Technical Report On

the

General Electric Company

8 x 8 Fuel Assembly

5 February 1974

Regulatory Staff

U. S. Atomic Energy Commission

Table of Contents

1.0 Introduction

The current fuel in General Electric Company boiling water reactors is sintered, slightly enriched uranium dioxide pellets sealed in Zircaloy tubes. Bundles of these fuel rods are contained within a square open-ended Zircaloy channel box to form fuel assemblies. The General Electric Co. has recently modified the design of these fuel assemblies and licensees propose to reload assemblies of this new design as replacements for depleted assemblies of the old type.[1, 2, 3, 4, This report presents the results of the Regulatory Staff's generic review of 8 x 8 fuel assemblies as used both in partial and full core reloads. As part of the staff's review of the General Electric Company BWR-6 class of reactors, which are currently under consideration for construction permits, the Staff is continuing its review of the 8 x 8 fuel assemblies used in these new reactor designs. The Staff's review of reload assemblies considered the effects that the changes in the fuel design have on normal operation, abnormal operational transients and accidents. However, the Staff review considered only generic aspects of the fuel design such as the adequacy of design methods, the comparative performance of the old and new fuel designs, and the applicability of accident analysis methods. The plant specific aspects of the review, such as compliance with the Interim Acceptance Criteria, including the effects of fuel pellet densification, any necessary revisions to Technical Specification requirements, and the radiological consequences of postulated accidents will be addressed in separate evaluation for the individual plants.

2.0 Mechanical Design

The reload fuel assemblies consist of 63 fuel rods and one unfueled, capture-spacer rod in a square 8 x 8 array within a square channel box. The rods are spaced and supported at the top and bottom by stainless steel tie plates. The rods are also held in alignment by spacer-guides located along the assembly. As shown in Table I the 8 x 8 fuel assembly is similar to the current 7 x 7 design. The major mechanical changes are the larger number of rods; the reduction in the rod diameter; the introduction of the asymetrically located unfueled spacer-capture rod; and the use of fully annealed, rather than cold worked, Zircaloy cladding. Other changes, which have also been incorporated in the most recent 7 x 7 designs include shorter, chamfered and undished pellets and a hydrogen getter. However, the designs of both assemblies have the same objective, that is maintainance of clad integrity during normal operation and abnormal transients. The designs of both are also based on the same stress criteria, that is, the ASME Boiler and Pressure Vessel Code, Section III. In evaluating the performance of the fuel, the design analyses considered stresses due to external coolant pressure, internal gas pressure, thermal effects, spacer contact, and flow induced vibration. Other effects which were considered included pellet-cladding mechanical interaction, stress corrosion cracking, fretting and densification. Verification of the adequacy of the design of the 8 x 8 assemblies is based on analysis, mechanical tests, operating experience of previous designs, in-pile tests of a prototypical fuel rod and similar fuel rods, and an out-of-pile test of an assembly of similar design.

TABLE I

MECHANICAL DESIGN COMPARISON

ASSEMBLY		
Rod Array	7 x 7	8 x 8
Number of Fueled Rods	49	63
Rod Pitch, In.	0.738	0.640
FUEL ROD		
Active Fuel Length, In.	144	144
Gas Plenum Length, In.	11.25	11.25
Fill Gas	He	He
FUEL		
Material	UO_2	UO_2
Pellet Diameter, In.	0.477	0.416
Pellet Immersion Density, % TD	95.0	95.0
CLADDING		
Material	Zr-2	Zr-2
Thickness, In.	0.037	0.034
Outside Diameter, In.	0.563	0.493
CHANNEL		
Material	Zr-4	Zr-4
Thickness, In.	0.080	0.080
Outside Dimension, In.	5.438	5.438
Length, In.	162 1/8	162 1/8
SPACERS		
Number	7	7
Material		
Grid	Zr-4	Zr-4
Springs	Inconel.	Inconel.

Much of the previous experience with fuel rods and assemblies is applicable to the 8 x 8 fuel assemblies.[(6)] These rods ranged in diameter from 0.344 to 0.593 inches, in clad thickness from 0.022 to 0.088 inches, and in pellet-clad diametral gap from 0.002 to 0.016 inches. Rods have been irradiated for up to 6 years and had peak exposure of 30,000 MWD/T. Although rods identical to the 8 x 8 design have not been tested by GE, the background of experience is sufficient to enable GE to design rods of new design with confidence in their durability.

Confidence that the vibration and fretting characteristics of the 8 x 8 assemblies are known, is based on rod vibration experiments[(7)] and the operating experience with other types of fuel assemblies in general and the 7 x 7 design in particular. The 7 x 7 and 8 x 8 assemblies are very similar in this regard. The fuel rods in both are of similar design, are made of the same material and have nearly the same natural frequency. The fuel rod spacer grids in both types of assembly also are of similar design, are made of the same materials and exert the same spring force. Both operate at the same pressure and temperature with nearly identical fluid velocities and quality.

Further verification of the adequacy of the design has been provided by the testing of an assembly of similar design for 7000 hours in high pressure, two-phase flow loop.[(8)] This test was performed by ASEA-Atom, a Swedish BWR manufacturer and a General Electric Company licensee, as part of a fuel development program.

A comparison of the significant parameters of this test assembly(8) and the GE 8 x 8 assemblies indicate that the wear and fretting characteristics would be similar. The most significant differences are that the test assembly had no unfueled spacer-capture rod, and had four latern springs supporting a fuel rod, where the GE assemblies have only two. However, the vibration and fretting in this test would be expected to be at least as severe as in a GE 8 x 8 assembly since the axial pitch of the spacers was larger and the rods thinner walled and smaller in diameter. Inspection at 1.5 month intervals and the conclusion of the test revealed no significant fretting wear.

Although the design of the unfueled spacer-capture rod is new, it is based on experience with similar designs. Five 6 x 6 fuel assemblies with eccentrically located fueled spacer-capture rods which have a locking tab design identical to the 8 x 8 design have operated in the Humboldt Bay reactor. Visual examination of these assemblies has revealed no deficiencies. Assemblies with eccentrically located fuel spacer rods with a different locking tab design have operated in the Dresden-1, KRB, Tarapur and Garigliano reactors. Twenty four assemblies with unfueled rod have operated in the Big Rock reactor.

A number of mechanical tests have been performed on 8 x 8 fuel assemblies and components in order to demonstrate their integrity. Dead weight loading of the 7 x 7 type assembly spacer grids has demonstrated that they are adequate to withstand all expected loads. Although, as GE has stated, the 8 x 8 assembly spacer-grids are stronger than 7 x 7

Accident induced loads and stresses have been calculated for both the 7 x 7 and 8 x 8 assemblies using the same methods. The limiting accident loads result from a steam line break. The pressure differenc following a steam line break are less than 10% greater than normal operating pressure differences. As in normal operation, the pressure differences in an 8 x 8 assembly following a steam line break are 5 to 10% greater than in a 7 x 7 assembly. The loads following a steam line break are well below the allowable loads.

The behavior of the two fuel designs under seismic loading is nea identical. This is so because the stiffness of the fuel channel and t weight of the fuel assembly are the same for both designs. Only these two parameters need to be considered s'nce the stiffness of the bundle of fuel rods is small compared to the channel, and the clearance betwe the channel and the rod bundle is small compared to the limiting de- flection of the channels. The predicted loads from the postulated saf shutdown earthquake are one-half the allowable loads.

We conclude that based on operating experience with similar fuel, the results of an out-of-pile test of a assembly of similar design, th increased thermal margins which the 8 x 8 fuel has, the Technical Spec cation requirements to monitor and limit off-gas and coolant activity, and the existence of a continuing fuel rod surveillance program which includes destructive and non-destructive post irradiation examirations the cladding integrity of the 8 x 8 fuel will be maintained during nor operation and abnormal operational transients and significant amounts

radioactivity will not be released. Furthermore, we conclude that accidents or earthquake induced loads will not result in an inability to cool the fuel and safely shutdown the reactor.

3.0 Nuclear Design

The nuclear design of the 8x8 reload assemblies is similar to that of the equivalent 7 x 7 reload assemblies as shown in Table II . The U-235 enrichments for the individual fuel rods, the number and distribution of fuel rods containing gadolinia, and the water-to-fuel ratio are similar in the two designs. However, two features which might effect the nuclear characteristics differ in the proposed 8 x 8 reload assemblies and the equivalent 7 x 7 reload assemblies. First, there are 64 rods in the 8 x 8 assembly, compared to 49 in the 7 x 7 assembly. Second, the 8 x 8 assembly has a water filled rod near the center of the assembly and the 7 x 7 does not.

The major items of interest from the standpoint of nuclear design of the 8 x 8 reload fuel assembly are the uncontrolled and controlled (all control rods in) reactivity, the change in reactivity of the assembly with burnup, the local peaking in the assembly, the Doppler reactivity coefficient, the delayed neutron fraction, and the void reactivity coefficient. Values of these parameters as a function of burnup for an infinite lattice of 8 x 8 reload assemblies were presented[1,2,3,4,5] and compared with values for an infinite lattice of 7 x 7

TABLE II

Nuclear Design Comparison

	8 x 8	7 x 7
Pellet Outside Diameter, in	0.416	0.487
Rod Outside Diameter, in.	0.493	0.563
Rod-to-Rod Pitch, in.	0.640	0.738
Water-to-Fuel Ratio	2.60	2.43
U Bundle Weight, lbs	404.6	427.8
Cladding Thickness, mils.	34	32
K_{∞}, cold uncontrolled	1.166	1.163
k_{∞}, cold-controlled	0.981	0.988
Max. Local Peaking Factor	1.22	1.24
Average U-235 content, %	2.62	2.63
Number Gadolinia containing pins	4	4
Relative gadolinia content of gadolinia containing pins	2	1
Number of water rods	1	0

2.59 w/o U-235 8 x 8 Assembly; 2.50 w/o U-235 7 x 7 Assembly

	8 x 8	7 x 7
Pellet Outside Diameter, in.	0.416	0.477
Rod Outside Diameter, in.	0.493	0.563
Rod-to-Rod Pitch, in.	0.640	0.738
Water to Fuel Ratio	2.60	2.53
U Bundle Weight, lbs.	404.6	412.8
Cladding Thickness, mils.	34	37
k_{∞}, cold uncontrolled	1.148	1.129
K_{∞}, cold controlled	0.966	0.960
Max. Locak Peaking Factor	1.22	1.30
Average U-235 content, %	2.50	2.50
Number gadolinia containing pins	4	4
Relative gadolinia Content of Gadolinia containing pins	1	1
Number of water rods	1	0

assemblies of similar enrichment. In general, the values for the 8 x 8 lattice differed by less than 10% from those of the 7 x 7 lattice.

The same calculational techniques were used in calculating the lattice parameters for the 8 x 8 reload assemblies and those equivalent 7 x 7 assemblies. The particulars of the design of the assembly do not directly enter reactor calculations since homogenized parameters for the assembly (e.g., few group cross-sections, diffusion coefficients) are used as input. The 8 x 8 reload assemblies are neutronically similar to the 7 x 7 assemblies (i.e., similar enrichment, water-to-fuel ratio and gadolinia content), and we believe the calculational techniques are of equivalent accuracy for an 8 x 8 assembly as for a 7 x 7 assembly. The local peaking factor for the 8 x 8 reload assemblies is reported to decrease monotonically with exposure, while that of the equivalent 7 x 7 assemblies is reported to decrease with an exposure of about 10 GWD/t, then increase slowly. This behavior was explained, in response to a staff question, in terms of differences in the shift in the position of the peak local power rod within the bundle as a function of exposure.

The effect of the water rod is to increase moderation in the interior of the bundle and reduce the rod to rod power peaking. Voiding of the water rod would decrease the reactivity of the bundle and would depress the flux in the center of the bundle. (Voiding of the water rod is equivalent to increasing the void fraction in the assembly of about 1%).

We have reviewed the nuclear design of the 8 x 8 reload fuel assemblies by comparing their properties with equivalent 7 x 7 assemblies and conclude that the nuclear design of the 8 x 8 reload assemblies is acceptable.

4.0 Thermal-Hydraulic Design

During normal operation and abnormal operational transients, the design objective for both types of assembly is to maintain clad integrity and prevent the release of significant amounts of radioactivity. The fuel damage limits and thermal-hydraulic criteria used to evaluate the performance of the fuel is the same for both designs. During normal steady state operation the Minimum Critical Heat Flux Ratio (MCHFR) is held above 1.9. For abnormal operational transients, the clad strain is limited to less than 1% and the MCHFR is maintained greater than 1.0. These design bases are the same as the design bases for fuel previously reviewed and accepted for boiling water reactors.

In general, the 8 x 8 fuel has greater thermal margins to these design limits than 7 x 7 fuel. The design value of linear heat generation rate for normal operation is 13.4 kw/ft for an 8 x 8 fuel and 17.5 to 18.5 kw/ft for 7 x 7 fuel. Based on previous experience, this lower thermal duty combined with the other design changes is expected to result in fewer clad perforations. During normal operation, the hot channel MCHFR in the 8 x 8 assemblies is expected

to be greater than 2.3 which is 11% greater than the hot channel MCHFR expected for 7 x 7 assemblies. The LHGR which is calculated to produce 1% strain in the cladding is 1.8 times the design value for 8 x 8 fuel and only 1.5 times the design value for 7 x 7 fuel. Similarly, the LHGR which produces fuel pellet center-line melting is 1.4 times the design value for 8 x 8 fuel as compared to 1.2 times the design value for 7 x 7 fuel.

Since the 8 x 8 assemblies are different than the 7 x 7 assemblies, we reviewed the thermal-hydraulic design methods to determine their applicability to the new fuel design. The differences are the modified flow geometry and the introduction of an unfueled rod. The portions of the thermal-hydraulic design methods which might be affected by these differences and which we reviewed are the techniques used to calculate flow rate and critical heat flux in the 8 x 8 assemblies.

The methods used to calculate flow and pressure drop in the 8 x 8 assemblies are the same as that used for the 7 x 7 assemblies. However, empirical constants are varied to adjust the results to the specific fuel design. Tests have been made to determine these empirical constants for an 8 x 8 geometry and to confirm the method of calculating friction,

acceleration and elevation pressure drop. Furthermore, the fuel assembly support cas..ng orifice is the major flow resistance and, therefore, the flow distribution between fuel assemblies is insensitive to differences in the hydraulic characteristics of the fuel assemblies. The methods of hydraulic analyses are the same as those previously reviewed and accepted for boiling water reactors and are equally applicable for 8 x 8 fuel assemblies.

The correlation used to calculate the critical heat flux in the 8 x 8 assemblies is the same Hench-Levy correlation used in evaluation of 7 x 7 assemblies. Introduced in 1966, the Hench-Levy correlation has been the accepted basis for determining thermal margin for a variety of General Electric boiling water reactors. The 8 x 8 fuel assembly is, except for the inclusion of an unheated rod and the change in hydraulic diameter, very similar in geometrical and thermal-hydraulic characteristics to the 7 x 7 fuel assembly.

We have previously reviewed (12) the effect of an unheated rod and the applicability of a CHF correlation such as the Hench-Levy correlation which is based on average fluid

conditions and concluded that the effect of the unheated rod is not significant. We have also reviewed the effect that the changes in subchannel hydraulic diameters might have on thermal performance and conclude that the subchannel flow in the 8 x 8 assembly is more balanced than in the 7 x 7 design and should result in improved thermal performance. Therefore, we conclude that the Hench-Levy correlation is equally applicable to both the 8 x 8 and the 7 x 7 assemblies.

Because the Hench-Levy correlation does not specifically account for non-uniform axial heat flux distributions and rod-to-rod variations in power, as exist in fuel assemblies, a lower limit line to the then existing critical heat flux data was chosen as the form of the correlation. In addition, for added conservatism, the steady state design CHF was to be such that it did not exceed the Hench-Levy CHF divided by 1.9.

In order to overcome these shortcomings of the Hench-Levy correlation and to provide a data base that is more representative of actual fuel assembly performance, General Electric constructed the ATLAS facility which has the capability to test full size, full power 8 x 8 rod bundles. Except for the

method of heating the rods (electrical resistance heating) and differences in grid spacer design, the 8 x 8 rod bundles tested in the ATLAS loop are similar to fuel assemblies. The large body of critical heat flux data obtained from the ATLAS facility for both 7 x 7 and 8 x 8 of rods in 16, 49, and 64 rod bundles has provided the foundation for developing a new correlation called GEXL (General Electric Critical Quality X_c - Boiling Length) which GE proposed as a replacement for the Hench-Levy correlation. A new thermal design method (GETAB, General Electric Thermal Analysis Basis), which uses GEXL and appropriate design parameters to determine the maximum power capability of a fuel assembly during normal operation and abnormal operational transients and accident conditions, is also proposed.

The Regulatory staff is now reviewing GEXL, GETAB, the Hench-Levy correlation, and the ATLAS rod bundle data. General Electric has informed the Regulatory staff that all operating BWR plants have been provided with GETAB with the instructions that, in the interim, operating thermal limits be determined by either the Hench-Levy correlation or GETAB, choosing the method that provides the more conservative result.

At this time the staff agrees that the operating plant thermal margins should be predicted on the basis of the method (i.e., either Hench-Levy or GETAB) which yields the more conservative result, on this basis, use of the Hench-Levy correlation for the 8 x 8 fuel design would be acceptable.

5.0 Abnormal Operational Transients

To assure the safety of the plant, the results of the analyses of abnormal operational transients are required to indicate that the fuel and the reactor coolant pressure boundary (RCPB) are not damaged. The fuel damage criteria are a minimum critical heat flux ratio (MCHFR) of unity and a cladding strain of one percent. The RCPB damage criteria is the system design pressure (as specified in the ASME Boiler and Pressure Vessel Code, Section III). These damage limits for 8 x 8 fuel are the same as previously reviewed and accepted for 7 x 7 fuel in boiling water reactors.

Abnormal operational transients are the result of single equipment failures or single operator errors that can reasonably be expected to occur during anticipated modes of station operation. The types of failures and errors considered are the same for both types of fuel. The transients resulting from these failures and errors can cause variations in both system parameters such as core flow, core power, pressure and coolant level, and in local parameters such as flow and power in a single assembly. System parameters are primarily a function of the core average nuclear, thermal and hydraulic characteristics.

Since the characteristics of the 8 x 8 assemblies are similar to those of the 7 x 7 assemblies, the 8 x 8 fuel has no significant effect on these transients. However, for the determination of local parameters, the characteristics of the 8 x 8 fuel may be significant. It has been reported(1) that the thermal margin of the hot assembly has been analyzed using the conservative fuel type and the results demonstrate that the fuel damage limits are not exceeded. The results of three limiting events, i.e., a seizure of one recirculation pump, the continuous withdrawal of a control rod, and the misorientation of an assembly indicate that the consequences of these events are less severe for 8 x 8 assemblies than for 7 x 7 assemblies. Analyses of all transients have been made(3) considering both the 7 x 7 and 8 x 8 assemblies and the results indicate that the fuel damage limits are not exceeded.

6.0 Accidents

Analyses of the design basis accidents are made to evaluate the capability of the engineered safety features to mitigate the consequences of postulated accidents and control the possible escape of fission products. The four postulated design basis accidents are the a) loss-of-coolant b) steam line break c) fuel handling and d) control rod drop accidents.

6.1 Rod Drop Accident

The rod drop accident analysis is not significantly affected by a change from a 7 x 7 to an 8 x 8 assembly. The kinetics model uses homogenized cross sections and is not directly involved with the details of the lattices. The local peaking factors of interest are also similar for both types of assemblies. Analyses of the rod drop accident demonstrate that the dropping of a maximum worth sequenced control rod will not result in a peak fuel pellet enthalpy which exceeds the damage limit of 280 cal/gm.

6.2 Refueling Accident

The method of determining the number of rods which might fail following the dropping of an assembly is equally applicable to both designs. Since the types of assembly are similar, the total amount of fission products released

Models including their Conservative Assumptions and Procedures" which is contained in the Commission's Interim Policy Statement, entitled "Criteria for Emergency Core Cooling Systems for Light-Water-Power Reactors" and published in the Federal Register on June 29, 1971. The Commission Rule "Acceptable Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors" dated December 28, 1973, is intended to replace the Interim Policy Statement. Conformance with this new rule, which includes revised criteria and revised features of the evaluation model, will require re-analysis of the ECCS performance. When the requisite evaluations are sumbitted to the Director of Regulation, as required by the implementation schedule contained in the rule, the staff will make its review and conclusions. Our current review is only concerned with compliance with the Interim Policy Statement. Since the 8 x 8 fuel assemblies are a different design than the 7 x 7 assemblies considered in the General Electric Evaluation Model described in NEDO-10329, and referenced in Part 2 of Appendix A to the Interim Policy Statement, the staff has reviewed the evaluation model to determine its applicability to the new fuel design. The features of the new fuel design which are different from the old design and significant in determining applicability

of the evaluation model are: a) smaller diameter fuel rods; b) larger number of fuel rods in each assembly, and c) an unfueled central rod. The features of the evaluation model which might be affected by these changes in the design of the fuel assembly and which we reviewed include applicability of the transient critical heat flux correlation, the thermal radiation and the spray cooling convective heat transfer in an 8 x 8 array, and the effect of the unfueled rod on heat transfer.

As discussed in a preceding section of this report, we have reviewed the differences in the thermal and hydraulic characteristics between an 8 x 8 fuel assembly and the 7 x 7 assembly, and concluded that the steady state critical heat flux correlation is equally applicable to both designs. In addition, GE has nearly completed an extensive series of steady-state critical heat flux tests on full-scale, 8 x 8 heater bundles with varying inlet conditions, and power distributions which are representative of expected conditions in a BWR. These tests will provide a large additional set of critical heat flux data applicable to the 8 x 8 fuel design. General Electric and the staff are now in the process of evaluating this data and its applicability to the conditions

following a loss-of-coolant accident. Upon the completion of this evaluation and during the review of the re-analysis required by the new rule, the staff will re-examine the acceptability of the current critical heat flux model.

We have also reviewed the differences in thermal radiation and spray cooling characteristics between the 8 x 8 and the 7 x 7 fuel assemblies and conclude that the procedures used to calculate the heatup of an 8 x 8 fuel assembly following a loss-of-coolant accident are consistent with the approved General Electric Evaluation Model. Our conclusion is based on independent calculations using a computer program developed for the staff(13) and the results of full-scale, stainless steel 8 x 8 rod array, heater bundle spray cooling and flooding tests.(14,15)

The adequacy of the thermal radiation model for an 8 x 8 fuel bundle has been verified by comparison of the predicitons of clad temperature using both the GE(14) and staff's(16) computer programs to the results of steady-state heater bundle tests which had not spray cooling. The staff's computer

program underpredicts the temperature of rods in the bundle by not more than 25°F, but overpredicted the temperature of some rods by as much as 150°F. The GE program predicted temperatures which were from 50 to 75°F lower than the staff's calculations. The temperature overprediction of the corner and unfueled rods may be due to local differences in emissivity. Although comparison of the gray body view factors for individual rods used in the two programs revealed no reason for the difference between the GE and staff results, the simpler nodalization of the heater rods in the GE program could account for the difference.

The adequacy of both the GE and staff heatup models, including both convective cooling to the spray and rod-to-rod radiation, was demonstrated by comparing predictions to the results from transient tests of the 8 x 8 stainless steel heater bundle. The predicitons were based in part on the conservative values of spray cooling convective heat transfer coefficient specifed in the IAC evaluation model. The other parameters, such as heat-generation, emissivity and thermal properties, were best estimate values. The staff's calculations are as much as 40°F lower,

and as much as 80°F higher than the measured temperatures. The predicitions reported by GE have approximately the same inaccuracy. These differences are within the uncertainties of the test results.

The General Electric Company has also completed a test witnessed by the staff on an 8 x 8 Zircaloy heater bundle, but has not yet reported the results. Previous tests have shown that a heatup model which is based on the results of tests with stainless steel rods can predict the thermal response of Zircaloy rods within the uncertainty of the experimental measurements. For most reactors which have jet pumps, the heatup transients are short, that is, approximately two minutes long, result in moderate temperatures, that is below 2000°F, and the degree of uncertainty is acceptably small. However, for transients which are longer and result in higher temperatures, such as occur in reactors without jet pumps, additional experimental verification of the applicability of analytical methods derived from stainless steel heater bundle tests to Zircaloy clad rods are required. Therefore, the results of this Zircaloy bundle test will be submitted and reviewed prior to use of fuel in reactors without jet pumps.

7.0 References

1. "Dresden 3 Nuclear Power Station, Second Reload License Submittal," General Electric Co., Nuclear Fuel Department, September 1973, and Supplement A, November 27, 1973; Supplement B, December 6, 1973; Supplement C, December 6. 1973; Supplement D, December 17, 1973; Supplement E, December 17, 1973 (Proprietary); Supplement F, January 9, 1974; Supplement G, January 9, 1974 (Proprietary); Supplement H, January 23, 1974.

2. "Nine Mile Point Unit 1 - Second Refueling," P. D. Raymond to A. Giambusso, September 14, 1973.

   "Nine Mile Point Unit 1 Safety Analysis for Type 5 and Type 6 Reload Fuel," Niagara Mohawk Power Corporation, October 15, 1973.

   "Nine Mile Point Unit 1, Part 1, Non-Proprietary Response and Part 2, Proprietary Response, January 15, 1974.

   "Nine Mile Point Unit 1, Analyses and Proposed Technical Specification Changes, January 22, 1974.

3. "Monticello Nuclear Generating Plant, Permanent Plant Changes to Accomodate Equilibrium Core Scram Reactivity Insertion Characteristics," January 23, 1974.

4. "Pilgrim Cycle-2 Licensing Submittal," M. J. Feldman to J. F. O'Leary, January 24, 1974.

5. NEDO-20103. "General Design Information for General Electric Boiling Water Reactor Reload Fuel Commencing in Spring, '74," September 1973.

6. H. E. Williamson and D. C. Ditmore, "Experience with BWR Fuel Through September 1971," NEDO-10505, May 1972.

7. GEAP-4059, "Vibration of Fuel Rods in Parallel Flow," E. P. Quinn, July 1962.

8. Letter J. A. Hinds to V. Moore, February 4, 1974.

9. NEDM-10735 "Densification Considerations in BWR Fuel Design and Performance," D. C. Ditmore and R. B. Elkins, December 1972, Supplement 2, "Response to AEC Questions, NEDM-10735 April 1973 (Proprietary), Supplement 2, "Response to AEC Questions, NEDM-`0735 Supplement 1," May 1973 (Proprietary), Supplement 3, "Response to AEC Questions, NEDM-10735, Supplement 1, June 1973 (Proprietary), Supplement 4, "Response to AEC Questions, NEDM-10735," July 1973, (Proprietary), Supplement 5, "Densification Considerations in BWR Fuel," July 1973 (Proprietary), Supplements 6, 7, and 8, "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel."

10. NEDO-20181, Supplement 1, December 3, 1973 (Proprietary); GEGAP-III, "A Model for the Prediction of Pellet Conductance in BWR Fuel Rods."

11. "Technical Report on Densification of General Electric Reactor Fuels," December 14, 1973.

12. "Change No. 17 for Oyster Creek, Docket No. 50-219, License DPR-16" Letter from D. Skovholt to Ivan Finfrock, Jersey Central Power Co., dated November 16, 1973.

13. "Sensitivity Study on BWR/6 Fuel Bundle Response to a Postulated LOCA," C. M. Moser and R. W. Griebe, December 1973.

14. NEDE-10801, "Modeling the BWR/6 Loss-of-Coolant Accident" Core Spray and Bottom Flooding Heat Transfer Effectiveness," J. D. Duncan and J. E. Leonard, March 1973, and "Response to AEC Request for Additional Information on NEDE-10801," May 1973. (Proprietary)

15. NEDO-10993, "Core Spray and Bottom Flooding Effectiveness in the BWR/6," J. D. Duncan and J. E. Leonard, September 1973.

16. "Core Thermal Analyses of a Stainless Steel Clad Heater Rod Bundle," C. M. Moser and R. W. Griebe, December 1973.

CHRONOLOGY

REGULATORY REVIEW OF GENERAL ELECTRIC COMPANY 8X8 FUEL ASSEMBLY

September, 1973	General Electric Company submits report "General Design Information for General Electric Boiling Water Reactor Reload Fuel Commencing in Spring, '74," NEDO-20103.
September, 1973	General Electric Company, Nuclear Fuel Department, submits report "Dresden 3 Nuclear Power Station, Second Reload License Submittal."
September 14, 1973	Memo to A. Giambusso, AEC from P. D. Raymond, "Nine Mile Point Unit 1 - Second Refueling."
October 15, 1973	Niagara Mohawk Power Corporation submits report "Nine Mile Point Unit 1 Safety Analysis for Type 5 and Type 6 Reload Fuel."
October 17, 1973	Memo to V. Stello, AEC, from D. Ross and T. Novak, "Review of GE 8X8 Reload Fuel Assemblies."
October 24, 1973	Memo to V. Stello, AEC, from W. Minners, AEC, "Review of GE 8X8 Reload Fuel Assemblies."
November, 1973	Northern States Power Company submits report "Monticello Nuclear Generating Plant - Second Reload Submittal."
November 16, 1973	Letter from D. Skovholt to Ivan Finfrock, Jersey Central Power Company, "Change No. 17 for Oyster Creek, Docket No. 50-219, License DPR-16."
November 17, 1973	General Electric Company, Nuclear Fuel Department submits Supplement A, "Dresden 3 Nuclear Power Station, Second Reload License Submittal."
December, 1973	Energy Incorporated submits "Sensitivity Study on BWR/6 Fuel Bundle Response to a Postulated LOCA," Part IV, C. M. Moser and R. M. Griebe.
December 6, 1973	General Electric Company, Nuclear Fuel Department submits

	Supplement B, "Dresden 3 Nuclear Power Station. Second Reload License Submittal."
December 6, 1973	General Electric Company, Nuclear Fuel Department submits Supplement C, "Dresden 3 Nuclear Power Station, Second Reload License Submittal."
December 6, 1973	Letter to J. O'Leary, AEC, from J. Abel, Commonwealth Edison, "Supplement B to Second Reload License Submittal."
December 6, 1973	Letter to J. O'Leary, AEC, from J. Abel, Commonwealth Edison, "Supplement C to Second Reload License Submittal and Proposed Change to Facility Operating License DPR-25."
December 14, 1973	Memo to V. Stello, AEC, from W. Minners, AEC, "General Electric 8X8 Reload Fuel Assemblies."
December 17, 1973	General Electric Company, Nuclear Fuel Department submits Supplement D, "Dresden 3 Nuclear Power Station, Second Reload License Submittal."
December 17, 1973	General Electric Company, Nuclear Fuel Department submits Supplement E, "Dresden 3 Nuclear Power Station, Second Reload License Submittal."
December 17, 1973	Letter to D. Ziemann, AEC, from J. Abel, Commonwealth Edison, "Supplement D to the Second Reload License Submittal."
December 17, 1973	Letter to D. Ziemann, AEC, from J. Abel, Commonwealth Edison, "Supplement E to the Second Reload License Submittal."
December 18, 1973	ACRS meeting on GETAB and applications to LOCA analyses for 8X8 assemblies.
January 8, 1974	ACRS Subcommittee on Fuels Meeting, Washington, D.C.
January 10, 1974	ACRS Meeting, Washington D. C.
January 24, 1974	ACRS Subcommittee on Fuels Meeting, Denver, Colorado.

January 30, 1974	AEC - General Electric Meeting.
February 5, 1974	Letter from J. A. Hinds to V. Moore.