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

May 1, 1984

Docket No. 50-244
LS05-84-05-001

See Correction to
Amdt 61 to DPR-18
dtd. 5/18/84

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: USE OF WESTINGHOUSE OPTIMIZED FUEL ASSEMBLY (OFA) AS RELOAD FUEL

Re: R. E. Ginna Nuclear Power Plant

The Commission has issued the enclosed Amendment No. 61 to Provisional Operating License No. DPR-18 for the R.E. Ginna Nuclear Power Plant. This amendment is in response to Rochester Gas and Electric Corporation (RG&E) application dated December 20, 1983.

The staff has reviewed the information submitted by RG&E for an amendment of the Ginna Technical Specifications. The staff finds that, starting with Cycle 14, mixed core operation with Westinghouse (W) and Exxon Nuclear Company (ENC) fuel with transition to a full core of OFA fuel is acceptable for the fuel system mechanical design, nuclear design, thermal-hydraulic design, the transients and accident analyses, and the Technical Specifications proposed. However, the acceptance requires that the sign of $F(\Delta I)$ in the overtemperature ΔT equation (page 2.3-3 of the revised Technical Specifications) be negative.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on March 21, 1983 (49 FR 10591). No request for hearing and no comments were received.

Add: H. Balukjian J. Collins Rm 314
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P PDR

A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's Monthly Notice publication in the Federal Register.

Sincerely,

for D. E. Lyons

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 61 to
   License No. DPR-18
2. Safety Evaluation

cc w/enclosures:
See next page

A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's Monthly Notice publication in the Federal Register.

Sincerely,

Original signed by James Lyons
for

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 61 to License No. DPR-18
2. Safety Evaluation

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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 PROVISIONAL OPERATING LICENSE

Amendment No. 61
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) notarized December 20, 1983 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 Provisional Operating License No. DPR-18 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 61, 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

for J. E. Lyons

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 1, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 61

PROVISIONAL 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 contain the captioned amendment number and marginal lines which indicate the area of changes.

REMOVE	INSERT
page 1-1	page 1-1
page 2.1-2	page 2.1-2
page 2.1-4	page 2.1-4
Figure 2.1-1	Figure 2.1-1
pages 2.3-2 through 2.3-3	pages 2.3-2 through 2.3-3
page 3.1-4b	page 3.1-4b
pages 3.1-17 through 3.1-19	pages 3.1-17 through 3.1-19
page 3.8-3 through 3.8-4	page 3.8-3 through 3.8-4
pages 3.10-3 through 3.10-4	pages 3.10-3 through 3.10-4
Figures 3.10-2, 3.10-3	Figures 3.10-2, 3.10-3

TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

The following terms are defined for uniform interpretation of the specifications.

1.1 Thermal Power

The rate that the thermal energy generated by the fuel is accumulated by the coolant as it passes through the reactor vessel.

1.2 Reactor Operating Modes

Mode	Reactivity $\Delta k/k\%$	Coolant Temperature (°F)
Refueling	≤ -5	$T_{avg} \leq 140$
Cold Shutdown	≤ -1	$T_{avg} \leq 200$
Hot Shutdown	≤ -1	$T_{avg} \geq 540$
Operating	≥ 0	$T_{avg} \sim 580$

1.3 Refueling

Any operation within the containment involving movement of fuel and/or control rods when the vessel head is unbolted.

1.4 Operable

Capable of performing all intended functions in the intended manner.

Amendment No. ~~43~~, 61
Proposed

boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB) and at this point there is a sharp reduction of the heat transfer coefficient which would result in high clad temperatures and the possibility of clad failure. DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters, thermal power, reactor coolant temperature and pressure have been related to DNB through the W-3 and/or WRB-1 DNB correlation. These DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB. A minimum value of the DNB ratio, MDNBR, is specified so that during steady state operation, normal operational transients and anticipated transients, there is a 95% probability at a 95% confidence level that DNB will not occur.(1) The curves of Figure 2.1-1 represent the loci of points of thermal power, coolant system pressure and average temperature for which this minimum DNB value is satisfied. The area of safe operation is below these lines.

Since it is possible to have somewhat greater enthalpy rise hot channel factors at part power than at full power due to the deeper control bank insertion which is permitted at part power, a conservative allowance has been made in obtaining the curves in Figure 2.1-1 for an increase in $F^N_{\Delta H}$ with decreasing power levels. Rod withdrawal block and load runback occurs before reactor trip set points are reached.

The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure and thermal power level that would result in there being less than a 95% probability at a 95% confidence level that DNB would not occur.[3]

(1) FSAR, Section 3.2.2

(2) FSAR, Section 3.2.1

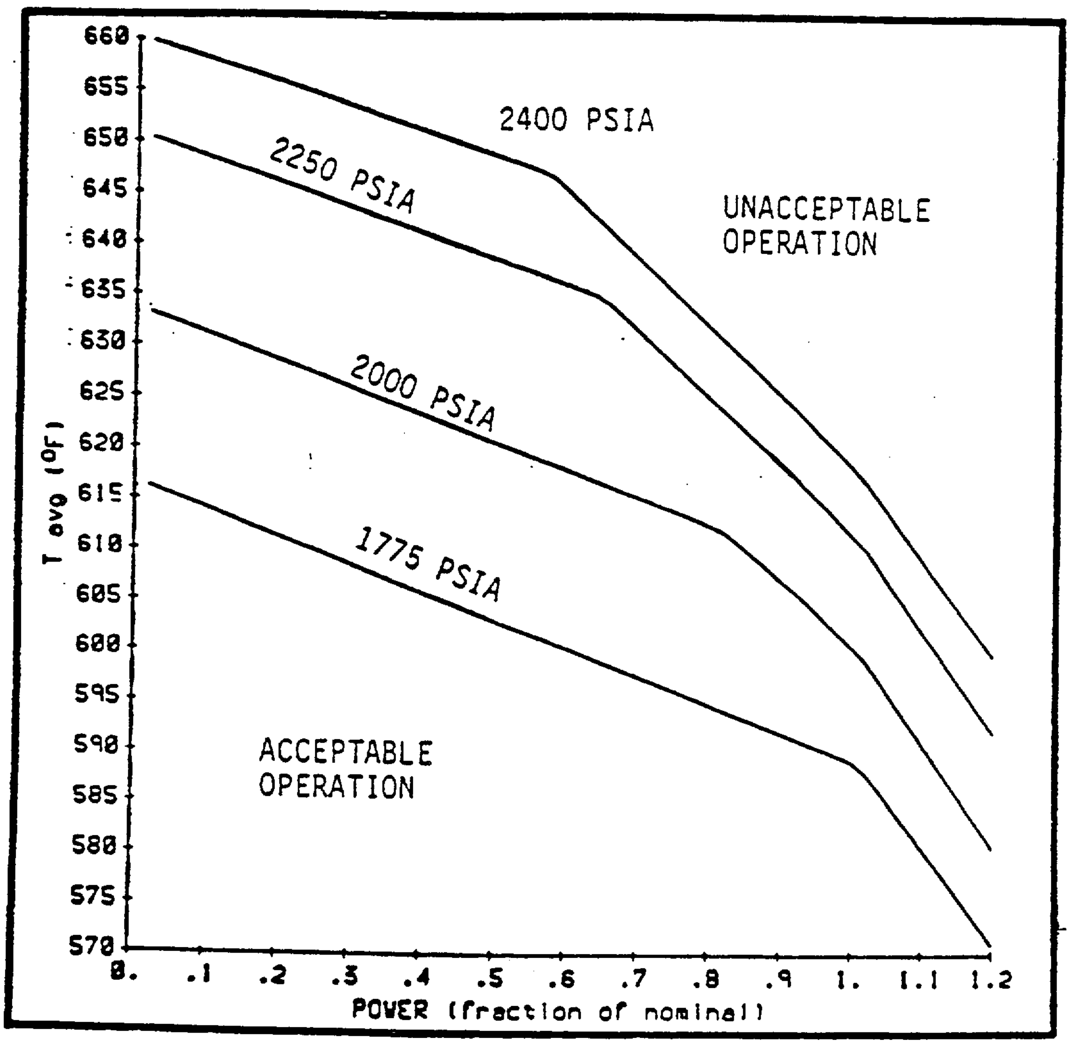
(3) FSAR, Section 14.1.1

Amendment No. ~~10~~, 61
March 30, 1976
Proposed

FIGURE 2.1-1

CORE DNB SAFETY LIMITS

2 LOOP OPERATION



d. Overtemperature ΔT

$$\leq \Delta T_o \; [K_1 + K_2(P-P^1) - K_3(T-T^1) \quad (\frac{1 + \tau 1S}{1 + \tau 2S})] \; -f(\Delta I)$$

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T^1 = 573.5°F

P = pressurizer pressure, psig

P^1 = 2235 psig

K_1 = 1.20

K_2 = .000900

K_3 = .0209

$\tau 1$ = 25 sec

$\tau 2$ = 5 sec

and f (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is the total core power in percent of rated power such that:

(i) for $q_t - q_b$ less than +21 percent, f (ΔI) = 0

Amendment No. ~~10~~, 61
March 30, 1976
Proposed

(ii) for each percent that the magnitude of $q_t - q_b$ is more positve than +21 percent, the ΔT trip set point shall be automatically reduced by an equivalent of 1.6 percent of rated power.

e. Overpower ΔT

$$\leq \Delta T_o \left[K_4 - K_5(T-T^1) - K_6 \frac{\tau 3\ ST}{\tau 3S + 1}\right] - f(\Delta I)$$

where

ΔT_o = indicated ΔT at rated power, °F

T = average temperature, °F

T^1 = indicated T avg at nominal conditions at rated power, °F

K_4 = 1.077

K_5 = 0.0 for $T<T^1$

= 0.0011 for $T \geq T^1$

K_6 = 0.0262 for increasing T

= 0.0 for decreasing T

$\tau 3$ = 10 sec

$f(\Delta I)$ = as defined in 2.3.1.2.d.

Amendment No. ~~10~~, 61
March 30, 1976
Proposed

3.1.1.5 Pressurizer

Whenever the reactor is at hot shutdown or critical the pressurizer shall have at least 100 kw of heaters operable and a water level maintained between 12% and 87% of level span. If the pressurizer is inoperable due to heaters or water level, restore the pressurizer to operable status within 6 hrs. or have the RHR system in operation within an additional 6 hrs.

Bases

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above the limit value during all normal

Change No. ~~12~~
Amendment No. ~~23~~, ~~26~~, ~~42~~, ~~43~~, 61
Proposed

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at a temperature below 500°F, and if the moderate temperature coefficient is more positive than

a. 5 pcm/°F (below 70 percent of rated thermal power)
b. 0 pcm/°F (at or above 70 percent of rated thermal power)

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit line shown on Figure 3.1-1 of these specifications.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

Basis

Previous safety analyses have assumed that for Design Basis Events (DBE) initiated from the hot zero power or higher power condition, the moderator temperature coefficient (MTC) was either zero or negative.(1)(2) Beginning in Cycle 14, the safety analyses have assumed that a maximum MTC of +5 pcm/°F can exist up to 70% power. Analyses have shown that the design criteria can be satisfied for the DBE's with this assumption.(3) At greater than 70% power the MTC must be zero or negative.

The limitations on MTC are waived for low power physics tests to permit measurement of the MTC and other physics design parameters of interest. During these tests special operating precautions will be taken.

The requirement that the reactor is not to be made critical above and to the left of the criticality limit provides increased assurance that the proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the specified shutdown margin is maintained, there is no possibility of an accidental criticality as a result of an increase in moderator temperature or a decrease of coolant pressure.

Reference

(1) FSAR Table 3.2.1-1

(2) FSAR Figure 3.2.1-8

(3) Safety Evaluation for R. E. Ginna Transition to 14 x 14 Optimized Fuel Assemblies, Westinghouse Electric Corporation, November 1983.

to public health and safety.[1] Whenever changes are not being made in core geometry one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat pump is used to maintain a uniform boron concentration.

The shutdown margin as indicated will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 230,000 gallons of borated water. The boron concentration of this water at 2000 ppm boron is sufficient to maintain the reactor subcritical by at least 5% $\Delta k/k$ in the cold condition with all rods inserted (best estimate of 10% subcritical), and will also maintain the core subcritical even if no control rods were inserted into the reactor.[2] Periodic checks of refueling water boron concentration insure the proper shutdown margin. Communication requirements allow the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

In addition to the above safeguards, interlocks are utilized during refueling to insure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time. In addition interlocks on the auxiliary building crane will prevent the trolley from being moved over storage racks containing spent fuel.

The operability requirements for residual heat removal loops will ensure adequate heat removal while in the refueling mode. The requirement for 23 feet of water above the reactor vessel flange while handling fuel and fuel components in containment is consistent with the assumptions of the fuel handling accident analysis.

<u>References</u>:

(1) FSAR - Section 9.5.2

(2) Reload Transition Safety Report, Cycle 14

(3) FSAR - Section 9.3.1

average power tilt ratio shall be determined once a day by at least one of the following means:

a. Movable detectors

b. Core-exit thermocouples

3.10.2.2 Power distribution limits are expressed as hot channel factors. At all times, except during low power physics tests the hot channel factors must meet the following limits:

$F_Q(Z) = (2.32/P)*K(Z)$ for $P \geq .5$

$F_Q(Z) = 4.64*K(Z)$ for $P \leq .5$

$F^N_{\Delta H} = 1.66\ [1 + .3(1-P)]$ for $0 \leq P \leq 1.00$

where P is the fraction of rated power at which the core is operating, K(Z) is the function given by Figure 3.10-3, and Z is the height in the core. The measured F_Q^N shall be increased by three percent to yield F_Q. If the measured F_Q or $F\Delta_H^N$ exceeds the limiting value, with due allowance for measurement error, the maximum allowable reactor power level and the Nuclear Overpower Trip set point shall be reduced on percent for each percent which $F\Delta_H^N$ or F_Q exceeds the limiting value, whichever is more restrictive. If the hot channel factors cannot be reduced below the limiting values within one day, the Overpower ΔT trip setpoint and the Overtemperature ΔT trip setpoint shall be similarly reduced.

3.10.2.3 Except for physics tests, if the quadrant to average power tilt ratio, exceeds 1.02 but is less than 1.12, then within two hours:

a. Correct the situation, or

b. Determine by measurement the hot channel factors, and apply Specification 3.10.2.2, or

c. Limit power to 75% of rated power.

3.10.2.4 If the quadrant to average power tilt ratio exceeds 1.02 but is less than 1.12 for a sustained period of more than 24 hours without known cause, or if such a tilt recurs intermittently without known cause, the reactor power level shall be restricted so as not to exceed 50% of rated power. If the cause of the tilt is determined, continued operation at a power level consistent with 3.10.2.2 above, shall be permitted.

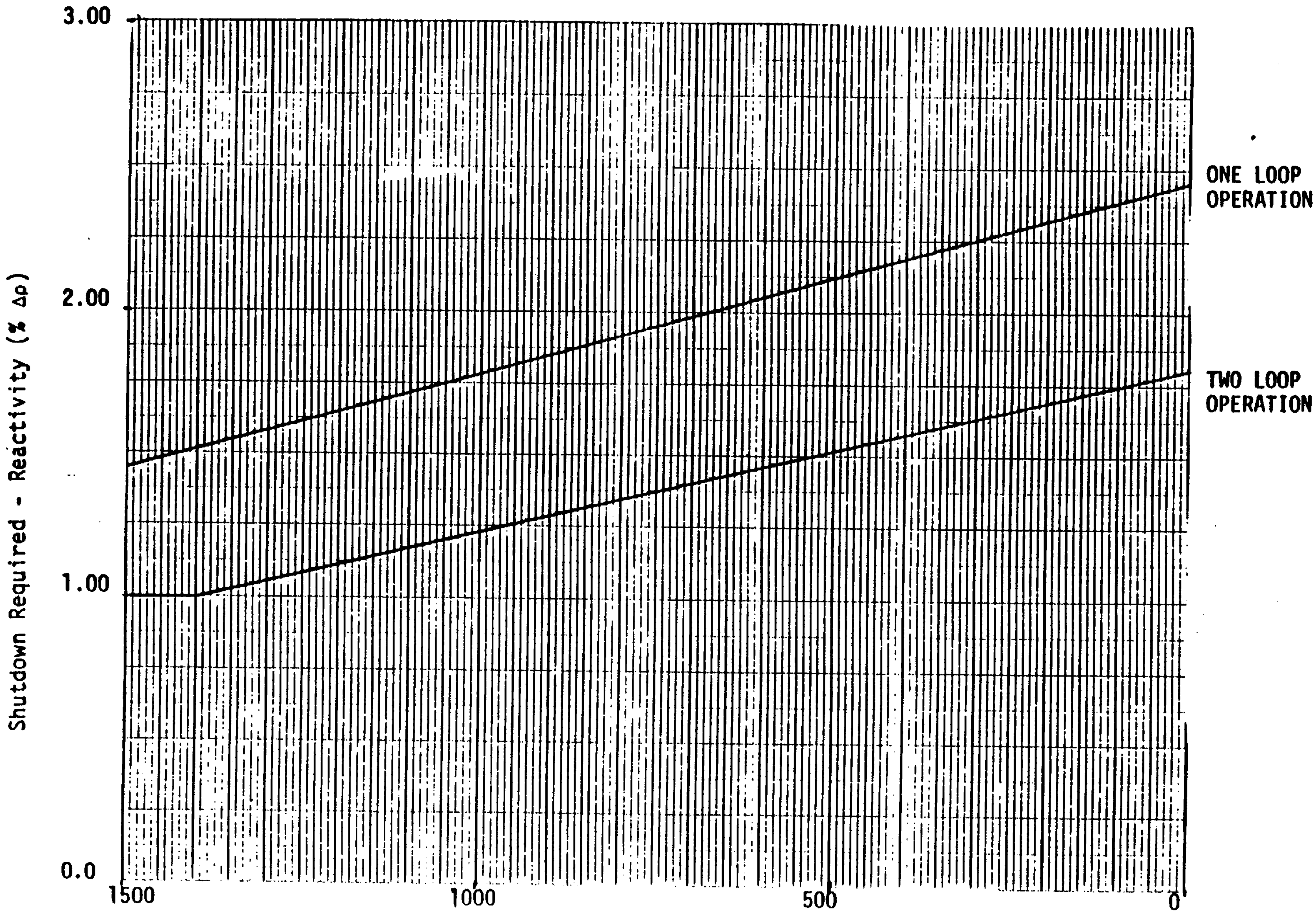
3.10.2.5 Except for physics test, if the quadrant to average power tilt ratio is 1.12 or greater, the reactor shall be put in the hot shutdown condition utilizing normal operating procedures. Subsequent operation for the purpose of measuring and correcting the tilt is permitted provided the power level does not exceed 50% of rated power and the Nuclear Overpower Trip "set point is reduced by 50%".

3.10.2.6 Following any refueling and at least every effective full power month thereafter, flux maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.2 are met.

3.10.2.7 The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux difference must be updated at least each equivalent full power month using a measured value or by linear interpolation using the most recent measured value and the predicted value at the end of the cycle life.

3.10.2.8 Except during physics tests, control rod exercises, excore detector calibration, and except as modified by 3.10.2.9 through 3.10.2.12, the indicated axial flux difference shall be maintained within ±5% of the target flux difference (defines the target band on axial flux difference). Axial flux difference for power distribution control is defined as the average value for the four excore detectors. If one excore detector is out of service, the remaining three shall be used to derive the average.

3.10.2.9 Except during physics tests, control rod exercises, or excore calibration, at a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band. The flux difference shall be returned to the target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.



Amendment No. ~~10~~, ~~40~~, 61
PROPOSED

REQUIRED SHUTDOWN MARGIN

FIGURE 3.10-2

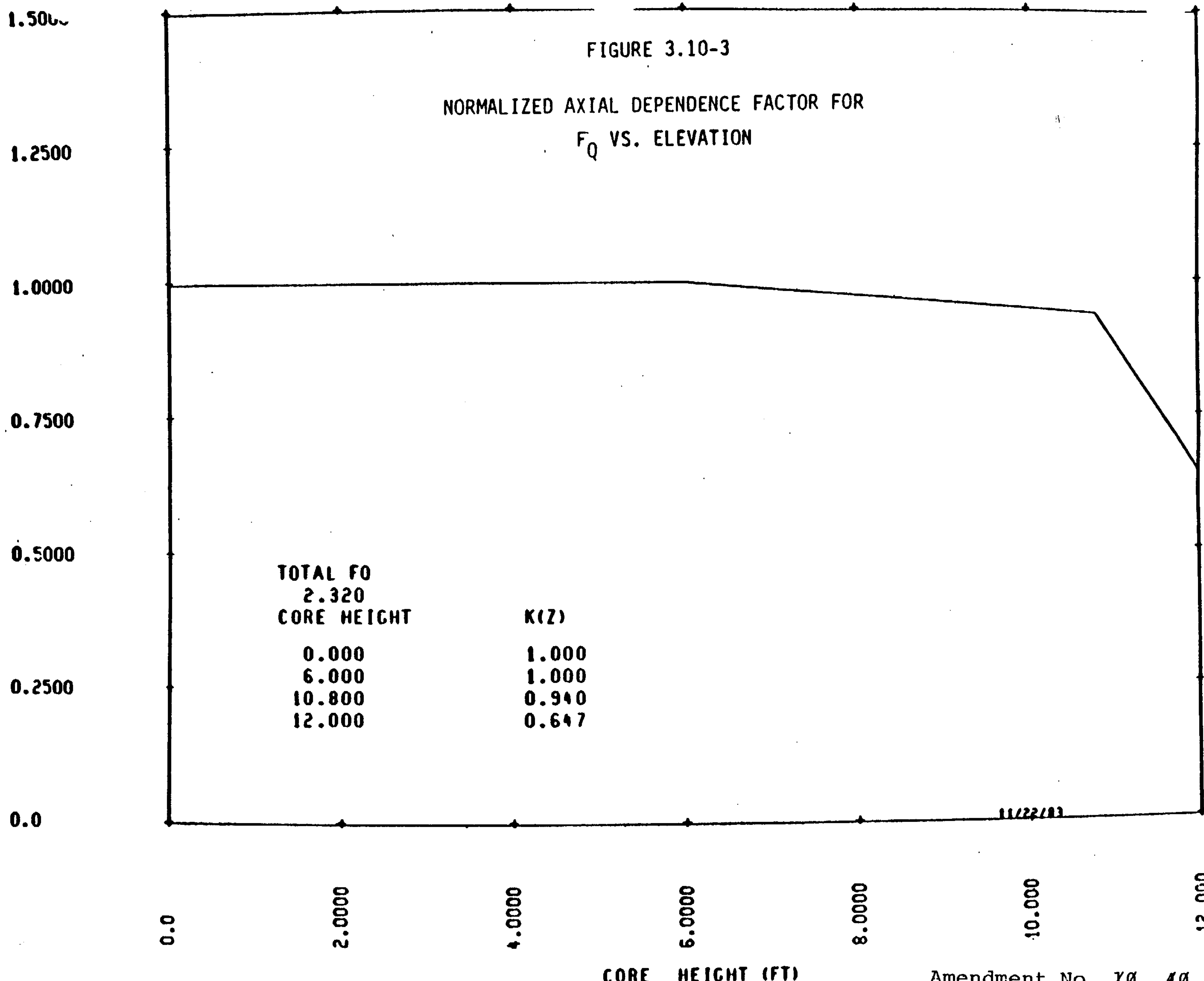


FIGURE 3.10-3

NORMALIZED AXIAL DEPENDENCE FACTOR FOR F_Q VS. ELEVATION

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 61 TO PROVISIONAL 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 December 20, 1983, Rochester Gas and Electric Corporation (RG&E) (the licensee) for the R. E. Ginna Plant, submitted a request (Ref. 1) for an amendment of the Technical Specifications. Commencing with Cycle 14 the licensee will change vendors for reload fuel from Exxon (ENC) to Westinghouse (W) with W performing the reload analysis. In support of the application, Attachments A, B and C were appended to RGE's submittal which set forth the requested change in Technical Specifications, its safety evaluation and the basis for determining that the change does not involve a significant hazards consideration. The "safety evaluation" (Attachment B to Reference 1) addressed the mechanical, nuclear, thermal-hydraulic and accident analysis considerations.

A Notice of Consideration of Issuance of Amendment and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on March 21, 1984 (49 FR 10591). A request for hearing and public comments were not received.

2.0 FUEL SYSTEM DESIGN

The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and Anticipated Operational Occurrences (AOO), (b) fuel system damage is never so severe as to prevent control rod insertion when required, (c) the number of fuel rod failures is not under-estimated for postulated accidents, and (d) coolability is always maintained. The staff's evaluation of the information provided in support of the proposed Technical Specification changes is described in this Safety Evaluation with regard to these review objectives.

2.1 Description

R. E. Ginna is a Westinghouse designed PWR and is currently operating with an all Exxon Nuclear Company (ENC) 14 x 14 fueled core except for four W Mixed Oxide (MOX) assemblies. R. E. Ginna was last supplied with W fuel during the cycle 7 reload. Cycle 14 will be the first cycle in a transition phase from ENC to W 14 x 14, 9 grid Optimized Fuel Assembly (OFA) fuel with core loadings ranging from approximately a 15% OFA and 85% ENC fueled core (actually 20 OFA, 97 ENC and 4 MOX assemblies) to eventually an all-OFA-fueled core. The OFA fuel is a new design but similar to W seven grid 14 x 14 low parasitic fuel (LOPAR) which has had substantial operating performance in a number of plants (Ref. 1).

The similarities between the OFA design and previous W fuel include the number of rods, grids, guide thimbles and instrumentation tube. Also the rod to rod spacing is the same. The materials of the top and bottom nozzles (stainless steel), fuel rod (Zircaloy), and top and bottom grids (Inconel) are the same in both the W OFA and initial fuel designs. The elevation of the centerline of each of the OFA grids matches that of the ENC grids in order to minimize cross flow during operation.

The design changes between the two designs include a reduction in the OFA fuel rod, guide thimble, and instrumentation tube diameters and cladding thickness. In addition to the reduction of the fuel rod diameter, 6.2 inches of natural uranium pellets replace the standard slightly enriched pellets at both ends of the fuel stack (axial blanket). Also, there is a change of material of the guide thimble and instrumentation tube from stainless steel to Zircaloy. Of the nine grids, the seven intermediate grids have been changed from Inconel to Zircaloy. To retain the required grid strength, the thickness and height of Zircaloy grids has been increased. The Zircaloy grid height is 2.25 inches as compared to Inconel height of 1.5 inches.

Another design change from previous W fuel is the OFA reconstitutable bottom nozzle feature which is similar to that introduced in other W plants such as Trojan, Farley Units 1 and 2, Salem Unit 1, North Anna Units 1 and 2. In this design a locking cup is used to lock the thimble screw of a guide thimble tube in place. This facilitates easy removal of the bottom nozzle from the fuel assembly.

The W OFA thimble tubes, fabricated from Zircaloy, have two sections with a large diameter and two with a smaller diameter. The larger diameter at the top permits rapid control rod insertion. Both of the reduced diameter sections produce a dashpot action near the end of the control rod travel to decelerate the control rod and reduce impact forces.

The instrumentation tube, also fabricated from Zircaloy, is of constant diameter and is designed to accept the R. E. Ginna incore instrumentation. The OFA instrumentation tube has a 0.004 inch diametral increase when compared to the ENC assembly instrumentation tube. There is sufficient diametral clearance for the instrumentation thimble to traverse the OFA instrumentation tube.

2.2 Design Evaluation

The design and safety analysis of the OFA fuel assembly is discussed in WCAP-9500 (Reference 3) which the staff has reviewed and found acceptable. However, the staff SER of WCAP-9500 requires that certain items be addressed on a plant specific basis. Reference 2 includes RG&E's responses to staff questions on these plant specific items and are discussed below.

2.2.1 Fuel Design Comparison

Table 1 compares fuel assembly, fuel rod and fuel pellet design information for the original W HIPAR fuel, the current Exxon fuel and the new W OFA fuel. Table 1 includes information on materials used and dimensions. The new

TABLE 1

Fuel Design Comparison

Comparison Basis	W-HIPAR 14 x 14 Fuel	ENC 14 x 14 Fuel	W-OFA 14 x 14 Fuel
Fuel Assemblies:			
Number of Fuel Assemblies	121	121	121
UO_2 Rods per Assembly	179	179	179
Rod Pitch, in.	.556	.556	.556
Assembly Pitch	7.803	7.803	7.803
Number of Grids Per Assembly	9	9	9
Material	SS-304	ZRC w INCONEL SPRINGS	7-ZRC 2-INCONEL
Fuel Rods:			
Number	21,659	21,659	21,652
Clad O.D., in.	.422	.424	.400
Diametral Gap, in.	.0075	.0075	.0070
Clad Thickness, in	.0243	.030	.0243
Clad Material	ZRC	ZRC	ZRC
Fuel Pellets:			
Material	UO_2	UO_2	UO_2
Density (% Theoretical)	94	94	95
Diameter, in	.3659	.3565	.3444
Length	-	.410	.5650

OFA fuel assemblies have the same configuration as the Exxon fuel. The fuel pellets are slightly smaller in diameter and slightly longer. The clad O. D., clad thickness and diametral gap is slightly smaller than for the ENC fuel.

2.2.3 Design for Seismic and LOCA Forces

The licensee stated that Westinghouse has analyzed the projected grid impact forces during a seismic event by postulating a mixed core of Westinghouse HIPAR (original fuel in Ginna) and OFA fuel. Because the estimated grid impact stiffness of an Inconel grid of the HIPAR design would be greater than that of the Exxon Zircaloy grid, this configuration would present an upper bound grid force condition for the OFA design. The resulting forces on the OFA were found to be acceptable.

The licensee stated that the upper bound grid force condition for the Exxon fuel would also be bounded by those resulting from that mixed core configuration with Westinghouse HIPAR fuel which existed over the previous six operating cycles. Therefore, this case was not reanalyzed.

The vibrational characteristics of the Exxon and Westinghouse designs are essentially equivalent. This was verified during the flow testing of an Exxon fuel assembly in the FACTS facility at the R. E. Ginna site.

Therefore the mixing of the Westinghouse OFA and Exxon fuel types with respect to seismic and LOCA loads is acceptable for Ginna.

2.2.4 Surveillance

Since this is the first substantial application of 14 x 14 nine grid W OFA fuel (excluding lead test assemblies) a visual surveillance is to be performed. This is to be conducted in the containment area for a reasonable number of OFA fuel assemblies until they complete their fuel cycles and are put into the spent fuel pool. We find this to be deisirable and acceptable.

2.2.5 ECCS Calculation LOCA Cladding Models

The licensee stated that the 1981 version of the ECCS evaluation model for Ginna includes NRC supplied LOCA cladding models as described in NUREG-0630, burst/blockage models. Additional information regarding the models are in WCAP-9220-P-A, Rev. 1, February 1982. The licensee has used updated approved models and, therefore, no supplemental calculations are necessary.

2.2.6 Initial Fuel Conditions for Transient Analysis

RG&E stated that the initial fuel temperatures used in the R. E. Ginna Cycle 14 transient and accident analyses were calculated using the NRC approved W fuel performance code, PAD-3.3 (Miller, J. V. [Ed]), "Improved Analytical Model Used in Westinghouse Fuel Rod Design Computations, "WCAP-8785, October 1976). In using PAD to generate fuel temperatures for input to safety analyses calculations, a conservative thermal safety model was used ("Westinghouse Revised PAD Code Thermal Safety Model, "WCAP-8720, Addendum 2 [Proprietary]). Calculations of initial fuel stored energy used in safety analyses were also based on the results of conservative fuel average temperature calculations at the time of maximum densification. As a result, fuel temperatures at the end of one cycle are significantly less than those occurring at the time of maximum denisification. RG&E stated that, considering the similarity of the W and ENC fuel, the initial fuel temperatures, calculated for W fuel and used in the transient analysis, envelopes the thermal condition of all fuel in the core for Cycle 14. The staff finds this acceptable as an approved code and as a conservative model used.

2.2.7 Predicted Clad Collapse Time

The RG&E evaluation was performed using WCAP-8377, "Revised Clad Flattening Model," R. A. George, July 1974. WCAP-8377 is an NRC approved report. The licensee stated that calculations based on WCAP-8377, will be performed for each cycle of operation and will be presented in the cycle specific RG&E Reload

Safety Evaluations. Calculations performed for the W 14 x 14 OFA fuel in cycle 14 confirmed that clad flattening criteria are met for the projected fuel residence time. The staff finds this acceptable.

3.0 NUCLEAR DESIGN

RG&E requested changes in the Technical Specfications which allow the gradual substitution of the present ENC assemblies with W 14 x 14 OFA starting with Cycle 14 until the complete conversion to a W core from the present ENC loading. The fuel mix will range from 15 percent OFA, 85 percent ENC to 100 percent OFA. Except for the moderator temperature coefficient (MTC) the nuclear characteristics of the ENC to W OFA conversion are generally within the range of the cycle-to-cycle variations observed in previous reload designs.

The nuclear design and analysis of the Ginna core were performed with the standard W reload safety evaluation methodology. No changes in the nuclear design methodology or models were necessary due to the transition to OFA. The most important nuclear design parameter change is the positive value of the MTC, for which the estimated maximum value is .6 pcm/°F expected to occur at the Beginning of Cycle (BOC) of the 100 percent W OFA core. The parameters were chosen to maximize the applicability of the transition evaluation of Cycle 14 and future cycles until the completion of the fuel conversion. In particular, conservatively positive values of the MTC were assumed in the accident evalutions. In general, the neutronic parameters used as input to the safety evalution were chosen to bound the values obtained from the transition cycles. Therefore, the analysis presented establishes a reference base for the conversion (cores containing any combination of ENC and OFA fuel) and operation of future equlibrium OFA loadings. The required shutdown margin (SDM) was computed using the negative temperature coefficient corresponding to the end of cycle life and assuming all but the most reactive rod inserted. The required value of the SDM was found to be 1.8 percent p. The licensee will perform whole core power distribution measurements at startup (in addition to administrative procedures) to assure against fuel misloading. Likewise the licensee will assure that future cycles comply with the calculated values and bounds of this analysis. The applicant includes a listing of the neutronic parameters used in the safety analysis to provide bounding values against which cycle dependent parameters may be compared. The staff concludes that the nuclear design is acceptable.

4.0 THERMAL-HYDRAULIC EVALUATION

4.1 Thermal-Hydraulic Design Comparison

The licensee supplied information on the thermal-hydraulic design comparison including Table 2, which compares values for the original 14 x 14 W HIPAR, ENC, and W OFA fuel. For the current Ginna core which will include both ENC and W OFA fuel, the following values are constant in Table 2: reactor heat input, system pressure, total flow rate, nominal reactor inlet temperature, average temperature rise in the vessel, average linear heat rate (kw/ft) and maximum thermal output (kw/ft). The active heat transfer surface area for the OFA fuel is smaller than for the Exxon fuel; however, the heat flux is higher. Also the effective flow rate for heat transfer is higher for the OFA fuel (fuel rods have smaller O. D.) than for the ENC fuel, but the average velocity along the fuel rods is lower.

4.2 Thermal-Hydraulic Compatibility

The W OFA and ENC fuel assemblies have been tested for hydraulic characteristics using the W Fuel Assembly Compatibility Systems (FACTS) loop (References 4 and 5). In this test, the regular seven grid OFA was used which has two less mixing vane grids than the nine grid OFA used for Ginna. Ginna also has a slightly shorter fuel length. These design differences were addressed in evaluating the hydraulic characteristics of the test assembly. The results of the evaluation

Table 2

Reactor Design Comparison

Comparison Basis	W-HIPAR 14 x 14 Fuel	ENC 14 x 14 Fuel	W-OFA 14 x 14 Fuel
Performance Characteristics:			
Reactor core heat output (Mwt)	1520	1520	1520
System pressure (psia)	2250	2250	2250
Minimum DNBR			
Typical cell	1.3	1.58*	1.34
Thimble cell	1.3	1.50*	1.33
Critical heat flux correlation	W-3	W-3 & ITDP	WRB-1 & ITDP
Coolant Flow:			
Total flow rate (10^6 lb/hr)	68.0	67.9	67.9
Effective flow rate for heat transfer (10^6 lb/hr)	64.9	64.8	67.8
Average velocity along fuel rods (ft/sec)	14.7	14.8	14.3
Coolant Temperature °F:			
Nominal reactor inlet	544.5	543.7	543.7
Average rise in vessel	58.0	59.6	59.6
Heat Transfer, 100% Power:			
Active heat transfer surface (ft^2)	28,715	28,450	27,200
Average heat flux (Btu/hr-ft^2)	176,700	177,030	189,440
Maximum heat flux (Btu/hr-ft^2)	409,944	410,710	439,501
Average linear heat rate (kW/ft)	5.36	5.4	5.4
Maximum thermal output (kW/ft)	12.4	12.6	12.6

*For Conditions I & II events, the Exxon fuel in the transition core has been evaluated using the W-3 correlation and ITDP methodology.

showed that the overall loss coefficient between the ENC and OFA fuel is less than 1 percent and that the two assemblies are, therefore, hydraulically compatible. RG&E has stated (Ref. 2) that the pressure drop difference between an all OFA core and an all ENC core for Ginna is approximately 0.2 psia and the change in flow rate is less than 0.1 percent. The measured primary system flow in the last cycle (cycle 13) was approximately 97,500 gpm per loop (195,000 gpm total for the two loops) which is well over the design flow of 87,000 gpm per loop (174,000 gpm total for both loops).

4.3 Fuel Assembly Hydraulic Lift-Off

From the precision flow calorimetric in Cycle 13, the value for the reactor system flow obtained was approximately 195,000 gpm. The hold-down springs of the W OFA assembly are designed to withstand lift-off of the assembly up to a flow rate of 100,000 gpm/loop or 200,000 gpm system flow and should therefore resist lift-off. Additional conservatism has also been built into the analysis to account for uncertainties in thermal and hydraulic parameters, fuel assembly hydraulic resistance, and worst case inlet flow maldistribution factors.

4.4 Thermal-Hydraulic Analysis

The thermal-hydraulic analysis of this mixed core was performed using the Improved Thermal Design Procedures (ITDP) (Ref. 6) and the THINC-IV code (Refs. 7 and 8). The WRB-1 (Ref. 9) and W-3 (Refs. 10 and 11) Critical Heat Flux (CHF) correlations were used for the W OFA and the ENC fuel assemblies, respectively. The ITDP and the THINC-IV code used with both CHF correlations have previously been approved by the staff. However, additional areas were examined regarding this transitional mixed core configuration. These areas are addressed as follows:

(a) The licensee supplied information in Reference 2 as required by WCAP-9500 for plants using the W ITDP. This included information on sensitivity factors. The sensitivity factors are different for the two different fuel types (W and ENC) because the WRB-1 DNB correlation is used for the W fuel and the W-3 DNB correlation is used on the ENC fuel. The S_i values used in the R. E. Ginna analyses are different than those used in WCAP-9500 because the WCAP-9500 sensitivity values are not applicable to 14 x 14 fuel geometries. The uncertainty allowance calculations for W and ENC fuel were provided. A generic W report including block diagrams and a supplemental attached table was provided which gave the various uncertainties used for Ginna. These uncertainties were stated to conservatively bound those associated with Ginna instrumentation.

The uncertainties that have been used in the DNB calculations were provided. These uncertainties were stated to conservatively bound actual Ginna plant parameters. The licensee stated that for R. E. Ginna, the THINC-IV code and the WRB-1 DNB correlation are the same as that used in WCAP-9500 for the W OFA fuel. The W-3 DNB correlation has been used for the ENC fuel. The licensee stated that all parameter values are within the ranges of the codes and correlations used, and sensitivity factors have been determined specific to the fuel type over the range of Ginna plant parameters. The staff has reviewed the information provided and has found it in conformance with that required in WCAP-9500 and acceptable.

(b) The WRB-1 correlation was approved for the 17 x 17 OFA, and 17 x 17 and 15 x 15 standard LOPAR fuel assemblies with a DNBR limit of 1.17 for the R-grid.

The licensee provided information to justify the use of the WRB-1 CHF correlation for the 9 grid 14 x 14 W OFA fuel assemblies. The 14 x 14 OFA DNB test results were provided to the NRC in reference 12 which contains supplement 1 to WCAP-8762 (Ref. 13). These test results were used to demonstrate that the WRB-1 CHF correlation correctly accounted for the geometry changes from the 0.422-inch R grid design to the 14 x 14 OFA design. The DNB safety analyses for Ginna have been performed with the grid spacing term in the WRB-1 correlation set equal to 22 inches, the longest grid spacing in the assembly. The WRB-1 correlation has been shown to accurately predict the 0.422 R grid CHF performance with grid spacings of 13 to 32 inches (reference 12). Although the review of reference 12 has not yet been completed, the review to date indicates the WRB-1 correlation is applicable to the Ginna 14 x 14 OFA fuel as the range of data covers the spacing for the 9 grid design for Ginna. The Cycle 14 analysis indicates an available thermal margin on the order of 10%. Therefore, we find the Ginna safety analysis for this reload to be acceptable.

(c) The use of ITDP for the analysis of a transitional mixed core has been previously reviewed by the staff and approved with a condition requiring a penalty on DNBR to account for the uncertainty associated with the interbundle cross-flow in the mixed core. The licensee has performed an analysis to determine the required penalty factor in the same manner approved for the 17 x 17 OFA/LOPAR mixed core analysis. The result shows that a 2% penalty is required on the OFA fuel and 1% on the Exxon fuel for the Cycle 14 transitional core.

(d) The licensee provided information on rod bow penalties for both the ENC and W fuel. The maximum projected assembly burnup for Cycle 14 for an Exxon assembly will not be greater than 41,000 MWD/MTU. Since the ENC fuel assembly has thicker cladding

(Ref. 14) it is expected to have less gap closure than the Westinghouse OFA fuel assembly. Using equations 3.2 and 3.4 of XN-NF-75-32, Supplement 1, the resulting predicted gap closure would be less than 40 percent. References 15 and 16, indicate there is no effect on DNB for gap closures less than 55%. Therefore, no rod bow penalty is required for the Exxon fuel.

The W OFA fuel assembly for R. E. Ginna has nine grids and an active fuel length of 141.4 inches. The fractional closure at any given burnup for Ginna can be compared to that of a 7-grid assembly using relevant parameters. The relevent parameters for making such a comparison are L^2/I (L = span length between grids, I = fuel rod moment of inertia) and the initial rod-to-rod gap. The L/I ratio is higher for the OFA, but the initial rod-to-rod gap is also larger, therefore, these effects offset each other. The fractional closure at any burnup for the 9-grid W OFA can be obtained by direct L^2 scaling from that of the 7-grid 14 x 14 assembly. The licensee supplied a table listing rod bow penalty vs burnup (MWD/MTU) and corresponding closure. The results indicated that there is a maximum rod bow penalty of 4.2% DNBR at a burnup of 33,000 MWD/MTU.

According to the approved topical report WCAP-8691, Revision 1 (ref. 17), by the time the fuel attains a burnup of 33,000 MWD/MTU it is not capable of achieving limiting peaking factors due to the decrease in fissionable isotopes and buildup of fission product inventory. This physical burndown effect is greater than the rod bow predicted at those burnups. Therefore, for the purpose of evaluating effects of rod bow, 33,000 MWD/MTU represents the maximum burnup of concern for which the rod bow penalty is 4.2% DBNR as stated above.

For the W OFA fuel assemblies sufficient margin (11.9%) between the safety analysis DNBR and the design limit DNBR, as shown below, is available to accommodate this penalty as well as the transition core DNB penalty.

	W 14 x 14 OFA		ENC 14 x 14	
	Typical	Thimble	Typical	Thimble
Correlation	WRB-1	WRB-1	W-3	W-3
Correlation Limit	1.17	1.17	1.30	1.30
Design Limit	1.34	1.33	1.58	1.50
Safety Analysis Limit	1.52	1.51	1.62	1.54

The DNBR margin is defined as:

$$\text{Safety analysis DNBR value} = \frac{\text{Design DNBR value}}{1 - \text{Margin}}$$

(e) The core thermal-hydraulic analysis was performed using 1520 MWT core power, 2250 psia system pressure, a nominal Tave of 573.5°F. and 174,000 gpm primary system thermal design flow. The DNBR design limits using ITDP are shown in the table above for both typical and thimble cells. For the Westinghouse OFA fuel the WRB-1 correlation with a DNBR limit of 1.17 was used and the safety limit for Ginna is 11.9% higher than the design limit. This margin is more than enough to account for the rod bow penalty and transitional core penalty. For the ENC fuel, the W-3 correlation with a DNBR limit of 1.30 was used, and the safety limit is approximately 2.5% higher than the design limit. This margin is more than enough to account for the transitional core penalty. There is no rod bow penalty associated with the ENC fuel. The staff concludes that the thermal-hydraulic analysis is acceptable.

5. TRANSIENT AND ACCIDENT ANALYSIS

Most of the non-LOCA transients and accidents were reanalyzed to include the major changes for R. E. Ginna, i.e., the OFA design, the $F_{\Delta H}$ multiplier and a positive MTC. The FSAR (Chapter 14) was revised to include the methodology, the results and the conclusions of each accident reanalyzed. Accidents and transients which were not reanalyzed were those which resulted in excessive heat removal from the reactor coolant system for which a negative MTC is conservative and those which are not sensitive to the moderator coefficient.

The main mechanical difference in the OFA design is the smaller fuel rod, which results in higher fuel rod temperature and lower coolant flow velocity (because of a larger hydraulic diameter) which in turn leads to lower DNBR. This DNBR penalty was offset with the use of the WRB-1 DNB correlation and the improved thermal design procedure. The proposed Technical Specification change for the MTC requires +5 pcm/°F MTC below 70 percent of rated power and 0 pcm/°F MTC above 70 percent of rated power. The transients which have been analyzed were based on +5 pcm/°F MTC, which was assumed to remain constant with temperature.

Exceptions to the above are the control rod withdrawal from subcritical and control rod ejection which are analyzed with TWINKLE (which has automatic temperature feedback) with an initial MTC of +5 pcm/°F, but less positive values at higher power levels. Finally the hot channel factor change to $F_{\Delta H} = 1.66\ (1.0 + .3(1-P))$ where P is the fraction of full power and .3 the power correction constant (adjusted from .2). The safety analysis is not affected by the power correction constant change because: (a) the effect on the accident analyses is through the core safety limits at high pressure and low power levels, for which plant protection is effected through steam generator (SG) safety valve settings, which have not changed and (b) the effect can also be manifested by its impact on the axial offset envelope which, however, has not been changed.

In addition to the large LOCA the following accident and transients have been reanalyzed:

- uncontrolled Rod Cluster Control Assembly (RCCA) withdrawal from a subcritical condition;
- uncontrolled RCCA bank withdrawal at power;
- RCCA drop;
- chemical and volume control system malfunction;
- startup of an inactive reactor coolant loop;
- reduction in feedwater enthalpy incident;
- excessive load increase incident;
- loss of reactor coolant flow/locked rotor;
- loss of external electrical load;
- loss of normal feedwater/station blackout;
- rupture of a steam pipe; and
- rupture of a control rod mechanism housing-RCCA ejection.

All the transients and accidents and the LOCA were reanalyzed using approved methods and acceptable initial conditions. The results presented were acceptable since they did not violate applicable criteria.

5.1 Non-LOCA Accident Analyses

A discussion of each transient listed above which shows sensitivity to the OFA and the proposed change in the MTC, follows:

5.1.1 Uncontrolled RCCA Withdrawal from a Subcritical Condition

This accident results in uncontrolled addition of reactivity due to control rod cluster withdrawal from a subcritical condition. The neutron flux (and power) response is characterized by a very fast rise that is limited by the prompt negative Doppler reactivity feedback. If the reactivity addition persists the transient will be terminated by one of the following automatic protection features: source level trip, intermediate range rod stop, intermediate range flux level trip, power range flux level trip low setting, or power range flux level trip high setting. This transient is analyzed with TWINKLE for the power rate generation and feedback effects and FACTRAN for the thermal heat flux transient. Finally the THINC code is used for DNBR calculations. Conservative values of all pertinent parameters and setting have been used. The maximum power achieved during this transient is estimated at 35 percent of nominal when it is terminated by reactor trip. The maximum DNBR at all times remains above the limit value. Likewise the average fuel temperature remains lower than the nominal full power value. Therefore, this transient does not violate the DNBR limit and is acceptable.

5.1.2 Uncontrolled RCCA Withdrawal from Power

An uncontrolled RCCA withdrawal from power will create a power mismatch with a coolant temperature rise that can result in DNB if the transient is not terminated. However, any of the following protection features will be activated: nuclear power range trip, overtemperature ΔT trip, overpower ΔT trip, overpressure trip or pressurizer level trip. This transient has been analyzed for several reactivity addition rates starting at several power levels from 10-100 percent of rated power. The analyses were carried out with the LOFTRAN code which simulates the major core parameters during the transient including MDNBR. The results, assuming conservative values of the pertinent procedures showed that the DNBR did not reach the limit value for any power level or reactivity addition rate. Therefore, the results are acceptable. This is the most limiting transient for Ginna.

5.1.3 RCCA Drop

Dropping a full-length RCCA at power would cause a power reduction and an increase in the hot channel factor. The reactor control system will attempt to restore the power level unless some protective trip has occurred. Return to the original power level could cause DNB. A dropped RCCA would be detected by a rod bottom signal or by an excore detector or both. These signals will also activate reduction of the turbine load and will block any further automatic rod withdrawal. For the analysis of this transient the LOFTRAN code is used in simulating the core transient. Turbine run-back and rod-block are also input. Conservative values of the RCCA reactivity worth, the Doppler coefficient and the moderator coefficient have been assumed. The results indicate that either a 100 pcm or a 800 pcm worth RCCA will result in a peak heat flux less than the equivalent of full power. It is concluded that DNB will not occur; therefore, the analysis and the results are acceptable.

5.1.4 Chemical and Volume Control System Malfunction

Reactivity addition can take place with dilution of the boron content in the reactor coolant system water. Such reactivity addition takes place much slower than RCCA withdrawal and numerous alarms and administrative procedures have been instituted to warn the operator of inadvertent dilution. The times required to reach criticality under refueling conditions and during startup have been estimated and found to be 48.8 minutes and 64.1 minutes, respectively. These times are adequate to warn the operator who could intervene to reverse or stop the process. Inadvertent dilution at power is equivalent to a RCCA withdrawal at a very slow rate. In this case the reactor protection system will respond with an overtemperature ΔT alarm and a turbine runback or a reactor protection trip. In summary, the reactor water dilution accident is sufficiently slow that the operator has sufficient time to intervene. Given that the estimated times are of the order of 50-60 minutes, the results of this analysis are acceptable.

5.1.5 Loss of Reactor Coolant Flow/Locked Rotor

Loss of coolant flow can result from loss of electrical power to one or more of the reactor coolant pumps or mechanical damage to these pumps such as a locked rotor. Loss of flow could result in a rapid increase of coolant temperature, DNB or even fuel damage. Therefore, it is necessary that the reactor be shutdown promptly. Such protection is provided by: low voltage on the pump power supply bus, pump circuit breaker opening and low reactor coolant flow. The most severe loss of flow accident can be caused by a simultaneous loss of electrical power to all reactor coolant pumps. The question then is whether the reactor trip and the coolant and rotating parts inertia is sufficient to prevent DNBR from falling below the limit value. This transient is analyzed using the LOFTRAN code to calculate coolant flows, temperatures and pressure and then using FACTRAN to estimate heat fluxes during the transient. Finally, the THINC code is used to calculate the DNBR during the transient. Nominal initial core conditions are assumed with conservative values of the Doppler coefficient and the maximum value of the MTC. The results indicate that DNBR remains above the limit value.

The locked pump rotor constitutes the other loss-of-flow transient. In this case the momentum of the pump rotating parts is not available to the cooling water, and in addition the locked rotor is an impediment in the circulation of that loop. It is, therefore, assumed that the circulation in the affected loop ceases immediately. Following pump seizure and reduced flow the reactor coolant will heat up, expand, increase the pressurizer level, actuate the pressurizer spray and open the pressurizer safety valves. The locked rotor pump transient is analyzed using the LOFTRAN code to calculate the flows and pressure; however, for conservatism the PORV and the pressurizer spray operation are not included in the analysis. The FACTRAN code is then used to estimate core hot spot parameters. The rotor seizure is not combined with other failures. The initial conditions are conservative with respect to the pressure transient. The maximum system pressure is estimated to be 2,836 psia which is less than the 120 percent design pressure and, hence, acceptable. With an F_Q = 3.0 it is assumed that there will be departure from nucleate boiling, and an evaluation of the zirconium-steam reaction indicates that it is less than 1 percent, hence, acceptable. Finally the peak clad surface temperature was estimated at 2176°F, i.e., less than 2200°F and, therefore, acceptable.

5.1.6 Loss of External Electrical Load

The plant has been designed to accept a loss of load up to 50 percent of its nominal power from any power level. The present analysis is for a complete loss of load from full power. The analysis is accomplished with LOFTRAN which simulates core nuclear characteristics, the reactor coolant system, the pressurizer, the pressurizer spray and relief valves and the steam generator and its safety valves. Conservatively it is assumed that there is no direct reactor trip. Maximum and minimum MTC and Doppler coefficient values are assumed in the analysis. The results show that the integrity of the core is maintained by the operation of the protection system and at no time is the minimum DNBR approached. The methods used and the results are acceptable.

5.1.7 Excessive Heat Removal Due to Feedwater Temperature Decrease

Reduction of feedwater temperature results in a primary coolant temperature reduction and reactivity insertion (when MTC < 0) which in turn can increase the power level above full power. In such a case the overpower ΔT and the overtemperature ΔT trips will trip the reactor preventing DNBR values below the limit. The most severe case of this transient can occur by inadvertent opening of the feedwater bypass valve of the low pressure feedwater heaters. This transient is analyzed using the LOFTRAN code which simulates the core and the coolant system behavior. One feedwater heater is assumed to be by-passed with conservative values of the MTC and control system response. The reactor response in this case is similar to the RCCA withdrawal transient which has been discussed previously. The results indicate that a reduction in DNBR is experienced but the value remains above the limit. The analysis and the results are acceptable.

5.1.8 Excessive Load Increase

An excessive load increase is defined as an increase beyond 10 percent of the nominal full power. The 10 percent increase can be accommodated without a reactor trip. The excess load increase transient is similar to the excessive heat removal transient reviewed above. With a steam generator and core power mismatch exceeding 10 percent, the reactor will be tripped by overpower or overtemperature ΔT. The turbine load limiter will keep the turbine load below 100 percent of rated power at all times. This transient was also analyzed using LOFTRAN for all combinations of reactivity feedback and manual or automatic control. The results indicated that for an excessive load increase and power mismatch the DNBR will not fall below the limit value.

5.1.9 Rupture of a Steam Pipe

For the steamline break (SLB) accident the W-3 DNBR correlation was used for the W OFA fuel rather than the WRB-1 correlation. The licensee stated (ref. 2) that this was because the minimum pressure falls below the range of the WRB-1 correlation (1440 P 2490 psia). The minimum pressure also falls below the pressure range given in most references (1000 psia) for the W-3 correlation. However, the licensee justified use of the W-3 correlation for lower pressure based on data presented (ref.18) for Prairie Island that showed no abnormality exists for pressure.
The pressure statepoint for Ginna is slightly below the range of data presented in reference 18. However, the data does not show trends in predicted and measured DNB heat fluxes as a function of pressure and, therefore, reinforces its acceptability. Also the results of the analysis performed by the applicant (ref. 25) show that the minimum DNBR value during the SLB accident is well above the limit of 1.3. On the basis of the data presented and the substantial DNBR margin available, we find the W-3 correlation acceptable for the SLB analysis presented for Ginna.

Rupture of a steam pipe is assumed to include any accident which involves inadvertent steam release from a steam generator. Under no load conditions, a negative temperature coefficient, and the most reactive rod stuck out of the core, the cooldown would result in reduction of the shutdown margin.

Return to power would be a potential problem to the extent that there is a large increase in the hot channel factor when the highest reactivity rod is fully withdrawn. A number of protection systems will be activated in case of steam pipe rupture such as: safety injection, overpower trips, isolation of the feedwater lines and trip of the steam line isolation valves. The transient analysis is accomplished using the LOFTRAN code to compute the reactor and coolant system status and the THINC IV code to compute whether the DNB ratio falls below the minimum value. Analyses were performed using a .018 reactivity shutdown margin, a negative temperature coefficient corresponding to the EOC with all but the most reactive rod inserted, injection capability corresponding to 2 out of 3 safety injection pumps, power peaking factors corresponding to one rod stuck out and three different sizes of the steam line break. The results indicate that following a steamline break the DNBR will remain higher than the limit value. Therefore, the assumed 1.8 percent Δ K reactivity shutdown margin is adequate and the results are acceptable.

5.1.10 Rupture of a Control Rod Mechanism Housing; RCCA Ejection

In the case of a control rod housing rupture the pressure differential would eject the control rod assembly very rapidly. The resultant transient would be limited by the Doppler reactivity feedback and be terminated by the reactor protection trip actuated by a high nuclear power signal. While normally the control rods are withdrawn during operation, on some occasions rods are inserted more than the normal amount. The rod insertion limit is a function of power level and assures, among other things, an adequate shutdown margin. The licensee proposes that less than 200 cal/gm fuel pellet enthalpy, hot spot clad temperature less than 2700°F, pressure within accpetable stress limits and fuel melting less than 10 percent in the hot spot be the criteria for the transient. These are acceptable and indeed are well within the 280 cal/gm requirement of SRP Section 15.4.8. The transient analysis is divided into two parts: hot spot and reactor transient. With the conservative assumption that the hot spot is at the same location before and after the ejection, the FACTRAN code is used for the hot spot analysis followed by TWINKLE which calculates the average core transient.

The THINC IV code is used to calculate the pressure transient, conservatively assuming no leakage through the failed rod housing. Conservative values of the ejected rod worth are estimated as a function of the power level. Similarly, values of the power peaking factors, the delayed neutron fraction and the delay in the initiation of rod insertion for the reactor trip are conservatively chosen. Four cases have been investigated, i.e., BOC and EOC at full and zero power each. The cladding and fuel peak temperatures and fraction of hot spot fuel melt were within the respective criteria. The results of the calculations are acceptable.

5.2 LOCA

Ruptures of the primary coolant piping which are limited to equivalent break areas of 1.0 ft^2 or less (13.54 inches in diameter) are classified as small break LOCAs. Larger breaks are classified as large break LOCAs. Whenever the leak rate is higher than the makeup flow of the charging pumps, depressurization and pressurizer level decrease will result in a reactor trip from pressurizer low pressure or low-low level. The consequences of a LOCA are limited by a reactor trip and the injection of borated water in quantities sufficient to keep the peak clad temperature within acceptable limits.

5.2.1 Small Break LOCA

The analysis of a small break LOCA is accomplished with the WFLASH code which permits a detailed representation of the reactor coolant system. For Ginna, (a two-loop plant) both loops are modeled through conservation equations for mass energy and momentum (Refs. 19 and 20). WFLASH permits a bubble rise model and the calculation of a core mixture height during the transient. The safety injection is explicitly modeled. Peak clad temperature analyses are performed with the LOCTA-IV (Ref. 21) code, with input determined by WFLASH (Ref. 19). The initial power distribution chosen is conservatively skewed toward the top of the core. (The upper part of the core is most likely to get uncovered). The results indicate that the worse small break LOCA (in terms of peak clad temperature and core uncovery) is the 6 inch diameter break. The maximum peak clad temperature (i.e., the peak clad temperature for the 6 inch diameter break) is estimated at 1092°F. Therefore, we conclude that the small break LOCA analysis performed with approved codes and conservative initial conditions, results in peak clad temperatures within the required limits of 10 CFR 50.46 and is acceptable.

5.2.2 Large Break LOCA

In a large break LOCA the pressurizer pressure will decrease rapidly to trip the reactor and initiate safety injection. The consequences of the accident are limited, because (a) of the rapid reduction of power and (b) of the injection of large amounts of borated water to prevent excessive clad temperatures. The conservative assumption is made that the accumulator water injection bypasses the core and exits through the break until the termination of the bypass consistent with the requirements of 10 CFR 50 Appendix K.

The power transient is evaluated with the code SATAN-VI (Ref. 22). The hydraulic and heat transfer aspects of the transient are evaluated with WREFLOOD (Ref. 23), LOCTA-IV (Ref. 21) and the containment pressure analysis with the code COCO (Ref. 24), which are NRC approved models. A 21°F peak clad temperature penalty was added due to the 14x14 OFA fuel.

The results of the analysis indicate that the peak clad temperature is below the 10 CFR 50.46 limit of 2200°F and the amount of fuel element cladding-steam reaction is less than the limit of 1 percent. The peak (localized) cladding oxidation does not exceed the limit of 17 percent. The core integrity is maintained and long term core cooling is maintained as required.

In conclusion, the large break LOCA has been analyzed for Ginna with the OFA Westinghouse fuel using an approved model and the results meet the required limits. The analysis and the results are acceptable.

6. TECHNICAL SPECIFICATIONS

Attachment A of reference 1 provided proposed changes to the Ginna Technical Specifications as follows to account for the use of the Improved Thermal Design Procedure (ITDP) and the introduction of the Westinghouse OFA fuel into the core.

6.1 Definition 1.2 - Reactor Operating Modes

For the refueling mode, the reactivity delta k/k% was changed from -10% to -5%. This is acceptable as it is a conservative value.

6.2 Pages 2.1-2 to 2.1-4

Starting with cycle 14, new W OFA fuel will be introduced into the core which was formerly fueled entirely with ENC fuel. The ENC fuel uses the W-3 DNB correlation. Since the OFA fuel uses the WRB-1 correlation both correlations are now mentioned and the reference to only the W-3 correlation is removed. This is acceptable.

6.3 Figure 2.1-1 - Core DNB Safety Limits

This figure has been modified to account for the ITDP. For the mixed core of ENC and W OFA fuel, the setpoints for accident analysis are based on the W-3 correlation with DNBR limit of 1.30 for the ENC fuel as this is the most limiting. This is acceptable.

6.4 Pages 2.3-2 to 2.3-3 Overtemperature ΔT

To account for the mixed core using ITDP, the constants K_1 to K_5 have been changed and (i), (ii) and (iii) showing the ranges for qt - qb have also been modified with (iii) being removed. There is, however, an error in the equation on page 2.3-3. The staff requires that the value f(ΔI) have a negative rather than the positive value shown. With this modification made the changes are acceptable.

6.5 Page 3.1 - 4b Bases

The reference to the "DNBR value above 1.30" has been removed and replaced to "DNBR above the limit value" to account for the two correlations used in mixed core. This is acceptable.

6.6 Pages 3.1-18 to 20 Minimum Conditions for Criticality and Bases

Ranges of pcm for the MTC were changed due to the introduction of the OFA fuel with an explanation in the Bases. The transient analysis shows this is acceptable.

6.7 Pages 3.8-3 to 4 Change in $\Delta k/k$

For the statement, "The boron concentration of this water at 2000 ppm boron is sufficient to maintain the reactor subcritical by approximately 12% $\Delta k/k$ in the cold condition with all rods inserted, ..." was changed such that the 12% value was reduced to 5%. From the analysis, this changed value is still conservative. Also, a reference was changed to reflect the current Reload Transition Safety Report for Cycle 14. This is acceptable.

6.8 Page 3.10-3 Nuclear Enthalpy Rise Hot Channel Factor-$F_{\Delta H}$

A new single equation covering the zero to full power range for $F_{\Delta H}$ was proposed which replaces two previous equations which each covered separate power ranges. The new equation is in the same form as in the current Westinghouse Standard Technical Specifications. The old and new equations give the same value at full power and nearly identical values down to 75% power. The new equation gives higher values than the old equations at powers below 75%. This is because of a multiplier in the equation which allows a linear increase of 30% at zero power. This same multiplier has been approved for a number of Westinghouse reactors in recent years and is therefore acceptable.

6.9 Page 3.10-4 Target Flux Difference

This section has been changed to eliminate target flux difference ranges for the beginning and end of life by using linear interpolation to determine values of the reference equilibrium indicated axial flux difference in the cycle life from the most recent measured value for each full power month.

6.10 Figure 3.10-2 Coolant Boron Concentration (PPM) Required Shutdown Margin

The proposed figure is very similar to the previous figure and the values are acceptable. It is noted, that parameters are given for one and two loop operation. One loop operation, as specified in Section 3.1 of the Ginna Technical Specifications, is only for operation at less than 8.5% power. This was previously reviewed and approved under the SEP program.

6.11 Figure 3.10-3 Normalized Axial Dependence Factor For Fq vs Elevation

This proposed figure is very close to the previous values and represents changes for the mixed core using ITDP. We find it acceptable.

7.0 SUMMARY

The staff has reviewed the information submitted by RG&E for an amendment of the R. E. Ginna Plant Technical Specifications. The staff finds that, starting with Cycle 14, mixed core operation (W and ENC fuel) and transition to a full core of W OFA fuel is acceptable for the fuel system mechanical design, nuclear design, thermal hydraulic design, the transients and accident analyses, and the Technical Specifications proposed.

However, as stated in Section 6.4, the acceptance requires that the sign of $f(\Delta I)$ in the overtemperature ΔT equation (TS page 2.3-3) be negative.

8.0 ENVIRONMENTAL CONSIDERATION

The staff has determined that the amendment does not authorize a change in effluent types or total amount nor an increase in power level and will not result in any significant environmental impact. Having made this determination, the staff has further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

9.0 CONCLUSION

The staff has further concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; 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.

10.0 ACKNOWLEDGEMENT

H. Balujian and L. Lois prepared this evaluation.

Dated: May 1, 1984

References

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2. Letter from R. W. Kober, RGE, to D. M. Crutchfield, NRC, March 19, 1984.

3. WCAP-9500, "Reference Core Report 17x17 Optimized Fuel Assembly," May 1982.

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5. 83 R.G. G-018, April 28, 1983 (Westinghouse internal document).

6. Chelemer, H., et al., "Improved Thermal Design Procedure," WCAP-8567, July 1975.

7. Chelemer, H., et al., "THINC IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

8. Hochreiter, L. E. et al., "Application of the THINC IV Program to PWR Design," WCAP-8054, September 1973.

9. Motley, F. E. et al., "New Westinghouse Correlation WRB-1- For Predicting Critical Heat Flux In Rod Bundles With Mixing Vane Grids," WCAP-8762, July 1976.

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16. Letter, J. H. Taylor to S. A. Varga, Subject: Status Report on R&D Programs described in Semi Annual Topical Report BAW-10097A; Rev. 2, November 13, 1978.

17. WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation", July 1979.

18. Prairie Island FSAR, Amendment 20, p. 14.2-30, Docket No. 50-282, August 4, 1972.

19. Esposito, V. L. et al., "WFLASH-A FORTRAN-IV, Computer Program for Simulation of Transients in a Multi-Loop PWR" WCAP-8261, Revision 1, July 1974.

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21. Borderlon, F. M., et al., "LOCTA-IV Program: Loss-of-Coolant Transient Analysis," WCAP-8301-P and WCAP-8305-NP, June 1974.

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25. Letter from R. W. Kober, Rochester Gas and Electric Corporation to D. M. Crutchfield, NRC, April 12, 1984 (Enclosure is proprietary)

April 25, 1984

Note to: George Dick

From: Mary Wagner

SUBJECT: GINNA AMENDMENT TO PERMIT USE OF WESTINGHOUSE OPTIMIZED FUEL ASSEMBLY AS RELOAD FUEL

As proposed, the tech spec amendments do not limit the acceptability of the mixed core to Cycle 14 but would appear, from the standpoint of the license, to permit all mixed cores up to all W fuel. That is not what the Staff intends, as evidenced by the language in the transmittal letter to Licensee, which states that approval is on an "interim" basis for fuel Cycle 14 only. Some limitation to that regard should be in the license itself.

If, as you indicated on the telephone yesterday, the Staff is now ready to accept Licensee's analysis without the condition that the changes be limited to Cycle 14, then the SER and transmittal letter should be modified to reflect that.

When you return this to me, I will "walk" the package through our office in order that you don't miss your scheduled deadline for issuance of this amendment.

Mary

Joe: This has now been done, + I have concurred. Mary

OK
[illegible]
4/26/84

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

4/20/84

Mary,

Attached is an amendment for Ginna for a fuel reload. The plant is currently down. I believe that the five days that ELD requires to review amendments should be sufficient to permit me to issue this in time. I will talk to the licensee on Monday (4/23/84) regarding the timing. If I run into a real crunch, I will let you know & get what priorities I can.

I appreciate your help on this.

Thanks
George Dick
X27215