Docket No.: 50-244

December 16, 1985

Mr. Roger W. Kober, Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Kober:

SUBJECT: STORAGE OF CONSOLIDATED FUEL

Re: R. E. Ginna Nuclear Power Plant

The Commission has issued the enclosed Amendment No.12 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your application dated February 27, 1985 as supplemented June 10, June 26 and July 11, 1985.

The amendment approves changes to Technical Specifications to authorize storage of consolidated fuel canisters in the existing spent fuel pool and to authorize use of the auxiliary building crane to move consolidated fuel canisters. The portion of your request to increase the capacity of the pool will be handled by a separate licensing action.

A copy of our related Safety Evaluation and of the notice of issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED BY

George E. Lear, Director
PWR Project Directorate #1
Division of PWR Licensing-A, NRR

Enclosures:
1. Amendment No.12 to License No. DPR-18
2. Safety Evaluation
3. Notice

cc w/enclosure:
See next page

DISTRIBUTION:
SEE ATTACHED PAGE

With noted changes; No issuance until EA is published in Fed Reg. & no requests for hearing have been received after notice period expires.

GL
PWR#1/DPWR-A
GLear
12/10/85

GL
George Lear
D/PD-1
12/16/85

PWR#1:DPWR-A	BWR#1/DBWR	BWR#1/DBWR	OELD	BWR#1/DBWR
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Mr. Roger W. Kober
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.12
License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Rochester Gas and Electric Corporation (the licensee) dated February 27, 1985, as supplemented June 10, June 26 and July 11, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. DPR-18 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No.12 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George E. Lear, Director
PWR Project Directorate #1
Division of PWR Licensing-A, NRR

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 16, 1985.

ATTACHMENT TO LICENSE AMENDMENT NO. 12

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1-8	1-8
3.11-2	3.11-2
3.11-4	3.11-4
5.4-2 to 5.4-4	5.4-2 to 5.4-4
	5.4-4a

1.18 Dose Equivalent I-131

The dose equivalent I-131 shall be that concentration of I-131 which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The dose conversion factors used for this calculation shall be those for the adult thyroid dose via inhalation, contained in NRC Regulatory Guide 1.109 Rev. 1 October 1977.

1.19 Reportable Event

A Reportable Event shall be any of those conditions specified in Section 50.73 to 10CFR Part 50.

1.20 Canisters Containing Consolidated Fuel Rods

Canisters containing consolidated fuel rods are stainless steel canisters containing the fuel rods of no more than two fuel assemblies which have decayed at least five years and are capable of being stored in a storage cell of the spent fuel pool.

e. Charcoal adsorbers shall be installed in the ventilation system exhaust from the spent fuel storage pit area and shall be operable.

3.11.2 Radiation levels in the spent fuel storage area shall be monitored continuously.

3.11.3 A load in excess of one fuel assembly and its handling tool shall never be stationed or permitted to pass over storage racks containing spent fuel.

3.11.4 The spent fuel pool temperature shall be limited to 150^0F.

3.11.5 The restriction of 3.11.3 above shall not apply to the movement of cannisters containing consolidated fuel rods if the spent fuel rack beneath the transported canister contain only spent fuel that has decayed at least 60 days since reactor shutdown.

Basis

Charcoal adsorbers will reduce significantly the consequences of a refueling accident which considers the clad failure of a single irradiated fuel assembly. Therefore, charcoal adsorbers should be employed whenever irradiated fuel is being handled. This requires that the ventilation system should be operating and drawing air through the adsorbers.

The desired air flow path, when handling irradiated fuel, is from the outside of the building into the operating floor area, toward the spent fuel storage pit, into the area exhaust ducts, through the adsorbers, and out through the ventilation system exhaust to the facility vent. Operation of a

The spent fuel pool temperature is limited to 150°F because if the spent fuel pool cooling system is not at that temperature, sufficient time (approximately 7 hours) is available to provide backup cooling, assuming the maximum anticipated heat load (full core discharge & previously stored fuel), until a temperature of 180°F is reached, the temperature at which the structural integrity of the pool was analyzed and found acceptable.

The requirement of 3.11.5 insures that should a handling accident occur during the movement of a consolidated fuel cannister (as described in 5.4.) the dose at the exclusion area boundary would satisfy the requirements of 10CFR100.

References

(1) FSAR - Section 9.3-1

(2) ANS-5.1 (N 18.6), October 1973

5.4.4 Cannisters containing consolidated fuel rods may be stored in either Region 1 or 2 provided that:

a. the average burnup and initial enrichment of the fuel assemblies from which the rods were removed satisfy the requirements of 5.4.2 and 5.4.3 above, and

b. the average decay heat of the fuel assembly from which the rods were removed is less than 2150 BTU/hr

5.4.5 The requirements of 5.4.4a may be excepted for those consolidated fuel assemblies of Region RGAF2.

5.4.6 The spent fuel storage pit is filled with borated water at a concentration to match that used in the reactor cavity and refueling canal during refueling operations whenever there is fuel in the pit.

Basis

The center to center spacing of Region 1 insures that Keff $\leq$ 0.95 for the enrichment limitations specified in 5.4.2[1], and for a postulated missile impact the resulting dose at the EAB would be within the guidelines of 10CFR100[2].

In Region 2, Keff $\leq$ 0.95 is insured by the addition of fixed neutron poison (boraflex) in each of the Region 2 storage locations, and a minimum burnup requirement as a function of initial enrichment for each fuel assembly design. The 60 day cooling time requirement insures that for a postulated missile impact the resulting dose at the EAB would be within the guidelines of 10CFR100.

The two curves of Figure 5.4-2 divide the fuel assembly designs into two groups. The first group is all fuel delivered prior to January 1, 1984. This incorporates all Exxon and Westinghouse HIPAR designs used at Ginna.[4] The second curve is for the Westinghouse Optimized Fuel Assembly design delivered to Ginna beginning in February 1984.[3]

The assembly average burnup is calculated using INCORE generated power sharing data and the actual plant operating history. The calculated assembly average burnup should be reduced by 10% to account for uncertainties. An uncertainty of 4% is associated with the measurement of power sharing. The additional 6% provides additional margin to bound the burnup uncertainty associated with the time between measurements and updates of core burnup. The curves of Figure 5.4-2 incorporate the uncertainties of the calculation of assembly reactivity.[3]

The calculations of fuel assembly burnup for comparison to the curves of Figure 5.4-2 to determine the acceptability for storage in Region 2 shall be independently checked. The record of these calculations shall be kept for as long as fuel assemblies remain in the pool.

The fuel storage cannisters are designed so that, normally, they can contain the equivalent number of fuel rods from two fuel assemblies in a close packed array, and can be stored in either Region 1 or Region 2 rack locations. The close packed array will insure the K_∞ of the rack configuration containing any number of cannisters will be less than that for stored fuel assemblies at the same burnup and initial enrichment. The exception of paragraph 5.4.5 is possible because the consolidated configuration is substantially less reactive than that of a fuel assembly. The maximum decay heat requirement will insure that local and film boiling will not occur between the close packed fuel rods if the pool temperature is maintained at or below 150°F. The decay heat of the assembly will be determined using ANS 5.1, ASB 9-2 or other acceptable substitute standards.

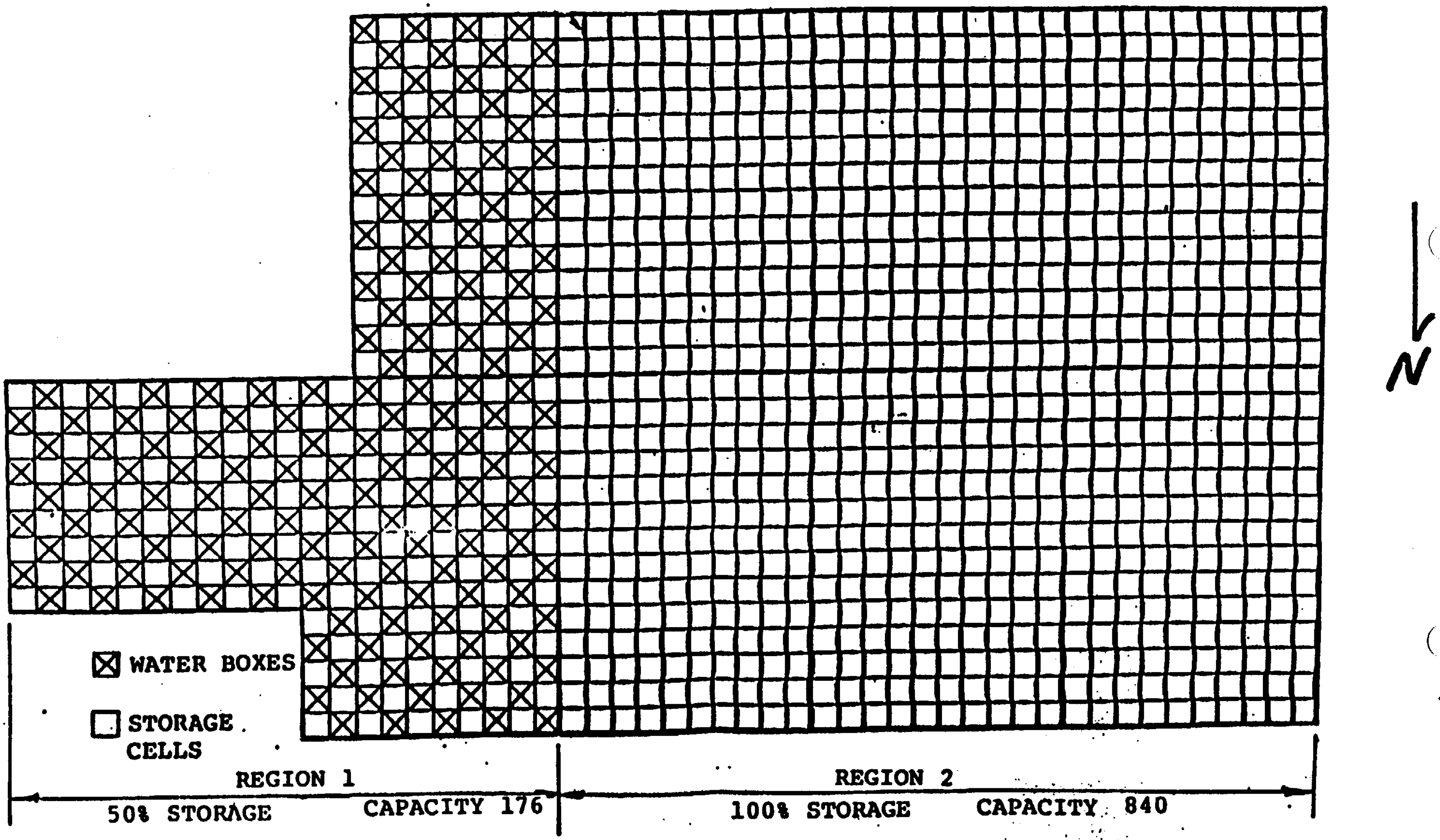
With the addition of the storage of consolidated fuel cannisters, the theoretical storage capacity of the pool would be increased to 2032 fuel assemblies (2x1016). However, due to limitation on the heat removal capability of the spent fuel pool cooling system, the storage capacity is limited to 1016 fuel assemblies.[5]

References

1. Letter, J.E. Maier to H.R. Denton, January 18, 1984.
2. Letter J.E. Maier to H.R. Denton, January 18, 1984.
3. Criticality Analysis of Region 2 of the Ginna MDR Spent Fuel Storage Rack, Pickard, Lowe and Garrick, Inc. March 8, 1984.
4. Letter, T.R. Robbins, Pickard, Lowe and Garrick, Inc. to J.D. Cook, RG&E March 15, 1984.
5. Letter, D.M. Crutchfield to J.E. Maier, November 5, 1981.

FIGURE 5.4-1

SPENT FUEL STORAGE RACKS



*TOTAL CAPACITY 1016 Fuel Assemblies

*Total Capacity includes provisions for storage of consolidated fuel.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO.12 TO FACILITY OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated February 27, 1985 as supplemented on June 10, June 26 and July 11, 1985, Rochester Gas and Electric Corporation (RG&E) requested a Technical Specification change to allow the storage of consolidated fuel in the spent fuel pool at Ginna Nuclear Power Plant. The proposed storage of consolidated spent fuel involves placing spent fuel containing, at most, all the rods from two standard spent fuel assemblies into one canister. Also, a new Technical Specification, 3.11.5, is to be established allowing the movement of canisters containing consolidated fuel rods over stored spent fuel which has decayed for at least 60 days since reactor shutdown.

2.0 DISCUSSION AND EVALUATION

2.1 Criticality Considerations

The consolidation process could theoretically double the capacity of the spent fuel storage facility, however, due to heat removal limitations the maximum total storage planned is 1,253 fuel assemblies by the year 2014. Consolidation in the context of this application means that the fuel rods of two fuel assemblies (358 rods) are stored in a stainless steel canister (in a packed triangular array) capable of being stored in a storage cell. The canister will accept only undamaged fuel rods. Bowed, broken or otherwise failed fuel rods will be stored first in a stainless steel tube of .75 inch outer diameter. Each canister will accommodate 110 such tubes (Ref. 1).

The design of the Ginna fuel storage racks has been described in an earlier submittal (Ref. 2) and it consists of two regions. Region I is for the storage of unirradiated or low burnup fuel assemblies and Region II is for assemblies which satisfy certain minimum burnup criteria. Region II has been designed as a high density configuration. Rochester Gas and Electric does not have nor do they contemplate to use fuel assemblies with burnable poisons, hence, the fresh assemblies which are the most reactive are to be stored in Region I (Ref. 3).

2.1.1 Analysis Methods

The previous criticality analyses for the Ginna spent fuel pool utilized the LEOPARD (Ref. 4), PDQ07 (Ref. 5) and CINDER (Ref. 6) codes. The Boraflex absorbers on the racks were treated with blackness theory. A similar methodology has been employed for the analysis of the consolidated fuel storage. The validation was based on experiments performed by Babcock and Wilcox Company for this purpose (Ref. 7). The tightly packed fuel rods present two modeling difficulties, namely the high metal/water and the related distortion of the neutron spectrum (compared to the Wigner Wilkins model) and the "rubber bond" method to define the amount of water to be included in the fuel region. This method provides a consistent definition of the fuel region for the analyses of the critical experiments as well. The critical experiments analyzed constituted an adequate test of the component-codes of the model. The results of the analyses were compared to the experiment as well as to KENO-IV calculations from Reference 7. The comparison indicated that the LEOPARD-PDQ07-CINDER model results in a closer prediction than KENO-IV, a previously approved model. Therefore, the LEOPARD-PDQ07-CINDER model is considered validated and acceptable.

The analytical model described above was supplemented with burnup capability in order to be used for spent fuel. The burnup code was part of the spent fuel pool modification resubmittal (Ref. 2) and has been approved. An allowance was made for geometrical manufacturing and thermal deviations. The licensee took advantage of existing calculations for 3.13 w/o U-235 Exxon fuel to support conservative criticality estimates for the West Valley New York fuel assemblies which had an initial enrichment of 2.8 w/o U-235. The calculation yields K_∞ vs burnup which is everywhere less than 0.8. This is true for either 358 rod/canister (i.e., 2 x 179 where 179 is the number of rods/assembly) or a reduced loading of 350 rods/canister (2 x 175). For the fuel rods stored in West Valley the average burnup is about 15,000 MWD/MTU and the K_∞ for consolidated fuel is 0.63. In view of the estimated low values for K_∞ it was not deemed necessary to calculate the specific uncertainties, therefore the corresponding estimates of the fuel pool storage capacity increase of April 2, 1984, were adopted (Ref. 2). The total reactivity adjustment to the calculated value for the consolidated rods is .056 ΔK. The above results do not include the effect of the 2,000 ppm of dissolved boron in the pool water. The results of these estimates assure that for fuel designs satisfying the enrichment-burnup criteria of Figure 5.4-2 of the Ginna Technical Specifications, the K_{eff} criteria of 0.95 is satisfied and, therefore, the proposed consolidated fuel storage is acceptable.

2.1.2 Accident Analysis

The accident analysis performed in the submittals of April 2, 1984 (Ref. 2) and February 23, 1983 (Ref. 8) are also applicable for the case of the consolidated fuel storage with respect to criticality. These submittals analyzed configurations that are analogous to those found in a fully packed canister and demonstrate that criticality cannot occur. In addition, however, the loss of containment of all of the rods in a canister and their subsequent relocation on a uniform and optimum pitch has been analyzed. In the particular case of the Exxon 3.13 w/o U-235 fuel, assuming the presence of 2,000 ppm of boron, the optimum pitch is .632 inches. On such a pitch a square array would be 12 inches on the side with a K_∞ of about 0.95. When the 2,000 ppm boron and a 15,000 MWD/MTU burnup is taken into account, the K_∞ reduces to 0.55. This particular fuel rod arrangement is thought to be extremely unlikely. The staff also has considered the case of a canister being not completely filled with the fuel rods since damaged fuel rods will be stored in a separate canister in individual steel tubes and the remaining undamaged rods may then not fill a canister. However, the neutron leakage from a partially filled canister will be even greater than that of a completely filled canister (i.e., 2 x 179 = 358 rods); hence, for a partially filled canister, the $K_\infty \leq 0.55$. In summary, none of the postulated accidents could result in criticality. In the analysis of the postulated accidents it was assumed that 2,000 ppm boron is diluted in the pool cooling water. Credit for this boron is allowed through the use of the double contingency principle. In view of the above, the accident analysis presented for the Ginna consolidated spent fuel storage is acceptable.

2.1.3 Conclusions

The staff concludes that the spent fuel pool, including the consolidated fuel, meets the General Design Criterion 62 as regarding criticality. This conclusion is based on the following considerations:

1. Acceptable calculational methods which have been verified by comparison to critical experiments have been used.

2. Assumptions regarding the enrichment of the fuel rods which have been analyzed are conservative.

3. A series of credible accidents has been considered.

4. Allowance for uncertainties in the estimation of the applicable multiplication factor is conservative.

5. The estimated final value of the multiplication factor meets the NRC acceptance criterion.

2.2 Materials

The staff has reviewed the materials compatibility and the corrosion degradation aspects of the storage canisters. The canisters are designed to store the equivalent number of fuel rods from two fuel assemblies and can be placed in either Region I or Region II rack locations of the spent fuel storage pool.

2.2.1 Evaluation

The canisters are made of stainless steel type 304, the same materials used to construct the spent fuel pool racks that hold the canisters. In a safety evaluation dated September 10, 1984, the staff concluded: (1) 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; (2) that the environmental compatibility and stability of the materials used in the spent fuel storage pool is adequate based on the test data and actual service experience in operating reactors; and (3) that the selection of appropriate materials of construction by the licensee meets the requirements of General Design Criterion 61 in Appendix A to 10 CFR Part 50.

2.2.2 Conclusion

Based on the above evaluation, the staff concludes that the proposed canisters for storage of consolidated fuel rods will have little significant corrosion degradation during the life of the plant, provide adequate material compatibility and stability with the environment in which they will be used, and meet the requirements of General Design Criterion 61, as related to fuel storage systems designed with appropriate confinement, and are, therefore, acceptable.

2.3 Spent Fuel Pool Cooling and Load Handling

2.3.1 Spent Fuel Pool Cooling and Decay Heat Load

In 1981, the staff reviewed and approved a proposed spent fuel pool cooling system modification for R. E. Ginna (Ref. 9). This modification will be implemented in 1986, and will consist of the addition of a new cooling loop in parallel with the existing loop which is sized to accommodate the maximum normal and abnormal heat loads. Further, the licensee has stated that the decay heat load on the spent fuel pool cooling system resulting from storage of consolidated fuel will remain below the previously approved spent fuel pool cooling system design capability of 16×10^6 Btu/hr. Additionally, the maximum pool water temperature will not exceed the Technical Specification limit of 150°F. Since the present capability of up to 1016 fuel assemblies will not be increased, the staff concludes that the previously approved spent fuel pool cooling system will acceptably handle the maximum normal and abnormal heat loads for the proposed storage of consolidated fuel.

As indicated above, the decay heat loads will not exceed those previously considered and approved during the pool cooling system modification review in 1981. Therefore, the staff concludes that the associated boiloff rate also will not exceed that which was previously accepted. Similarly the staff concludes that demands on pool water makeup will not exceed those previously reviewed and approved and, therefore, the makeup capability is acceptable.

2.3.2 Load Handling

The canisters containing consolidated fuel are considered a heavy load per NUREG-0612 criteria and will be transported within the pool using a special tool suspended from a 5 ton hook of the 40 ton auxiliary building crane. In a safety evaluation report dated October 1, 1984, the staff reviewed and approved modifications to the auxiliary building crane in order to meet the crane single-failure criteria of NUREG-0612 and NUREG-0554. Therefore, handling of consolidated fuel will be performed in accordance with the guidelines of NUREG-0612 with regard to limiting the chance of an unacceptable heavy load drop.

2.3.3 Conclusions

Based on the above review, the staff concludes that the proposed change to R. E. Ginna Technical Specification 5.4.4 regarding storage of consolidated fuel is in accordance with the applicable guidelines of SRP Section 9.1.2, 9.1.3 and 9.1.5 and is, therefore, acceptable.

2.4 Occupational Radiation Exposure

The staff has estimated the increment in onsite occupational dose during normal operations considering the proposed storage of consolidated spent fuel. This estimate is based on information supplied by the licensee 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. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed storage of consolidated spent fuel should add less than one percent to the total annual occupational radiation exposure at the plant. The small increase in radiation exposure should not affect the licensee's ability to maintain individual occupational dose to as low as is reasonably achievable levels (ALARA) and within the limits of 10 CFR Part 20. Thus, the staff concludes that storing additional fuel in the SFP will not result in any significant increase in dose received by workers.

2.5 Radiological Consequences of Accident Involving Postulated Mechanical Damage to Spent Fuel

For evaluation of offsite radiological consequences of accidents involving consolidated spent fuel stored in the spent fuel pool, four types of accidents were considered; a cask drop or tip, a tornado missile impact, a standard fuel assembly drop, and a fully-loaded consolidated spent fuel canister (classified as a heavy load) drop while handling standard fuel assemblies and/or spent fuel canisters. These are discussed below.

2.5.1 Cask Drop/Tip Accidents

The staff, in its Safety Evaluation of November 14, 1984, has judged that the auxiliary building crane meets the intent of Guideline 7 of NUREG-0612, Section 5.1.1. The staff, therefore, does not postulate a cask drop or tip accident which could damage stored spent fuel.

2.5.2 Tornado Missile Accidents

The design tornado missile, established in the staff review of Systematic Evaluation Program (SEP) Topics III-2, Wind and Tornado Loadings, and III-4.A, Tornado Missiles, is a 1490 lb. wooden pole, 35 feet in length and 13.5 inches in diameter, which could impact the racks with a vertical velocity of 70 ft/sec. The minimum decay time of consolidated spent fuel rods contained in storage canisters located in the pool is five years. The staff judges that the worst position for impact of this missile would be that centered on a fuel storage location where a total of nine fuel storage cells could be damaged in reracked pool sections (staff SER of November 14, 1984), or two total assemblies in unreracked pool sections. In either case, the offsite radiological consequences due to the release of volatile gap activity (almost totally ^{85}Kr) due to missile impact are bounded, due to the five year minimum decay time of consolidated fuel assemblies, by the consequences determined in the November 14, 1984 staff SER. After long decay time periods, such as 5 years or more, the volatile gap radionuclides have decayed to insignificant levels, except for ^{85}Kr which has a 10.8 year half-life. These (0-2 hr.) bounding offsite radiological consequence values are: 1) 63 rem thyroid* and 0.1 rem whole body for impact with stored assemblies in the unreracked section of the pool; and 2) 2 rem thyroid* and 0.1 rem whole body for impact with stored assemblies in the reracked section of the pool. Both limiting sets of consequences are well within the guideline values of 10 CFR Part 100, and are, therefore, acceptable.

2.5.3 Standard Fuel Assembly Drop

The offsite radiological consequences of the drop of a standard fuel assembly are bounded, due to the five year minimum decay time of consolidated fuel assemblies, by the consequences determined for the Fuel Handling Accident in the Staff SER of November 14, 1984. These (0-2 hr.) bounding offsite radiological consequence values are: 1) 44 rem thyroid* and 0.1 rem whole body in the unreracked section of the pool; and 2) 1 rem thyroid* and 0.1 rem whole body in the reracked section of the pool. Both limiting sets of consequences are well within the guidelines value of 10 CFR Part 100.

2.5.4 Consolidated Spent Fuel Canister Drop

The movement of canisters of consolidated spent fuel over spent fuel stored in the pool requires a change in the Technical Specifications because the fully loaded canister weight will be approximately 2300 lbs. This exceeds the 2000 lb weight of a standard fuel assembly and its handling tool. The fully loaded consolidated spent fuel canister must thus be classified as a heavy load.

*The key radionuclide producing thyroid dose, ^{131}I, with an 8.05 day half-life, has decayed to negligible concentrations at 5 years cooldown time. Thus the bounding thyroid doses are far beyond any expected thyroid doses resulting from the accident.

Present Technical Specification 3.11.3 states that "A load in excess of one fuel assembly and its handling tool shall never be stationed or permitted to pass over storage racks containing spent fuel." A new Technical Specification is proposed (TS 3.11.5) which states that "The restriction of 3.11.3 above shall not apply to the movement of canisters containing consolidated fuel rods if the spent fuel racks beneath the transported canister contain only spent fuel that has decayed for at least 60 days since reactor shutdown." This proposed Technical Specification allows canisters containing consolidated fuel rods to be transported over either standard stored spent fuel assemblies with 60 days decay or stored canisters containing spent fuel with at least five years decay and will result in very small (0-2 hr) thyroid and whole body doses (0.1 rem). In the case of a canister dropped onto a standard stored spent fuel assembly, which will have at least 60 days decay (new T.S. 3.11.5), the staff judges that the (0-2 hr) offsite radiological consequences due to the postulated release of the volatile gap activities of both the canister (at least five years decay) and the stored assembly are bounded by the consequences of the tornado missile impact onto nine standard assemblies in the reracked section of the pool, as discussed in the staff SER of November 14, 1984. These bounding consequences are 2 rem thyroid and 0.1 rem whole body, both well within the guidelines of 10 CFR Part 100.

2.5.5. Conclusions

Since the staff has concluded that the auxiliary building crane meets the intent of Guideline 7 of NUREG-0612, Section 5.1.1, a cask drop or tip accident which could damage stored spent fuel is sufficiently unlikely that it need not be evaluated.

The staff also concludes that a tornado missile accident resulting in damage to either two standard and/or consolidated stored spent fuel assemblies in unreracked pool sections (staff SER of November 14, 1984), or nine standard and/or consolidated assemblies in reracked pool sections, will result in atmospheric radionuclide releases with (0-2 hr) offsite radiological consequences which are well within the guidelines of 10 CFR Part 100.

The staff concludes, additionally, that the (0-2 hr) offsite radiological consequences of the drop of a standard fuel assembly are bounded, due to the five year minimum decay time of consolidated spent fuel assemblies, by the consequences determined for the Fuel Handling Accident in the staff SER of November 14, 1984, and are, therefore, well within the guideline values of 10 CFR Part 100.

Finally, the staff judges that the (0-2 hr) offsite radiological consequences due to the postulated release of the volatile gap activities of both a dropped fully loaded consolidated spent fuel canister and a stored standard or consolidated fuel assembly are bounded by the (0-2 hr) offsite radiological consequences of the tornado missile impact onto nine standard stored spent fuel assemblies in the reracked section of the pool, as discussed in the staff SER of November 14, 1984. These bounding consequences are well within the guidelines of 10 CFR Part 100.

2.5.6 Structural Evaluation

References 2 and 10 document the structural analysis performed and the staff evaluation for the Ginna spent fuel storage racks under the loads due to storage of consolidated fuel. This evaluation determined that the structural integrity of the racks would be maintained under a seismic event.

The cannisters will be fabricated from SS304. All Welding will be in accordance with ASME Section 3, subsection NF requirements. The design loads will satisfy the criteria for a seismic category 1 component. Based on the above, the staff concludes the proposed change to be acceptable.

3.0 OVERALL CONCLUSION

Based on the review, the staff concludes that the licensee's proposed storage of consolidated fuel assemblies is acceptable. In addition, the proposed Technical Specifications are acceptable.

The staff concludes, 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 ACKNOWLEDGEMENT

This Safety Evaluation was prepared by the following NRC staff: C. Miller, J. Kelly, M. Wohl, J. Wing, L. Lois, and A. Singh.

5.0 REFERENCES

1. R. W. Kober (RG&E) to J. A. Zwolinski (NRC), SUBJECT: Response to NRC Staff Questions, dated June 26, 1985.

2. R. W. Kober (RG&E) to Director, NRR, dated April 2, 1984.

3. R. W. Kober (RG&E) to J. A. Zwolinski (NRC), SUBJECT: Responses to NRC Questions on the February 27, 1985 Submittal, dated July 2, 1985.

4. R. F. Barry, "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094," WCAP-3269, (September 1963).

5. W. R. Caldwell, "PDQ-7 Reference Manual," WAPD-TM-678, (January 1967).

6. Electric Power Research Institute, "Fission Product Data for Thermal Reactors, Part 1 and Part 2: Data Set for EPRI-CINDER and Users Manual for EPRI-CINDER Code and Data," EPRI NP-356, Final Report (1976).

7. G. S. Hoovler, et al., "Critical Experiments Supporting Underwater Storage of Tightly Packed Configurations of Spent Fuel Pins," BAW-1645-4, (November 1981).

8. J. E. Maier (RG&E) to H. R. Denton (NRC), dated February 23, 1983.

9. D. M. Crutchfield (NRC) to J. E. Maier (RG&E), SUBJECT: Spent Fuel Pool Cooling System Modifications (Ginna), dated November 3, 1981.

10. J.A. Zwolinski (NRC) to R. W. Kober(RG&E), SUBJECT: Increase of the Spent Fuel Pool Capacity, dated November 14, 1984.

Dated: December 16, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 14

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1 - 8	1 - 8
3.11-2	3.11-2
3.11-4	3.11-4
5.4-2 to 5.4-4	5.4-2 to 5.4-4
	5.4-4a

2.1.1 Analysis Methods

The previous criticality analyses for the Ginna spent fuel pool utilized the LEOPARD (Ref. 4), PDQ07 (Ref. 5) and CINDER (Ref. 6) codes. The Boraflex absorbers on the racks were treated with blackness theory. A similar methodology has been employed for the analysis of the consolidated fuel storage. The validation was based on experiments performed by Babcock and Wilcox Company for this purpose (Ref. 7). The tightly packed fuel rods present two modeling difficulties, namely the high metal/water and the related distortion of the neutron spectrum (compared to the Wigner Wilkins model) and the "rubber bond" method to define the amount of water to be included in the fuel region. This method provides a consistent definition of the fuel region for the analyses of the critical experiments as well. The critical experiments analyzed constituted an adequate test of the component-codes of the model. The results of the analyses were compared to the experiment as well as to KENO-IV calculations from Reference 7. The comparison indicated that the LEOPARD-PDQ07-CINDER model results in a closer prediction than KENO-IV, a previously approved model. Therefore, the LEOPARD-PDQ07-CINDER model is considered validated and acceptable.

The analytical model described above was supplemented with burnup capability in order to be used for spent fuel. The burnup code was part of the spent fuel pool modification resubmittal (Ref. 2) and has been approved. An allowance was made for geometrical manufacturing and thermal deviations. The licensee took advantage of existing calculations for 3.13 w/o U-235 Exxon fuel to support conservative criticality estimates for the West Valley New York fuel assemblies which had an initial enrichment of 2.8 w/o U-235. The calculation yields K_∞ vs burnup which is everywhere less than 0.8. This is true for either 358 rod/canister (i.e., 2 x 179 where 179 is the number of rods/assembly) or a reduced loading of 350 rods/canister (2 x 175). For the fuel rods stored in West Valley the average burnup is about 15,000 MWD/MTU and the K_∞ for consolidated fuel is 0.63. In view of the estimated low values for K_∞ it was not deemed necessary to calculate the specific uncertainties, therefore the corresponding estimates of the fuel pool storage capacity increase of April 2, 1984, were adopted (Ref. 2). The total reactivity adjustment to the calculated value for the consolidated rods is .056 ΔK. The above results do not include the effect of the 2,000 ppm of dissolved boron in the pool water. The results of these estimates assure that for fuel designs satisfying the enrichment-burnup criteria of Figure 5.4-2 of the Ginna Technical Specifications, the K_{eff} criteria of 0.95 is satisfied and, therefore, the proposed consolidated fuel storage is acceptable.

2.1.2 Accident Analysis

The accident analysis performed in the submittals of April 2, 1984 (Ref. 2) and February 23, 1983 (Ref. 8) are also applicable for the ~~b~~case of the consolidated fuel storage with respect to criticality. In addition, however, the loss of containment of all of the rods in a canister and their subsequent relocation on a uniform and optimum pitch has been analyzed. In the particular case of the Exxon 3.13 w/o U-235 fuel, assuming the presence of 2,000 ppm of

These submittals analyzed configurations that are analogous to those found in a fully rodded canister and demonstrate that criticality cannot occur.

partially filled for a ~~half~~ canister,

boron, the optimum pitch is .632 inches. On such a pitch a square array would be 12 inches on the side with a K∞ of about 0.95. When the 2,000 ppm boron and a 15,000 MWD/MTU burnup is taken into account, the K∞ reduces to 0.55. This particular fuel rod arrangement is thought to be extremely unlikely. The staff also has considered the case of a ~~half~~ canister being not completely filled with the fuel rods ~~in an optimum or near optimum spacing may be more likely. The reason is that the~~ since damaged fuel rods will be stored in a separate canister in individual steel tubes and the remaining ~~number of~~ undamaged rods may then not fill ~~exactly~~ a ~~half~~ canister. However, the neutron leakage from a ~~half~~ partially filled canister will be even greater than that of a ~~whole~~ completely filled canister (i.e., 2 x 179 = 358 rods); hence, the K∞ ≤ 0.55. In summary, none of the postulated accidents could result in criticality. In the analysis of the postulated accidents it was assumed that 2,000 ppm boron is diluted in the pool cooling water. Credit for this boron is allowed through the use of the double contingency principle. In view of the above, the accident analysis presented for the Ginna consolidated spent fuel storage is acceptable.

2.1.3 Conclusions

The staff concludes that the spent fuel pool, including the consolidated fuel, meets the General Design Criterion 62 as regarding criticality. This conclusion is based on the following considerations:

1. Acceptable calculational methods which have been verified by comparison to critical experiments have been used.
2. Assumptions regarding the enrichment of the fuel rods which have been analyzed are conservative.
3. A series of credible accidents has been considered.
4. Allowance for uncertainties in the estimation of the applicable multiplication factor is conservative.
5. The estimated final value of the multiplication factor meets the NRC acceptance criterion.

2.2 Materials

The staff has reviewed the materials compatibility and the corrosion degradation aspects of the storage canisters. The canisters are designed to store the equivalent number of fuel rods from two fuel assemblies and can be placed in either Region I or Region II rack locations of the spent fuel storage pool.

2.2.1 Evaluation

The canisters are made of stainless steel type 304, the same materials used to construct the spent fuel pool racks that hold the canisters. In a safety evaluation dated September 10, 1984, the staff concluded: (1) 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 to galvanic potential between them and have a high resistance ~~of~~ general corrosion, localized corrosion, and galvanic corrosion; (2) that the environmental compatibility and stability of the materials used in the spent fuel storage pool is adequate based on the test data and actual service experience in operating reactors; and (3) that the selection of appropriate materials of construction by the licensee meets the requirements of General Design Criterion 61 in Appendix A to 10 CFR Part 50.

2.2.2 Conclusion

Based on the above evaluation, the staff concludes that the proposed canisters for storage of consolidated fuel rods will have little significant corrosion degradation during the life of the plant, provide adequate material compatibility and stability with the environment in which they will be used, and meet the requirements of General Design Criterion 61, as related to fuel storage systems designed with appropriate confinement, and are, therefore, acceptable.

2.3 Spent Fuel Pool Cooling and Load Handling

2.3.1 Spent Fuel Pool Cooling and Decay Heat Load

In 1981, the staff reviewed and approved a proposed spent fuel pool cooling system modification for R. E. Ginna (Ref. 9). This modification will be implemented in 1986, and will consist of the addition of a new cooling loop in parallel with the existing loop which is sized to accommodate the maximum normal and abnormal heat loads. Further, the licensee has stated that the decay heat load on the spent fuel pool cooling system resulting from storage of consolidated fuel will remain below the previously approved spent fuel pool cooling system design capability of 16×10^6 Btu/hr. Additionally, the maximum pool water temperature will not exceed the Technical Specification limit of 150°F. Since the present capability of up to 1016 fuel assemblies will not be increased, the staff concludes that the previously approved spent fuel pool cooling system will acceptably handle the maximum normal and abnormal heat loads for the proposed storage of consolidated fuel.

As indicated above, the decay heat loads will not exceed those previously considered and approved during the pool cooling system modification review in 1981. Therefore, the staff concludes that the associated boiloff rate also will not exceed that which was previously accepted. Similarly the staff concludes that demands on pool water makeup will not exceed those previously reviewed and approved and, therefore, the makeup capability is acceptable.

2.3.2 Load Handling

The canisters containing consolidated fuel are considered a heavy load per NUREG-0612 criteria and will be transported within the pool using a special tool suspended from a 5 ton hook of the 40 ton auxiliary building crane. In a safety evaluation report dated October 1, 1984, the staff reviewed and

2.5.2 Tornado Missile Accidents

The design tornado missile, established in the staff review of Systematic Evaluation Program (SEP) Topics III-2, Wind and Tornado Loadings, and III-4.A, Tornado Missiles, is a 1490 lb. wooden pole, 35 feet in length and 13.5 inches in diameter, which could impact the racks with a vertical velocity of 70 ft/sec. ~~The licensee has stated that~~ The minimum decay time of consolidated spent fuel rods contained in storage canisters located in the pool is five years. The staff judges that the worst position for impact of this missile would be that centered on a fuel storage location where a total of nine fuel storage cells could be damaged in reracked pool sections (staff SER of November 14, 1984), or two total assemblies in unreracked pool sections. In either case, the offsite radiological consequences due to the release of volatile gap activity (almost totally ^{85}Kr) due to missile impact are bounded, due to the five year minimum decay time of consolidated fuel assemblies, by the consequences determined in the November 14, 1984 staff SER. After long decay time periods, such as 5 years or more, the volatile gap radionuclides have decayed to insignificant levels, except for ^{85}Kr which has a 10.8 year half-life. These (0-2 hr.) bounding offsite radiological consequence values are: 1) 63 rem thyroid* and 0.1 rem whole body for impact with stored assemblies in the unreracked section of the pool; and 2) 2 rem thyroid* and 0.1 rem whole body for impact with stored assemblies in the reracked section of the pool. Both limiting sets of consequences are well within the guideline values of 10 CFR Part 100, and are, therefore, acceptable.

2.5.3 Standard Fuel Assembly Drop

The offsite radiological consequences of the drop of a standard fuel assembly are bounded, due to the five year minimum decay time of consolidated fuel assemblies, by the consequences determined for the Fuel Handling Accident in the Staff SER of November 14, 1984. These (0-2 hr.) bounding offsite radiological consequence values are: 1) 44 rem thyroid* and 0.1 rem whole body in the unreracked section of the pool; and 2) 1 rem thyroid* and 0.1 rem whole body in the reracked section of the pool. Both limiting sets of consequences are well within the guidelines value of 10 CFR Part 100.

2.5.4 Consolidated Spent Fuel Canister Drop

The movement of canisters of consolidated spent fuel over spent fuel stored in the pool requires a change in the Technical Specifications because the fully loaded canister weight will be approximately 2300 lbs. This exceeds the 2000 lb weight of a standard fuel assembly and its handling tool. The fully loaded consolidated spent fuel canister must thus be classified as a heavy load.

*The key radionuclide producing thyroid dose, ^{131}I, with an 8.05 day half-life, has decayed to negligible concentrations at 5 years cooldown time. Thus the bounding thyroid doses ar far beyond any expected thyroid doses resulting from the accident.
e

Present Technical Specification 3.11.3 states that "A load in excess of one fuel assembly and its handling tool shall never be stationed or permitted to pass over storage racks containing spent fuel." A new Technical Specification is proposed (TS 3.11.5) which states that "The restriction of 3.11.3 above shall not apply to the movement of canisters containing consolidated fuel rods if the spent fuel racks beneath the transported canister contain only spent fuel that has decayed for at least 60 days since reactor shutdown." This proposed Technical Specification allows canisters containing consolidated fuel rods to be transported over with standard stored spent fuel assemblies with 60 days decay or stored canisters ~~both~~ containing spent fuel with at least five years decay, will result in very small (0-2 hr) thyroid and whole body doses (0.1 rem). and In the case of a canister dropped onto a standard stored spent fuel assembly, which will have at least 60 days decay (new T.S. 3.11.5), the staff judges that the (0-2 hr) offsite radiological consequences due to the postulated release of the volatile gap activities of both the canister (at least five years decay) and the stored assembly are bounded by the consequences of the tornado missile impact onto nine standard assemblies in the reracked section of the pool, as discussed in the staff SER of November 14, 1984. These bounding consequences are 2 rem thyroid and 0.1 rem whole body, both well within the guidelines of 10 CFR Part 100.

2.5.5. Conclusions

Since the staff has concluded that the auxiliary building crane meets the intent of Guideline 7 of NUREG-0612, Section 5.1.1, a cask drop or tip accident which could damage stored spent fuel is sufficiently unlikely that it need not be evaluated.

The staff also concludes that a tornado missile accident resulting in damage to either two standard and/or consolidated stored spent fuel assemblies in unreracked pool sections (staff SER of November 14, 1984), or nine standard and/or consolidated assemblies in reracked pool sections, will result in atmospheric radionuclide releases with (0-2 hr) offsite radiological consequences which are well within the guidelines of 10 CFR Part 100.

The staff concludes, additionally, that the (0-2 hr) offsite radiological consequences of the drop of a standard fuel assembly are bounded, due to the five year minimum decay time of consolidated spent fuel assemblies, by the consequences determined for the Fuel Handling Accident in the staff SER of November 14, 1984, and are, therefore, well within the guideline values of 10 CFR Part 100.

Finally, the staff judges that the (0-2 hr) offsite radiological consequences due to the postulated release of the volatile gap activities of both a dropped fully loaded consolidated spent fuel canister and a stored standard or consolidated fuel assembly are bounded by the (0-2 hr) offsite radiological consequences of the tornado missile impact onto nine standard stored spent fuel assemblies in the reracked section of the pool, as discussed in the staff SER of November 14, 1984. These bounding consequences are well within the guidelines of 10 CFR Part 100.

For further details with respect to this action see (1) the application for amendment dated February 27, 1985 as supplemented on June 10, June 26, and July 11, 1985, (2) Amendment No. 12 to Facility Operating License No. DPR-18, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Rochester Public Library, 115 South Avenue, Rochester, New York 14610. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of PWR Licensing-A, NRR.

Dated at Bethesda, Maryland, this 16th day of December 1985.

FOR THE NUCLEAR REGULATORY COMMISSION
ORIGINAL SIGNED BY

George E. Lear, Director
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