0.94189
3.8/2.5	0/17,955	0.88561	0.00455	0.94130
3.8/1.4	0/0	0.89106	0.00550	0.94760
3.8/water	0/NA	0.87295	0.00654	0.93056

Note: All enrichments are in wt% U-235 and all burnups are in MWD/MTU. See Figure 5 for checkerboard arrangement.

For these cases, since burned fuel assemblies were being modeled, the 13 group NULIF/NUTAN cross section library was used and, consequently, the NULIF/NUTAN reactivity credit and penalty were applied. Also, the burnable poison penalty of 0.006 delta k was added. Using the first case as an example, the final k-effectives were calculated in the following manner:

$$\text{Final K-effective} = 0.88949 + 0.023 + 0.006 + 0.00499$$

$$+ 0.01319 + 0.00200 - 0.01583$$

$$+ 2[(0.00566)^2 + (0.00600)^2 + (0.00595)^2 + (0.00145)^2 + (0.00555)^2]^{0.5}$$

$$= 0.94619$$

It should be noted that the burnable poison penalty was applied to the last two cases, even though no burned fuel was involved, thus making these cases very conservative. The two cases run with 3.1 wt% fuel were used to interpolate to a final k-effective for 3.1 wt% of 0.947, yielding a burnup of 26,085 MWD/MTU. Also, the final case demonstrates that water holes (rack locations with no fuel assemblies) will also serve to hold down k-effective below 0.95.

The use of this type of loading pattern allows the possibility of another type of accident case, that is, accidentally misloading a 3.8 wt% fresh assembly in a location intended for a low reactivity fuel assembly. This possibility was examined using KENO-IV with a 9x9 checkerboard in which the low reactivity fuel assembly (3.8 wt% with 34,144 MWD/MTU of burnup) in the center of the array has been replaced with a fresh 3.8 wt% assembly. This is a conservative approach because, for the Davis-Besse SFSR, this scenario is approximately the equivalent of having 9 misloaded assemblies uniformly spaced in the spent fuel storage racks simultaneously. The results of this case were compared back to the "normal" case above and are shown in the following table:

Geometry	K-effective	Uncertainty
Misloaded	0.89846	0.00433
Normal	0.88949	0.00566

Penalty = 0.00897 delta k plus or minus 0.00713

Since this penalty is smaller than the penalty for the "T-bone" accident, and, since only one accident had to be considered, the "T-bone" accident is bounding and no additional penalty has to be applied for a misloaded assembly accident.

The final uncertainty to be considered was the uncertainty on the burnup of the fuel assemblies themselves. The B&W Nuclear Reliability Factor Topical Report [14] quotes an uncertainty on measured radial power distributions of 5 percent. Since the uncertainty on assembly burnup would be directly proportional to the uncertainty in the radial power distribution, a burnup uncertainty penalty of 5 percent of the total accumulated burnup was considered. In the most extreme case, which is burned fuel with an initial enrichment of 3.8 wt% U-235, this penalty amounts to 1,706 MWD/MTU. However, from the two cases run above that each use burned 3.1 wt% fuel, it was possible to quantify the effect upon reactivity that a given amount of burnup contributes. From these two cases, it was shown that a change in burnup of 1,789 MWD/MTU in the irradiated fuel assemblies would cause a one percent change in the reactivity of the array. Since it would bound the worst-case 5 percent uncertainty (1,704 MWD/MTU for 3.8 wt% fuel), a 1,800 MWD/MTU penalty was added to the assembly burnups shown in the above table.

A quadratic least squares fit was performed to obtain the equation of the line shown in Figure 7. The largest negative absolute error of the equation, when compared to the fitting points, was 18 MWD/MTU, and this quantity was added to the intercept term to produce an equation which conservatively bounded all of the points from which it was fitted. The equation for the line is:

$$B = -26640 + 22584 * E - 1610 * E^2$$

where B is assembly burnup in MWD/MTU and E is the initial fuel enrichment in wt% uranium-235.

Figure 7 defines three types of fuel. Category "A" fuel assemblies, which are above the burnup-enrichment line, may be stored in any location in the SFSR. Category "B" fuel assemblies, which lie below the burnup-enrichment line but have initial enrichments greater than 3.3 wt% U-235, may be stored only in locations that are directly adjacent to locations containing either Category "A" assemblies, water holes, or some combination or Category "A" assemblies and water holes. Category "C" fuel assemblies, which also lie below the burnup-enrichment line but instead have initial enrichments of less than 3.3 wt% U-235, may be stored anywhere in the SFSR except that they may not be stored adjacent to locations containing Category "B" fuel assemblies. It is possible to define Category "C" fuel by noting that the SFSR were previously licensed to 3.3 wt% U-235 without taking credit for burnup [2]. As was noted above, the new analysis verified the results of the previous analysis with respect to the upper limit of 3.3 wt% when no burnup credit is taken [2]. Category "B" fuel assemblies may not be stored directly adjacent to each other or directly adjacent to Category "C" fuel in the SFSR. Enrichments higher than 3.8 wt% U-235 will not be permitted.

UNREVIEWED SAFETY QUESTION EVALUATION

The proposed action would not increase the probability of an accident previously evaluated in the USAR because the only USAR accident that is applicable is the dropped fuel assembly accident [4]. Since the mass of the fuel assembly does not increase when the fuel enrichment is increased, the probability of the accident occurring does not change. Also, as was stated above, any changes to the nuclear properties of the reactor core due to an increase in fuel enrichment would be analyzed with respect to the USAR accidents in the appropriate reload analysis. (10CFR50.59(a)(2)(i))

The proposed action would not increase the consequences of an accident previously evaluated in the USAR because fission product inventories in a fuel assembly are not a significant function of initial fuel enrichment. Any changes in fission product inventories that may result from the higher fuel burnups attainable with more highly enriched fuel would be analyzed with respect to the USAR accidents in the appropriate reload analysis for the fuel cycle in which the higher enrichment fuel batch was introduced. (10CFR50.59(a)(2)(i))

The proposed action would not increase the probability of a malfunction of equipment important to safety because no physical changes to the facility, including the fuel handling equipment, will be made with the exception of increasing the uranium-235 enrichment of the fuel pellets in the fuel assemblies. Since the mass of the fuel assemblies will not change, the probability of the new fuel storage racks, the spent fuel storage racks, or the reactor core fuel assembly support structures failing during a seismic or other off-normal event will not increase. (10CFR50.59(a)(2)(i))

The proposed action would not increase the consequences of a malfunction of equipment important to safety because the fission product inventories in the fuel assemblies would not change significantly due to an increase in the fuel enrichment. Any changes in fission product inventories that may result from the higher fuel burnups attainable with more highly enriched fuel would be analyzed with respect to the USAR accidents in the appropriate reload analysis for the fuel cycle in which the higher enrichment fuel batch was introduced. (10CFR50.59(a)(2)(i))

The proposed action would not create the possibility for an accident of a different type than any analyzed in the USAR since the only possible accident that could be created through an increase in fuel enrichment would be a criticality accident, which is already addressed in the USAR in the design bases for the NFSR and SFSR. If any new type of accident or condition could be created in the reactor core due to the increase in U-235 enrichment, that accident or condition would be analyzed as part of the reload analysis for the fuel cycle in which the higher enrichment fuel batch was introduced. (10CFR50.59(a)(2)(ii))

The proposed action would not create a possibility for a malfunction of equipment of a different type than any evaluated previously in the USAR because the only condition that would change with the increase of fuel enrichment would be criticality in the NFSR and SFSR. The above analysis has demonstrated that even with a "mist" condition in the NFSR and the racks loaded with 64 3.8 wt% U-235 fuel assemblies and another 3.8 wt% assembly lying on top of the NFSR cover, criticality could not occur. Similarly, under worst-case conditions in the SFSR (single dropped assembly or single misloaded assembly of the highest possible reactivity), criticality could not occur, even with no soluble boron in the water in the spent fuel pool. (10CFR50.59(a)(2)(ii))

The proposed action would not reduce any margin of safety as defined in the basis for any Technical Specification because the criticality analysis demonstrates that both the NFSR and the SFSR will always be at least 5 percent subcritical under flooded conditions even when credible accidents, such as the "T-bone" accident or a single misloaded assembly (spent fuel storage racks only), are accounted for. The margin of safety in the NFSR will be, indeed, greatly increased since "mist" accidents were not considered in the past and "mist" accidents are much more restricting in terms of criticality. (10CFR50.59(a)(2)(iii)

CONCLUSIONS

Implementation of the change to Technical Specification 5.3.1 to allow an increase in the maximum fuel enrichment from 3.3 wt% U-235 to 3.8 wt% U-235 will have no impact on safety if the fuel is loaded into the NFSR according to the geometry in Figure 1 and if the fuel is loaded into the Spent Fuel Storage Racks using the geometry shown in Figure 5 using the criteria shown in Figure 7. The restrictions on fuel assembly placement in the SFSR will be incorporated into Technical Specification 3.9.13. The above evaluation demonstrates that there are no unreviewed safety questions involved with the implementation of this change.

REFERENCES

[1] Babcock And Wilcox Document No. 86-1170592-00, "Davis-Besse Unit 1 New and Spent Fuel Storage Rack Criticality Analysis", October 2, 1987.

[2] FCR 77-290, "Safety Evaluation of the Spent Fuel Storage Capacity Modification for Davis-Besse Nuclear Power Station Unit 1", December 5, 1977.

[3] Babcock And Wilcox Proposal No. A4-360, July 3, 1987.

[4] Davis-Besse Nuclear Power Station No. 1 Updated Safety Analysis Report, latest revision 5, 7/87.

[5] Davis-Besse Nuclear Power Station Unit 1 Technical Specifications, Appendix "A" To License No. NPF-3, April 22, 1977, last amendment 104.

[6] USNRC Standard Review Plan Section 9.1.1, New Fuel Storage, Rev. 2, July 1981.

[7] USNRC Standard Review Plan Section 9.1.2, Spent Fuel Storage, Rev. 3, July 1981.

[8] USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Revision 1, December 1975.

[9] ANSI/ANS-57.2-1983, Design Requirements For Light Water Reactor Spent Fuel Storage Facilities At Nuclear Power Plants.

[10] ANSI/ANS-57.3-1983, Design Requirements For New Fuel Storage Facilities At Light Water Reactor Plants.

[11] ANSI/ANS-57.9-1984, Design Criteria For An Independent Spent Fuel Storage Installation (Dry Storage Type).

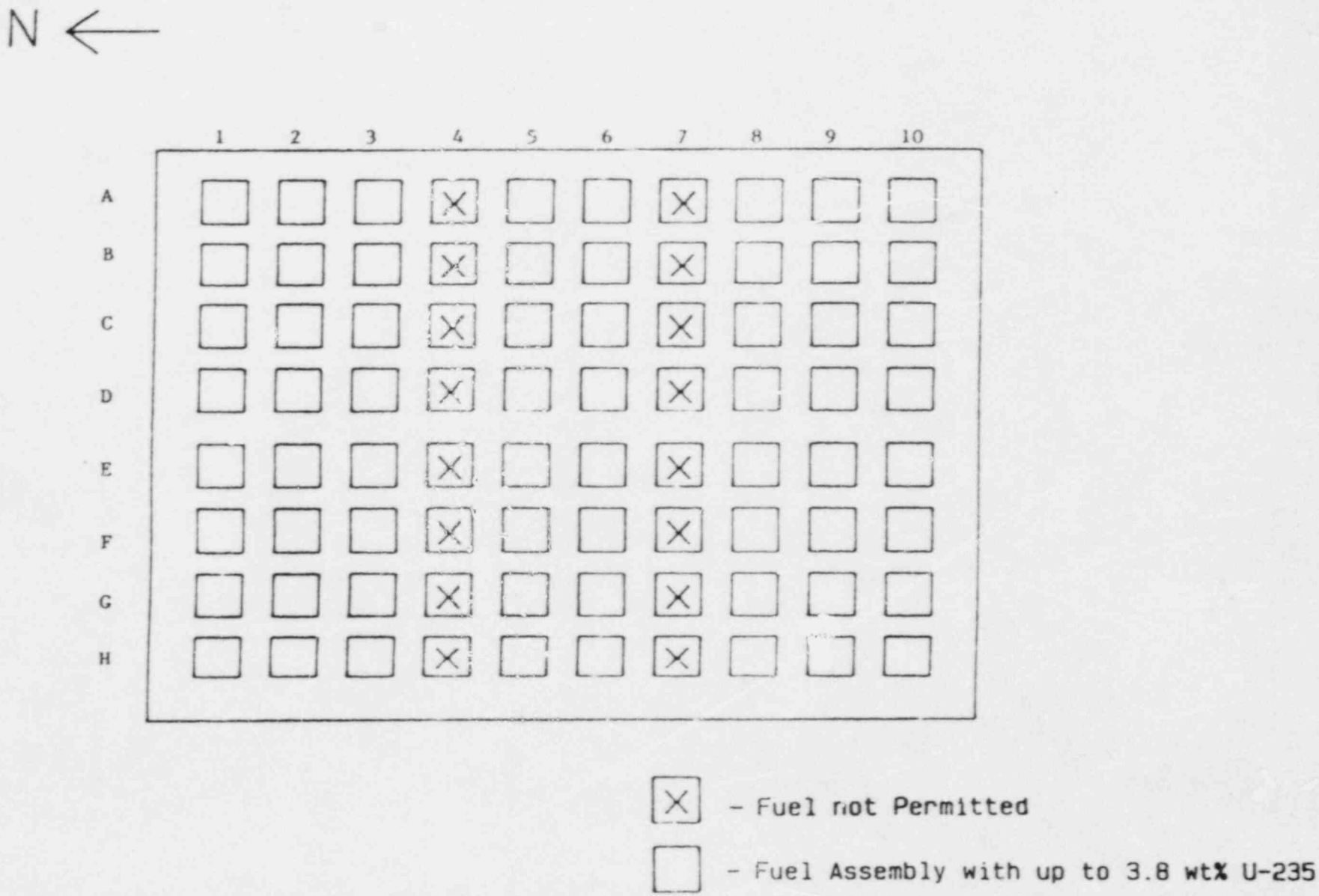
[12] ANSI N16.9-1976, Validation Of Calculational Methods For Nuclear Criticality Safety.

[13] Babcock And Wilcox Topical Report No. BAW-1484-7, "Critical Experiments Supporting Close Proximity Water Storage Of Power Reactor Fuel", July 1979.

[14] Babcock And Wilcox Topical Report No. BAW-10119P-A, "Power Peaking Nuclear Reliability Factors", February 1979.

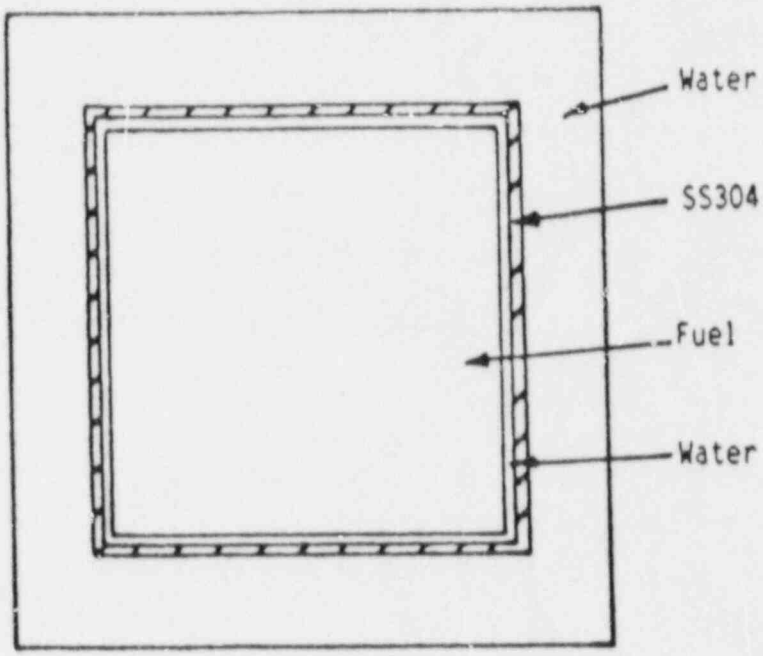
[15] Babcock And Wilcox Specification 08-1116-06, "Pellet Type Zircaloy Clad Fuel", April 26, 1983.

Figure 1
New Fuel Storage Rack
Loading Geometry



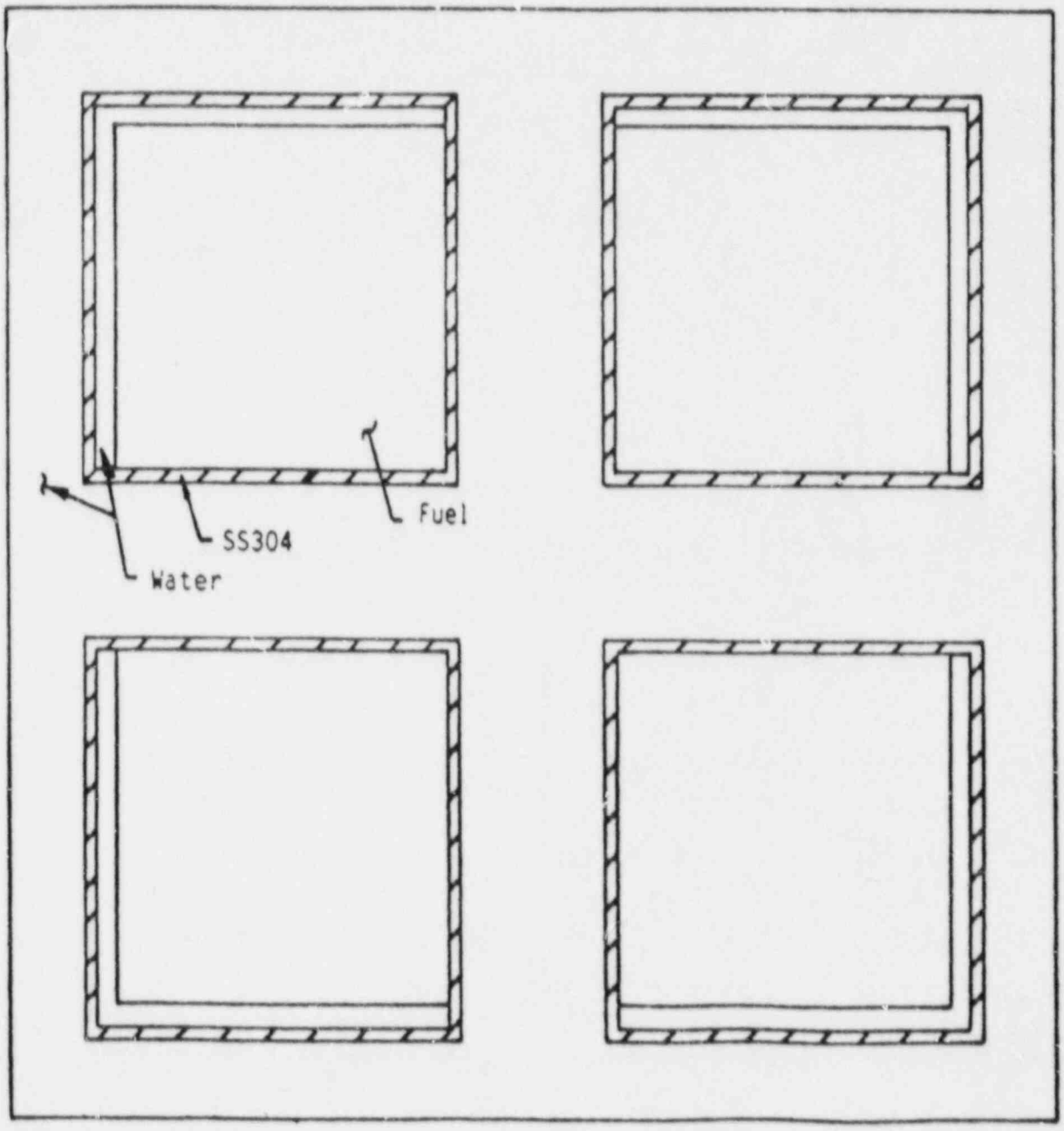
Docket No. 50-346
License No. NPF-3
Serial No. 1479
Attachement 1
Figure 2

Figure 2
Basic Fuel Assembly Representation in the Spent Fuel Pool Storage Rack



Docket No. 50-346
License No. NPF-3
Serial No. 1479
Attachement 1
Figure 3

Figure 3
Off-Centered Assembly Representation in the Spent Fuel Pool Storage Rack



Docket No. 50-346
License No. NPF-3
Serial No. 1479
Attachement 1
Figure 4

Figure 4
T-Bone Accident Representation On the Spent Fuel Pool Rack

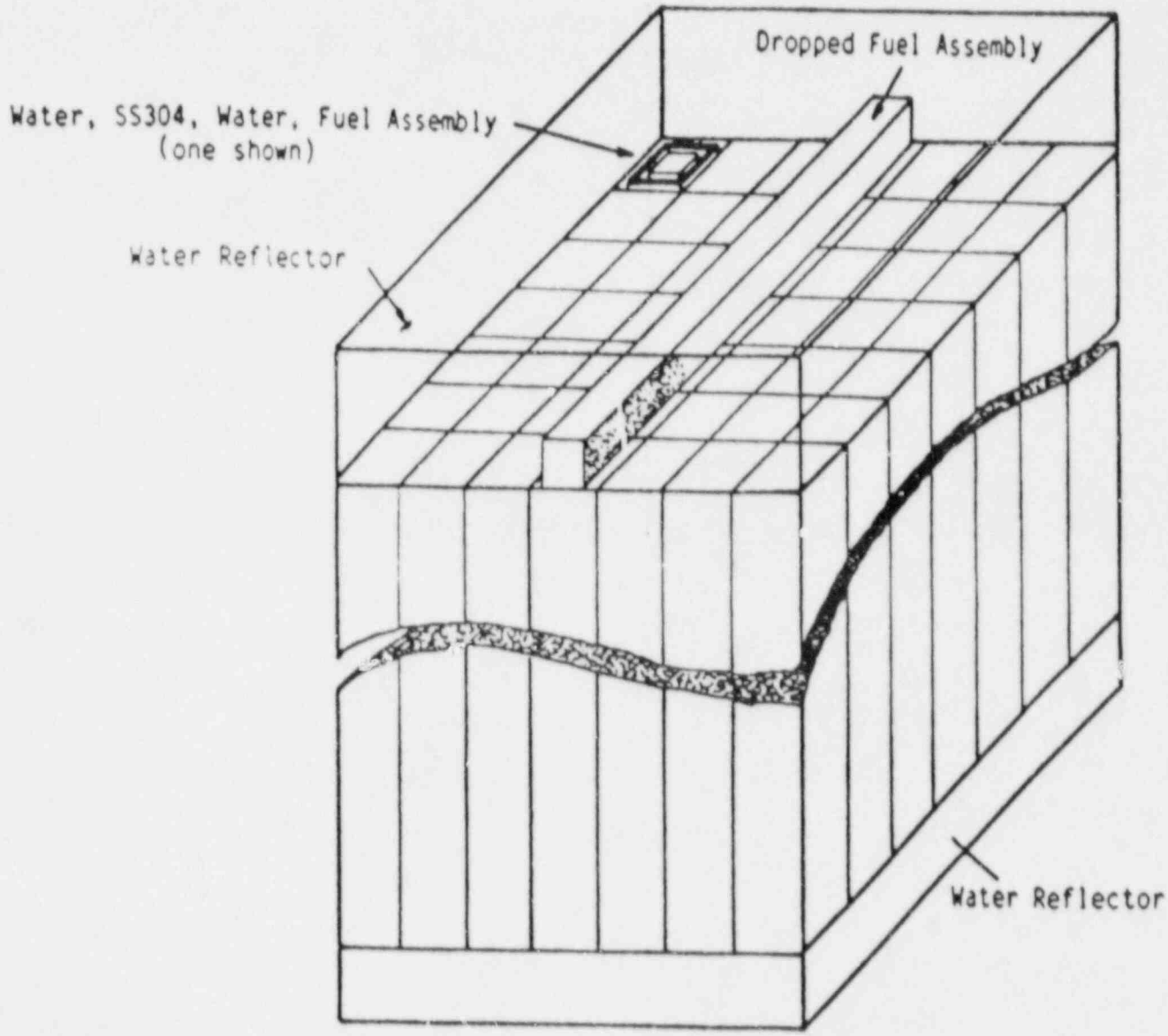


Figure 5
Spent Fuel Pool Checkerboard Pattern

A	B	A	B	A
B	A	B	A	B
A	B	A	B	A
B	A	B	A	B
A	B	A	B	A

A - Low Reactivity Assembly
B - 3.8 wt% U-235, 0 Burnup

Docket No. 50-346
License No. NPF-3
Serial No. 1479
Attachement 1
Figure 6

Figure 6
KENO-IV Geometry for Checkerboard Spent Fuel Storage Rack

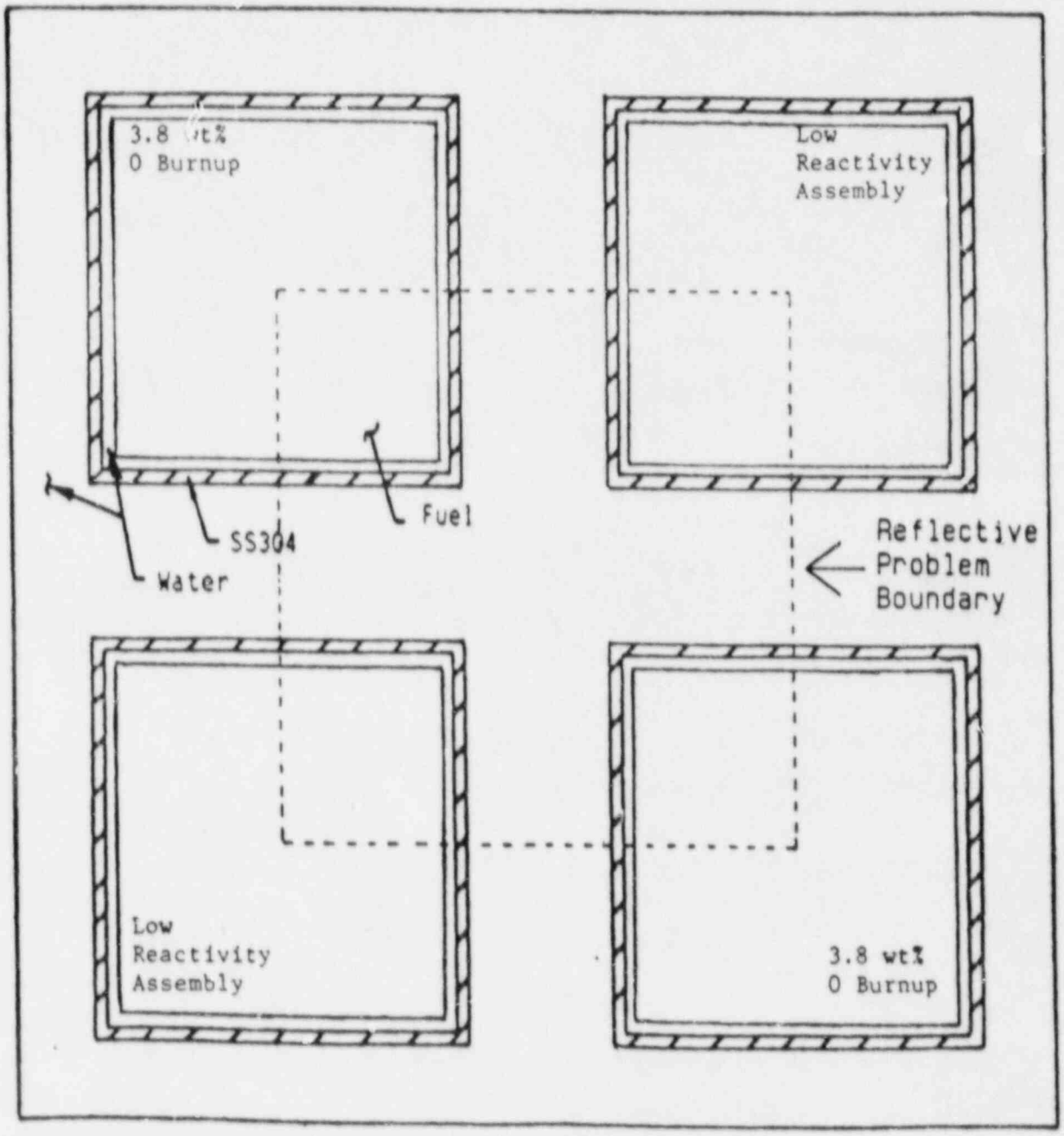
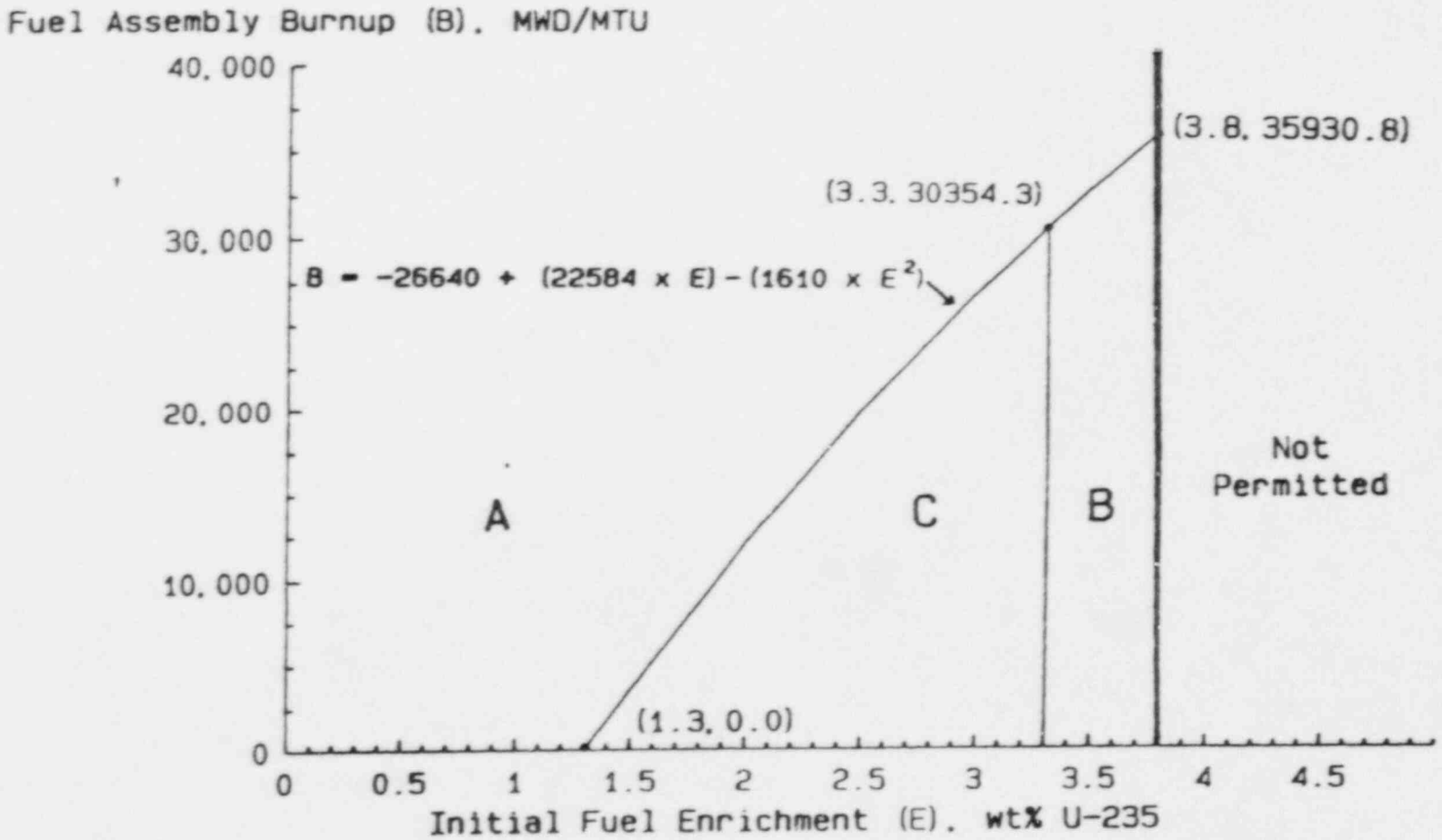


Figure 7
Burnup vs. Enrichment Curve for Davis-Besse Spent Fuel Storage Racks



Category "A" Fuel - May be located anywhere within the storage racks

Category "B" Fuel - Shall only be located adjacent to Category "A" Fuel or water holes within the storage racks

Category "C" Fuel - Shall not be located adjacent to Categor, "B" Fuel

SIGNIFICANT HAZARDS CONSIDERATION

DESCRIPTION OF PROPOSED ACTIONS

The purpose of this Significant Hazards Consideration is to review proposed changes to Technical Specification 5.3.1 to allow an increase in the maximum uranium-235 enrichment in the nuclear fuel from 3.3 wt% U-235 to 3.8 wt% U-235. Since the maximum allowable enrichment for the fuel is determined by and limited by criticality considerations in the Spent Fuel Storage Racks and New Fuel Storage Racks, Technical Specification 5.6.1.2 must also be amended to include the new criticality criterion for optimum moderation or "mist" conditions in the New Fuel Storage Racks. A new Limiting Condition for Operation, 3.9.13, along with its associated Applicability, Actions, Surveillance Requirements, and Basis, will be added to address the restrictions on fuel assembly location in the Spent Fuel Storage Racks. Also, the reference to "FSAR" in Section 5.6.1.2a has been corrected to read "USAR".

Increasing the maximum allowable U-235 enrichment in the fuel will permit longer fuel cycles with smaller fuel assembly feed batch sizes. This results in more efficient uranium utilization and fewer spent fuel assemblies to be stored on site. The latter is especially important considering the limited capacity of the Spent Fuel Storage Racks. It is anticipated that enrichments above the current limit of 3.3 wt% U-235 will be required at Davis-Besse for Cycle 7 (fuel batch 9).

SYSTEMS AND COMPONENTS AFFECTED

Reactor Core
Nuclear Fuel
Spent Fuel Storage Racks (SFSR)
New Fuel Storage Racks (NFSR)

DOCUMENTS AFFECTED

USAR Section 1.2.7, Fuel Handling

USAR Section 9.1.1, New Fuel Storage

USAR Section 9.1.2, Spent Fuel Storage

USAR Section 9.1.4, Fuel Handling System

USAR Section 15.4.7, Fuel-Handling Accident

Technical Specification 5.3.1, Fuel Assemblies

Technical Specification 5.6.1, Criticality

SAFETY FUNCTIONS OF THE AFFECTED SYSTEMS AND COMPONENTS

The nuclear fuel in the reactor core produces heat through the fissioning of uranium and plutonium. This heat is ultimately used to produce steam which drives the turbine to produce electricity. The safety functions performed by the reactor core and the nuclear fuel assemblies are to retain the fuel in an appropriate geometry for heat removal, and to prevent the migration of radioactive fission products away from the uranium dioxide fuel pellets by encapsulating the fuel pellets in Zircaloy cladding.

The NFSR are designed to store new, non-irradiated nuclear fuel assemblies in a dry, vertical configuration. The safety functions of the NFSR are to prevent damage to the fuel assemblies through seismic or other similar events, and to maintain the fuel assemblies in a non-critical configuration. The NFSR are Seismic Class I structures. These racks were designed with a nominal 21 inch center-to-center spacing between fuel assemblies which maintains a k-effective of less than 0.95 even if the entire rack and assembly configuration is flooded with non-borated water [4].

The SFSR are designed to store either new, non-irradiated fuel or burned, irradiated fuel in a vertical configuration under water. The safety functions of the SFSR are to prevent damage to the fuel assemblies during seismic and other similar events, and to maintain the fuel assemblies in a non-critical configuration. The SFSR are Seismic Class I structures. These spent fuel racks are designed with a 12-31/32 inch nominal center-to-center fuel assembly spacing in one direction, and a 13-3/16 inch nominal spacing in the other orthogonal direction. This spacing, along with the "flux trap" construction of the racks, is designed to maintain a k-effective of less than 0.95 when immersed in non-borated water [4].

EFFECTS ON SAFETY

Increasing the uranium-235 enrichment of the fuel to 3.8 wt% U-235 will in no way adversely affect the safety functions performed by the reactor core and the nuclear fuel assemblies. Increasing the fuel enrichment does not increase the mass of the fuel assembly and, therefore, has no effect on the ability of the structural materials in the core to maintain the fuel assemblies in the proper configuration. Any changes in the nuclear properties of the reactor core as a result of increasing the fuel enrichment would be analyzed in the corresponding reload analysis for that core. Also, increasing the fuel enrichment would have little, if any, effect on fission product inventories in the core or in individual fuel assemblies since fission product inventories are mainly a function of fuel assembly power and burnup and not initial fuel enrichment. Any changes in fission product inventories that might result from the higher burnups achievable using higher enriched fuel in the reactor core would also be analyzed as part of the reload analysis for the core.

As was stated above, increasing the U-235 enrichment of the fuel does not increase the mass of the fuel. Therefore, increasing the enrichment of the fuel will have no effect on the ability of the NFSR or the SFSR to maintain the fuel in a proper configuration during normal and off-normal events. The amount of decay heat produced by the fission products in the spent fuel will be within the capability of the spent fuel pool cooling systems.

The only major effect that increasing the fuel enrichment will have is on criticality in the NFSR and the SFSR. The criticality analysis for both storage facilities has been re-evaluated [1] and is discussed below.

CRITICALITY ANALYSIS

1. Background

Criticality defines the condition of a fissile system when the number of nuclear fissions produced by one generation of neutrons equals the number of fissions produced by the next generation of neutrons, which indicates that a fission chain reaction is occurring. This condition is highly undesirable in a fuel storage facility since the chain reaction will produce large amounts of heat and neutron and gamma radiation. Such storage facilities are generally designed such that, under worst case conditions, the system is always at least 5 percent subcritical (k-effective < 0.95) when flooded with non-borated water, and at least 2 percent subcritical (k-effective < 0.98) when immersed in a hydrogenous "mist" that provides optimum moderation [6,7]. The latter criterion applies only to dry storage facilities, such as the NFSR [11].

2. Methodology

The primary computer code used for performing this analysis was the KENO-IV Monte Carlo code. KENO-IV allows the modeling of relatively complex geometries with a simple input description and calculates a k-effective and associated uncertainty for the fissile system. The KENO-IV code was developed at Oak Ridge National Laboratory and is widely used throughout the nuclear industry for criticality analyses.

Cross sections for KENO-IV were obtained from either the standard 123 energy group XSDRN cross section library or from the NULIF/NUTAN 111 energy group library. It was necessary to use the NULIF/NUTAN library to model irradiated fuel since the 123 group library did not contain adequate fission product cross section data for this purpose. The NULIF/NUTAN codes were developed by Babcock & Wilcox and have been used to process cross sections for past analyses on Davis-Besse cores. The NITAWL code was used to generate resonance self-shielding corrections for the 123 group cross sections. The XSDRNPM code was used to generate a representative flux spectrum for spatially weighting the cross sections from the 123 group library. The XSDRN library, as well as the NITWAL and XSDRNPM codes, was developed at Oak Ridge.

3. Biases And Uncertainties

Aside from the statistical uncertainty associated with each KENO-IV calculation, the primary uncertainty in this analysis was the accuracy of KENO-IV when compared against actual critical experiments. B&W has benchmarked KENO-IV against several critical experiments conducted at its Lynchburg Research Center [13]. When using the standard 123 group cross section library, it was determined that KENO-IV exhibited a non-conservative bias in k-effective that was a function of the spacing between fuel assemblies. When adjusted for the actual spacing between assemblies in the Davis-Besse spent fuel pool, this bias, combined with its associated uncertainty, had a value of 0.023 delta k. This bias was determined to be bounding for all fuel assembly spacings and water densities that could be present in either the SFSR or the NFSR. Therefore, this bias of 0.023 delta k was added to all results generated by KENO-IV.

When using the NULIF/NUTAN cross section libraries, NUTAN collapsed the cross sections from 111 groups to 13 groups. When the 13 group model was benchmarked against the 123 group model, it was discovered that the 13 group NULIF/NUTAN cross sections produced an overly conservative k-effective. Therefore, a reactivity credit was defined and used in cases where the 13 group NULIF/NUTAN library was employed in KENO-IV. This credit had a value of 0.01583 delta k and had an associated 1 sigma uncertainty of 0.00555 delta k. This uncertainty was statistically combined with all other uncertainties when calculating the final k-effective for a particular case. The 13 group NULIF/NUTAN cross section library was used only in cases where it was necessary to model irradiated fuel, and, therefore, this reactivity credit was applied only in those cases. An additional set of benchmarks was performed by comparing pin cell KENO-IV cases run with the NULIF code itself. It was shown that KENO-IV slightly underpredicted the k-effective calculated by NULIF, and, consequently, an additional reactivity penalty of 0.002 delta k with a 1 sigma uncertainty of 0.00145 delta k was applied to all KENO-IV cases that used the 13 group NULIF/NUTAN cross section library.

When burnable poison rod assemblies (BPRAs) are inserted into fuel assemblies, the neutron spectrum seen in these fuel assemblies is hardened, thus leading to less U-235 utilization and more plutonium production. Once the BPRA is removed from the fuel assembly, that fuel assembly will have a higher reactivity (k-effective) than an assembly with an identical fuel burnup but with no BPRA history. Since the 13 group NULIF/NUTAN cross section library generated for the KENO-IV cases assumed that no BPRAs were present in the fuel, a reactivity penalty had to be defined to conservatively account for this effect. A special NULIF depletion to 18,577 MWD/MTU was performed assuming the fuel contained a BPRA with 1.6 wt% boron carbide and the reactivity penalty was determined to be 0.006 delta k. This value is conservative for two reasons: one, BPRAs with greater than 1.4 wt% boron carbide are not anticipated to be

used at Davis-Besse and the effect increases with the boron carbide enrichment; and, two, fuel assemblies with BPRA loadings this large do not reach 18,577 MWD/MTU of burnup in one eighteen month cycle, and the effect increases with the amount of burnup incurred while the BPRA was present. This reactivity penalty was applied only to those KENO-IV cases that modeled irradiated fuel.

4. Assumptions

The analysis was done to determine whether an increase in fuel enrichment to 3.8 wt% U-235 could be tolerated. This value was chosen because it bounded any enrichment that would be used in an 18 month fuel cycle, even if the reload feed batch size was reduced to 60 assemblies. Further, in order to address the potential uncertainty in the actual enrichment of the fuel pellets, a one percent tolerance (per B&W fuel pellet specification [15]) was added to the enrichment. Therefore, as is required per ANSI/ANS-57.2-1983 [9] and ANSI/ANS-57.3-1983 [10], an enrichment of at least 3.838 wt% U-235 was used for the "most reactive fuel assembly". Throughout this text, nominal values for enrichment are quoted even though the one percent tolerance was included in the actual calculations.

For the NFSR, the following assumptions were used:

a) The center-to-center spacing between fuel assemblies is the minimum possible based upon dimensional tolerances for the NFSR and for the fuel assemblies themselves. This minimum spacing, which is 20.57 inches, will conservatively bound, with respect to k-effective, any other possible spacing.

b) No burnable poison, control rod components, rack structural materials, assembly spacer grids, upper or lower assembly end fittings, or other structural neutron absorbers were modeled in the criticality calculations.

c) Criticality was examined for both flooded conditions (water density of 1.00 grams per cubic centimeter) and for optimum moderation or "mist" conditions. The "mist", as described in ANSI/ANS-57.9-1984 [11] could be created as a result of a water or steam pipe break or as a result of using a foam-type fire retardent in a fire-fighting situation in the vicinity of the NFSR. Analysis for the type of criticality phenomenon is required per USNRC Standard Review Plan Section 9.1.1 [6]. For the "mist" condition, sufficient cases were run to determine the moderator density which provided the highest value of k-effective.

d) Due to the design of the NFSR, it was not necessary to examine the effects of dropping or misplacing a fuel assembly into the racks in such a manner as to compromise the minimum assembly spacing discussed above.

For the SFSR, the following assumptions were used:

a) No structural materials in the SFSR were considered with the exception of the 0.125 inch thick 304 stainless steel can surrounding each fuel assembly.

b) No credit was taken for soluble boron in the water in the spent fuel pool. This is required per ANSI/ANS-57.2-1983 [9] and per USNRC Standard Review Plan Section 9.1.2 [7].

c) To conservatively account for the effects of fuel densification, the fuel pellets were assumed to be of nominal diameter and length, but of maximum density.

d) The SFSR were considered to be infinite in the horizontal X-Y directions, and to have a twelve inch water reflector immediately above and below the active fuel length. This conservatively ignores neutron leakage in the horizontal directions and provides sufficient distance for thermal spectrum decoupling in the vertical directions.

e) No burnable poisons, control rod components, assembly spacer grids, or upper and lower assembly end fittings were modeled in the criticality calculations.

f) For irradiated fuel, it was assumed that all xenon had decayed away before storage in the spent fuel racks and that samarium had been allowed to accumulate for seven days before storage. The latter assumption is slightly non-conservative since current Technical Specifications allow fuel movement within three days of shutdown. Subsequent calculations indicated that the increase in reactivity between seven days of decay and three days of decay was very small and could be adequately accounted for in the extra conservatism incorporated in the burnable poison reactivity penalty defined above for irradiated fuel.

g) To account for dimensional tolerances in the construction of the SFSR and in the fuel assemblies themselves, the center-to-center assembly spacings were assumed to be 13-1/8 inches in one direction and 12-29/32 inches in the other orthogonal direction.

h) The temperature of the water in the spent fuel pool was assumed to be that which produced the highest value of k-effective. Sufficient cases were examined to determine this optimum water temperature.

5. New Fuel Storage Rack Criticality

When the NFSR were modeled as being completely filled (80 assemblies) with 3.8 wt% fuel and in a flooded condition, KENO-IV yielded a k-effective of 0.9018 plus or minus 0.0042. When combined with the bias of 0.023 defined above and the statistical uncertainty, the final k-effective was 0.9332. This is well under the acceptance criterion of 0.95, as is mandated by USNRC Standard Review Plan 9.1.1 [6], ANSI/ANS-57.3-1983 [10], and Technical Specification 5.6.1.2 [5]. However, it was determined that 80 fuel assemblies of this enrichment in the NFSR would, under "mist" conditions, become critical. Therefore, it was necessary to reduce the number of fuel assemblies that could be stored in the NFSR so as to meet the "mist" criticality criterion of k-effective less than 0.98 (USNRC Standard Review Plan 9.1.1 [6]).

Various geometries were examined and it was determined that 64 fuel assemblies of 3.8 wt% U-235 or less could be stored in the new fuel storage racks if no fuel assemblies were loaded into rows 4 or 7 (see Figure 1). Administrative controls will be implemented to ensure that fuel assemblies will not be placed in any locations in rows 4 and 7 of the NFSR. Using this scheme, several KENO-IV cases were run to determine the optimum moderator density, which was found to be 0.07 grams per cubic centimeter. From a KENO-IV run for the stated conditions and geometry, a k-effective of 0.9352 plus or minus 0.0041 was obtained. Statistically combining the KENO-IV bias and the case uncertainty yielded a maximum k-effective of 0.9663. This value is also well below the acceptance criterion of 0.98, as defined above.

A worst-case scenario was examined by assuming 64 3.8 wt% assemblies in the above geometry under "mist" conditions with an additional dropped fuel assembly of 3.8 wt% lying horizontally on the new fuel storage rack cover above rows 5 and 6. The KENO-IV run for this scenario produced a k-effective that was statistically identical to the case with no dropped fuel assembly, and, therefore, it was not necessary to define a reactivity penalty for this type of accident.

6. Spent Fuel Storage Rack Criticality

In order to perform the spent fuel storage rack criticality analysis, it was necessary to define several penalties to account for various off-normal and accident conditions that could occur in the spent fuel pool.

First, the optimum temperature of the spent fuel pool water with respect to the system k-effective was determined using KENO-IV and this analysis assumed the racks to be fully loaded with 3.9 wt% fuel. The optimum water temperature was determined to be 90 degrees Fahrenheit and all final analysis cases were run at that temperature.

Since the stainless steel cans in the SFSR are slightly larger than the fuel assemblies, it was necessary to examine the effects of a worst-case off-center loading of fuel. Two KENO-IV cases were run, one with the fuel assemblies centered in the cans (see Figure 2) and a second with the fuel assemblies positioned as shown in Figure 3. The fuel assemblies all contained 3.9 wt% U-235 and the following table shows the results:

Geometry	K-effective	Uncertainty
Off-Center	0.94126	0.00421
Normal	0.93627	0.00428

Penalty = 0.00499 delta k plus or minus 0.00600

The penalty is obtained by taking the difference in the two k-effective values. The 1 sigma penalty uncertainty is calculated by statistically combining the uncertainties of the two cases. As was calculated here, this penalty was bounding for all off-center configurations and was added to all final k-effective values for the SFSR.

It was necessary to define a bounding criticality accident for the SFSR and to define the reactivity penalty associated with that bounding accident. Due to the construction of the SFSR, it is physically impossible to load a fuel assembly anywhere that was not meant to have fuel assemblies [2]. Also, since the SFSR are Seismic Class I structures, there was no possibility of the racks themselves physically moving. These design features precluded all but the dropped assembly or "T-bone" accident, in which a fuel assembly is dropped from some fuel transfer mechanism and lies horizontally on top of the racks. This accident configuration was analyzed with KENO-IV using the geometry shown in Figure 4. This analysis was conservative in that the dropped fuel assembly was modeled as lying directly on top of the active fuel in the vertical assemblies in the racks as opposed to approximately 8 inches above the active fuel in reality. Also, the geometry used in the model assumes that every seventh row in the spent fuel racks has a dropped assembly on top of it, instead of just one dropped assembly in the entire pool, and that each dropped fuel assembly is infinite in length. All of the fuel assemblies in this analysis contained 3.9 wt% U-235 fuel. The accident case was compared with the "normal" case described above, and the results are shown in the following table:

Geometry	K-effective	Uncertainty
"T-bone"	0.94946	0.00413
Normal	0.93627	0.00428

Penalty = 0.01319 delta k plus or minus 0.00595

This penalty was bounding for all accident conditions and was applied to all final results for the SFSR.

The first portion of the criticality analysis was to determine if 3.8 wt% unburned fuel could be loaded into all locations in the spent fuel storage racks without exceeding a k-effective of 0.95. This would be the most ideal approach since it would not require any administrative controls on fuel assembly location in the spent fuel racks. An initial KENO-IV case using 3.5 wt% fuel and a pool temperature of 90 degrees Fahrenheit yielded a k-effective of 0.91836 plus or minus 0.00775. When combined with the above penalties and uncertainties and the KENO-IV bias, a final k-effective of 0.98247 was obtained. The following equation demonstrates how the final k-effective was calculated:

$$\begin{aligned}\text{Final k-effective} = {} & 0.91836\ (\text{Base K-eff}) + 0.023\ (\text{KENO-IV Bias}) + \\ & 0.00499\ (\text{Off-Center Penalty}) + \\ & 0.01319\ (\text{Dropped Assembly Penalty}) \\ & + 2[(0.00775\ (\text{Base Uncert.})^2 + \\ & (0.00600(\text{Off-Center Uncert.})^2 + \\ & (0.00595\ (\text{Dropped Assembly Uncert.})^2]^{0.5} \\ = {} & 0.98247\end{aligned}$$

It should be noted that by statistically combining the various calculational uncertainties to obtain an overall 2 sigma uncertainty and then adding this 2 sigma uncertainty to the final result, a 95/95 confidence level is achieved, which is required for criticality analyses [12].

The k-effective calculated above was clearly unacceptable and indicated that it would not be possible to store 3.8 wt% fuel in the SFSR without some kind of administrative control. It was determined that the highest enrichment that could be stored in the SFSR without administrative controls would be approximately 3.36 wt% U-235, which was very close to the value (3.30 wt%) determined in the previous safety evaluation for these racks [2].

The administrative control decided upon was to use a checkerboard pattern which interspersed the fresh 3.8 wt% U-235 fuel assemblies with fuel assemblies of a lower reactivity, i.e., lower initial enrichment and/or a specified amount of burnup. The type of checkerboard pattern used is shown in Figure 5. The fuel assemblies with combinations of initial enrichment and assembly burnup that could be stored adjacent to the fresh 3.8 wt% fuel assemblies while maintaining a final k-effective of less than 0.95 were then determined.

The geometry for the KENO-IV cases for this type of analysis is shown in Figure 6. The KENO-IV cases defined a curve that began at a low enrichment with no burnup extending to 3.8 wt% enrichment with a large burnup. The following table shows that results of these cases:

Checkerboard Enrichments (B/A)	Checkerboard Burnups (B/A)	K-effective	Uncertainty	Final K-effective
3.8/3.8	0/34,114	0.88949	0.00566	0.94619
3.8/3.1	0/25,136	0.89603	0.00565	0.95273
3.8/3.1	0/26,932	0.88599	0.00477	0.94189
3.8/2.5	0/17,955	0.88561	0.00455	0.94130
3.8/1.4	0/0	0.89106	0.00550	0.94760
3.8/water	0/NA	0.87295	0.00654	0.93056

Note: All enrichments are in wt% U-235 and all burnups are in MWD/MTU. See Figure 5 for checkerboard arrangement.

For these cases, since burned fuel assemblies were being modeled, the 13 group NULIF/NUTAN cross section library was used and, consequently, the NULIF/NUTAN reactivity credit and penalty were applied. Also, the burnable poison penalty of 0.006 delta k was added. Using the first case as an example, the final k-effectives were calculated in the following manner:

Final K-effective = 0.88949 (Base K-eff) + 0.023 (KENO-IV Bias)
+ 0.006 (Lumped Burnable Poison)
+ 0.00499 (Off-Center Penalty)
+ 0.01319 (Dropped Assembly Penalty)
+ 0.00200 (NULIF Bias) - 0.01583 (NULIF Bias)
+ $2[(0.00566\ (\text{Base Uncert.}))^2 + (0.00600\ \text{Off Center Uncert.})^2$
+ $(0.00595\ (\text{Dropped Assembly Uncert.}))^2$
+ $(0.00145\ (\text{NULIF Bias Uncert.}))^2$
+ $(0.00555\ (\text{NULIF Bias Uncert.}))^2]^{0.5}$
= 0.94619

It should be noted that the burnable poison penalty was applied to the last two cases, even though no burned fuel was involved, thus making these cases very conservative. The two cases run with 3.1 wt% fuel were used to interpolate to a final k-effective for 3.1 wt% of 0.947, yielding a burnup of 26,085 MWD/MTU. Also, the final case demonstrates that water holes (rack locations with no fuel assemblies) will also serve to hold down k-effective below 0.95.

The use of this type of loading pattern allows the possibility of another type of accident case, that is, accidentally misloading a 3.8 wt% fresh assembly in a location intended for a low reactivity fuel assembly. This possibility was examined using KENO-IV with a 9x9 checkerboard in which the low reactivity fuel assembly (3.8 wt% with 34,144 MWD/MTU of burnup) in the center of the array has been replaced with a fresh 3.8 wt% assembly. This is a conservative approach because, for the Davis-Besse SFSR, this scenario is approximately the equivalent of having 9 misloaded assemblies uniformly spaced in the spent fuel storage racks simultaneously. The results of this case were compared back to the "normal" case above and are shown in the following table:

Geometry	K-effective	Uncertainty
Misloaded	0.89846	0.00433
Normal	0.88949	0.00566

Penalty = 0.00897 delta k plus or minus 0.00713

Since this penalty is smaller than the penalty for the "T-bone" accident, and, since only one accident had to be considered, the "T-bone" accident is bounding and no additional penalty has to be applied for a misloaded assembly accident.

The final uncertainty to be considered was the uncertainty on the burnup of the fuel assemblies themselves. The B&W Nuclear Reliability Factor Topical Report [14] quotes an uncertainty on measured radial power distributions of 5 percent. Since the uncertainty on assembly burnup would be directly proportional to the uncertainty in the radial power distribution, a burnup uncertainty penalty of 5 percent of the total accumulated burnup was considered. In the most extreme case, which is burned fuel with an initial enrichment of 3.8 wt% U-235, this penalty amounts to 1,706 MWD/MTU. However, from the two cases run above that each use burned 3.1 wt% fuel, it was possible to quantify the effect upon reactivity that a given amount of burnup contributes. From these two cases, it was shown that a change in burnup of 1,789 MWD/MTU in the irradiated fuel assemblies would cause a one percent change in the reactivity of the array. Since it would bound the worst-case 5 percent uncertainty (1,704 MWD/MTU for 3.8 wt% fuel), a 1,800 MWD/MTU penalty was added to the assembly burnups shown in the above table.

A quadratic least squares fit was performed to obtain the equation of the line shown in Figure 7. The largest negative absolute error of the equation, when compared to the fitting points, was 18 MWD/MTU, and this quantity was added to the intercept term to produce an equation which conservatively bounded all of the points from which it was fitted. The equation for the line is:

$$B = -26640 + 22584 \times E - 1610 \times E^2$$

where B is assembly burnup in MWD/MTU and E is the initial fuel enrichment in wt% uranium-235.

Figure 7 defines three types of fuel. Category "A" fuel assemblies, which are above the burnup-enrichment line, may be stored in any location in the SFSR. Category "B" fuel assemblies, which lie below the burnup-enrichment line but have initial enrichments greater than 3.3 wt% U-235, may be stored only in locations that are directly adjacent to locations containing either Category "A" assemblies, water holes, or some combination or Category "A" assemblies and water holes. Category "C" fuel assemblies, which also lie below the burnup-enrichment line but instead have initial enrichments of less than 3.3 wt% U-235, may be stored anywhere in the SFSR except that they may not be stored adjacent to locations containing Category "B" fuel assemblies. It is possible to define Category "C" fuel by noting that the SFSR were previously licensed to 3.3 wt% U-235 without taking credit for burnup [2]. As was noted above, the new analysis verified the results of the previous analysis with respect to the upper limit of 3.3 wt% when no burnup credit is taken [2]. Category "B" fuel assemblies may not be stored directly adjacent to each other or directly adjacent to Category "C" fuel in the SFSR. Enrichments higher than 3.8 wt% U-235 will not be permitted.

SIGNIFICANT HAZARDS ANALYSIS

The Commission has provided standards in 10CFR50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an Operating License for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences 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 Company has reviewed the proposed change and determined that:

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated because there is no increase in the probability of a dropped fuel assembly accident since the mass of the fuel assembly does not increase when the fuel enrichment is increased. The fission product inventories in the fuel assemblies would not change significantly due to an increase in the fuel enrichment. Any change to the nuclear properties of the reactor core and any changes in fission product inventories that may result from the higher fuel burnups due to an increase in the fuel enrichment would be analyzed in the appropriate reload analysis for the fuel cycle in which the higher enrichment fuel batch was introduced. (10CFR50.92(c)(1))

The proposed amendment does not create the possibility of a new or different kind of accident than previously evaluated because the only possible accident that could be created through an increased fuel enrichment would be a criticality accident, which is already addressed in the USAR in the design bases for the NFSR and SFSR. If any new type of accident or condition could be created in the reactor core due to the increase in U-235 enrichment, that accident or condition would be analyzed as part of the reload analysis for the fuel cycle in which the higher enrichment fuel batch was introduced. (10CFR50.92(c)(2))

The proposed amendment does not involve a significant reduction in a margin of safety because the criticality analysis demonstrates that both the NFSR and SFSR will always be at least five (5) percent subcritical under flooded conditions even when credible accidents are accounted for. (10CFR50.92(c)(3))

Based on the above reasoning, Toledo Edison has determined that the proposed amendment does not involve a significant hazards consideration.

REFERENCES

[1] Babcock And Wilcox Document No. 86-1170592-00, "Davis-Besse Unit 1 New and Spent Fuel Storage Rack Criticality Analysis", October 2, 1987.

[2] FCR 77-290, "Safety Evaluation of the Spent Fuel Storage Capacity Modification for Davis-Besse Nuclear Power Station Unit 1", December 5, 1977.

[3] Babcock And Wilcox Proposal No. A4-360, July 3, 1987.

[4] Davis-Besse Nuclear Power Station No. 1 Updated Safety Analysis Report, latest revision 5, 7/87.

[5] Davis-Besse Nuclear Power Station Unit 1 Technical Specifications, Appendix "A" To License No. NPF-3, April 22, 1977, last amendment 104.

[6] USNRC Standard Review Plan Section 9.1.1, New Fuel Storage, Rev. 2, July 1981.

[7] USNRC Standard Review Plan Section 9.1.2, Spent Fuel Storage, Rev. 3, July 1981.

[8] USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Revision 1, December 1975.

[9] ANSI/ANS-57.2-1983, Design Requirements For Light Water Reactor Spent Fuel Storage Facilities At Nuclear Power Plants.

[10] ANSI/ANS-57.3-1983, Design Requirements For New Fuel Storage Facilities At Light Water Reactor Plants.

[11] ANSI/ANS-57.9-1984, Design Criteria For An Independent Spent Fuel Storage Installation (Dry Storage Type).

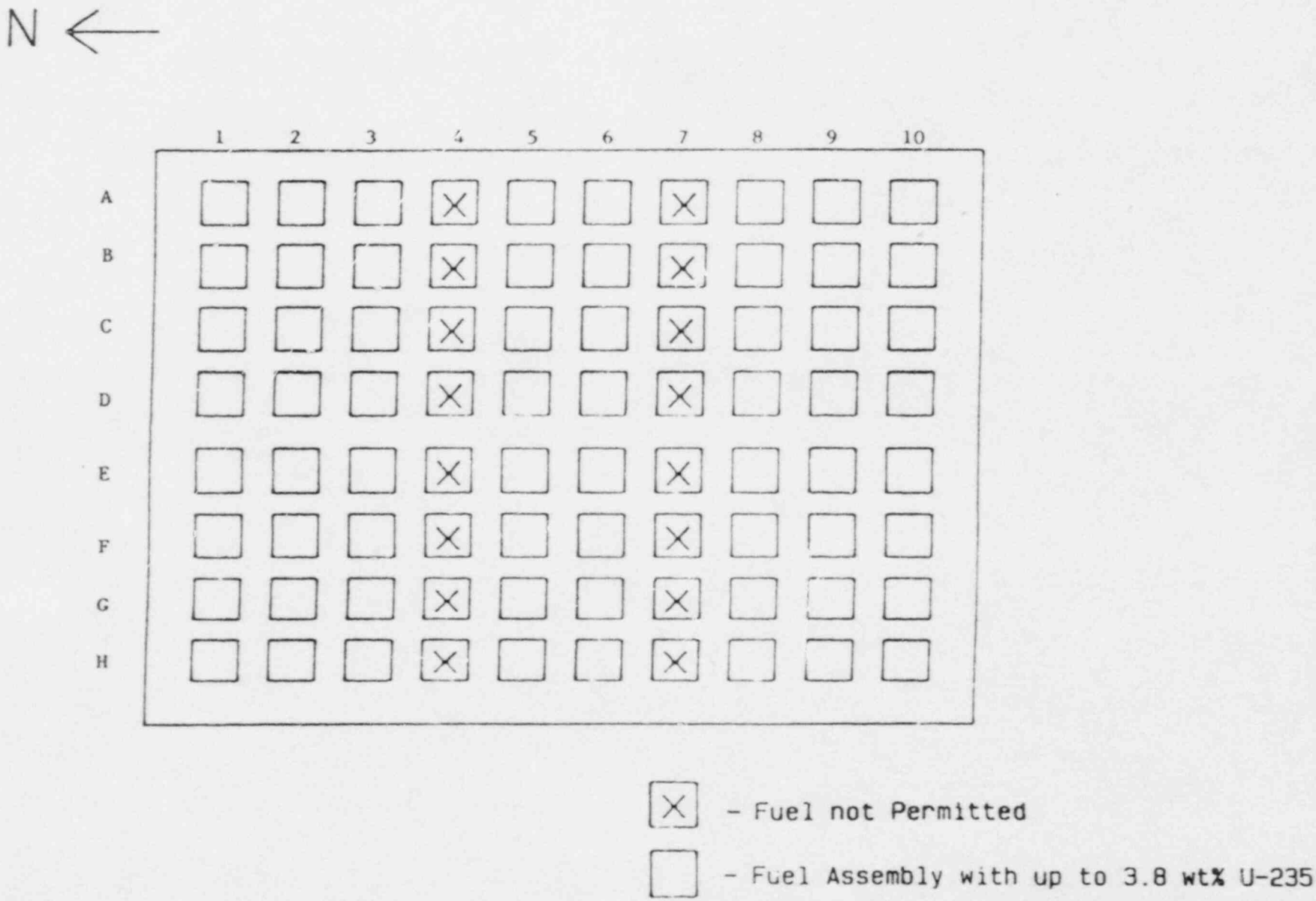
[12] ANSI N16.9-1976, Validation Of Calculational Methods For Nuclear Criticality Safety.

[13] Babcock And Wilcox Topical Report No. BAW-1484-7, "Critical Experiments Supporting Close Proximity Water Storage Of Power Reactor Fuel", July 1979.

[14] Babcock And Wilcox Topical Report No. BAW-10119P-A, "Power Peaking Nuclear Reliability Factors", February 1979.

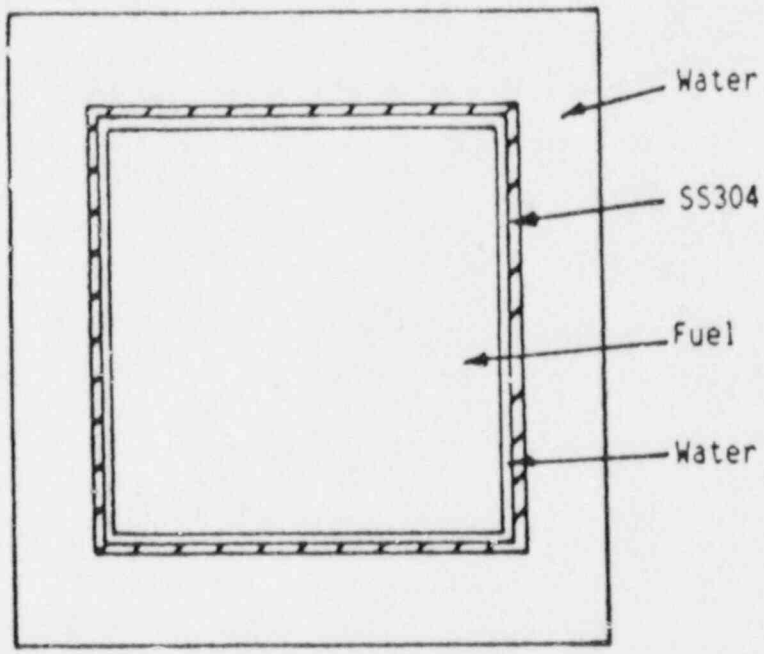
[15] Babcock And Wilcox Specification 08-1116-06, "Pellet Type Zircaloy Clad Fuel", April 26, 1983.

Figure 1
New Fuel Storage Rack
Loading Geometry



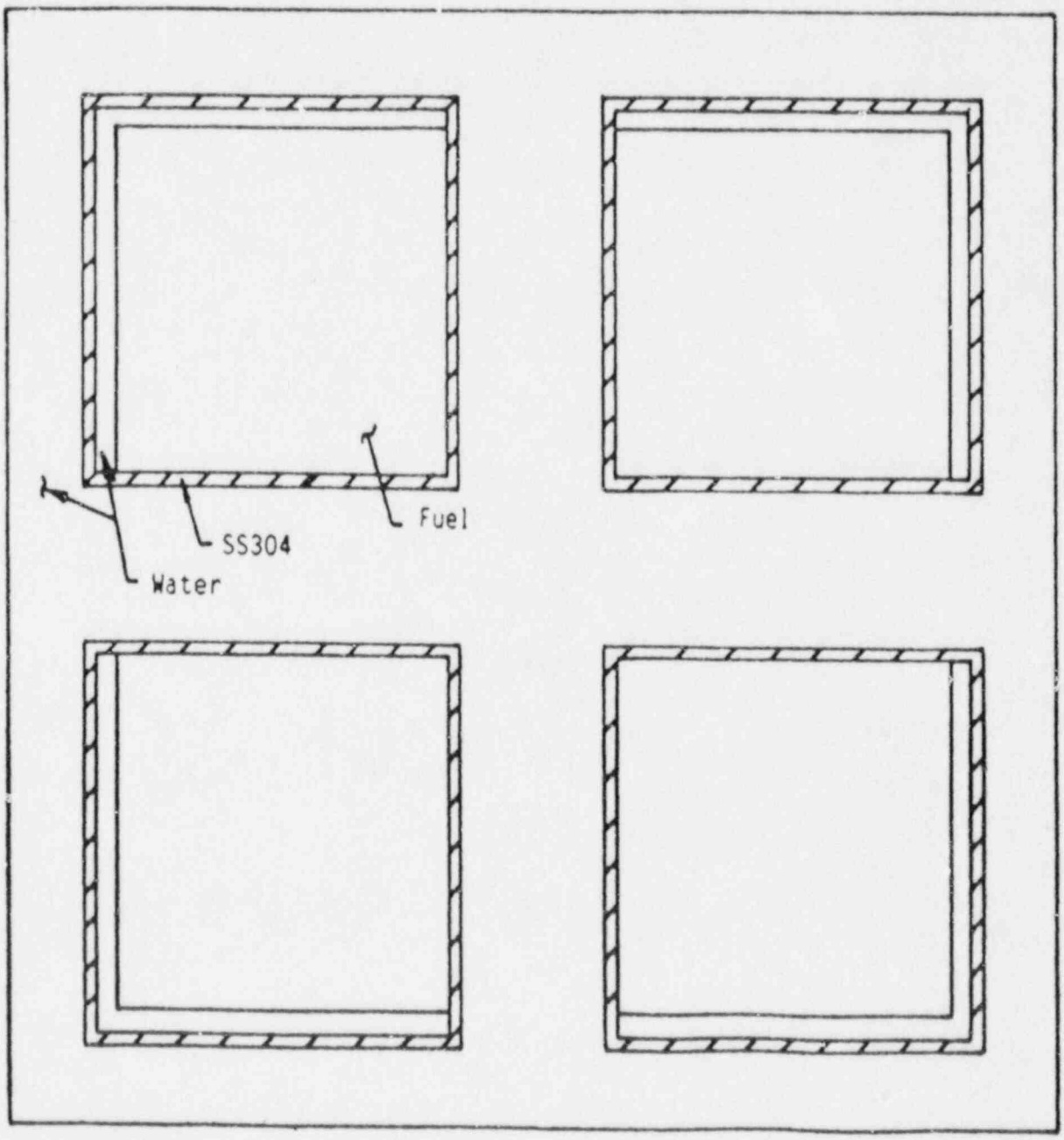
Docket No. 50-346
License No. NPF-3
Serial No. 1479
Attachement 2
Figure 2

Figure 2
Basic Fuel Assembly Representation in the Spent Fuel Pool Storage Rack



Docket No. 50-346
License No. NPF-3
Serial No. 1479
Attachement 2
Figure 3

Figure 3
Off-Centered Assembly Representation in the Spent Fuel Pool Storage Rack



Docket No. 50-346
License No. NPF-3
Serial No. 1479
Attachement 2
Figure 4

Figure 4
T-Bone Accident Representation On the Spent Fuel Pool Rack

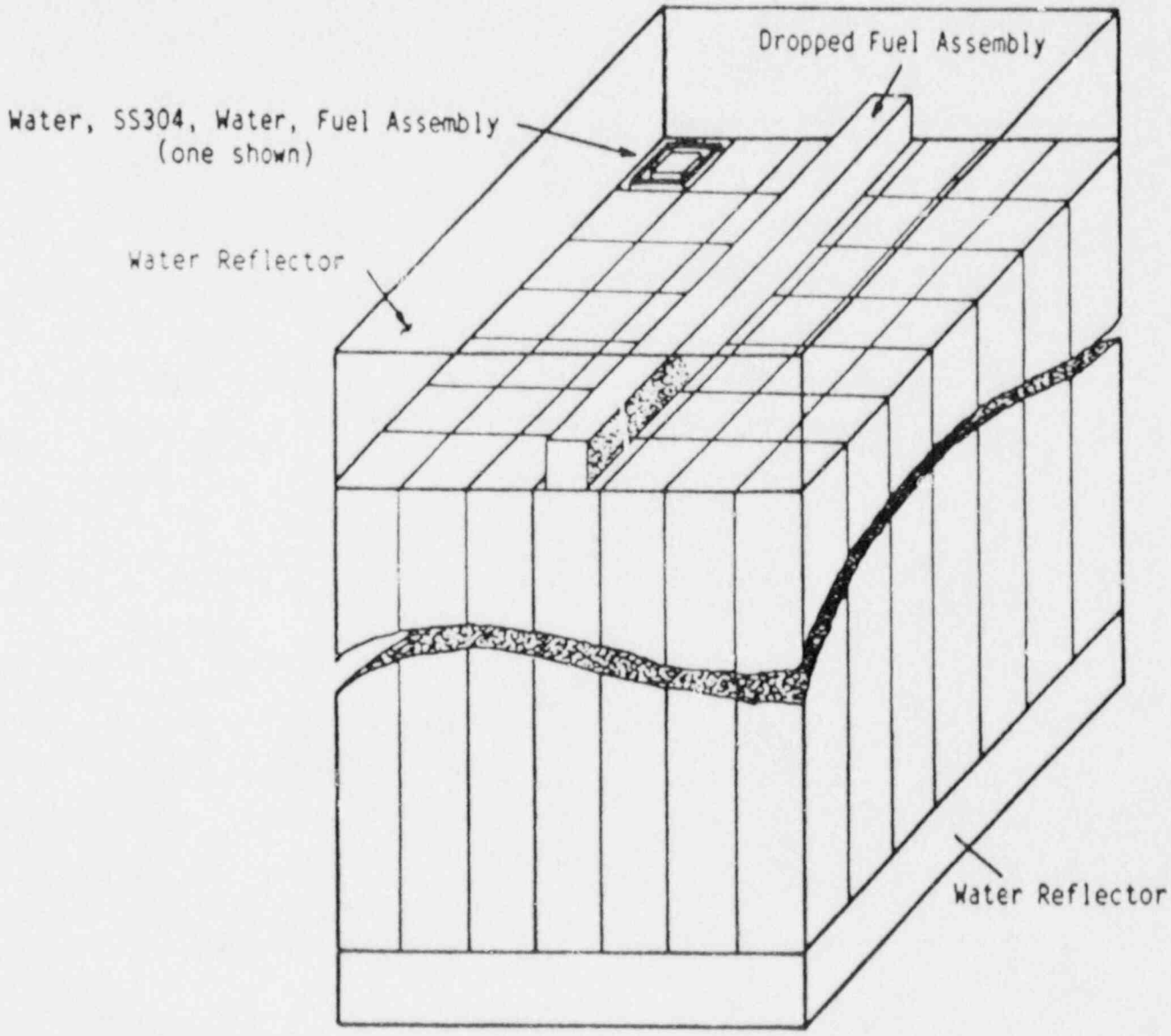


Figure 5
Spent Fuel Pool Checkerboard Pattern

A	B	A	B	A
B	A	B	A	B
A	B	A	B	A
B	A	B	A	B
A	B	A	B	A

A - Low Reactivity Assembly
B - 3.8 wt% U-235, 0 Burnup

Docket No. 50-346
License No. NPF-3
Serial No. 1479
Attachement 2
Figure 6

Figure 6
KENO-IV Geometry for Checkerboard Spent Fuel Storage Rack

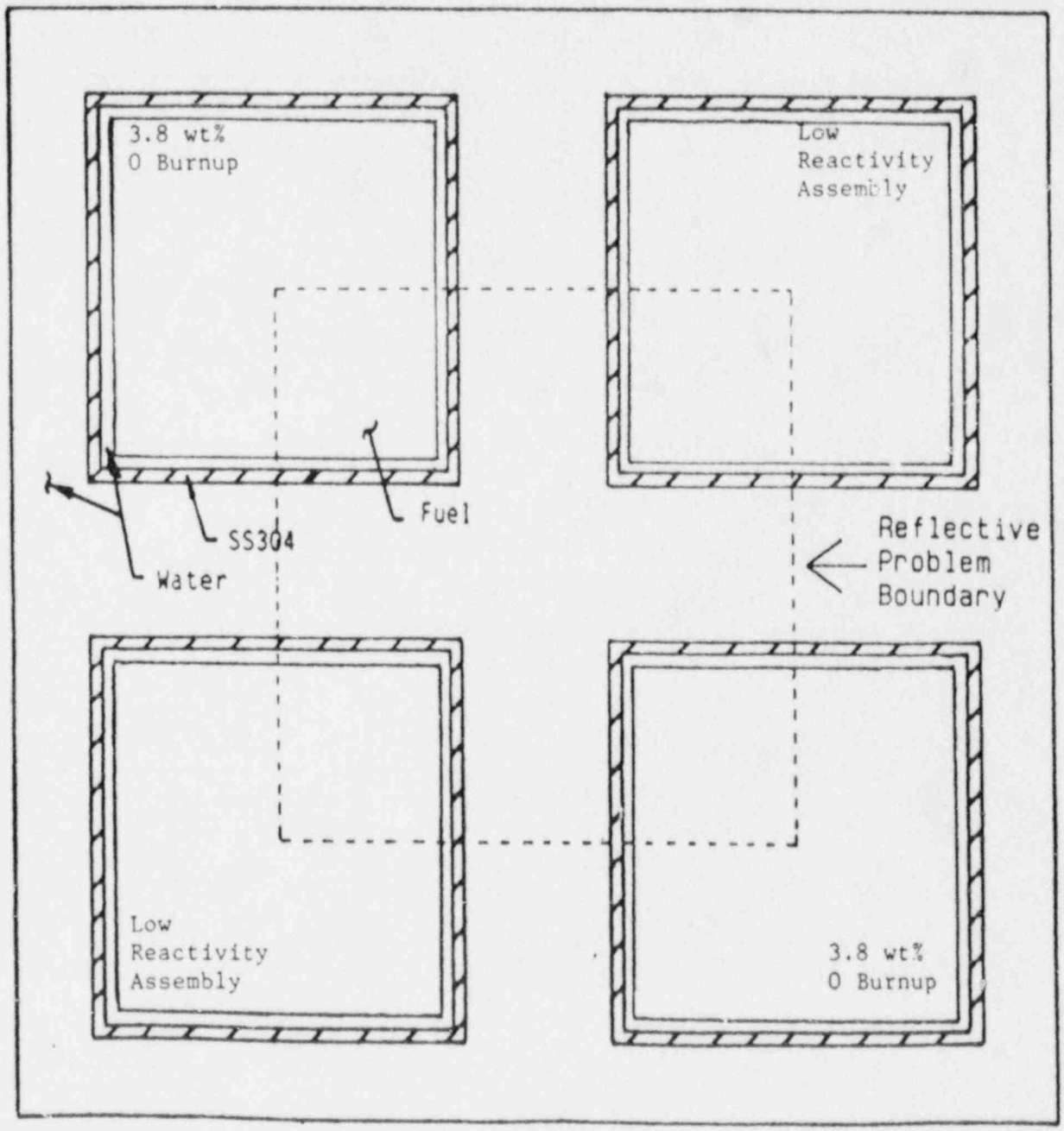
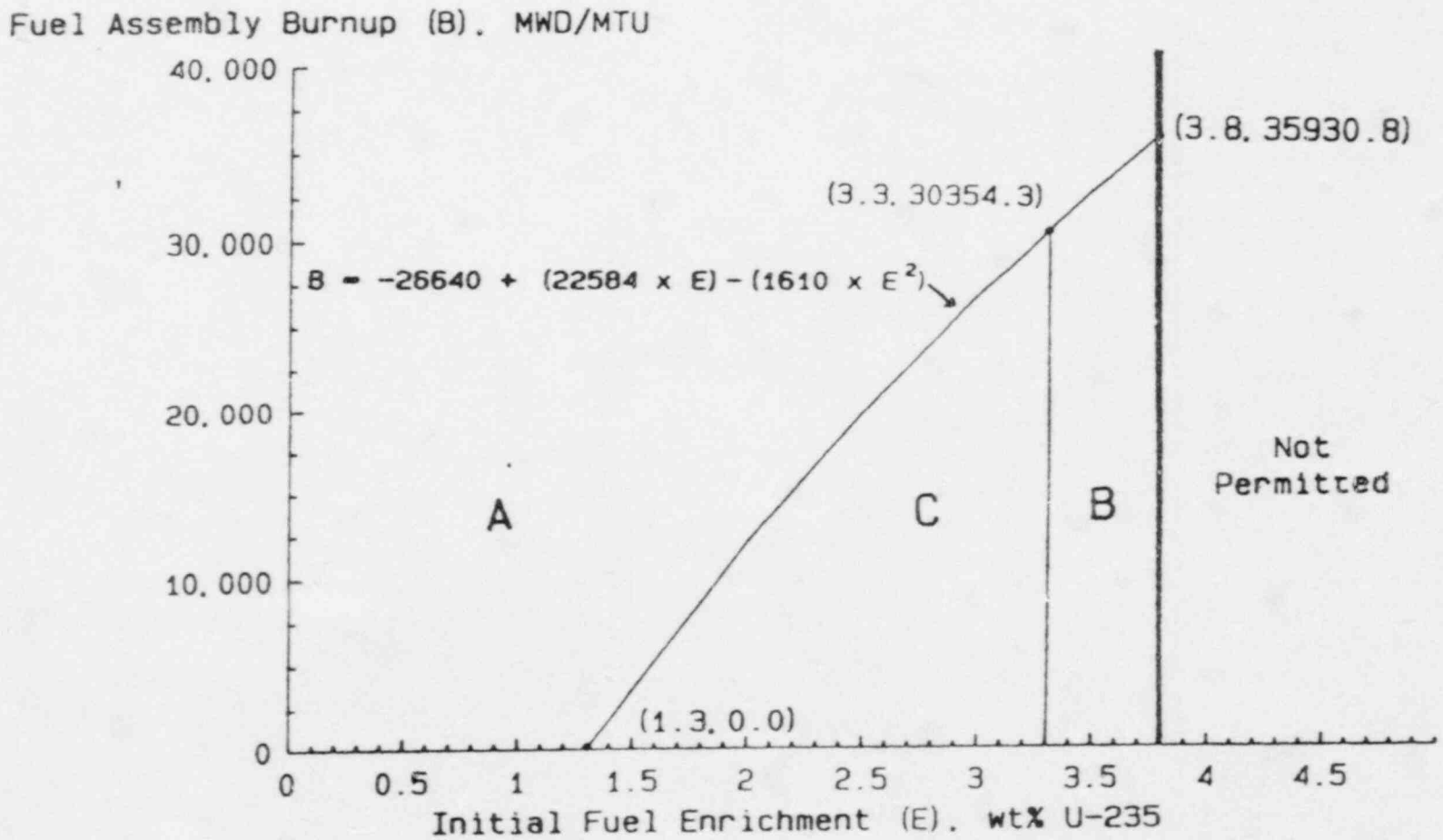


Figure 7
Burnup vs. Enrichment Curve for Davis-Besse Spent Fuel Storage Racks



Category "A" Fuel - May be located anywhere within the storage racks

Category "B" Fuel - Shall only be located adjacent to Category "A" Fuel or water holes within the storage racks

Category "C" Fuel - Shall not be located adjacent to Category "B" Fuel