# APPENDIX 9.1A

# FUEL STORAGE RACK ANALYSIS

## 9.1A.1 THE HIGH DENSITY RACK (HDR) DESIGN CONCEPT

### 9.1A.1.1 Introduction

Historically, spent fuel rack designs have been based on conservative assumptions that could easily be accommodated since it was not planned to store large numbers of high exposure spent fuel assemblies on-site. Previously it was anticipated that only small amounts of high exposure fuel assemblies (1/4 to 1/2 of a full core load) would normally be stored in the spent fuel pool at any one time. Additionally it was anticipated that occasionally (e.g., for inservice inspection of the reactor vessel internals) the entire core would be unloaded and temporarily stored in the initial spent fuel pool. Therefore, the spent fuel storage rack design was based on the conservative assumption that all fuel rack storage positions would be occupied by fresh unirradiated fuel assemblies of the highest initial enrichment that was foreseen as being usable in that facility.

The penalty in achievable fuel storage density associated with this conservative design assumption was relatively small under the circumstances anticipated and easily accommodated by a conservative fuel rack design. The potential penalty associated with this conservative design basis is no longer small when long-term on-site storage of spent fuel is a necessity.

There is no situation where more than one full core load of fresh unirradiated fuel assemblies is to be stored in the fuel storage pool. Therefore, it is unnecessary and wasteful to base the entire fuel storage rack design on the assumption of fresh unirradiated fuel of the highest initial enrichment.

In the previous maximum density rack (MDR) design concept utilized by Wolf Creek, the spent fuel pool was divided into two separate and distinct regions which, for the purpose of criticality considerations may be considered as separate pools. Suitability of this design assumption regarding pool separability was assured through appropriate design restrictions at the boundaries between Region 1 and Region 2. The smaller region, Region 1, of the pool was designed on the basis of accepted conservative criteria which allowed for the safe storage of a number of fresh unirradiated fuel assemblies (including a full core loading if that should prove necessary). The larger region of the pool, Region 2, was designed to store irradiated fuel assemblies. The change in criteria was the recognition of actual fuel and fission product inventory, accompanied by a system for checking fuel prior to moving any fuel assembly from Region 1 to Region 2.

In the HDR design concept, currently utilized by Wolf Creek, the rack modules for the fuel storage pool are designed for storage of both new fuel and spent fuel. Spent fuel storage is designated into Regions based upon initial enrichment and accumulated burnup. (b)(4)

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9.1A.1.2 Design Bases

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Moderator is unborated water at a temperature that results in the highest reactivity (4°C, corresponding to the maximum possible moderator density).

No soluble poison or control rods are assumed to be present for normal operations, although the additional margin due to the presence of soluble boron is identified.

The effective multiplication factor of an infinite radial array of fuel assemblies was used except for the assessment of peripheral effects and certain abnormal/accident conditions where neutron leakage is inherent.

Neutron absorption in minor structural members is conservatively neglected, i.e., spacer grids are replaced by water.

Depletion calculations assume conservative operating conditions; highest fuel and moderator temperature and an allowance for the soluble boron concentrations during incore operation.

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9.1A.1.3 Design Description

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9.1A.2.2.3.1 Fuel Burnup Calculations and Uncertainties

CASMO-3 was used for burnup calculations in the hot operating condition. CASMO-3 has been extensively benchmarked (References 7 and 9) against cold, clean, critical experiments (including plutonium-bearing fuel), Monte Carlo calculations, reactor operations, and heavy element concentrations in irradiated fuel. In addition to burnup calculations, CASMO-3 was used for evaluating the small reactivity increments (by differential calculations) associated with manufacturing tolerances and for determining temperature effects.

In the CASMO-3 geometric model, each fuel rod and its cladding were described explicitly and reflective boundary conditions were used at the centerline of the Boral and steel plates between storage cells. These boundary conditions have the effect of creating an infinite array of storage cells in the X-Y plane and provide a conservative estimate of the uncertainties in reactivity attributed to manufacturing tolerances.

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9.1A.2.2.3.2 <u>Effect of Axial Burnup Distribution</u>

Initially, fuel loaded into the reactor will burn with a slightly skewed cosine power distribution. As burnup progresses, the burnup distribution will tend to flatten, becoming more highly burned in the central regions than in the upper and lower regions. At high burnup, the more reactive fuel near the ends of the fuel assembly (less than average burnup) occurs in regions of high neutron leakage. Consequently, it is expected that over most of the burnup history, fuel assemblies with distributed burnups will exhibit a slightly lower reactivity than that calculated for the uniform average burnup. As burnup progresses, the distribution, to some extent, tends to be self-regulating as controlled by the axial power distribution, precluding the existence of large regions of significantly reduced burnup. Among others, Turner (Reference 10) has provided generic analytic results of the axial burnup effect based upon calculated and measured axial burnup distributions. These analyses confirm the minor and generally negative reactivity effect of the axially distributed burnup.

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9.1A.2.2.4 Criticality Analyses Uncertainties and Tolerances

A number of tolerances result in reactivity uncertainties which must be considered in the criticality analyses.

9.1A.2.2.4.1 Nominal Design

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9.1A.2.2.4.2 Uncertainties Due to Manufacturing Tolerances

The uncertainties due to manufacturing tolerances are summarized in Table 9.1A-5 and discussed below.

9.1A.2.2.4.2.1 Boron Loading Tolerances

The Boral absorber panels are manufactured with a tolerance limit in B-10 content which assures that at any point, the minimum B-10 areal density will not be less than (b)(4) (b)(4)

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9.1A.2.2.4.2.2 <u>Boral Width Tolerance</u>

The differential (b)(4) calculated reactivity uncertainty is ±0.0010 Δk, when the reference storage cell design has the minimum tolerance for Boral panel thickness.

9.1A.2.2.4.2.3 <u>Tolerances in Cell Lattice Spacing</u>

The differential (b)(4) calculations determine an uncertainty of ±0.0016 Δk in the calculated reactivity when the minimum manufacturing tolerance on the inner box dimension is used. The minimum manufacturing tolerance on the inner box dimension directly affects the storage cell lattice spacing between fuel assemblies.

9.1A.2.2.4.2.4 <u>Stainless Steel Thickness Tolerances</u>

The nominal stainless steel thickness for the stainless steel box also has an impact on the calculation of reactivity. The reactivity uncertainty of the expected stainless steel thickness tolerances was calculated with (b)(4) (b)(4)

9.1A.2.2.4.2.5 <u>Fuel Enrichment and Density Tolerances</u>

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9.1A.2.2.4.3 <u>Water-Gap Spacing Between Modules</u>

The water-gap between modules, (b)(4), constitutes a neutron flux-trap for the storage cells of facing racks. (b)(4) (b)(4) were made with the reference (b)(4) to determine the uncertainty associated with a water-gap tolerance. Due to the asymmetries in the MZTR pool configuration, the effect of the horizontal and vertical water gaps (see Figure 9.1A-1) were calculated separately. (b)(4) (b)(4) (b)(4) The racks are constructed with the base plate extending beyond the edge of the cells which assures that the minimum spacing between storage modules is maintained under all credible conditions.

9.1A.2.2.4.4 <u>Eccentric Fuel Positioning</u>

The fuel assembly is assumed to be centered in the storage rack cell. Calculations were made using KENO5a assuming the fuel assemblies were located in the corners of the storage rack cells (four-assembly clusters at the closest possible approach). These calculations indicated that the reactivity effect is small and negative. Therefore, the reference case in which the fuel assemblies are centered is controlling and no uncertainty for eccentricity is necessary.

9.1A.2.2.5 Abnormal and Accident Conditions

The reactivity effects of abnormal and accident conditions are summarized in Table 9.1A-3.

9.1A.2.2.5.1 Temperature and Water Density Effects

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9.1A.2.2.5.2 Lateral Rack Movement

The possibility of reductions in the rack-to-rack gaps and the resulting criticality consequences have also been reviewed. Criticality evaluations are sensitive to these gap dimensions, since the inter-rack gaps provide a flux trap which reduces the reactivity. Rack-to-rack gap reductions are a concern subsequent to dynamic events which are severe enough to displace the racks laterally or produce fuel-to-rack cell wall impacts of sufficient magnitude to exceed cell wall material yield strength (i.e., produce plastic deformation).

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9.1A.2.2.5.5 Dropped Fuel Assembly

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9.1A.2.2.6 Benchmark Calculations

The methodologies for determining criticality safety have been verified by comparison with critical experiment data for configurations that impose a stringent test of the capability of the analytical methodologies. These benchmark calculations have been made on selected critical experiment, chosen, in so far as possible to bound the range of variables in fuel storage rack designs, including the Wolf Creek high density racks.

9.1A.2.2.6.1 Summary

Two independent methods of analysis were used in performing the Wolf Creek fuel storage rack criticality safety analyses. These two methods differ in cross section libraries and in the treatment of the cross sections. (b)(4) 4
(b)(4) is a continuous energy Monte Carlo code and (b)(4)
(b)( uses group-dependent cross sections. For the (b)(4) analyses reported here, the 238-group library was chosen, processed through the NITAWL-II (b)(4) program to create a working library and to account for resonance self-shielding in uranium-238 (Nordheim integral treatment). The 238-group library was chosen to avoid or minimize the errors (trends) that have been reported (e.g., References 19-21 ) for calculations with collapsed cross section sets. Small but observable trends (errors) have been reported for calculations with the (b)(4) These errors are probably due to the use of a single collapsing spectrum when the spectrum should be different for the various cases analyzed, as evidenced by the spectrum indices.

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Table 9.1A-7 summarizes results of the benchmark calculations for all cases selected and analyzed, as referenced in the table. The effect of the major variables are discussed in subsequent sections below. It is important to note that there is obviously considerable overlap in parameters since it is not possible to vary a single parameter and maintain criticality; some other parameter or parameters must be concurrently varied to maintain criticality.

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Figures 9.1A-7 and 9.1A-8 show the calculated $k_{eff}$ for the benchmark critical (b)(4) respectively ($UO_2$ fuel only). The scatter in the data (even for comparatively minor variation in critical parameters) represents experimental error in performing the critical experiments within each laboratory, as well as between the various testing laboratories. A classical example of experimental error is the corrected enrichment in the PNL experiments, first as an addendum to the initial report and, secondly, by revised values in subsequent reports for the same fuel rods. The B&W critical experiments show a larger experimental error than the PNL criticals. This would be expected since the B&W criticals encompass a greater range of critical parameters than the PNL criticals.

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9.1A.2.2.6.4 Miscellaneous and Minor Parameters

9.1A.2.2.6.4.1 Reflector Material and Spacings

PNL has performed a number of critical experiments with thick steel and lead reflectors. Analysis of these critical experiments are listed in Table 9.1A-11 (subset of data in Table 9.1A-7). There appears to be a small tendency toward over prediction of $k_{eff}$ at the lower spacing, although there are an insufficient number of data points in each series to allow a quantitative determination of any trends. The tendency toward over prediction at close spacing means that the rack calculations may be slightly more conservative than otherwise.

9.1A.2.2.6.4.2 Fuel Pellet Diameter and Lattice Pitch

The critical experiments selected for analysis cover a range of fuel pellet (b)(4) 4

(b)(4) for BWR fuel. Thus, the critical experiments analyzed provide a reasonable representation of power reactor fuel. Based on the data in Table 9.1A-7, there does not appear to be any observable trend with either fuel pellet diameter or lattice pitch, at least over the range of the critical experiments applicable to rack designs.

9.1A.2.2.6.4.3 Soluble Boron Concentration Effects

Various soluble boron concentrations were used in the B&W series of critical experiments and in one PNL experiment, with boron concentrations ranging up to (b)(4)

show a tendency to slightly over predict reactivity for the three experiments (b)(4) In turn, this would suggest that the evaluation of the racks with higher soluble boron concentrations could be slightly conservative.

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9.1A.3 THERMAL AND HYDRAULIC ANALYSES

The Wolf Creek reracked fuel storage pool (spent fuel pool and cask loading pool with fuel storage racks installed) and the Fuel Pool Cooling and Cleanup System (FPCCS) comply with the provisions of Section III of the USNRC "OT Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications", (April 14, 1978). The methods, models, analyses, and numerical results are summarized below.

The thermal-hydraulic qualification analyses for the rack arrays fall into the following categories:

1. Evaluation of the maximum decay heat load limit as a function of the bulk temperature limit for the postulated discharge scenario.

2. Evaluation of the postulated loss-of-forced cooling scenarios to establish that pool boiling will not occur.

3. Determination of the maximum temperature difference between the pool local temperature and the bulk pool temperature at the instant when the bulk temperature reaches its maximum value.

4. Evaluation of the maximum temperature difference between the fuel rod cladding temperature and the local pool water temperature to establish that nucleate boiling at any location around the fuel is not possible with forced cooling available.

Because the thermal and hydraulic analyses bound both the Callaway and Wolf Creek fuel storage pools, the pool water volume is conservatively based on the minimum east-west and north-south dimensions of the two pools. This conservatism results in a lower bound thermal inertia and outer periphery downcomer dimension in the thermal-hydraulic calculations.

FPCCS at Wolf Creek is described in Section 9.1.3. (b)(4) (b)(4) (b)(4) The decay heat generated by the stored fuel in the pool is transferred from the fuel pool cooling system (b)(4) Normal makeup water to the fuel storage pool is supplied by the reactor makeup water system. An alternate source of makeup water is the RWST via the fuel pool cleanup pumps. Emergency makeup water is supplied from the Essential Service Water system. Boron addition to the fuel storage pool is normally accomplished by supplying borated water from the boric acid tanks via the boric acid blending tee. Boron may also be added by using the RWST as the source of makeup water to the fuel storage pool. Isolation of non-safety related portions of the FPCCS is a manual action.

The fuel pool cleanup system provides the capability for purification of the water in the spent fuel pool, the cask loading pool, the transfer canal, the refueling pool, and the RWST. The cleanup system is an essential adjunct to the FPCCS system to maintain clarity and water chemistry control in the fuel storage pool.

Consistent with the current plant practice, two discharge scenarios are postulated when considering fuel storage pool cooling:

i. partial core offload

ii. full-core offload

In lieu of prescribing a batch size and cooling period for the partial core offload, the maximum pool heat load is determined for a scenario of only one cooling train operating and a limit on the steady state bulk pool temperature (b)(4)

Similarly, the full core offload scenario is required to be executed so that the maximum pool heat load will not allow for bulk pool boiling at the (b)(4) (b)(4) (b)(4) More specifically, the bulk water temperature is (b)(4) after two hours of pool heat-up in the absence of all forced cooling paths.

Evaluation of these two scenarios provides maximum flexibility in batch sizes and cooling periods prior to offload into the pool. In both scenarios, the component cooling water (CCW), used to remove heat from the fuel pool cooler, is assumed to be at its maximum design temperature. During the partial core offload scenario CCW flow is assumed to be at its nominal rate. During full core offload conditions, CCW flow is assumed to be at its design basis flow rate. With the thermal effectiveness of the fuel pool cooler thus fixed, the requirement of the ceiling on the bulk pool temperature essentially translates into a limit on the total heat generation rate in the pool.

Finally an evaluation is performed for a loss of cooling accident occurring some time after restart. This evaluation considers a four hour long loss of forced cooling in the FPCCS followed by a twenty hour long period with cooling provided at one-half the normal coolant flow rate. Under this scenario, the fuel pool does not reach the bulk boiling temperature (b)(4) For this evaluation, the component cooling water to the heat exchanger is assumed to be at an elevated temperature and reduced flow rate.

### 9.1A.3.1 Decay Heat Load Limit

The heat load imposed on the pool is from the decay heat generated by fuel assemblies discharged into the pool. The primary safety function of the FPCCS is to adequately transport this heat load to the CCW system and thereby maintain the bulk pool temperature within specified limits. Compliance with the limiting heat load will be ensured through adjustments to the cooling system performance and/or adjustments to the fuel offload rate. Commonly used decay heat calculation methods based upon ASB 9-2, ANS 5. 1, or ORIGEN2 are used to provide conservative estimates of decay heat values for specific fuel pool inventories.

#### 9.1A.3.1.1 Decay Heat Load Calculations and Conservatisms

The following conservatisms are applied in the decay heat load limit calculations.

- FPCCS heat exchanger thermal performance is based on the design maximum fouling and plugging level. This will conservatively minimize the heat rejection capability of the FPCCS.

- Thermal inertia induced transient effects resulting in a lag in bulk pool temperature response are neglected. This conservatively lowers the calculated decay heat load limit by forcing the peak decay heat load to coincide with the peak pool temperature.

- In calculating the fuel storage pool evaporation heat losses, the building housing the fuel storage pool (b)(4) (b)(4) humidity. This minimizes the evaporative heat loss component, maximizing the heat duty burden on the pool cooling system.

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### 9.1A.3.3 Local Pool Water Temperature

A single conservative evaluation for a bounding amalgam of conditions was performed to evaluate the local pool water temperature. The result of the single evaluation is a bounding temperature difference between the maximum local water temperature and the bulk pool temperature.

In order to determine an upper bound on the maximum local water temperature, a series of conservative assumptions are made. The most important of these assumptions are:

- With a full core discharged into the racks farthest from the coolant water inlet, the remaining cells in the spent fuel pool are postulated to be occupied with previously discharged fuel.

- The hottest assemblies, located together in the pool, are assumed to be located in "pedestal" cells of the racks. These cells have a reduced water entrance area, caused by the pedestal blocking the baseplate hole, and a correspondingly increased hydraulic resistance.

- The coolant water inlet temperature, and therefore the bulk pool temperature, is minimized to conservatively maximize the fluid viscosity. This assumption will maximize the head losses for water flowing through the fuel racks and fuel assemblies.

- No downcomer flow is assumed to exist between the rack modules.

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9.1 A.3.3. 1 <u>Local Temperature Evaluation Methodology</u>

The inlet piping which returns cooled pool water from the FPCCS terminates above the level of the fuel racks. To demonstrate adequate cooling of hot fuel in the pool, it is necessary to rigorously quantify the velocity field in the pool created by the interaction of buoyancy driven flows and water injection/egress. A Computational Fluid Dynamics (CFD) analysis for this demonstration is required. The objective of this study is to demonstrate that the principal thermal-hydraulic criteria of ensuring local subcooled conditions in the pool is met for all postulated fuel discharge/cooling alignment scenarios. The local thermal-hydraulic analysis is performed such that partial cell blockage and slight fuel assembly variations are bounded. An outline of the CFD approach is described in the following.

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The fuel storage pool geometry required an adequate portrayal of large scale and small scale features, spatially distributed heat sources in the fuel storage racks and water inlet/outlet configuration. Relatively cooler bulk pool water normally flows down between the fuel rack outline and pool wall liner clearance known as the downcomer. Near the bottom of the racks, the flow turns from a vertical to horizontal direction into the bottom plenum supplying cooling water to the rack cells. Heated water issuing out of the top of the racks mixes with the bulk pool water. An adequate modeling of these features on the CFD program involves meshing the large scale bulk pool region and small scale downcomer and bottom plenum regions with sufficient number of computational cells to capture the bulk and local features of the flow field.

The distributed heat sources in the fuel storage pool racks are modeled by identifying distinct heat generation zones considering full-core discharge, bounding peak effects, and presence of background decay heat from old discharges. Three heat generating zones were modeled. The first consists of background fuel from previous discharges, the remaining two zones consist of fuel from a bounding full-core-discharge scenario. The two full core discharge zones are differentiated by one zone with higher than average decay heat generation and the other with less than average decay heat generation. The background decay heat load is determined such that the total decay heat load in the pool is equal to the calculated decay heat load limit. This is a conservative model, since all of the fuel with higher than average decay heat is placed in a contiguous area. A uniformly distributed heat generation rate was applied throughout each distinct zone.

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9.1A.3.3.2 Local Water and Fuel Cladding Temperatures

Consistent with the approach to make conservative assessments of temperature, the local water temperature calculations are performed for a pool with decay heat generation equal to the maximum calculated decay heat load limit. Thus, the local water temperature evaluation is a calculation of the temperature increment over the theoretical spatially uniform value due to local hot spots (due to the presence of a highly heat emissive fuel bundle).

The CFD study has analyzed a single bounding local thermal-hydraulic scenario. In this scenario, a bounding full-core discharge is considered in which the 193 assemblies are located in the pool, farthest from the cooled water inlet, while the balance of the rack cells are postulated to be occupied by fuel from old discharges.

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### 9.1A.3.5 Decay Heat Load Limits

The calculated decay heat load limit is summarized in Table 9.1A-17. Because all transient effects were excluded from the evaluations, this decay heat load corresponds to the invariant heat load which results in a steady-state bulk pool temperature which will not exceed the temperature limit for either the partial core or full core offload scenario.

This calculated decay heat load limit is not based on any specific discharge conditions, but is a mathematically derived quantity. Any conservative decay heat calculation used to determine the operational limits (i.e. in-core hold time requirement) necessary to avoid exceeding this decay heat load provides conservative operational limits. The operational limits are determined based on the decay heat load limit in Table 9.1A-17. Based on this limit, the fuel storage pool cooling system will remain in compliance.

## 9.1A.4 STRUCTURAL AND SEISMIC CONSIDERATIONS

The structural adequacy of the high density spent fuel racks are considered under all loadings postulated for normal, seismic, and accident conditions. The fuel storage racks must remain fully functional during and after a seismic disturbance. The seismic adequacy is demonstrated in response to both a Safe Shutdown Earthquake (SSE) and the Operational Design Basis Earthquake (OBE). The analyses undertaken to confirm the structural integrity of the racks are performed in compliance with the USNRC Standard Review Plan (Reference 42) and the OT Position Paper (Reference 43).

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(b)(4) Some unique attributes of rack dynamic behavior include a large fraction of the total structural mass in a confined rattling motion, friction support of rack pedestals against lateral motion, and large fluid coupling effects due to deep submergence and independent motion of closely spaced adjacent structures. Whole Pool Multi-Rack (WPMR) analysis simulates the dynamic behavior of the storage rack structures. (b)(4)

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### 9.1A.4.1 Analysis Methodology

An accurate simulation is obtained by direct integration of the nonlinear equations of motion with three pool slab acceleration time-histories applied as the forcing functions acting simultaneously. Reliable assessment of the stress field and kinematic behavior of the rack modules incorporates key attributes of the actual structure in a conservative dynamic model. The model must have the capability to execute the concurrent motion forms compatible with the free-standing installation of the modules.

Calculations must incorporate momentum transfers due to the rattling of fuel assemblies inside storage cells; the lift-off and subsequent impact of support pedestals with the pool liner (or bearing pad); and quantification of fluid coupling due to water mass in the interstitial spaces around rack modules. In short, there are a large number of parameters with potential influence on the rack kinematics. The comprehensive structural evaluation must deal with all of these without sacrificing conservatism.

The model must be capable of effecting momentum transfers which occur due to rattling of fuel assemblies inside storage cells and the capability to simulate lift-off and subsequent impact of support pedestals with the pool liner (or bearing pad). The contribution of the water mass in the interstitial spaces around the rack modules and within the storage cells must be modeled in an accurate manner. During dynamic rack motion, hydraulic energy is either drawn from or added to the moving rack, modifying its submerged motion in a significant manner. Therefore, the dynamics of one rack affects the motion of all others in the pool.

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### 9.1A.4.1.2 Synthetic Time-Histories

The synthetic time-histories in three orthogonal directions (N-S, E-W, and vertical) are generated in accordance with the provisions of SRP 3.7.1 (Reference 48). A preferred criterion for the synthetic time-histories in SRP 3.7.1 calls for both the response spectrum and the power spectral density corresponding to the generated acceleration time-history to envelope their target (design basis) counterparts with only finite enveloping infractions. The time-histories for the pools have been generated to satisfy this preferred (and more rigorous) criterion. The seismic files also satisfy the requirements of statistical independence mandated by SRP 3.7.1.

Figures 9.1A-12 through 9.1A-16 provide plots of the time-history accelerograms (b)(4) respectively. These artificial time-histories are used in all non-linear dynamic simulations of the racks.

Results of the correlation function of the three time-histories are given in Table 9.1A-19. Absolute values of the correlation coefficients are shown to be less than 0.15, indicating the statistical independence of the three data sets.

### 9.1A.4.2 WPMR Methodology

The WPMR methodology incorporates both stress and displacement criteria. The following summarizes the sequence steps undertaken for model development:

a. Suitable 3-D dynamic models for a time-history analysis of the new maximum density racks are prepared. These models include the assemblage of all rack modules in each pool. Include all fluid coupling interactions and mechanical coupling appropriate to performing an accurate non-linear simulation. This 3-D simulation is referred to as a Whole Pool Multi-Rack model.

b. 3-D dynamic analyses are performed on various physical conditions (such as coefficient of friction and extent of cells containing fuel assemblies). Appropriate displacement and load outputs from the dynamic model for post-processing are archived.

c. A stress analysis of high stress areas for the limiting case of all the rack dynamic analysis is performed to demonstrate compliance with ASME Code Section III, Subsection NF limits on stress and displacement.

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The time-varying lateral (horizontal) and vertical forces on the extremities of the support pedestals produce stresses at the root of the pedestals in the manner of an end-loaded cantilever. The stress field in the cellular region of the rack is quite complex, with its maximum values located in the region closest to the pedestal. The maximum magnitude of the stresses depends on the severity of the pedestal end loads and on the geometry of the pedestal/rack baseplate region.

Alternating stresses in metals produce metal fatigue if the amplitude of the stress cycles is sufficiently large. In high density racks designed for sites with moderate to high postulated seismic action, the stress intensity amplitudes frequently reach values above the material endurance limit, leading to expenditure of the fatigue "usage" reserve in the material.

Because the locations of maximum stress (viz., the pedestal/rack baseplate junction) and the close placement of racks, a post-earthquake inspection of the high stressed regions in the racks is not feasible. Therefore, the racks are engineered to withstand multiple earthquakes without reliance of nondestructive inspections for post-earthquake integrity assessment. The fatigue life evaluation of racks is an integral aspect of a sound design.

A time-history analysis was performed to provide the means to obtain a complete cycle history of the stress intensities in the highly stressed regions of the rack.

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9.1A.4.3.5.5 <u>Weld Stresses</u>

Weld locations subjected to significant seismic loading are at the bottom of the rack at the baseplate-to-cell connection, at the top of the pedestal support at the baseplate connection, and at cell-to-cell connections. Bounding values of resultant loads are used to qualify the connections.

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b. <u>Baseplate-to-Pedestal Welds</u>

The weld between the baseplate and the support pedestal is checked using finite element analysis to determine the maximum stress under a Level B or Level D event. The calculated stress values are below the allowable values.

c. <u>Cell-to-Cell Welds</u>

Cell-to-cell connections are by a series of connecting welds along the cell height. Stresses in storage cell to cell welds develop due to fuel assembly impacts with the cell wall. These weld stresses are conservatively calculated by assuming that fuel assemblies in adjacent cells are moving out of phase with one another so that impact loads in two adjacent cells are in opposite directions; this tends to separate the two cells from each other at the weld. Table 9.1A-20 gives results for the maximum allowable load that can be transferred by these welds based on the available weld area. An upper bound on the load required to be transferred is also given in Table 9.1A-20 and is much lower than the allowable load. This upper bound value is very conservatively obtained by applying the bounding rack-to-fuel impact load from any simulation in two orthogonal directions simultaneously, and multiplying the result by 2 to account for the simultaneous impact of two assemblies. An equilibrium analysis at the connection then yields the upper bound load to be transferred. It is seen from the results in Table 9.1A-20 that the calculated load is well below the allowable.

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9.1A.4.3.5.7 <u>Level A Evaluation</u>

The Level A condition is not a governing condition for spent fuel racks since the general level of loading is far less than Level B loading. To illustrate this, the heaviest spent fuel rack is considered under the dead weight load. It is shown below that the maximum pedestal load is low and that further stress evaluations are unnecessary.

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9.1A.4.4 Fuel Pool Structure Integrity

The Wolf Creek fuel storage pool is a safety related, seismic category 1, reinforced concrete structure. Spent fuel is to be placed within storage racks located in the fuel storage pool. The fuel storage pool includes the spent fuel pool and the cask loading pool with fuel storage racks installed. The area is collectively referred to in this section as the fuel pool structure. An analysis was performed to demonstrate the structural adequacy of the pool structure, as required by Section IV of the USNRC OT Position Paper (Reference 63).

The fuel storage pool regions are analyzed using the finite element method. Results for individual load components are combined using factored load combinations mandated by SRP 3.8.4 (Reference 64) based on the "ultimate strength" design method. It is demonstrated that for the critical bounding factored load combinations, structural integrity is maintained when the pools are assumed to be fully loaded with spent fuel racks, as shown in Figure 9.1-2 with all storage locations occupied by fuel assemblies.

The regions examined in for the fuel storage pool include the floor slabs and the highly loaded wall sections adjoining the slabs. Both moment and shear capabilities are checked for concrete structural integrity. Local punching and bearing integrity of the slab in the vicinity of a rack module support pedestal pad is evaluated. All structural capacity calculations are made using design formulas meeting the requirements of the American Concrete Institute (ACI).

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4. The Cask Handling Crane and Spent Fuel Handling Machine (Refueling Platform) are designed to move along the N-S direction. The dead weight and the rated lift weight of these cranes are considered as dead load and live load, respectively.

5. The hydrostatic water pressure.

9.1A.4.4.2.2 Seismic Induced Loads

1. Vertical loads transmitted by the rack support pedestals to the slab during a SSE or OBE seismic event.

2. Hydrodynamic inertia loads due to the contained water mass and sloshing loads (considered in accordance with (Reference 66)) which arise during a seismic event.

3. Hydrodynamic pressures between racks and pool walls caused by rack motion in the pool during a seismic event.

4. Seismic inertia force of the walls and slab.

9.1A.4.4.2.3 Thermal Loading

Thermal loading is defined by the temperature existing at the faces of the pool concrete walls and slabs. Two thermal loading conditions are evaluated: The normal operating temperature and the accident temperature. The effect of gamma heating on the concrete was also considered and requires the implementation of administrative controls to maintain concrete temperatures within acceptable ranges, as discussed in section 9.1A.4.4.5.

9.1A.4.4.2.4 Pool Water Loading

The loadings described above were considered for two possible scenarios: one considers the Cask Loading Pool full of water and the other considers the Cask Loading Pool empty.

9.1A.4.4.3 Analysis Methodology

9.1A.4.4.3.1 Finite Element Analysis Model

The finite element model encompasses the entire Spent Fuel Pool and three other reinforced concrete structures located immediately adjacent to the Spent Fuel Pool (the Cask Loading Pool, the Transfer Canal and the Cask Washdown Pit). The interaction with the rest of the Fuel Building reinforced concrete, which is not included in the finite-element model, is simulated by imposing appropriate boundary conditions. The structural area of interest for the fuel storage pool includes only two, the spent fuel pool and the cask loading pool. However, by augmenting these areas of interest with the addition of the Transfer Canal and the Cask Washdown Pit, the constructed finite-element model and numerical investigation are enhanced because the perturbation induced by the boundary conditions on the stress field distribution for the area of interest is minimized.

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To simulate the interaction between the modeled region and the rest of the Fuel Building a number of boundary restraints were imposed upon the described finite-element model.

The behavior of the reinforced concrete existing in the structural elements (walls, slab and mat) is considered elastic and isotropic. The elastic characteristics of the concrete are independent of the reinforcement contained in each structural element for the case when the un-cracked cross-section is assumed. This assumption is valid for all load cases with the exception of the thermal loads, where for a more realistic description of the reinforced concrete cross-section including the assumption of cracked concrete is used. To simulate the variation and the degree of cracking patterns, the original elastic modulus of the concrete is modified in accordance with Reference (Reference 65).

9.1A.4.4.3.2 Analysis Technique

The structural region isolated from the Fuel Building and comprised of four pools (the Spent Fuel Pool, the Cask Loading Pool, the Transfer Canal and the Cask Washdown Pit) is numerically investigated using the finite element method. The pool walls and their supporting reinforced concrete mat are represented by a 3-D finite-element model.

The individual loads considered in the analysis are grouped in five categories: dead load (weight of the pool structure, dead weight of the rack modules and stored fuel, dead weight of the reinforced concrete Fuel Building upper structure, the dead weight of the Cask Handling Crane (CHC) and the Spent Fuel Handling Machine (SFHM), and the hydro-static pressure of the contained water), live loads (CHC and SFHM suspended loads), thermal loads (the thermal gradient through the pool walls and slab for normal operating and accident conditions) and the seismic induced forces (structural seismic forces, interaction forces between the rack modules and the pool slab, seismic loads due to self-excitation of the pool structural elements and contained water, and seismic hydro-dynamic interaction forces between the rack modules and the pool walls for both OBE and SSE conditions). The dead and thermal loads are considered static acting loads, while the seismic induced loads are time-dependent.

The material behavior under all type of loading conditions is described as elastic and isotropic representing the un-cracked characteristics of the structural elements cross-section, with the exception of the thermal load cases where the material elasticity modulus is reduced in order to simulate the variation and the degree of the crack patterns. This approach (Reference 65) acknowledges the self-relieving nature of the thermal loads. The degree of reduction of the elastic modulus is calculated based on the average ultimate capacity of the particular structural element.

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9.1A.4.4.6 Conclusions

Regions affected by loading the fuel pool completely with high density racks are examined for structural integrity under bending and shearing action. It has been determined that adequate safety margins exist assuming that all racks are fully loaded with a bounding fuel weight and that the factored load combinations are checked against the appropriate structural design strengths. It is also shown that local loading on the liner does not compromise liner integrity under a postulated fatigue condition and that concrete bearing strength limits are not exceeded.

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