UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 109 TO DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

1.0 INTRODUCTION

By letter dated July 24, 1985, Northeast Nuclear Energy Company (NNECo) submitted an application to increase the storage capacity of the spent fuel pool (SFP) by replacing the existing racks with new storage racks. By letters dated September 16, 1985, October 17 and 28, 1985, November 25 and 27, 1985 and December 3, 1985, NNECo provided additional clarification in response to the NRC staff's requests. This represents the second re-racking of this storage facility. The first reracking was approved on June 30, 1977 and increased the storage capacity from 301 fuel assemblies to 667 fuel assemblies.

The amendment would authorize the licensee to increase the current capacity by installing high density racks to bring the capacity up to 1112 fuel assemblies. This will provide storage until 1993 with space for a full core offload assuming reloads of a third of a core.

2.0 DISCUSSION AND EVALUATION

2.1 Criticality Considerations

NNECo has requested approval of an application to install high density spent fuel storage racks at Millstone Unit 2. The proposed reracking will replace the present racks and increase the storage capacity from 667 to 1112 fuel assemblies. The new racks have been designed and analyzed for criticality by Combustion Engineering (CE). These racks are divided into two regions, one of which will accept unburned fuel with up to 4.5 percent U-235 enrichment and can accommodate offloads of the reactor, and the other which will receive fuel which has achieved a required (initial enrichment dependent) burnup. Similar high density storage, with multiple storage regions of different characteristics and with credit for fuel burnup, has been reviewed and approved for numerous reactors over the past several years. Recent examples have been Virgil C. Summer and St. Lucie Unit 2. The latter in particular was also designed and analyzed by CE, and the review approved CE methods and analyses.

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Region 1 will have 384 storage locations which can hold 1.7 reactor cores and thus can be used for offloading, if necessary. The cell nominal center-to-center spacing is 9.8 inches, is made of stainless steel, and uses a flux trap and Boroflex absorbers with a minimum B-10 loading of 0.030 gm/cm^2. It is designed to hold unburned Millstone 14 x 14 fuel assemblies with an enrichment up to 4.5 percent U-235. The nuclear design is similar to other approved high density storage racks using fixed boron absorber.

Region 2 will have 962 cells with 788 useable storage locations. The cell nominal center-to-center spacing is 9.0 inches, is made of stainless steel and has no fixed boron absorber. One of every four cells will not contain fuel and is blocked to prevent inserting fuel assemblies. This blocking of cells is similar to other approved designs for high density storage. This storage array is designed to hold fuel which has experienced sufficient burnup such that storage in Region 1 is not required. A required amount of burnup as a function of initial U-235 enrichment has been developed and will become part of the Technical Specifications. As an example, for 4.5 percent enrichment the assembly average burnup must be at least 34 GWD/T.

2.1.1 Analysis Methods, Benchmarking Uncertainties

The criticality analyses for the storage was done by CE using the same methods as used, reviewed and approved for previous analyses, e.g., for St. Lucie 2. The primary analysis method is DOT-2W, a two-dimensional discrete ordinates transport theory code for reactivity determination. Cross sections are calculated by the CEPAK lattice code with correction factors to account for heterogeneous lattice effects calculated by the NUTEST two-dimensional integral transport theory code. Calculated regions are assumed to be infinite in extent, both laterally and in length.

NNECo has provided benchmark qualification analyses of the CE calculation model and methods. These provided uncertainty and bias values for the calculations. In addition, calculations were performed to evaluate reactivity effects of mechanical tolerances, offcenter placements, and temperature changes. Boron-10 density in the Boroflex was taken at a minimum. The pool temperatures were taken at effectively maximum conditions for reactivity by adding uncertainty values to the relatively conservative temperatures used in the nominal calculations to reach peak (region dependent) non-accident and accident design limit conditions. The resulting total uncertainites, which are at least at a 95/95 confidence level, are 0.020 and 0.014 delta k for Regions 1 and 2, respectively. These are similar to other comparable rack analysis uncertainties previously reviewed.

The effect of non-uniform axial burnup was investigated by analyzing extreme axial burnup shapes and comparing to the uniform burnup shape assumed in the rack calculations. In most cases, non-uniform burnup is less reactive but the maximum positive reactivity effect found was used as a positive basis of 0.0114 delta k and was converted into an equivalent burnup to be applied to the Region 2 required burnup vs. enrichment limit curve.

2.1.2 Criticality Results

The results of the analyses for Region 1 give an effective multiplication factor for 4.5 percent enriched fuel of 0.943 (for unborated water), including the uncertainty factors at a conservative temperature of 68°F (calculated at 90°F and extended to 68°F). This is within the acceptance criterion of 0.95 at a 95/95 limit.

For Region 2, a family of curves of multiplication vs. burnup for a range of enrichments is generated. The values include the uncertainties. The curves are then used to define the minimum burnup for fuel of a given initial enrichment which will result in a multiplication less than 0.95 when Region 2 is fully loaded with assemblies of this type. These data points are used to plot the curve of required minimum average burnup vs. initial enrichment which forms a part of the Technical Specifications. The uncertainty associated with axial burnup shapes is factored into this curve. The calculations account for maximum pool temperature up to 236°F (higher temperatures are conservative for Region 2), which includes maximum abnormal and accident design conditions.

2.1.3 Accident Analysis

The reactivity effects of postulated abnormal or accident conditions have been considered. These include misloading an assembly into an incorrect region (e.g., unburned 4.5 percent enriched fuel into Region 2), dropped fuel assemblies (e.g., inserting an unburned fuel assembly into a blocked location in Region 2, completing a four-out-of-four geometry), pool temperature variations, and dropping heavy loads into the regions. None of these violate the acceptance criterion of 0.95. For these accidents, the assumption is made (in accordance with staff guidelines) that it is not necessary to assume concurrently two unlikely independent events to ensure protection against a criticality accident (double contingency principle). Therefore, the assumption of the minimum boron concentration in the fuel pool required by Technical Specifications (800 ppm) is used.

2.1.4 Procedures

During a core reload in which fuel is to be placed in Region 2, the procedures which are being written for the new fuel pool operations (in response to NRC staff questions) will indicate that the freshly discharged assemblies from the reactor will, as a preferred method, be placed first in Region 1 and later transferred, after careful checks of burnup records, to Region 2. There will be direct transfers to Region 2 (from the core) only if no alternative exists. This procedure is intended to preclude loading errors. Procedures are also being written to address related questions of recordkeeping and accountability. The staff finds these concepts acceptable.

2.1.5 Technical Specifications

In addition to the changes to the fuel pool, it is also proposed that the following related Technical Specifications and corresponding Bases be changed or added.

3/4.9.16 and Basis: This specifies that the boron concentration of the pool water be at least 800 ppm when a shielded cask is to be moved. The 800 ppm value is consistent with the value used in the relevant accident analysis and is thus acceptable.

3/4.9.17 and Basis: This similarity specifies the 800 ppm boron concentration is required in Region 2 when a fuel assembly is moved over the Region 2 racks. This too is consistent with the accident analyses and is acceptable.

3/4.9.18 and Basis: This specification requires the fuel to be loaded into Region 2 (defined via Figure 3.9-2) to meet the burnup vs. enrichment limits of Figure 3.9-1. This is the limit relationship previously discussed and is acceptable.

5.6.1: This specification (in Parts b and c) describes the dimensions and limits of Regions 1 and 2 and is in agreement with the previous discussions. Part a presents the limits for the new (dry) fuel storage racks and is in accordance with the original design limits of these racks. The staff finds these descriptions acceptable.

2.1.6 Conclusions

The staff concludes that the proposed storage racks meet the requirements of General Design Criterion 62 as regards criticality. This conclusion is based on the following considerations:

1. Both regions meet criticality criteria for a specific range of enrichments and burnup combinations.

2. Several staff-accepted spent fuel pool designs involving multiple storage regions have taken credit for burnup in a similar manner.

3. State-of-the-art methods have been used and have been verified by comparison with experiments.

4. Conservative assumptions have been made about the fuel, rack material, dimensional conditions, and the pool conditions.

5. Suitable uncertainties have been considered in determining the multiplication status.

6. Suitable procedures have been developed to minimize misplacement events.

7. Credible accidents and conditions have been considered.

8. The various effective multiplication factors meet staff acceptance criteria.

The staff has also concluded that the modifications to the Millstone Unit 2 Technical Specifications are acceptable to allow operation with the proposed expansion of spent fuel pool storage capacity.

2.2 Spent Fuel Pool Cooling

2.2.1 Decay Heat Loads

NNECo's calculated spent fuel discharge heat load to the pool, which was determined in accordance with Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," and the Standard Review Plan Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System," indicates that the expected maximum normal heat load following the last refueling is 15.2 MBTU/Hr. This heat load results in a maximum bulk pool temperature of 188°F with operation of one spent fuel pool cooling train. With the single active failure of one spent fuel pool cooling system pump, the flow from the operating pump can be pumped through both spent fuel pool cooling system heat exchangers with a resulting maximum bulk pool water temperature of 160°F. Since this is higher than the 140°F specified in the Standard Review Plan (SRP) Section 9.1.3, NNECo agreed, in their November 27, 1985 letter, to submit appropriate changes to the Technical Specifications during the second quarter of 1986.

Preliminary calculations made by NNECo indicate that a requirement of 21 days (504 hours from the reactor going subcritical) of decay time for the off-loaded fuel would satisfy the SRP requirement. The forthcoming Technical Specification change would add a requirement that the decay time for the fuel off-loaded during a refueling outage will be considered along with cooling system operability before proceeding with a reactor startup.

This change will provide assurance of the availability of a loop of the shutdown cooling system for spent fuel pool cooling in the event that the pool temperature rises above 140°F. After 21 days, the bulk pool water temperature for the normal heat load case is less than 140°F with one pump and both heat exchangers.

The expected maximum heat load following a full core discharge is 33.0 MBTU/Hr. This abnormal heat load results in a maximum bulk pool temperature of 212°F with both spent fuel pool cooling trains operating. However, with the reactor defueled, the shutdown systems are available to supplement the spent fuel pool cooling systems and maintain the pool water temperature below 140°F.

2.2.2 Spent Fuel Pool Cooling System

The spent fuel pool water is cooled by the reactor building component cooling water system, which in turn is cooled by the service water system. The licensee proposed no modifications to these two systems as part of this spent fuel pool expansion project. Although each spent fuel pool heat exchanger has a design capacity of only 5.65 MBTU/Hr, an independent pool water temperature calculation was performed which verified that the pool water temperature will remain within acceptable limits. Therefore, the staff concludes that this system is capable of removing the additional heat in conjunction with one loop of the shutdown cooling system.

2.2.3 Conclusion

Based on the staff review of the proposed spent fuel pool expansion program for Millstone 2, the staff concludes the following:

1. The calculated maximum normal and abnormal heat loads have been properly determined and are acceptable.

2. The existing spent fuel pool cooling capability can maintain the fuel pool water temperature for the maximum normal and abnormal heat loads within the limits indicated in the criteria of SRP Section 9.1.3 provided Technical Specification changes are submitted in accordance with NNECO's November 27, 1985 letter.

Therefore, the proposed spent fuel expansion is acceptable.

2.3 Installation of Racks and Load Handling

NNECo provided drawings in their September 16, 1985 submittal which show the sequence of events for the relocation of spent fuel within the spent fuel pool, the order of rack replacement, and the load paths which will be used for moving the racks. No spent fuel storage racks will be carried over spent fuel or racks containing spent fuel. Thus, a load drop of a spent fuel storage rack would not damage spent fuel and therefore there will be no resulting radiological consequences. The postulated rack drop would not change the separation distance between the stored fuel assemblies or the concentration of boron. Therefore, the margin of safety to criticality will not be affected by a rack drop accident. The racks will enter and exit the building by means of a temporary construction crane. The racks will be maneuvered inside the fuel building by means of the auxiliary building crane. There are no safety related components on or above the fuel handling floor. Therefore, a drop of a spent fuel rack will not have any adverse consequences as identified in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and is, therefore, acceptable.

2.4 Structural Design

The structural aspects of the proposed modification are based on a review performed by the staff's consultant, Franklin Research Center (FRC). The FRC Technical Evaluation Report, TER-C5506-586, is appended to this Safety Evaluation as Appendix A.

2.4.1 Description of the High Density Racks and Spent Fuel Pool.

The new high density racks are stainless steel "egg-crate" structures with each cell capable of holding one spent fuel assembly. A typical rack consists of approximately 60 to 100 cells. The racks are each free-standing on the pool floor with a gap between the racks and between the racks and pool wall so as to preclude impact during earthquakes. The weight of the rack and fuel is transmitted to the floor of the pool through supporting legs.

Spent fuel storage is located in the auxiliary building, which is a multi-story, reinforced structure with flat slabs and shear walls.

2.4.2 Applicable Codes, Standards and Specification.

Load combinations and acceptance criteria were found to be consistent with those in the "Staff Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and amended January 18, 1979. The existing concrete pool structure was evaluated for the new loads and the results were compared against ACI-349-80.

2.4.3 Seismic and Impact Loads.

Seismic loads for the rack design are based on the original design floor acceleration response spectra calculated for the plant at the licensing stage. The seismic loads were applied to the model in three orthogonal directions. Loads due to a fuel bundle drop accident were considered in a separate analysis. The postulated loads from these events were found to be acceptable.

2.4.4 Design and Analysis Procedures

a. Design and Analysis of the Racks

A non-linear time-history analysis of the rack module model was performed. The model included mass, spring, damping, and gap elements and accounts for sliding, tipping and potential rack-to-rack interaction in order to determine stresses and strains within the racks. A three dimensional finite element model was used to determine a final stress in the rack modules. This finite element model was also used to generate an equivalent stiffness for the simplified two dimensional non-linear dynamic model.

Calculated stresses for the rack components were found to be well within allowable limits. The racks were found to have adequate margins against tipping and impacting. An analysis was conducted to assess the potential effects of a dropped fuel assembly on the racks and results were considered satisfactory.

An analysis was conducted to assess the potential effects of a stuck fuel assembly causing an uplift load on the racks and a corresponding downward load on the lifting device as well as a tension in the fuel assembly. Resulting stresses were found to be within acceptance limits.

b. Analysis of the Pool Structures

The Millstone 2 auxiliary building, which houses spent fuel pool, is a multi-story concrete structure. The fuel storage area consists of a reinforced concrete pool lined with a 1/4-inch thick stainless steel liner.

The analysis of the spent fuel pool and associated components of the auxiliary building to accommodate the loadings associated with increased storage capacity was accomplished with the use of a large finite element model. The finite element model included the entire spent fuel pool, fuel transfer canal, and cask laydown area. Original plant resource spectra and damping values were used in consideration of the seismic loadings. The concrete sections were checked against criteria set forth in ACI-349-80. Consequently, the existing spent fuel pool structure has been determined to safely support the loads generated by the new fuel racks.

2.4.5 Conclusion

Based on the above, the staff concludes that the proposed rack installation will satisfy the requirements of 10 CFR Part 50, Appendix A (GDC 2, 4, 61 and 62), as applicable to structures.

2.5 Materials

The spent fuel racks in the proposed expansion would be constructed entirely of Type 304 stainless steel, except for the neutron absorber material in the poisoned modules. The existing spent fuel pool liner is constructed of stainless steel. The high density poisoned spent fuel storage racks will utilize Boroflex sheets as a neutron absorber. Boroflex consists of boron carbide powder in a silicone based polymeric homogeneous matrix. The spent fuel storage rack configuration is composed of individual storage cells interconnected to form an integral structure.

The space which contains the Boroflex is vented to the pool. Venting will allow gas generated by the chemical degradation of the silicone polymer binder during heating and irradiation to escape, and will prevent bulging or swelling of the stainless steel tube.

The pool contains oxygen-saturated demineralized water. The water chemistry control of the spent fuel pool has been reviewed and found to meet NRC recommendations.

2.5.1 Evaluation

The pool liner, rack lattice structure, and fuel storage tubes are stainless steel which is compatible with the storage pool environment. In this environment of oxygen-saturated borated water, the corrosive deterioration of the Type 304 stainless steel should not exceed a depth of 6.00×10^{-5} inches in 100 years, which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, and the Inconel and the Zircaloy in the spent fuel assemblies will not be significant because the materials are either similar or the materials are protected by highly passivating oxide films and are, therefore, at similar potentials. The Boroflex is composed of non-metallic materials and therefore, will not develop a galvanic potential in contact with the metal components. Boroflex has undergone extensive testing to study the effects of gamma irradiation in various environments, and to verify its structural integrity and suitability as a neutron absorbing material. The evaluation tests have shown that the Boroflex is unaffected by the pool water environment and will not be degraded by corrosion [1]. Tests were performed at the University of Michigan,[2] exposing Boroflex to 1×10^{11} rads of gamma radiation with substantial concurrent neutron flux in deionized water. These tests indicate that Boroflex maintains its neutron attenuation capabilities after being subjected to an environment of gamma irradiation. Irradiation will cause some loss of flexibility, but will not lead to break up of the Boroflex.

The annulus space in each cell which contains the Boroflex is vented to the pool. Venting of the annulus will allow gas generated by the chemical degradation of the silicone polymer binder during heating and irradiation to escape, and will prevent bulging and swelling of the inner stainless steel tube.

The tests[1] have shown that neither irradiation, environment, nor Boroflex composition has a discernible effect on the neutron transmission of the Boroflex material. The tests also show that Boroflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions are reached regarding the leaching of elemental boron from the Boroflex. Boron carbide of the grade normally present in the Boroflex will typically contain 0.1 wt percent of soluble boron. The test results have confirmed the encapsulation function of the silicone polymer matrix in preventing the leaching of soluble species from the boron carbide.

To provide added assurance that no unexpected corrosion or degradation of the materials will compromise the integrity of the racks, the licensee has committed to conduct a long term fuel storage cell inservice surveillance program. Surveillance samples are in the form of removable stainless steel clad Boroflex sheets, which are proto-typical of the fuel storage cell walls. These specimens will be removed and examined periodically over the expected service life.

2.5.2 Conclusion

Based on the above, the staff concludes that the corrosion that will occur in the spent fuel storage pool environment should be of little significance during the life of the plant. Components in the spent fuel storage pool are constructed of alloys which have a low differential galvanic potential between them and have a high resistance to general corrosion, localized corrosion, and galvanic corrosion. Tests under irradiation and at elevated temperatures in deionized water indicate that the Boroflex material will not undergo significant degradation during the expected service life.

The staff further concludes that the environmental compatibility and stability of the materials used in the expanded spent fuel storage pool is adequate, based on the test data cited above and the actual service experience in operating reactors.

The staff has reviewed the surveillance program and concludes that the monitoring of the materials in the spent fuel storage pool, as proposed by NNECo, will provide reasonable assurance that the Boroflex material will continue to perform its function for the design life of the pool. The materials surveillance program delineated by NNECo will reveal any instances of deterioration of the Boroflex that might lead to the loss of neutron absorbing power during the life of the new spent fuel racks. The staff does not anticipate that such deterioration will occur. This monitoring program will, however, ensure that in the unlikely situation that Boroflex deterioration occurs NNECo and the NRC will be aware of it in sufficient time to take corrective action.

Therefore, the staff finds that the implementation of an inservice surveillance program and the selection of appropriate materials of construction by NNECo meet the requirements of 10 CFR Part 50, Appendix A, Criterion 61, having a capability to permit appropriate periodic inspection and testing of components, and Criterion 62, preventing criticality by maintaining structural integrity of components and of the boron neutron absorber and are, therefore, acceptable.

2.5.3 References

1. J. S. Anderson, "Boroflex Neutron Shielding Material -- Product Performance Date," Brand Industries, Inc., Report 748-30-1, (August 1979).

2. J. S. Anderson, "Irradiation Study of Boroflex Neutron Sheilding Materials ," Brand Industries, Inc., Report 748-10-1, (August 1981).

2.6 Spent Fuel Pool Cleanup System

Water chemistry and optical clarity will be maintained by the existing spent fuel pool cleanup system. The cleanup system is a non-safety related system and has been designed to non-Seismic Category I requirements. Isolation capabilities from the Category I portion of the fuel pool cooling system have been provided by Seismic Category I isolation valves. The cleanup system consists of two refueling water purification pumps, two filters, and a demineralizer and associated valves and piping. In addition, the spent fuel pool has been provided with a skimmer pump and two filter assemblies to facilitate the removal of accumulated surface debris. The cleanup system has been designed to process water through the purification loop from the refueling pool and refueling water storage tank.

The staff expects only a small increase in radioactivity and other contaminants to be released to the pool water as a result of the proposed modification, and concludes that the spent fuel pool cleanup system is adequate for the proposed modification and will continue to keep the concentration of radioactivity and other contaminants in the pool water to acceptably low levels.

2.7 Occupational Radiation Exposure

The staff has reviewed the licensee's plan for reracking and modifying the spent fuel pool (SFP) with respect to occupational radiation exposure.

NNECo will use divers to disconnect the old racks and components. Because of the exposure to the divers and increased exposure to other personnel, the occupational exposure commitment from the rerack activities will be about 3-8 man-rem. The staff considers this to be a reasonable estimate. This estimate represents a small fraction of the total 500 man-rem burden from occupational exposure at the plant.

The licensee plans to take the following actions to keep occupational radiation exposure to as low as is reasonably achievable (ALARA). These actions include: (a) vacuum cleaning of SFP floors; (b) maximum water shielding to reduce dose rates to divers; (c) underwater radiation surveys; (d) calibrated alarming dosimeters and personnel monitoring for divers; (e) hydrolasing and cleaning of old spent fuel racks; (f) the use of remote operations for rack removal and replacement operations; and (g) visible work zone barriers for diving operations 7 feet from the nearest filled spent fuel rack.

Based on the above actions, the staff concludes that SFP modification exposure to workers is ALARA and is acceptable.

The licensee estimates the additional occupational radiation exposure for operation of the reracked pool at less than 1 man-rem/yr. The licensee also has provided a description of contained and airborne radioactivity sources related to the SFP water and which may become airborne as a result of failed fuel and evaporation. The staff has reviewed these source terms and finds them acceptable. The additional dose is less than 0.2% of the average annual occupational dose of 500 person-rems for all plant operations at Millstone Unit 2. The small increase in radiation dose should not affect the licensee's ability to maintain individual occupational doses to ALARA levels and within the limits of 10 CFR Part 20. Therefore, the staff concludes that storing additional fuel in the SFP will not result in any significant increase in doses received by workers.

2.8 RADIOACTIVE WASTE TREATMENT

Millstone Unit 2 contains radioactive waste treatment systems designed to collect and process the gaseous, liquid, and solid wastes that might contain radioactive material. The radioactive waste treatment systems were evaluated in the Safety Evaluation Report, dated June 1973, in support of the issuance of Operating License No. DPR-65 and in supplements thereto. There will be no change in the radioactive waste treatment systems or in the conclusions given regarding the evaluation of these systems because of the proposed spent fuel pool rerack.

2.8.1 Conclusions

The staff evaluation of the radiological considerations supports the conclusion that the proposed installation of new spent fuel storage racks at Millstone Unit 2 is acceptable because the conclusions of the evaluation of the radioactive waste treatment systems, as found in the Millstone Unit 2 Safety Evaluation Report, are unchanged by the installation of new spent fuel storage racks.

2.9 RADIOLOGICAL CONSEQUENCES OF ACCIDENTS INVOLVING POSTULATED MECHANICAL DAMAGE TO SPENT FUEL

The staff has reviewed the NNECo submittal for the expansion of the storage capacity of the SFP at Millstone Unit 2. The review was conducted according to the guidance of Standard Review Plan 15.7.4, and 15.7.5, NUREG-0612, and Regulatory Guide 1.25 with respect to accident assumptions.

2.9.1 Evaluation

The reracking operations at Millstone 2 will be conducted according to procedural controls and Technical Specifications which prohibit the movement of heavy objects, including fuel racks, over spent fuel stored in the pool.

The conservative bounding case, relative to potential fuel and load handling accidents during and following rerack operations, has been determined to be a spent fuel cask tip/drop in the Millstone 2 SFP. In order to mitigate the radiological consequences of such an accident, there is a Technical Specification requirement for a specified decay time of 120 days for fuel stored within a distance L from the center of the spent fuel cask set down area. The distance L is equal to the major dimension of the shielded cask. Assuming conservatively that the 587 assemblies within the distance L are damaged during a cask drop, the dose estimates, using NUREG-0612, are a small fraction (less than 3 rem thyroid and 1 rem whole body) of the guideline values specified in Standard Review Plans 15.7.4 and 15.7.5.

2.9.2 Conclusion

Based on the evaluation by both the staff and NNECo of the radiological consequences of potential fuel and load handling accidents at Millstone 2 SFP, the doses would be well within 10 CFR Part 100 values for the proposed design. The staff concludes, therefore, that the proposed modifications are acceptable.

3.0 CONCLUSIONS

In conclusion, the staff finds the proposed changes to the Millstone Unit 2 Technical Specifications to be acceptable and, 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, and (2) such activities will be conducted in compliance with the

Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Attachment: Technical Evaluation Report prepared by Franklin Research Center

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