UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 172

TO FACILITY OPERATING LICENSE NO. DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated May 14, 1993 (Reference 1), and supplemented by letters dated June 10 (Reference 2), July 16 (Reference 3), November 30 (Reference 4), December 1, 1993 (Reference 5), and January 27, 1994 (Reference 6), the Northeast Nuclear Energy Company (NNECO or the licensee) requested an amendment to change the Technical Specifications (TS) for the Millstone Nuclear Power Station, Unit No. 2. The change would modify the spent fuel pool (SFP) by introducing neutron absorbing (poison) rodlets (pins) into the stored fuel and increase the required burnup in Region C to permit removal of the cell blockers, thus increasing by 234 fuel assemblies the storage capacity of the SFP.

The November 30 and December 1, 1993, and January 27, 1994, submittals provided information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The existing racks in the Millstone 2 SFP are divided into three Regions according to the enrichment and burn-up limits of fuel assemblies. With the issuance of Amendment 109 (Reference 7), the storage racks in Region C were licensed for 75% storage occupancy, i.e., 234 cells out of total of 962 cells in Region C were blocked-off so that they could not be used for storage of fuel assemblies. Fuel assemblies are stored in a three-out-of-four array, with blocking devices installed to prevent inadvertent placement of a fuel assembly in the fourth location. In Amendment 117 (Reference 8), the NRC allowed storage of up to 5 canisters of consolidated fuel assemblies (a total of 10 fuel assemblies), and allowed the licensee to unblock the 234 cells, as required, for future storage of consolidated fuel assemblies. In Amendment 128 (Reference 9), the NRC deleted the limitation of Amendment 117 for storage of consolidated fuel. Amendment 158 reduced the storage capability of the SFP by adding cell blocking devices to 40 spent fuel storage locations in Region B, thus decreasing the total capacity of the SFP and thus limiting the time to 1994 at which the full off-load capability would be reached without further fuel consolidation. Therefore, to preclude this situation, and to ensure that sufficient spent fuel storage capacity continues to exist without fuel consolidation, the proposed amendment would introduce neutron absorbing rodlets into the stored fuel and increase the required burnup for fuel assemblies without rodlets in Region C to permit removal of cell blockers, thus reclaiming 234 blocked spent fuel storage locations. Three rodlets would be placed into each fuel assembly as required - one in the center control rod guide tube and the other two in opposite diagonals. However, the licensee emphasizes that the total fuel storage capacity, considering the removal of cell blockers continues to be less than that allowed as a result of consolidation. The proposed modification would increase the SFP storage capacity to 1306 storage locations which would carry the unit through the year 2000 with full off-load capability without fuel consolidation.

3.0 EVALUATION

3.1 Reactivity Analysis

The analysis of the reactivity effects of fuel storage in Region C was performed with the CASMO-3 transport theory computer code. Independent verification calculations were made with the KENO-5a Monte Carlo code using the 27-group SCALE cross-section library. Since the KENO-5a code package does not have burnup capability, depletion analyses and the determination of small reactivity increments due to manufacturing tolerances were made with CASMO-3. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. These experiments simulate the Millstone 2 spent fuel racks as realistically as possible with respect to parameters important to reactivity such as enrichment, assembly spacing, and absorber thickness. These two independent methods of analysis (KENO-5a and CASMO-3) showed good agreement both with experiment and with each other. The intercomparison between different analytical methods is an acceptable technique for validating calculational methods for nuclear criticality safety. To minimize the statistical uncertainty of the KENO-5a calculations, a minimum of 500,000 neutron histories in 1000 generations of 500 neutrons each were accumulated in each calculation. Experience has shown that this number of histories is sufficient to assure convergence of KENO-5a reactivity calculations. The staff concludes that the analysis methods used are acceptable and capable of predicting the reactivity of the Millstone 2 storage racks with a high degree of confidence.

The minimum burnup of spent fuel required for safe storage of fuel enriched to 4.5 weight percent (w/o) U-235 in every cell of Region C of the spent fuel pool, including the space under the present cell-blockers, was evaluated. The calculations were made at 150°F. The temperature coefficient of reactivity in Region C is positive and the Millstone 2 TS require the spent fuel temperature to be no greater than 140°F. Therefore, the use of 150°F in the criticality analyses is conservative and acceptable. Uncertainties due to tolerances in fuel enrichment and density, lattice spacing, and stainless steel thickness were accounted for as well as uncertainties in the depletion calculations. These uncertainties were appropriately determined at the 95/95 probability/confidence level. In addition, a calculational bias and uncertainty were determined from the benchmark calculations. The final Region C design, when fully loaded with fuel enriched to 4.5 w/o U-235 which has attained a burnup of at least 55.72 MWD/KgU, resulted in a k_{eff} of 0.946 when combined with all known uncertainties. This meets the staff's criterion of k_{eff} no greater than 0.95, including all uncertainties at the 95/95 probability/confidence level, and is, therefore, acceptable. The curve in TS Figure 3.9-1A gives the combination of initial enrichment and cumulative burnup required for spent fuel storage in Region C.

As an alternative, Region C was also evaluated for the minimum burnup required for safe storage in every cell with the use of borated steel poison rodlets inserted into the spent fuel assemblies. Each of these assemblies was assumed to contain three poison rodlets with a linear orientation, one in the center guide tube and any two diagonally opposite guide tubes. These rodlets were 152.5 inches long with a 0.87-inch O.D. and composed of 2 w/o natural boron in stainless steel. Approximately 9% reactivity is held down by the rodlets. The rodlets require a unique tool for removal and, therefore, cannot be inadvertently removed from the fuel assemblies once inserted. They would be secured by the weight of the rodlets (22 pounds buoyant weight). The licensee maintains surveillance of the rodlets through procedural controls. The rodlets can be verified to be in position by visual inspection from above. Therefore, the staff considers the rodlets to essentially be an integral part of the fuel assembly and acceptable for reactivity hold down.

As seen in the curve of TS Figure 3.9-1B, fuel enriched to 4.5 w/o U-235 with poison rodlets require a minimum burnup of 44.26 MWD/KgU for storage in every cell location of Region C. In addition to the uncertainties considered above, uncertainties due to rodlet diameter and boron loading tolerances were also incorporated. The resulting maximum k_{eff} was 0.940, which meets the 0.95 criterion.

Criticality calculations also confirmed that there are no adverse reactivity effects at the interfaces between any of the rack regions. In addition, criticality calculations for the new fuel elevator area of the pool showed that there is virtually no neutronic interaction between the fresh fuel in the elevator and the spent fuel in the storage racks. Consequently, the new fuel elevator may be used without any restrictions other then the enrichment limit of 4.5 w/o U-235.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, such as the accidental misloading of a fresh fuel assembly of the highest permissible enrichment into a cell in Region C. However, for such events, credit may be taken for the presence of approximately 800 ppm of boron in the pool water required by TS since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (Double Contingency Principle). The reduction in k_{eff} caused by the soluble boron more than offsets the reactivity addition caused by credible accidents.

Based on the staff's review, the staff finds the criticality aspects of the proposed fuel assembly storage changes in Region C of the Millstone 2 spent fuel pool are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. The staff concludes that fuel from Millstone 2 may be safely stored in Region C of the spent fuel pool provided that the U-235 enrichment does not exceed 4.5 w/o and that it meets the burnup requirements specified in TS Figures 3.9-1A or 3.9-1B.

3.2 Materials/Chemical Engineering Review

Three rodlets will be inserted into the guide tubes of each fuel assembly stored in Region C of the pool and whose enrichment-burnup characteristics conform to the requirements given in Figure 3.9-1b of the modified TS. The rodlets are made of borated stainless steel containing 2 w/o of boron.

The poison rodlets for Millstone 2 consist of solid cylinders, 0.870 inches in diameter and 152.5 inches long with a spherical tip and a 1.6 inch long counterbore at its top for inserting a special installation tool. Each rodlet weighs between 25 and 26 pounds. The size of the rodlet is very close to that of the individual fingers in the control element assembly. Its diameter is smaller by 0.08 inches and it is 1.75 inches longer. Since nominal internal diameter of the guide tube is 1.035 inches over most of its fuel length, a clearance of 82 mils exists between the rodlet and the guide tube walls with the exception of the dashpot region where the clearance is 48 mils. This clearance is larger than the corresponding clearance for the fingers of the control elements assembly and any crud accumulated in the gap should not interfere with removal or insertion of the rodlets.

The rodlets were made from borated stainless steel, Type 304 B7, Grade A, manufactured by Carpenter Technology Corporation and containing 2 w/o of boron. They were manufactured in accordance with the requirements of standards ASTM A 887-89 and ASTM A 484-91.

Borated stainless steel is a two phase alloy, composed of a complex boride phase in an austenitic chromium-nickel-iron matrix. In addition to the U.S., this material is also manufactured in Austria, Great Britain, and Japan with boron content ranging from 0.2 to 2.25 w/o. It is extensively used by foreign nuclear industry for making spent fuel storage racks and spent fuel transportation casks. In the U.S., it was used in the Indian Point Nuclear Station as a poison plate material in their spent fuel pool.

In general, the physical properties of the material resemble those of 304 austenitic stainless steel. However, the yield strength, ultimate tensile strength and hardness increase with increasing levels of boron and ductility and impact strength are decreased. These properties also vary with the exposure to neutron fluency, but no significant changes occur for neutron fluences below 10^{17} n/cm^2. This value is much higher than 10^{12} n/cm^2 which is the maximum anticipated neutron fluency reached during the lifetime of the SFP in Millstone 2. Since poison rodlets do not carry any loads when inserted in the guide tubes, mechanical properties of the material are not of primary importance.

Although intergranular corrosion resistance of borated stainless steel exposed to acidic conditions decreases with increased boron content, long term tests with borated stainless steel have indicated that in the SFP environment no measurable corrosion effects take place. It is not expected, therefore, that any meaningful corrosion degradation of poison rodlets will occur during their service life. However, in order to have assurance that at all times there is enough poison material for reactivity control, the licensee committed to institute a surveillance program where, at 5 year intervals, 1% of the rodlets will be visually inspected for any material degradation.

Based on the above evaluation, the staff concludes that the poison rodlets made of borated stainless steel, with the material characteristics described in the licensee's submittals, will resist material degradation in the SFP environment. Verification of their conditions by periodic inspections will provide an assurance that the integrity of the neutron absorbing material required for reactivity control will not diminish from that assumed in the analysis. The selection of appropriate material and commitment to a material surveillance program meet the requirements of 10 CFR 50, Appendix A, General Design Criterion 61 regarding the capability to permit appropriate periodic inspection, and testing of the components and General Design Criterion 62 regarding prevention of criticality by use of neutron absorbers. The proposed use of poison rodlets made of borated stainless steel is, therefore, acceptable.

3.3 Radiation Protection Review

3.3.1 Occupational

The licensee stated in its application that all work required to remove the existing cell blockers will be performed using remote handling tools. The licensee further stated (and subsequently committed via teleconference and in its response to request for additional information dated December 1, 1993) that it does not intend to use divers for the SFP modification. In the unforeseen circumstance that divers would be needed during the modification, the licensee has committed to stop SFP modification operations and submit to NRC a proposal addressing the radiological safety precautions to be utilized during diving operations. This submittal shall address the "Procedures for Diving Operations in High and Very High Radiation Areas" as stated in Appendix A of Regulatory Guide 8.38, "Control of Access to High and Very High Radiation Areas in Nuclear Power Plants."

The licensee stated in a teleconference held on December 10, 1993, that all phases of the SFP modification will result in a total of less than one person-rem of exposure.

The licensee will prepare and follow specific procedures that are consistent with good as low as is reasonably achievable (ALARA) principles and practices. The modification will be worked under specific radiological work permits that will require appropriate levels of protective clothing and dosimetry to keep employee exposures ALARA. The licensee further stated that all cell blockers will be decontaminated and monitored under water prior to the removal from the SFP to minimize the potential of any not particle exposure.

The licensee stated that external radiation fields in the area near the pool surface range between 1 and 3 mrem per hour, and are not expected to increase during the SFP modification. During all evaluations, when radioactive material is removed from the pool, there will be continuous health physics job coverage. Further, the licensee will perform air grab samples during the modification and will provide continuous air sampling of the work area when radioactive material is removed from the pool.

Based on the staff's review of the licensee's application, the staff finds the proposed radiation protection aspects of the SFP modification acceptable.

3.3.2 Design Basis Accident Analysis (DBA)

In its application, the licensee evaluated the possible consequences of postulated accidents, included means for their avoidance in the design and operation of the facility, and provided means for mitigation of their consequences should they occur. The licensee has evaluated the effect of the changes on the calculated consequences of a spectrum of postulated design basis accidents (i.e., Fuel Handling accidents and Spent Fuel Cask Drop accidents) and concludes that the effect of the proposed TS change is small and that the calculated consequences are within regulatory requirements and staff guideline dose values. The addition of poison pins or removal of blocking devices will not have any effect on the probability of occurrence of either of these two accidents. Since the licensee proposes to utilize extended burnup fuel, the staff reevaluated the fuel handling accident for Millstone 2 to consider the effects of increased burnup.

In its Final Safety Analysis Report for Millstone 2, issued on May 10, 1974, the staff conservatively estimated offsite doses due to radionuclide releases to the atmosphere from a fuel handling accident. The staff concluded that the plant mitigative features would reduce the doses for this DBA to below the doses specified in Standard Review Plan (SRP) Section 15.7.4.

Since the licensee intends to utilize extended burnup fuel, the staff reanalyzed the fuel handling DBA for this case. According to NUREG/CR-5009, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors" (February 1989), increasing fuel enrichment to 5.0 weight percent U-235 with a maximum burnup of 60,000 MWD/T increases the doses for a fuel handling accident by a factor of 1.2. Therefore, the 1.2 factor increase in dose, displayed in Table 1 below, bounds the dose consequences of the licensee's proposal. In Table 1, the new and old DBA doses are presented and compared to the guidelines doses in SRP Section 15.7.4 (established based on 10 CFR Part 100).

Table 1
Radiological Consequences for Fuel Handling Design Basis Accident (rem)

	Exclusion Area	Low Population Zone
	Thyroid	Thyroid
Staff Evaluation May 10, 1974	2.8	< 1
Bounding Estimates for Extended Burnup Fuel	3.4	1.2
Regulatory Requirement (NUREG-0800) Chapter 15.7.4)	75	75

[1]Factor of 1.2 greater than original estimate for iodine.

The staff concludes that the only potential increased dose resulting from the fuel handling accidents with extended burnup fuel is the thyroid doses; these doses remain well within the dose limits set forth in NUREG-0800 and are, therefore, acceptable.

3.4 Plant Systems Review

3.4.1 Heavy Loads Concern

The weight of the rodlet is approximately 25 pounds, therefore of no concern regarding movement of heavy loads since a rodlet weighs much less than a single fuel assembly.

3.4.2 Thermal/Hydraulic Concerns

3.4.2.1 Spent Fuel Pool Coolant and Stored Fuel Assemblies

While the storage capacity of fuel elements has increased by 234 assemblies, the overall storage capacity is less than that allowed by Amendment 117. Therefore, the thermal/hydraulic results affecting SFP coolant and stored fuel assemblies which were found to be acceptable in Amendment 117 remain acceptable and are consistent with the proposed changes.

3.4.2.2 Rodlet Cooling

The only outstanding thermal/hydraulic issue is that of cooling the rodlets. The licensee reported that heat generation would be 0.2 watt/cm^3 at a time 72 hours after shutdown. At this heat generation rate, the internal temperatures of the rodlet would be only a few degrees above the SFP coolant temperature.

The anticipated SFP coolant and spent fuel assembly temperatures are consistent with those previously found satisfactory by the staff and are, therefore, acceptable. The central temperature of a rodlet is expected to be only a few degrees higher than that of the coolant surrounding it and is, therefore, acceptable.

Therefore, the staff finds the licensee's proposal to be acceptable with regard to heavy loads and thermal/hydraulic concerns.

3.5 Civil/Structural Engineering Review

In the original submittal (Reference 1), and in the presentation of the content of the application, the licensee did not provide adequate technical basis regarding the structural integrity of the affected hardware. However, as a result of the request for additional information, the licensee provided the prior analyses and made qualitative conclusions regarding the structural integrity of the racks and the SFP. The staff utilized this information to arrive at the safety conclusion to accept the proposed amendment.

In the Safety Evaluation attached to Amendment 109 (Reference 7), the staff had concluded that the proposed reracking, with high-density, free-standing racks and single fuel assemblies in all cell locations, was acceptable from the standpoint of structural integrity of the racks and the SFP. In the Safety Evaluation attached to Amendment 117 (Reference 8), the staff had concluded that the structural considerations associated with the storage of consolidated spent fuel, including those associated with the consolidated fuel storage boxes (canisters), have been adequately addressed by the licensee.

The weight of a single fuel assembly is less than 1200 pounds, that of the consolidated fuel assemblies, together with the storage canister, is considered as 2500 pounds. The three rodlets (poison pins) added to a single fuel assembly increases its dry weight by less than 78 pounds. A cursory review of the prior structural analyses (References 10 and 11) of the racks indicates that the licensee has considered various fuel loading cases for single fuel assemblies and consolidated fuel assemblies. In each case the licensee has considered fully loaded, partially-unsymmetrically loaded and empty rack modules. Thus, the staff agrees with the licensee's reasoning that the structural analysis of the proposed modification is covered by the previously accepted analyses.

Even though the seismic analyses methods used in the licensee's analyses are not as current as the present state-of-the-art, the assumptions made are conservative and the resulting hardware dimensions (e.g., cell-wall thicknesses, pedestal dimensions) are comparable to the current designs. The staff also agrees with the licensee that, from the standpoint of structural integrity, various fuel drop accident cases (for the proposed modification) are enveloped by the previously accepted cases of the consolidated fuel drop analyses.

Based on the review of the licensee's submittal, the relevant portions of the prior reracking amendments (Amendment Nos. 109, 117 and 128), and responses to the staff's request for additional information, the staff concludes that the proposed modification to the existing Region C storage racks will not adversely affect the conclusions drawn regarding structural integrity of the racks and the SFP with the issuance of the prior amendments.

3.6 Mechanical Engineering Review

The licensee provided the NRC staff with an evaluation which concluded that the original structural analysis performed for the current spent fuel rack configuration remains valid for the proposed modification (Reference 1). With regard to the structural considerations, the licensee concluded that the change does not involve a significant reduction in margin of safety because the mechanical properties and weight of the fuel assemblies remain essentially unchanged. The fuel racks are freestanding and, with the inclusion of the weight of the three rodlets per assembly, the original analyses of the fuel assembly/fuel rack and fuel pool building interfaces previously approved by the NRC staff remain valid and conservative.

In the submittal dated June 10, 1993 (Reference 2), the licensee provided two reports to support their conclusions. The reports described the original seismic and structural analyses of the spent fuel racks performed in 1985.

The CESHOCK computer code was used to analyze nonlinear two-dimensional lumped mass models of the fuel rack loaded with spent fuel. The dynamic characteristics of the rack modules were determined from more detailed linear three-dimensional finite element models using the SAP IV computer code. The CESHOCK models incorporate the nonlinear characteristics of the system including friction and gaps and accounts for hydrodynamic coupling effects. Since the SFP contains fuel racks of different sizes and fuel load conditions, different models and load cases were developed in an attempt to determine the enveloping loads, displacements and stresses. The modules analyzed were a Region I 8 x 10 module, a Region II 7 x 8 module, and a Region II 7 x 9 module (the current submittal refers to the original Region II as Region C and the original Region I as Region A and B). Fully-loaded, partially-loaded and empty modules were considered in the analyses.

The results of the seismic analyses provided the loads that were used in the stress calculations described in the spent fuel rack structural report. A set of load multiplication factors was determined for each load direction for application to the three-dimensional SAP IV stress model. The component stress on each element resulting from the application of each directional load factor was combined by the square-root-of-the-sum-of-the-squares method. The resulting stresses associated with the Safe Shutdown Earthquake (SSE) loading were compared with the allowable stress limits for both the SSE and Operating Basis Earthquake (OBE) loadings. The allowable stress limits were based on the ASME Code, Section III, Subsection NF, 1983 Edition.

In a submittal dated July 16, 1993 (Reference 3), the licensee provided additional technical information. A section of another report describing the original design and analysis procedures was submitted (J. F. Opeka letter to USNRC, "Millstone Nuclear Power Station, Unit 2, Proposed Change to Technical Specifications Storage of Consolidated Spent Fuel," dated May 21, 1986). The report provided more information on the load cases used in the original analysis. The licensee stated that the only case that was not considered in the original analysis was the fuel assembly with poison pins. For this case the licensee stated that the additional weight of the pins does not adversely affect the frequency response of the fuel assembly and is therefore bounded by the cases of the fuel assembly without poison pins and the consolidated fuel assembly. The report also provided information on the original fuel handling accident evaluations. Both fuel assembly and consolidated box drop scenarios had been considered and evaluated relative to the racks and the pool liner. The licensee stated that the effects of the increased weight due to the poison pins in fuel assemblies was considered and was determined to be enveloped and bounded by the original evaluations. The licensee also provided the original Technical Evaluation Report prepared by the Franklin Research Center, TER-C5506-586, dated December 17, 1985, which reviewed the structural adequacy of the original Millstone 2 spent fuel rerack submittal and concluded that it was acceptable.

The staff, with the assistance of its consultants at the Brookhaven National Laboratory (BNL), reviewed the licensee's submittals to determine whether the original analysis bounds the proposed configuration but found the level of detail in the licensee's reports inadequate. The reports presented only the worst case maximum stress, load and displacement results. In order to perform an adequate independent evaluation, the staff requested the licensee to provide a more comprehensive description of the analysis including the following information: a complete list of cases analyzed along with the results for each individual case; a description of the differences between the consolidated and intact fuel assembly model dynamic characteristics such as the natural frequencies of the normal fuel assembly, consolidated fuel, and fuel racks in air and water, hydrodynamic mass and coupling parameters, and fuel to rack cell gaps; a description of how the cell blocking devices were considered in the analysis; and clarification on whether the analysis of a fully-loaded rack with the standard fuel considered that only 75% of the total cells would be used. In order to assess the effect of the additional weight of the poison rods, the licensee was asked to provide the percentage weight increase of the fuel assembly with poison rods. The licensee provided this additional detailed analysis in a submittal dated November 30, 1993 (Reference 4).

A total of nine CESHOCK load cases were identified. There were six cases of Region II 7 x 8 modules, two cases of Region II 7 x 9 modules, and one case of a Region I 8 x 10 module. In four load cases, the module was fully loaded with consolidated fuel canisters (CFCs). Two cases were partially loaded with CFCs (an outside row). One case was fully loaded with fuel assemblies and two cases were of empty racks. One load case assumed a sliding rack base. The other eight cases assumed a fixed base. SSE loads were applied to eight cases and OBE loads were applied to one. A summary of loads and displacements for each load case was provided. The licensee indicated that the most limiting load cases were selected to perform the detailed stress analysis. The full loaded 8 x 10 rack with the CFCs was found to develop the maximum stress. The licensee provided detailed descriptions of the different CESHOCK models that were analyzed. Quantitative model parameters included weights, stiffnesses, frequencies, gaps and hydrodynamic coupling parameters were given. The licensee stated that cell blocking devices were not considered in the analyses. The analyses of fully-loaded racks assumed that 100% of the total number of cells were occupied. The licensee also indicated that each poison rodlet weighs 26 pounds (dry), for a total increase of 78 pounds per candidate fuel assembly. The percentage weight increase to a single fuel assembly is 6%.

The information in Reference 4 clarifies a number of areas relative to the seismic adequacy of the spent fuel racks. Several different fuel rack models were developed and analyzed in order to determine bounding rack responses for the various possible configurations and loading conditions. It was noted that different CESHOCK dynamic models were developed for racks with consolidated fuel canisters and for racks with normal fuel assemblies. A review of the input parameters showed that differences in weights, stiffnesses, gaps and hydrodynamic coupling parameters were appropriately considered. This resulted in significantly different natural frequencies between the CFC loaded racks and the fuel assembly loaded racks.

The load and displacement results of the nine load cases were reviewed to identify bounding conditions. Of the nine CESHOCK loaded cases analyzed, eight represented typical Region C (originally designated Region II). The proposed modification affects only Region C fuel racks. The 7 x 8 racks were analyzed for the most significant parameter variations. The 7 x 8 rack load cases include a fully-loaded CFC case, a fully-loaded fuel assembly case, two partially-loaded CFC cases, and two empty rack cases. One of the empty rack cases assumed a sliding base and the remaining five cases assumed a fixed base. By comparing the load and displacement results of the six 7 x 8 rack load cases, it was observed that the responses of the CFC loaded racks enveloped the responses of the fuel assembly loaded rack. Base shear in the fully-loaded CFC rack was nearly twice the magnitude of the fully-loaded fuel assembly rack. CFC to rack cell impact loads for fully- and partially-loaded CFC racks were significantly larger than fuel to rack cell impact loads. Displacements of the fully-loaded CFC rack were also larger than those of the fully-loaded fuel assembly rack. The partially-loaded CFC racks showed significantly higher tipping displacements. As expected, the two empty rack cases resulted in minimum values of base shear with the sliding empty rack having the lowest value. The sliding empty rack load case provided the largest rack displacement. The remaining three load cases investigated the responses of the 7 x 9 module and 8 x 10 module fully loaded with CFCs. For SSE seismic loads, the load and displacement results were of similar magnitude as the 7 x 8 fully-loaded CFC loaded rack results and enveloped the loads and displacement of the 7 x 8 fully-loaded fuel assembly rack.

The staff determined that the original analyses of the 7 x 8 module fully loaded with fuel assemblies closely approximates the proposed configuration. Although the racks were licensed to contain cell blocking devices that would limit the fuel load capacity to 75% of the total number of cells, the analysis assumed that 100% of the cells were occupied. Thus, the only difference between the proposed modified configuration and the originally analyzed fully-loaded fuel assembly load case is the additional weight of the poison rodlets which increase the weight of a single fuel assembly by only 6%. The effect of the additional weight is not expected to be significant. As discussed above, the overall load and stress results are bounded by the CFC loaded racks by a significant margin.

With regard to rack displacements, the sliding rack load case provided the maximum displacement. However, only an empty rack load case was analyzed. It was not evident to the staff that the empty rack would have the largest sliding displacement. The staff, therefore, requested additional information to clarify why a fully- or partially-loaded sliding rack should not experience larger sliding displacement with greater potential for rack impact. In addition, the seismic report stated that impacts between adjacent racks will not occur because the maximum relative displacement between racks is less than the 2-inch nominal spacing between modules. From the displacement results of the nine load cases analyzed, the staff could not determine how the relative displacement was computed.

The licensee provided additional information in response to staff questions in a submittal dated January 27, 1994 (Reference 5). The licensee explained that in addition to the nine load cases discussed in Reference 4, there were eight load cases that were analyzed before the time history seismic input motions were finalized. Among the preliminary eight load cases, two cases considered sliding fuel racks. One rack was full loaded and one rack was empty. The results of the preliminary sliding rack analyses showed that the lateral displacement of the empty rack was much larger than that of the fully-loaded rack. Based on these comparative preliminary load case results, the empty rack load case was judged to be the bounding case for sliding and was included in the set of final load cases. The staff found this acceptable.

In response to the question on relative displacements, the licensee stated that maximum relative displacements between adjacent modules were obtained by considering the time-phased motions of two adjacent modules. Five different pairs of combinations of fuel load conditions (fully loaded, partially loaded and empty) in two adjacent racks were considered. The staff requested clarification of the methodology used to consider the time-phased motions of adjacent racks. From the displacement information that was provided the staff could not verify the maximum relative displacement was less than the spacing between the racks. In a telephone conversation between BNL, NRC and the licensee on January 14, 1994, the licensee explained that the relative displacements were determined from the response time histories of the adjacent rack modules for rack pairs considered. Using this information, the licensee was able to determine relative displacement as a function of time for each rack pair. The maximum relative displacement occurred between a fully-loaded rack and a partially-loaded rack. The staff agrees with this approach and accepted the licensee's results.

The staff reviewed the stress results presented in the structural reports and found that in the original analysis there were generally significant margins to stress allowables. In most cases, stresses associated with the SSE condition were below OBE allowable stresses. The smallest margin was associated with weld stresses (23,353 psi vs. 24,000 psi allowable). The location of these stresses was not given. Considering the small margin, the staff requested the licensee to provide more detailed weld stress information for review.

In response to this request (Reference 4), the licensee stated that the maximum weld stress occurs at the connection between the adjustable foot support block and the rack cell to which it is attached. The forces applied to the weld were determined from the maximum load found in that joint from the finite element analysis. The licensee explained that the loads for the weld stress calculation were obtained from the CFC rack configuration. Stresses for the normal fuel assembly load cases were not computed because the loads were well below the reported values. As discussed above, the staff concurs with the licensee's position that the loads for the CFC loaded rack configuration are expected to be significantly higher than the loads for the normal fuel assembly rack configuration. Since the proposed modification involves only normal fuel storage, the weld stress margin will be much larger and, therefore, acceptable.

In reviewing the stress results, the staff also noted that the licensee did not provide or address the evaluation of the fuel to rack storage cell impact loads. Appendix D to SRP 3.8.4 requires that these loads be considered for local as well as overall effects on the rack walls and supporting framework, and that it be demonstrated that the loads do not lead to fuel damage. The staff requested the licensee to provide this information for review in order to verify that the original results are applicable to the revised configuration.

In response (Reference 4), the licensee reported the fuel/CFC to rack cell impact loads and adequately addressed the evaluation of resulting rack stresses but did not demonstrate that the impact loads do not result in fuel damage. The licensee was therefore asked to provide the maximum allowable impact load on fuel assemblies. In Reference 5, the licensee provided the value of maximum allowable lateral impact load on a fuel assembly. The value was substantially larger than the maximum value predicted in the analysis. This adequately resolved the concern.

The staff reviewed the description of the analysis of the fuel assembly drop accident in Reference 3. Both the case of a normal fuel assembly and a consolidated fuel assembly drop were considered. The number or type of fuel assemblies stored in the fuel rack is not a factor in the analysis. The increased weight of a normal fuel assembly due to the addition of poison pins is clearly bounded by the weight of a consolidated fuel assembly. Therefore, the staff agrees with the licensee's conclusion that the original fuel drop calculation envelops the proposed modification.

Based on a review of the information provided by the licensee as discussed in this safety evaluation (SE), the staff concludes that the licensee adequately demonstrated that the seismic response and the structural evaluation of the proposed configuration is bounded by the original analysis. Therefore, the original staff-approved seismic qualification of the spent fuel racks remains valid for the proposed modified fuel storage configuration of the Region C modules.

4.0 TECHNICAL SPECIFICATION CHANGES

The following TS and Surveillance Requirement (SR) changes have been proposed as a result of the reanalysis of the Millstone 2 SFP. The staff finds these changes acceptable as well as the associated Bases changes.

(1) Definition 1.39, STORAGE PATTERN is currently defined for a cell blocking device in every 4th rack location for Region B and C. This is being changed to delete the reference to Region C, since the cell blocking devices are being removed for Region C.

(2) SR 4.9.18.1 is being modified to satisfy either:

(a) Fuel assembly enrichment and burnup are within the limits of Figure 3.9-1a, or

(b) Fuel assembly enrichment and burnup are within the limits of Figure 3.9-1b and borated stainless steel poison pins are installed in the assembly's center guide tube and in two diagonally opposite guide tubes.

(3) Figure 3.9-1 is being changed to Figure 3.9-1a, "Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region C," to reflect the revised burnups required of non-poisoned fuel assemblies for storage in Region C.

(4) Figure 3.9-1b, "Minimum Required Fuel Assembly Exposure as a Function of Initial Enrichment to Permit Storage in Region C with Poison Pins Installed," is being added to reflect the burnups required of poisoned fuel assemblies for storage in Region C.

(5) Figure 3.9-2 is being changed to remove the cell blocking devices previously required in Region C and to correct a typographical error in Region A.

(6) TS 3.9.19.1 is being deleted since all of the cell blocking devices are being removed from Region C.

(7) TS 5.6.1(d) is being changed to reflect the revised burnup requirements for fuel storage into Region C and to define the poison rodlets size and composition.

(8) TS 5.6.1(e) is being modified to reflect the allowable placement of consolidated fuel in compliance with Figure 3.9-3.

(9) TS 5.6.3 is being changed to replace the reference of "1237" storage locations to "1306" to reflect the removal of cell blocking devices from Region C. The value of locations remaining blocked now refers only to the 40 blocked cells in Region B.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Connecticut State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on February 18, 1994 (59 FR 8278). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. J. F. Opeka letter to USNRC, "Millstone Nuclear Power Station, Unit No. 2, Proposed Revision to Technical Specifications Spent Fuel Pool Modifications," dated May 14, 1993

2. J. F. Opeka letter to USNRC, "Millstone Nuclear Power Station, Unit No. 2 Response to Request for Additional Information," dated June 10, 1993.

3. J. F. Opeka letter to USNRC, "Millstone Nuclear Power Station, Unit No. 2 Response to Request for Additional Information," dated July 16, 1993.

4. J. F. Opeka letter to USNRC, "Millstone Nuclear Power Station, Unit No. 2 Response to Request for Additional Information," dated November 30, 1993.

5. J. F. Opeka letter to USNRC, "Millstone Nuclear Power Station, Unit No. 2 Response to Request for Additional Information," dated December 1, 1993.

6. J. F. Opeka letter to USNRC, "Millstone Nuclear Power Station, Unit No. 2 Response to Request for Additional Information," dated January 27, 1994.

7. Letter from D. B. Osborne (NRC) to J. F. Opeka (NNECO), "Issuance of Amendment No. 109," dated Jan. 15, 1986.

8. Letter from D. H. Jaffe (NRC) to E. J. Mroczka (NNECO), "Issuance of Amendment No. 117," dated June 2, 1987.

9. Letter from D. H. Jaffe (NRC) to E. J. Mroczka (NNECO), "Issuance of Amendment No. 128," dated March 31, 1988.

10. NNECO Submittal on Reracking of Spent-Fuel Pool, dated July 24, 1985.

11. NNECO Submittal on Storage of Consolidated Spent Fuel, dated May 21, 1986.

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