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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. TO LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

Introduction

By submittal dated August 17, 1977, as supplemented by letters dated September 12 (license amendment request), December 8, December 14, 1977, January 3, January 30, March 10, March 16, and March 28, 1978, Northern States Power Company (NSP or licensee) requested NRC approval of a proposed increase in the spent fuel pool (SFP) storage capacity at the Monticello Nuclear Generating Plant from 740 to 2237 fuel assemblies. Notice of Consideration of Proposed Modification to Facility Spent Fuel Storage Pool issued to Northern States Power Company was published in the FEDERAL REGISTER on September 19, 1977 (42 FR 46963).

Discussion

The proposed amendment would modify paragraph 2.B of the license to delete extraneous or superseded information and to incorporate information related to this licensing action to increase the spent fuel storage capacity. Specifically, the reference to NSP letters dated May 30, 1975 and July 1, 1975 has been deleted. The May 30, 1975 letter, which dealt with an increase in the storage of special nuclear material from 2300 to 3200 kilograms was addressed in Amendment 10 of July 8, 1975 but has been superseded by this latest action. The letter of July 1, 1975, which was addressed by License Amendment No. 11 dated September 17, 1975, only concerned sealed radioactive source leak testing and is not pertinent to the use of special nuclear material as fuel. Source leak testing is addressed as Technical Specifications 3.11 and 4.11. The proposed change to the Specification deletes references which are no longer applicable and adds references to reflect the latest licensing action.

We also have reviewed in detail the engineering and environmental aspects of this proposed action. The remainder of this evaluation, and the Environmental Impact Appraisal which follows, deals with these aspects.

The SFP at the Monticello facility contains 616 spent fuel assemblies at the present time. Spent fuel has been stored in the pool since the first core refueling. Since there is storage space for only 740 spent fuel assemblies and since the core contains 484 fuel assemblies, Monticello cannot, with the

existing spent fuel storage racks, accommodate removal and storage in the SFP of all the fuel assemblies in the core.

The proposed increase in spent fuel storage capacity from 740 fuel assemblies will (1) provide storage for all spent fuel assemblies removed from the core between the present time and 1990, (2) provide sufficient additional fuel assembly storage capacity so that the entire core (484 fuel assemblies) can be removed from the reactor vessel and stored in the SFP to about 1987, and (3) continue to accommodate one fuel assembly shipping cask for offsite shipping of spent fuel assemblies from the Monticello SFP when offsite spent fuel shipment is resumed at some indefinite future date within the next 12 years.

Our review and evaluation considered the following:

1. Structural adequacy of the proposed spent fuel racks and pool

2. Criticality considerations

3. Spent fuel pool cooling capacity

4. Fuel handling and installation of the modified spent fuel racks

5. Occupational radiation exposure and radioactive waste treatment

Evaluation

1. Structural Adequacy of the Proposed Spent Fuel Racks and Pool

The current fuel storage racks have a storage capacity of 740 fuel assemblies. The proposed SFP modification consists of installation of new fuel storage modules. Each module is composed of fuel storage tubes arranged in a 13 X 13 array. Thirteen such modules, one existing fuel storage rack, and two of the existing control rod/defective fuel storage racks will provide storage locations for 2237 assemblies. The new modules and the three existing racks comprise the proposed High Density Fuel Storage System.

The fuel storage tube is fabricated by forming an outer and inner sheet of 304 stainless steel sandwiching a core of Boral (clad by aluminum) into a single rectangular tube. The inner and outer walls of the storage tube are welded together at each end, which isolated the Boral from direct contact with fuel pool water. Except for the Boral and aluminum, all structural material used in fabrication of the new modules is type 304 stainless steel.

The module design, material, and fabrication are in accordance with the requirements set forth in Section III, Subsection NF of the ASME Boiler and Pressure Vessel Code. The modules are designed to remain within Code allowed stress limits for both Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) conditions. The modules were analyzed as cantilever beams attached to a rigid base using qualified computer codes to derive loads in a water filled rectangular pool. These loads were derived for horizontal and vertical accelerations specified in the General Electric BWR Systems Department seismic criteria document and were compared to the allowable stresses. The analysis indicates that the derived loads do not overstress the modules since the Monticello accelerations at the fuel pool elevation are 0.2g (SSE) and the analysis was performed for 3g (SSE). Added damping due to fluid effects was conservatively neglected. Stresses due to seismic loading in the three orthogonal directions were combined by the Square Root of the Sum of the Squares Method as outlined in Regulatory Guide 1.92.

The module design is free-standing, transferring shear forces to the pool slab through friction resistance provided by the normal force of the weight of the module through the support columns resting on the pool floor liner. NSP has used a minimum value for the coefficient of friction in the sliding analysis, a value which was verified by recent tests of stainless steel materials. The coefficient of friction used was sufficient to ensure that sliding will not occur for earthquake motions corresponding to OBE and SSE. An additional non-linear analysis for sliding was performed to determine relative displacements if the coefficient of friction were less than the minimum value used. This analysis gives added assurance that there should be no interaction between modules as a consequence of the SSE.

The NSP has re-evaluated the fuel pool structural capacity for the High Density Fuel Storage System and has shown that the existing structure is capable of supporting the increased load.

Since the possibility of long term storage of spent fuel exists, we are investigating the effects of the pool environment on the modules, fuel cladding and pool liner. Based upon our preliminary review and previous operating experience, we have concluded that at the pool temperature and the quality of the demineralized water, and taking no credit for inservice inspection, there is reasonable assurance that no significant corrosion of the modules, the fuel cladding or the pool liner will occur over the lifetime of the plant. However, if the results of the current generic review indicate that additional protective measures are warranted to protect the modules, the fuel cladding and/or the liner from the effects of corrosion, the necessary steps and/or inspection programs will be required to assure that an acceptable level of safety is maintained. Any conceivable problems which could be uncovered are of a long term nature and warrant no need for immediate concern.

The criteria used in the analysis, design, and construction of the High Density Fuel Storage System to account for the anticipated loadings and p[illegible]lated conditions that may be imposed on the structures during their service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the NRC Staff. The new modules meet the standards for seismic Category I components and are designed to maintain the spent fuel assemblies in a safe configuration through all environmental and abnormal loadings. Therefore, we find that the proposed expansion is acceptable from the aspect of mechanical, material, and structural considerations.

2. Criticality Considerations

The proposed spent fuel assembly racks are to be made up of alternating stainless steel containers. Thus, there will be only one container wall between adjacent spent fuel assemblies. Each container wall is to have a core of Boral sandwiched between 0.036 inch insi and 0.090 inch outside stainless steel containers. The containers will be about 14 feet long and will have a square cross section with an outer dimension of 6.563 inches and a total wall thickness of 0.2015 inches. The nominal pitch between fuel assemblies will be 6.563 inches.

The Boral core is made up of a central segment of a 0.056 inch thick dispersion of boron carbide in aluminum. This central segment is clad on both sides with 0.010 inches of aluminum. NSP states that the minimum homogeneous concentration of the boron-ten isotope will be 0.013 grams per square centimeter of the Boral plate. This is equivalent to 0.78×10^{21} boron-ten atoms per square centimeter. These Boral plates are to be sealed between two stainless steel containers by welding.

The NSP fuel pool criticality calculations are based on an unirradiated BWR fuel assembly with no burnable poison and a fuel loading of 15.2 grams of uranium-235 per axial centimeter of fuel assembly. The General Electric Company (GE) performed the criticality analyses for NSP. GE made the calculations with the MERIT Monte Carlo program with cross sections which were processed from ENDF/B-IV data. The accuracy of this calculational method was assessed by using it to calculate the following experiments: (1) thermal reactor benchmark experiments TRX-1 through 4 of the Cross Section Evaluation Work Group; (2) the Babcock and Wilcox UO_2 critical assemblies; and (3) the Oyster Creek BWR experiments with boron curtains. From this qualification program, GE determined that this calculational method underpredicts k_{eff} by 0.5 per cent Δk.

GE first used these computer programs to calculate the neutron multiplication factor for an infinite array of fuel assemblies in the nominal storage lattice at 20°C with the minimum boron concentration in the Boral, i.e., 0.013 grams of boron-ten per square centimeter. Since

the outside dimension of these storage containers is the same as the lattice pitch; i.e., 6.563 inches, GE assumed that this is essentially a close packed configuration and that the average pitch could not be less than 6.563 inches. Thus this calculation was for minimum boron and minimum lattice pitch. GE found the k_∞ for this configuration to be 0.87 after the 0.5 per cent experimental bias was included.

GE then calculated the k_∞'s for the following conditions: (1) increasing the temperature to 65°C; (2) increasing the lattice pitch; (3) locating every four fuel assemblies as close together as possible; and (4) reducing the density of the water. GE found that all of these changes resulted in a decrease in k_∞.

Because of the alternating lattice design, wherein there will be only one storage container for every two fuel assemblies, there will be spaces on the periphery of the rack modules which will not have Boral plates. Thus it will be possible for two rack modules to be put together so that adjacent fuel assemblies will not have a Boral plate between them. GE calculated the effect of these missing Boral plates for the minimum attainable gap between rack modules and found that it would not increase the maximum k_∞ of 0.87. GE also analyzed the situation where a fuel assembly is moved as close as possible to an unpoisoned location on the periphery of a filled storage rack and found that the neutron multiplication factor would not increase above 0.90.

In its February 13, 1978 submittal, NSP stated that neutron source testing at the Monticello Plant will be performed to verify the presence of the Boral plates in the fabricated fuel storage modules. NSP also stated that since calculations have demonstrated a K_{eff} of less than 0.90 at a 95 percent confidence level with any four complete Boral plates missing, any module with more than four missing Boral plates will be rejected.

We have evaluated the results of these criticality calculations, and find that all factors that could affect the neutron multiplication factor in this pool have been conservatively accounted for and that the maximum neutron multiplication factor in this pool with the proposed racks will not exceed 0.95. This is NRC's acceptance criterion for the maximum (worst case) calculated neutron multiplication factor in a spent fuel pool. This 0.95 acceptance criterion is based on the uncertainties associated with the calculational methods and provides sufficient margins to preclude criticality in the fuel pool. We also find that the uncertainty in the results of the criticality calculations will increase if the fuel loading in the fuel assemblies is increased. We, therefore, have requested, and NSP has agreed to provide for the NRC's review and approval, a revised criticality analysis of the fuel pool whenever and prior to storing fuel assemblies in it which have a fuel loading greater than 15.2 grams of uranium-235 in any axial centimeter.

With regard to NSP's onsite neutron radiography testing of the Boral plates, we find that with the quality assurance program procedures in effect there should be no Boral plates missing from the prescribed locations in the fabricated fuel storage modules. If NSP finds any Boral plates missing they should specifically note and document this finding in the test report.

In summary, we find that when any number of the fuel assemblies which NSP described in these submittals, which have no more than 15.2 grams of uranium-235 per axial centimeter of fuel assembly are loaded into the proposed racks, the neutron multiplication factor will be less than 0.95.

On this basis, we conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the use of the proposed racks.

3. Spent Fuel Pool Cooling Capacity

The licensed thermal power for Monticello is 1670 MWt. In their submittal, NSP assumed an 18 month refueling cycle. This will require the replacement of about 141 of the 484 assemblies in the core every 18 months. In its September 12, 1977 submittal, NSP assumed a four day (96 hour) time interval between reactor shutdown and the time 141 spent fuel assemblies have been transferred to the spent fuel pool and a 150 hour time interval between reactor shutdown and the time a full core offload is completed. For these cooling times, NSP states that the maximum heat load to the SFP due to normal eighteen months refuelings will be 11.3×10^6 BTU/hr and that the maximum heat load due to a full core offload will be 27.2×10^6 BTU/hr.

As indicated in Table 10-2-1 of the FSAR, the SFP cooling system consists of two pumps and two heat exchangers in parallel. Each pump is designed to pump 450 gpm (2.25×10^5 pounds per hour). Each heat exchanger is designed to transfer 2.87×10^6 BTU/hr from 125°F fuel pool water to the 95°F water in the Reactor Building Closed Cooling Water System. For higher heat loads, such as the full core offload, NSP states that the residual heat removal system (RHR) will be used in conjunction with the SFP cooling system. In their September 12, 1977 submittal, NSP stated that the RHR system has a capacity for removing 57.5×10^6 BTU/hr.

Section 10 of the FSAR indicates that instrumentation is provided in the spent fuel cooling system which will monitor pool water level, water temperature, and system flow. A loss of system flow will actuate an alarm in the reactor building.

Section 10 of the FSAR describes the 3500 gpm capacity Emergency Service Water System. Since this is piped to the RHR system, it could also be used for emergency makeup water for the SFP. NSP's calculated decay heat loads are based on a specific operating power of 30 KW/kgU. This is conservative because the presently licensed average, specific power of the Monticello Plant is about 19 KW/kgU. NSP's assumed fuel pool transfer times of 96 hours for 141 fuel assemblies and 150 hours for 484 assemblies are conservatively short. NSP used the ORIGEN computer program to calculate the decay heat loads. NSP's calculated decay heats are larger and hence more conservative than those obtained by using the method identified on pages 9.2.5-8 through 14 of the NRC Standard Review Plan.

We find that the maximum incremental heat load in this SFP that will be added by increasing the number of fuel assemblies stored in the pool from 740 to 2,237 assemblies will be 1.3×10^6 BTU/hr. This is the difference in peak heat loads for full core offloads that essentially fill the present and modified pools.

NSP's calculated fuel pool outlet water temperatures are consistent with the stated flow rates and the design of the heat exchangers. We calculate that with both spent fuel cooling pumps operating at design capacity and with NSP's peak heat load for any refueling (i.e., 11.3×10^6 BTU/hr), the maximum SFP water temperature will be less than 140°F. The 57.5×10^6 BTU/hr capacity of the RHR system is adequate to remove the maximum full core heat load of 27.2×10^6 BTU/hr and maintain the SFP outlet water temperature below 125°F.

Assuming a maximum fuel pool temperature of 150°F, the minimum possible time to achieve bulk pool boiling after any credible accident will be 5.6 hours. After bulk boiling commences, the maximum evaporation rate will be 56 gpm. We find that 5.6 hours would be sufficient time for NSP to establish a 56 gpm makeup rate from the Emergency Service Water System. We also find that under bulk boiling conditions the temperature of the fuel will not exceed 350°F. This is an acceptable temperature from the standpoint of fuel element integrity and surface corrosion.

We find that the present cooling capacity in the SFP of the Monticello Plant will be sufficient to handle the incremental heat load that will be added by the proposed modifications. We also find that this incremental heat load will not alter the safety considerations of SFP cooling from that which we previously reviewed and found to be acceptable. We conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the use of the proposed deisgn.

4. Fuel Handling and Installation of the Modified Spent Fuel Racks

In its September 12, 1977 submittal, NSP stated the following:

1) The reactor building crane has been modified to satisfy Regulatory Guide 1.104 as applicable to operating plants. This modification was reviewed and found acceptable by the NRC staff;

2) Procedures will be written that will prevent loads which weigh more than a fuel assembly from being moved over new or spent fuel during the rack replacement program;

3) Administrative controls will be established which will prohibit the movement of loads which weigh more than a fuel assembly from being moved over new or spent fuel during the rack replacement; and

4) Proposed modification will not increase the consequences or probability of the design basis fuel handling accident.

NSP has upgraded the Monticello reactor building crane to satisfy the provisions of Regulatory Guide 1.104, as far as practical for an operating facility. The use of this upgraded crane along with NSP's stipulation that racks will not be taken over fuel assemblies present in the pool will make the probability for an empty rack falling on a loaded rack in the pool acceptably small.

After the racks are installed in the pool, the fuel handling procedures in and around the pool will be the same as those procedures that were in effect prior to the proposed modifications.

We conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the installation and use of the proposed racks.

5. Occupational Radiation Exposure and Radioactive Waste Treatment

We have reviewed the licensee's plan for the removal, disassembly and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. The occupational radiation exposure for this operation is estimated by the licensee to be about 22 man-rem. We consider this to be a conservative estimate. This operation is expected to be performed only once during the lifetime of the station and will therefore represent a small fraction of the total man-rem burden from occupational exposure.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee and by utilizing relevant assumptions for occupancy times and for dose rates in the spent fuel area from radionuclide concentrations in the SFP water. The spent fuel assemblies themselves contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. The occupational radiation exposure resulting from the proposed action represents a negligible burden. Based on present and projected operations in the SFP area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at this facility. The small increase in radiation exposure should not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable and within the limits of 10 CFR 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. The waste treatment systems were evaluated in the Safety Evaluations (SE) dated March 1970 and February 1973. Although there have been system improvements, such as the installation of an augmented offgas system, since the issuance of the latest SE, there will be no change in the waste treatment systems or in the conclusions of the evaluation of these systems, as described in Section 11.0 of the SE, because of the proposed modification.

Our evaluation supports the conclusion that the proposed modification to the SFP at Monticello is acceptable because:

(1) The increase in occupational radiation exposure to individuals due to the storage of additional fuel in the SFP would be negligible.

(2) The installation and use of the new fuel racks does not alter the consequences of the design basis accident for the SFP, i.e., the rupture of a fuel assembly and subsequent release of the assembly's radioactive inventory within the gap.

(3) The overhead handling system is provided with a sufficient degree of redundancy to preclude cask and/or load handling accidents.

Conclusion

We have concluded, based on the considerations discussed above, that: (i) 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 14, 1978