WASH-1236

TECHNICAL REPORT ON
DENSIFICATION OF LIGHT WATER REACTOR FUELS

Date: November 14, 1972

REGULATORY STAFF
U. S. ATOMIC ENERGY COMMISSION

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1.0 Introduction and Summary

The fuel in all current-generation light water nuclear power plants is uranium oxide, UO_2 , in the form of small cylindrical pellets. The uranium is slightly enriched in the fissionable isotope, U-235, and typically contains 1 to 4 percent U-235 with the balance of the uranium being U-238. (Naturally occurring uranium contains 0.7 percent U-235.) In the manufacturing process, uranium oxide powder is compacted into pellet form and is then sintered at high temperatures to form a ceramic-like solid. The fabrication process yields pellets with densities that are less than the maximum possible density of void-free uranium oxide (called the "theoretical" density). Typical pellet densities for light water reactor (LWR) fuels are in the range of 90 to 95 percent of theoretical density. The void volume in the pellets, amounting to 5 to 10 percent of the total pellet volume, is distributed in many small voids, or "pores", in the oxide matrix. The density of the oxide pellets is adjusted by the manufacturer to achieve a value considered most suitable for reactor service, based on irradiation experiments and prior reactor experience. The presence of some porosity in the fuel oxide is desirable to offset the tendency of the pellets to swell during service as fission products accumulate in the oxide matrix.

The cylindrical fuel pellets are stacked end-to-end in thin-walled tubes of Zircaloy, an alloy of zirconium. The Zircaloy tube is called the fuel "cladding". A typical fuel rod is about 13 feet long and contains a 12-foot-long column of stacked oxide pellets. The pellet diameters are precisely ground to provide a carefully controlled radial clearance space

between the pellets and the cladding tube wall. Typical fuel rods for a pressurized water reactor (PWR) are 0.42 to 0.43 inches in outside diameter, have a cladding wall thickness of 0.022 to 0.024 inches, and have an as-manufactured, room temperature diametral clearance space of 0.0060 to 0.0085 inches. Typical fuel rods for a boiling water reactor (BWR) are 0.56 inches outside diameter, have a cladding wall thickness of 0.032 inches, and have an as-manufactured, room temperature diametral clearance space of 0.012 inches. The fuel rods stand vertically in the reactor, with the oxide pellet columns resting on the bottom seal plugs of the rods. The pellet columns are pressed downward by springs in the top ends (plenum spaces) of the cladding tubes, above the pellet columns. The springs are provided to assure that the pellets do not move during handling and shipping operations. Once the fuel rods are placed in the reactor, the springs are not needed to maintain a force on the fuel pellets and, in fact, after a short period of operation, the spring tension is relaxed. Groups of rods, typically 179 to 204 for a PWR and 49 for a BWR, are arranged together in fuel "assemblies".

The space within a fuel rod that is not occupied by fuel pellets is filled with a gas, generally helium. Fuel rods are characterized as "unpressurized" or "pressurized" (sometimes "unprepressurized" or "prepressurized") depending upon the pressure of the gas at room temperature, before the fuel rod is placed in service. Unpressurized fuel rods have a room temperature initial gas pressure that is close to atmospheric pressure. Pressurized rods have initial gas pressures of 200 to 400 psi (pounds per square inch). A number of initial core loadings for current-generation PWRs have been composed

partly or entirely of unpressurized fuel rods. Several years ago, PWR fuel manufacturers began to pressurize fuel rods and all current PWR fuel production is of pressurized rods.

The change to pressurized PWR fuel was in recognition of an increasing body of data on the rate of "creep" (the time-dependent permanent deformation of a material under continued stress) of irradiated Zircaloy at reactor operating temperatures. The coolant water surrounding the fuel rods in a reactor is under substantial pressure; typically 2200 psi in a PWR. The fuel rods are thus subjected to a large external pressure during operation. The benefit of internal pressurization of the fuel rods is to counterbalance the external pressure to some degree, and by reducing the net pressure force on the rod cladding to reduce the inward creep rate of the cladding. Complete balancing of the pressures is not practical since an allowance must be made for the gases evolved from the fission process during operation.

BWR fuel rods are of the unpressurized type. All BWRs operate at substantially lower primary coolant water pressures than PWRs; 1050 psia is a typical value for BWR system pressure, compared to 2200 psia for PWRs. The BWR operating characteristics are also such that the fuel cladding has a somewhat lower temperature than in PWRs. The cladding temperature difference is not great, being of the order of 40°F (at about 600°F wall temperature), but the creep rate of Zircaloy is quite sensitive to temperature in this range and there is a significant reduction in creep rate at the lower BWR cladding temperatures. The lower external pressure also reduces the creep rate, and the combined effects of lower temperature, thicker cladding, and

lower pressure reduce the expected creep rate of BWR fuel cladding enough so that internal pressurization is not needed.

A considerable amount of data and experience with oxide fuel had been amassed prior to April, 1972. The occurrence of an increase in fuel oxide density, called "densification", had been noted as a result of micro-structural changes in the oxide at very high temperatures. In reactor service, the temperatures required for these changes are limited to the hotter central regions of fuel pellets operated at high linear power ratings. (Fuel rod power ratings are given in terms of the power generated per unit length of rod, kW/ft.) The associated changes in overall pellet dimensions from densification due to restructuring are small, since the density changes are localized in the central region of the pellet and are masked by other physical changes in the pellet that occur during high temperature operation. Densification effects also had been observed as a result of the disappearance of small pores from the oxide matrix during irradiation. Again, the overall effect on pellet size was observed to be small because the pores involved were in the one-micron-or-less size class and the total porosity volume in this class is typically small in fuel oxides. Neither of these two previously reported phenomena was believed to cause significant changes in gross fuel pellet dimensions or fuel column length. A general bibliography on these phenomena and other pertinent effects is given in Section 6.0 of this report.

During refueling of the Ginna reactor in April and May, 1972, it was observed that some fuel rods had short flattened or "collapsed" sections. The cladding collapses were found to have resulted from the occurrence of

gaps in the fuel pellet column within the rods. All of the rods with flattened sections were of the unpressurized type. For these rods, the inward creep of the cladding during operation was expected to continue until the cladding was supported by the fuel pellets. In those sections of rods where gaps in the pellet column occurred, the inward cladding creep was not arrested by the pellets, and continued until essentially complete flattening had taken place. The Ginna core contained a number of assemblies with pressurized fuel rods and none of these were found to have collapsed sections. Subsequent examination of neutron flux measurements in other operating reactors (Point Beach Unit 1, Robinson) revealed small flux peaks that could be ascribed to the local power peaks calculated to occur in the immediate vicinity of a gap in the pellet column of a fuel rod. The peaks, or "spikes", in the flux traces were found in both pressurized and unpressurized fuel regions of these reactors. The Staff later learned that similar cladding collapses had been found in unpressurized fuel rods in Beznau Unit 1 in Switzerland at the first refueling of that reactor. Point Beach 1 has recently (October 1972) been shut down for refueling, and preliminary observations of the fuel conditions show that collapsed sections are present in unpressurized fuel rods, but are not found in pressurized rods. The number, location, and type of collapsed sections at Point Beach 1 are all consistent with the observations at Ginna.

An intensive review of the probable causes of gap occurrence in reactor fuel rods and of effects on reactor operation was started by the Regulatory Staff when the Ginna observations were made known. The Staff has received

data from reactor fuel fabricators, including the results of post-irradiation measurements on fuel rods from a number of Westinghouse-designed plants. The results of the Staff review to date are included in this report. Descriptions of the Ginna and Point Beach 1 fuel behavior, of the observations of flux spikes in other reactors, and of various possible mechanisms for fuel column gap formation are given in Section 2. A discussion of fuel densification and associated matters is given in Section 3. One of the important results of the Staff review is the designation of the essential elements of analyses for determining the effects of fuel densification on reactor normal operating modes, anticipated transients, and possible accident sequences. These elements are described in Section 4. The findings and conclusions of the Staff are given in Section 5.

In the course of the Staff review it has become clear, based upon the information available at present from all sources, that densification of oxide fuel pellets and the associated phenomena are matters that must be taken into account in the design and operation of LWRs. It is expected that future developments in fuel pellet manufacture, most likely in the areas of greater initial density, distribution of porosity, and more complete sintering in the fabrication process, will reduce the impact of fuel densification on reactor operation. Until such future developments are fully established by successful irradiation experience, the analyses outlined in this report can be used to account for the effects of densification. As a point of perspective, it is useful to note that as of the shutdown on September 30, 1972, for the first scheduled refueling, Point Beach Unit 1 had operated for about 13,000 effective

full power hours over a 23-month period with no significant problems in spite of the fact that fuel column gaps were present for most of this time, and cladding collapses were probably present over the last third of the operating period. Ginna returned to service after the April 1972 refueling shutdown at a reduced power level and with a number of fuel assemblies with collapsed sections. Operation has been without incident or difficulty from the fuel condition. These experiences suggest, and the Staff review confirms, that fuel densification is one of that class of engineering problems that can be accommodated by appropriate operating limits in the near term, and for which there is good expectation that reasonable design changes can eliminate the problem in the long term.

2.0 Background

2.1 Description of Ginna Fuel Behavior

The Ginna Nuclear Plant was first made critical on November 9, 1969. Although other plants had operated with fuel which was clad with Zircaloy, this was the first licensed pressurized water reactor to commence operation in the United States with a full core of Zircaloy-clad fuel. During early operation of the Ginna Plant with the first core loading (Cycle 1), it was found that the radioactivity in the primary coolant system increased with time. This increase was indicative of fuel cladding leaks. The primary coolant activity gradually increased during the first five months of reactor operation, after which the activity levels remained nearly constant for the remaining eight months of operation. On February 28, 1971, the reactor was shut down for turbine warranty inspection and many fuel assemblies were inspected to locate the leaking fuel. The source of the leaking fuel was determined to be several assemblies in scattered locations in Region 3 of the core.* No leaking fuel assemblies were found in the Region 1 or Region 2 areas. The description of fuel in each region is given in Table 2.1.A. At the time of the February shutdown there was no evidence of cladding collapse which would have indicated the presence of gaps in the fuel columns. Twelve assemblies from Region 3 were replaced with new assemblies. The region of the core comprising the new fuel assemblies was designated Region 4A.

* In nuclear reactors it is common practice to load fuel of different enrichments and/or fuel density into the core. Fuel assemblies having the same design characteristics are called "regions" of the core. A typical reactor core may have three or more regions.

The cause of the leaks in the Region 3 fuel was determined to be local hydriding of the cladding caused by moisture retained in the fuel during fabrication. When the 12 fuel assemblies were replaced, it was recognized that some leaking fuel remained in the core. After the refueling outage, the reactor operated at a power level of 1300 MW(t) until March 1972. Subsequently, during operation of the plant at power levels up to 1520 MW(t), the coolant activity again increased, indicating additional fuel cladding leakage. On April 14, 1972, the Ginna Nuclear Plant was removed from service to maintain and refuel the reactor. Examination of the fuel assemblies revealed leaking of fuel rods, bowed fuel rods, and an appreciable number of rods with sections of collapsed cladding. None of the fuel rods with sections of collapsed cladding were observed to be bowed and the two phenomena do not appear to be related. Information on the fuel behavior in the Ginna reactor was reported in submittals to the Atomic Energy Commission dated June 12, June 20, and June 22, 1972.* These submittals contained information on the observations of the fuel as discharged from Ginna, the operating limits proposed for resumption of operations, and additional surveillance requirements to be performed during operation. The non-proprietary versions of these submittals are attached as Appendices A, B, and C to this report. The reported

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- * 1. Letter from R. Koprowski of Rochester Gas and Electric to D. J. Skovholt of AEC dated June 12, 1972.
2. Letter from R. Koprowski of Rochester Gas and Electric to D. J. Skovholt of AEC dated June 20, 1972.
3. Letter from R. Koprowski of Rochester Gas and Electric to D. J. Skovholt of AEC dated June 22, 1972.

lengths of the collapsed fuel rod sections varied from 0.5 to 4.0 inches and were distributed over the top 40% of the fuel rod lengths. The observations of the fuel discharged from Regions 1, 2, and 3 of the Ginna reactor are summarized in Table 2.1.B.

After the reactor shutdown, some new fuel was loaded into the reactor for resumption of operation (Cycle 2). The characteristics of the fuel in the reactor core for Cycle 2 are tabulated in Table 2.1.C.

The Rochester Gas and Electric Company and the reactor designer, Westinghouse Electric Corporation, developed an evaluation model based on the observations from Ginna to evaluate the potential consequences of the local power peaking that results from fuel column gaps. This model was prepared to provide a basis for proposed resumption of operations of the Ginna Station. Based upon review of this interim model, the Regulatory Staff concluded that the Ginna Station could be returned to service at 83% of full power subject to additional limitations on peak linear power (kW/ft), on permitted rates of power changes, and the performance of additional surveillance during reactor operations.

Additional discussions of the interim evaluation of the fuel problems for Ginna, Point Beach Unit 1, Robinson Unit 2, and Palisades are contained in a staff report dated November 3, 1972, and attached as Appendix D to this report. This evaluation included consideration of the changes in analyses for the loss-of-coolant accident, limits for departure from nuclear boiling, and overpower limits to preclude centerline fuel melting.

On September 29, 1972, the Rochester Gas and Electric Corporation

informed the Regulatory Staff that the Ginna Nuclear Plant would be shut down in October, 1972, to remove all 48 of the remaining fuel assemblies from the initial core loading and replace them with prepressurized fuel assemblies. Preliminary results from observations at the October refueling are available at this writing for only the 12 prepressurized assemblies of Region 4A. No indications of collapsed sections were found in these assemblies, which had reached an exposure of about 10,000 effective full power hours at the time of shutdown.

Table 2.1.A

GINNA FUEL CHARACTERISTICS, CYCLES 1A AND 1B

	Region			
	1	2	3	4A
Number of Assemblies				
Cycle 1A (Initial fuel loading)	41	40	40	--
Cycle 1B (Loading after Feb, 1972 shutdown)	41	40	28	12
Rods per Assembly	179	179	179	179
Enrichment, % of U-235	2.44	2.78	3.48	3.20
Clad OD, In	0.422	0.422	0.422	0.422
Diametral Fuel-to-Clad Gap Thickness, mils	6.5	6.5	6.5	6.5
Pellet Density, % Theoretical	94	92	90	92
Helium Prepressurized	No	No	No	Yes

Table 2.1.B

SUMMARY OF OBSERVATIONS OF GINNA FUEL AT APRIL, 1972 SHUTDOWN

	Region			
	1	2	3	4A
Number of Assemblies Examined	20	22	11	1
Number of Fuel Rods Examined	1040	1144	572	52
Number of Rods with Collapsed Sections	21	83	20	0
Percent of Rods with Collapsed Sections	2.0	7.3	3.5	0

Table 2.1.C

GINNA FUEL CHARACTERISTICS CYCLE 1B AND CYCLE 2

	1	2	3	Region 4A	4	4B
Number of Assemblies						
Cycle 1B (Feb '72-Apr '72)	41	40	28	12	0	0
Cycle 2 (After April '72)	38	10	0	12	40	21
Rods per Assembly	179	179	179	179	179	179
Enrichment	2.44	2.78	3.48	3.20	3.14	2.90
Pellet Density	94	92	90	92	94	94
Helium Prepressurized	No	No	No	Yes	Yes	Yes

2.2 Description of Point Beach Unit 1 Fuel Behavior

The Point Beach Unit 1 reactor core contains the same number of fuel assemblies and fuel rods as the Ginna reactor. The characteristics of the fuel are summarized in Table 2.2. On September 30, 1972, the Point Beach Unit 1 was shut down for refueling. Initial observations indicated that fuel discharged from Region 1 of the core had collapsed sections while none of the pressurized fuel rods examined exhibited any abnormal indications. As of this writing, 39 of the 41 assemblies in Region 1 have been examined by observation with binoculars of the 52 outer fuel rods in each assembly. A total of 66 fuel rods, distributed among 20 assemblies were found to have collapsed sections. No rod had more than one collapsed section, and the collapsed section lengths are all judged to be less than the maximum 4-inch length found at Ginna. The fraction of rods with collapsed sections, among those examined, is $66/2028$, or 3.2%, closely matching the Ginna results. Of the Region 2 and 3 fuel, 39 assemblies from Region 2 and 36 from Region 3 have been examined. No collapsed sections were found. The fuel exposure for the core was about 13,000 effective full power hours at shutdown. Calculations based on the Westinghouse cladding creep and collapse model had predicted that cladding collapses would occur at about 12,000 effective full power hours for Region 2 and 3 fuel at locations where fuel column gaps had formed. The presence of fuel column gaps had been indicated by neutron flux spikes in flux traverses made before shutdown.

Preliminary results are available at this writing on a survey to find leaking fuel rods. A total of 20 assemblies of the Region 1 fuel have been

found to have leaking fuel rods, out of 30 assemblies surveyed. Two additional assemblies of the 30 may have leaks, but further measurements will be necessary to confirm the preliminary results. In Regions 2 and 3, 39 and 36 assemblies, respectively, have been surveyed. One leaking assembly was found in Region 2 fuel and 2 assemblies are suspected of having leaks in the Region 3 fuel.

The licensee has proposed a program for post-irradiation examination of fuel discharged from the Point Beach Unit 1. The program includes gamma scanning of the fuel rods to characterize more fully the location and size of axial fuel gaps occurring in the fuel. This information will be used to refine the analytical models used to predict fuel performance.

Table 2.2

POINT BEACH UNIT 1 FUEL CHARACTERISTICS

	Region		
	1	2	3
Number of Assemblies	41	40	40
Rods per Assembly	179	179	179
Enrichment	2.27	3.03	3.40
Pellet Density, % Theoretical	94	92	91
Helium Prepressurized	No	Yes	Yes

2.3 Observations of Flux Anomalies

In addition to the cladding collapses observed at Ginna and Point Beach Unit 1, in-core measurements of neutron flux distributions at these two reactors and at the H. B. Robinson Unit 2 reactor have indicated the presence of local anomalies in the neutron flux distribution (power spikes). The absence of fuel in a local region causes a decrease in the absorption of thermal neutrons and a consequent increase in local flux and power. The in-core neutron measurements can locate the axial position of such power spikes and give some indication of the magnitude of the local power perturbation. However, the magnitude of the observed perturbation cannot be related uniquely to the magnitude of the actual power spike in the core since there is no way of determining the radial distance from the instrument thimble to the fuel rod which contains the gap.

A summary of results of the in-core flux traverses from Point Beach and Robinson is contained in Figure 3-3 of Appendix E to this report.

The Staff has examined actual flux traverse chart recordings to evaluate the data on the observed frequency of power spikes that have been presented. Although some very small indications in the recorded traverses that might not be actual power spikes were included in the power spike count, it is not always evident that others of comparable magnitude were included. The Staff estimates that including all such small indications in the power spike count might increase the total count by about 75 percent.

Westinghouse has presented calculations to demonstrate that their power spike count data indicate that on the average there is less than one gap per

fuel rod. These calculations assumed that only the eight rods immediately surrounding the flux detector contributed to the observed power spike. Since rods further away, e.g., the 16 rods in the next rows, could also contribute to the power spike indications on the traverses (especially to the very small indications), the Staff concluded that a 75 percent increase in the observed power spike count would not increase the calculated number of gaps per rod above one, when the effect of the additional, more distant rods is taken into account.

2.4 Observations at Other Reactors

After the April 1972 shutdown of the Ginna Plant, the Staff learned that the Westinghouse-designed and fueled Beznau Unit No. 1 pressurized water reactor in Switzerland had experienced cladding collapses similar to those found at Ginna. Beznau I has a power rating about three quarters that of Ginna, and a proportionately smaller core. The initial fuel loading at Beznau 1 was of unpressurized, Zircaloy-clad fuel, with three enrichment regions. At the first refueling and maintenance outage of Beznau 1, in the summer of 1971, fuel rods in Region 2 of the core were found to have sections of collapsed cladding. About 2 percent of the fuel rods examined from Region 2 of the core had collapsed cladding. A portion of the data on irradiated fuel received by the Staff derives from the examinations carried out on Beznau 1 fuel.

Information on the fuel condition in Zorita Unit 1 reactor was also received from Westinghouse, the designer and fuel supplier for the unit. Zorita 1 is a pressurized water reactor of about one-third the power rating

of Ginna. The initial core loading at Zorita 1 was of Zircaloy-clad, unpressurized fuel rods. Examination of Zorita 1 fuel at the first and second refueling shutdowns revealed no signs of collapsed cladding sections. Zorita 1 is now in operation in the third fuel cycle.

The Staff understands that an examination of fuel conditions was made at the Obrigheim pressurized water reactor in West Germany at a shutdown after the Beznau 1 shutdown. No cladding collapses were found. Fuel for the Obrigheim reactor was fabricated by Siemens.

Older pressurized water reactors in the United States, Indian Point Unit 1, Yankee Rowe, San Onofre Unit 1, and Connecticut Yankee were all fueled initially with stainless steel-clad unpressurized fuel rods. A substantial amount of operating experience has been obtained with stainless steel-clad fuel in these reactors and no instances of cladding collapse have been reported. Fuel pellet densification has occurred and post-irradiation measurements of pellet density have been made. The results are consistent with other data on densification. Some fuel column gaps may have formed, since small flux spikes have been observed at Connecticut Yankee (the only reactor of the 4 to have an in-core flux measurement system capable of detecting the spikes). Some of the older reactors are converting to Zircaloy-clad fuel. Yankee Rowe has just completed a fuel cycle with a core loading in which about one-half of the fuel is Zircaloy-clad. No results of the fuel condition at Yankee Rowe are available at this writing. Several assemblies containing Zircaloy-clad fuel rods have been used in Connecticut Yankee to qualify supply sources for future core loadings. Four such fuel assemblies, two fabricated by Numec,

and two by Gulf United Nuclear Fuels, have operated for two and one fuel cycles, respectively, without cladding collapses. Most of the fuel rods in these assemblies are of the unpressurized type.

Other current-generation pressurized water reactors in the United States now in operation include the Combustion Engineering-designed and fueled Palisades and Maine Yankee reactors and the Westinghouse-designed and fueled Point Beach 2, Surry 1, and Turkey Point 3 reactors. Maine Yankee, Surry 1, and Turkey Point 3 are in the early stages of startup and have not accumulated any significant fuel exposure at this writing. Point Beach 2 has operated at 20 percent of full power for several months, but has not as yet shown any evidence on the in-core flux traces of flux spikes that might indicate fuel column gap formation. Palisades has operated at 60 percent of full power for a long enough time so that fuel column gaps, but not cladding collapses, might have formed. No indications of such gaps have been reported, but the evidence is inconclusive since the in-core monitoring system at Palisades, being of the fixed position detector type, is not capable of unambiguously detecting the flux spikes from short fuel column gaps.

A number of boiling water reactors, designed and fueled by General Electric, are in operation in this country and abroad. Post-irradiation examination of BWR fuel has been a standard practice by General Electric, so that a large body of data on BWR fuel performance is available. No instances of cladding collapse have been found. Densification of fuel pellets has been observed, of an amount generally consistent with the Westinghouse densification data. The effect is relatively small in BWR fuel, probably due to the

high initial density of the pellets. Some instances of gaps between fuel pellets have been found, but these have all been too small to be of significance with regard to cladding behavior or power spiking.

2.5 Causes of Fuel Column Gaps

The first phase of the Regulatory Staff's review of the Ginna fuel observations was an investigation of the possible causes of formation of axial gaps in fuel pellet columns. The Staff examined (1) the fuel fabrication processes of Westinghouse (the supplier of fuel for the Ginna, Point Beach, and Robinson reactors) and the associated quality control procedures, (2) the possible contributions of cladding length changes arising from irradiation growth of the Zircaloy cladding, and (3) the potential for fuel-cladding mechanical interactions that could cause "hangup" of pellets and prevent settling of the pellet column as a unit. A summary of these investigations follows.

The Staff has considered the significant factors involved in fuel fabrication to determine whether the observed fuel column gaps could have been the result of the manufacturing and assembly techniques used by Westinghouse. The Staff concluded that the Westinghouse rod-loading procedures provided reasonable assurance that the pellet stack is continuous, that the correct stack length has been inserted into the rod, and that the plenum chamber length is within design tolerance. In manufacturing, these process controls were checked by rod weighing and supplemented by full-rod x-rays for any rod falling below the established limits. The absence of gaps at the completion of fabrication is further supported by data obtained by x-raying

returned fuel rods from a reactor site. The x-raying technique also demonstrated that fuel handling and shipping following fabrication did not alter the continuity of the fuel column. In-core instrumentation data from Point Beach Unit 1 and Robinson indicated that gaps were not initially present in the core after loading, but developed during irradiation. In addition, x-ray and gamma scan examinations of well-characterized, individual, special test fuel rods irradiated in the Saxton reactor showed no pre-irradiation fuel column fabrication gaps and no post-irradiation increases of cladding length. These fuel rods did display fuel column length decreases, the approximate magnitudes of which were consistent with the volume densification increases measured by immersion density methods.

To assess the possible contribution of an increase in cladding length relative to a hypothetical unchanged fuel pellet column length, caused by fuel-cladding mechanical interaction or irradiation growth of the Zircaloy, or both, the Staff examined data on the total fuel cladding length changes of irradiated fuel rods. These data showed the fractional mean rod growth to be much less than the fuel column fractional shrinkage that would be required to explain the observed amount of axial gap formed. In comparisons made between adjacent collapsed and uncollapsed rods, a mean bias of the order of a few mils was noted, in which the collapsed rod length was shorter than the uncollapsed.

The current Westinghouse design model for irradiation growth of Zircaloy is based on approximately 2000 data points in which the total changes in rod lengths of power reactor fuel rods were measured, and includes data obtained

from non-fueled Zircaloy tubing, such as water tubes and dummy rods with alumina pellets. The Staff finds that corrections imposed by Westinghouse in their power spike model for the Zircaloy growth contribution to gap sizes are consistent with the data and are inconsistent with the hypothesis that fuel rod cladding length changes could account for a significant amount of the length of axial gaps.

The observed occurrence of axial gaps in fuel pellet columns could also arise from the densification of the fuel pellets and resulting axial shrinkage of the pellet column, if mechanical interaction between the cladding and the pellets were to cause the hangup of a fuel pellet in the cladding tube, thus preventing the settling of that pellet and those above it onto the top of the reduced-length pellet column below. The formation of gaps by such a mechanism depends primarily on whether or not the shrinkage and settling of the fuel column is completed before cladding creepdown prevents pellet movement. The time required for clad creepdown to produce contact between fuel pellets and cladding with sufficient force to cause fuel pellet hangup and prevent fuel column settling depends, in turn, on a number of parameters. As discussed in Section 3.2.1, these include cladding thickness and diameter, initial ovality, system pressure, internal fuel rod pressure, fast neutron flux, temperature, and initial cold fuel-cladding radial clearance. In examining the location of collapsed cladding areas there was one instance in 124 of adjacent rods having collapsed at the same axial elevation. At the present time it does not appear that there is any significant tendency for the formation of adjacent coplanar fuel column gaps. Pellet settling may be

prevented by other mechanisms such as frictional holdup of pellets, or pellets that become cocked in the cladding or are trapped by pellet chips, but the Staff believes these mechanisms to be more random in occurrence and of lesser importance in promoting fuel column gap formation than cladding-pellet contact by creepdown.

For fuel pellets contained in a free-standing cladding with a large radial clearance gap (as is the case in BWRs), it is much less likely that pellet movement would be prevented by cladding creepdown, friction, or other mechanisms. Pellet densification should then result in an overall shrinkage of the fuel column as a unit. This has been observed in many cases and the observed shrinkage is consistent with the estimated densification. These conditions are thought to be typical of current BWR fuel designs for the irradiation times of interest in the formation of fuel column gaps.

As a result of the investigations summarized above, the Staff has concluded that the observed occurrence of axial gaps in fuel pellet columns arises from the densification of the fuel, the resulting axial shrinkage of the pellet column, and the hangup of a pellet which prevents settling of the pellet column. The Staff arrived at this conclusion after examining fuel rod fabrication process and associated quality control procedures, the possible contribution of cladding length changes arising from irradiation growth of zirconium, and fuel-cladding mechanical interactions. The conclusion is also based on a comparison of pre-and post-irradiation radiographs of entire fuel rods.

In retrospect, it seems likely that more extensive fuel testing for current-generation light water plants, coupled with an adequate and continuing

post-irradiation examination of fuel rods from older reactors, would have revealed the nature of the fuel densification phenomenon and its possible effects.

3.0 Discussion of Fuel Densification and Associated Phenomena

3.1 Densification

3.1.1 Densification Mechanism

The observation of fuel column gaps in operating reactor fuel rods, and the suggestion that the gaps result from densification of the fuel pellets during irradiation, raise the question of why such densification had not been observed previously in irradiation testing or operating experience. In addressing this question, it is important to note that most of the previous experience had been obtained with fuel rods much shorter than those in current-generation reactor cores (i.e., three-foot-long or less compared with twelve-foot-long rods). Also, the earlier fuel pellets had a higher initial density than the pellets for fuel rods in which gaps have been observed (i.e., initial densities of about 95% of theoretical density compared to initial densities of as low as 90% of theoretical in some recent fuels).

These two differences in fuel rod design have a significant effect on the ability to observe changes in fuel column length as a result of irradiation. For example, for a three-foot-long test fuel rod with an initial pellet density of 95% of theoretical, if the pellet density increases under irradiation to 97%, and if the consequent volume decrease results in an isotropic reduction in pellet dimensions, then the 2% volume decrease will produce a change of less than 1% in the fuel column length, i.e., a change in length of the column of less than 0.4 inch. On the other hand, for a twelve-foot-long fuel rod with an initial pellet density of about 90%, if the pellet density increases to 97%, the 7% volume decrease could result in an isotropic

dimensional change of 2.3%, corresponding to a change in the 12-foot column length of more than 3 inches. Thus, it is entirely possible that the densification mechanism responsible for the recently observed behavior of reactor fuels could have been present in previous tests and have escaped detection.

The densification thought to be the cause of the observed fuel column gaps is a different phenomenon from the thermally-induced densification of fuel which occurs during fuel irradiations at high temperature. The structural changes that are normally assumed to occur in sintered UO_2 during irradiation are largely the result of the high temperatures and steep thermal gradients that are present in uranium oxide fuel pellets during operation.* Fuel rods operated at relatively low power levels retain their pre-irradiation structure except for radial cracks formed by thermal stresses. If uranium oxide fuel operates at relatively higher power levels, and consequently at higher temperatures, equi-axed grain growth occurs. If the linear power is high enough, columnar grains frequently form within the ring of equi-axed grains. These columnar grains are a result not only of the temperature but of the thermal gradient. They are formed when pores initially present in the fuel migrate up the thermal gradient by sublimation of the oxide, digesting the poly-crystalline matrix ahead of the pore, and developing a denser single crystal on the cooler side of the void.

The effects of these thermally-derived structural changes can be interpreted in terms of the concomitant porosity changes which occur due to thermal

* See, for example, the discussion in J. A. L. Robertson, "Irradiation Effects In Nuclear Fuels," ANS-USAEC Monograph, Gordon and Beach, Inc.

gradient and temperature effects. The restructuring of the fuel in the columnar grain growth region results in densification ranging up to approximately 97 or 98% of theoretical density. However, this is usually accompanied by formation of a central void due to the migration of the as-fabricated porosity in the columnar grain growth region. Restructuring during irradiation results in growth of equi-axed grains in the fuel, but little, if any, net change of porosity is observed in regions of equi-axed grain growth. In the equi-axed region the porosity is enlarged, locally redistributed to grain boundaries, and tends to be spheroidized.

Thermally-induced densification has also been observed to occur as a result of in-reactor sintering, but this process requires temperatures of about 1300 to 1500°C, and a structurally metastable fuel. A structurally metastable fuel is one composed of pellets in which the sintering process was interrupted before it had gone to completion, or was carried out at too low a temperature for the times involved to complete the sintering process.

The structural changes observed in the Ginna fuel occurred in a relatively low temperature region, 400-1000°C, in the fuel. In the Ginna core (which is typical of most light water reactor cores) almost all of the fuel operates in the low temperature range since the average linear heat generation rate in the hottest fuel rod for the plant is approximately 8.5 kW/ft, with a core-wide rod-average of about 5 kW/ft. Thus, most of the fuel operated at temperatures at which little or no thermal restructuring occurred and at which thermally-activated processes were so slow as to be insignificant. However, evidence from a number of power reactors showed that fuel

pellet densification occurred in fuel operated at linear powers ranging from 1.4 to 17 kW/ft, in both pressurized and unpressurized fuel. Examinations of metallographic cross-sections of irradiated fuel and immersion density measurements of irradiated fuel operated at these low temperatures confirm that individual pellets have densified, and that this process occurs by annihilation of pores.

References in the literature (see Bibliography, Section 6.0) describe the results of experiments in which annihilation of small pores in UO_2 fuel has occurred as a result of low temperature neutron irradiation for relatively short periods of time. These include both fission gas pores and the closed porosity initially present in the as-sintered material. However, these changes generally were believed to occur with pores or bubbles on the order of one micron or less in diameter. More recent information shows that more and larger pores are removed by irradiation, and suggests that pore morphology, (i.e., size, distribution, and shape) of sintered UO_2 fuel pellets could have a profound effect on irradiation induced densification.

The exact mechanism by which the densification occurs is as yet unclear. Two mechanisms have been suggested as the cause of pore annihilation:

- (1) Re-solution-related mechanisms: In this case, fission fragments passing close to the surface of a pore aid in trapping lattice vacancies in the surface. These migrate away from the pore, thus causing mass transfer of atoms to the pore and resulting in the eventual disappearance of the pore.

- (2) Fission spike-related mechanisms: In this case, interstitial atoms and lattice vacancies are generated by the fission process. The interstitial atoms, having greater mobilities at the temperatures of interest, quickly migrate to sinks, leaving an excess of vacancies in the lattice which increase the diffusion rates of the atoms in the fuel.

Both of these mechanisms can lead to enhanced diffusion of uranium within the UO_2 lattice which could result in (1) sintering driven by pore surface energy at temperatures at which it would ordinarily not be expected, or (2) creep of UO_2 at temperatures below 1000°C at a rate dependent on the fission rate, but relatively insensitive to temperature. The creep could be the result of stresses imposed by the cladding or by the weight of the fuel column. Both of these processes could result in pore annihilation. The Staff favors the theory of sintering driven by pore surface energy, wherein the rate of the process is enhanced by the generation of excess vacancies during fission events.

3.1.2 Kinetics of Densification

Examinations of density changes in irradiated fuel by Westinghouse show that for exposure times of less than 14 hours at full power operation, no temperature-independent densification has occurred, but that after 2000 hours of full power reactor operation, fuel densification probably has been completed. This timing is also supported by observations of in-core neutron flux distribution measurements in which indications of local power spikes appear in the first few hundred hours. If it is assumed that pore annihilation

occurs by enhanced sintering or by a re-solution process, as discussed in Section 3.1.1, the kinetics of densification can be considered to determine qualitatively if the observed densification times are compatible with theoretical analysis. For example, as discussed in Section 3.1.1, the in-reactor kinetics of diffusion-controlled processes in oxide fuels (e.g., creep) are enhanced and become observable at temperatures well below those at which one would expect to observe them out-of-pile. Uranium moves by a volume diffusion mechanism (although grain boundary diffusion of vacancies undoubtedly is also important). The diffusion mechanism controls the rate of sintering or creep of oxide fuels, and the diffusion rate is sensitive to the availability of adjacent vacancies.

If, as is likely, the processes that operate in densification involve vacancy diffusion, then a source and a sink for vacancies are required. The source is provided by the fission spikes (as discussed in Section 3.1.1) which provide a supersaturation of vacancies at a given temperature. A sink for the vacancies is provided by grain boundaries, free surfaces, and dislocations. Only in this way can a net flux of atoms into the pore be achieved to obtain the observed densification. Values have been obtained* for in-pile uranium diffusion from experiments on low density UO_2 compacts irradiated at temperatures of less than 400°C . The Staff has estimated the in-reactor diffusion coefficient for uranium to be $2.5 \times 10^{-17} \text{ cm}^2/\text{sec}$ at

* M. O. Marlow, "Fission Sintering and Irradiation Induced Diffusion in UO_2 " American Ceramic Society Bulletin, 51 (4) (1972) 385

a fission rate of 1.8×10^{12} fissions/cm³ sec. The fission rates of interest for power reactors range from 3 to 5×10^{13} fissions/cm³-sec. Staff estimates of the enhanced diffusion rates for these fission rates suggest that an enhanced, in-pile diffusivity for uranium is obtained which is roughly equivalent to an out-of-pile, thermally-activated uranium diffusion coefficient commensurate with a temperature of 1400°C, a temperature only slightly below those used in many UO₂ sintering processes. Thus, within the uncertainties associated with the in-pile diffusion values and fission rates, it appears reasonable that most small-sized porosity could be removed by a diffusion-controlled process in the times that have been observed for densification.

3.1.3 Effects of Powder, Pellet, and Fabrication History

The as-fabricated properties of UO₂ pellets in reactor fuel assemblies are dependent on the many variables which exist in the complete conversion cycle for the UO₂ raw material from

- (a) the formation of the intrinsic powder particles, to
- (b) the preparation of the press feed material, to
- (c) the compaction of the green (unfired) pellets, to
- (d) the sintering to achieve the design density of the pellets.

The effects of the many variables in these steps of the conversion cycle may be additive or offsetting and many fuel manufacturers choose to control pellet properties by performance of product tests, and specify that the major influencing characteristics be within tolerance bands. In reviewing these effects, the Staff has assessed the characterization data for the UO₂ raw material

powder, compared the current and previous UO_2 pellet manufacturing processes and product specifications, compared the sintering atmospheres for the production of UO_2 fuel pellets from the various sources, and estimated the effects of pellet microstructural and physical properties on densification.

In the Westinghouse manufacturing process, the specifications for the powder are checked primarily through performance tests. In 1967, the Westinghouse specifications required densities consistently in the range of 93 to 95% of theoretical density. They specified that the powder, when slugged at 4.5 ± 0.1 gram/cc and pressed at 5.5 ± 0.05 gram/cc with 0.3 percent Sterotex addition, and sintered at 1600°C with a range of $+25$ to -100°C for a maximum of three hours, should have a minimum density of 93%. In 1968, as a result of engineering studies, densities ranging from 90 to 95% were specified for the various regions of the reactor core. The powder specification was modified accordingly. Four types of powder were specified. These powder types were required to meet the following requirements subsequent to identical preparation, as earlier described: Type I, 90% TD; Type II, 92%; Type III, 93%; Type IV, 95%. An additional requirement was that the slope of the sintered density versus sintering temperature curve should fall within a given range as determined by sintering at 100 to 150°C above and below the nominal 1600° sintering temperature. Prior to powder production at the Westinghouse Columbia Plant, UO_2 powder for power reactor assemblies was obtained primarily from Numec and Nuclear Fuel Services. After the Columbia Plant started producing powder, it became apparent that the characteristics of this powder differed from previously used powder, and minor modifications

were made to the processing parameters. A comparison of current and previous UO_2 pellet manufacturing processes shows that until late 1967 the wet process of pellet manufacture was used. Subsequent to this time, Westinghouse pellet manufacture has been exclusively by the dry process (that is, no binder addition). The only process changes that have taken place during this period are that a rotary press is now used instead of a double-acting single-station press and that bottled, rather than bulk, hydrogen is used as the sintering atmosphere.

An examination of the sintering times and temperatures for the production of UO_2 pellets produced from powders obtained from various sources was made. In analyzing the process data, the Staff considered how the manufacturing techniques could affect densification. Westinghouse has presented data regarding pellet attributes considered important in controlling the amount and rate of fuel densification. The Staff has examined the proprietary data that forms a basis for the Westinghouse conclusions and agrees that the conclusions seem to be consistent with the evidence. However, it is not clear that the pellet attributes alleged to control densification are not also subtly dependent upon some of the other factors of the UO_2 conversion cycle. This is important because certain aspects of the Westinghouse pellet manufacturing processes are similar to those used by other fuel vendors. Even if all fuels were susceptible to densification (and one could conclude this from the Westinghouse analysis of the effects of pellet attributes and their interpretation of the pore annihilation mechanism), all fuels would not necessarily undergo the same initial-to-final density changes. However, until more definite irradiation

experience is obtained from irradiation tests, the Staff believes that differences in pellet manufacturing processes do not provide a sufficient basis for differentiating between fuels of different vendors for purposes of safety analyses, and that it is prudent to assume for the present that all light water reactor (LWR) oxide fuels will densify to an extent consistent with present observations.

3.1.4. Effects of Design and Environmental Factors

Some of the parameters Westinghouse has examined relative to densification are initial density, peak power, burnup, fission rate, and internal gas pressure. The effects of these parameters and their interrelationships are not completely established. However, some preliminary conclusions can be drawn as to their role in densification.

The trend of data regarding the amount of observed fuel column shrinkage as a function of initial fuel density is shown in Figure 3.1.4 a. There is a clear tendency for increased fuel column shrinkage with decreased initial density.

The trend of data obtained by mercury pycnometry measurements for the effect of fuel rod linear power on fuel density, for irradiation times from 1500 to 30,000 hours, is presented in Figure 3.1.4 b. It should be noted that densification has occurred for linear power levels which range from 1.5 to 17.0 kW/ft. The lack of a clear trend to the data may be due to the large differences in burnup of the various samples and the consequent effect of fuel swelling. As noted in Section 3.1.2, the densification process is nearly complete in 2000 hours, whereas some of the specimens in Figure 3.1.4 b range up to 30,000 hours exposure. This is further borne out by Figure 3.1.4 c

which is a plot of initial and final fuel density as a function of burnup. Again, there is no apparent trend. There are no clear relationships of densification to power or burnup from which to determine a fission rate dependence of densification from the limited data. Since there are no data within the approximately 2000 hours of full power operation in which full densification occurs, one can only assume from the proposed mechanisms (See Section 3.1.1) that the dependence exists and that the amount of densification is dependent on both burnup and fission rate during the first few thousand hours of full power operation.

Figure 3.1.4 d illustrates the relationship between fuel densification and internal pressure. No apparent influence of an isostatic hot pressing effect is observed.

Metallographic examination of the fuel indicates that shrinkage is greater in the outer portions of the pellet than toward the center. This is shown in Figure 3.1.4 e, a plot of cumulative porosity versus pellet radius for a fuel which, based on exposure time, is presumed to have undergone complete fuel shrinkage. Since the data were taken from high power pellets, the indicated porosity increase at the pellet center is probably related to fuel swelling.

The trend of data for fuel pellet length change as a function of pellet volume change, for irradiation times of 3000 to 10,000 hours is presented in Figure 3.1.4 f. If the process were isotropic, the fractional change in radius and length would both be equal to $1/3$ of the fractional change in volume, but the data show a bias toward greater densification taking place in

the axial direction than would be expected from an assumption of isotropy. This may be an effect of the greater shrinkage of the outer pellet periphery noted above and the fact that the ends of the pellets were dished, which amplifies the length change. The Staff's interpretation of the data trend leads to the conclusion that it is appropriate to assume that axial shrinkage is greater than radial shrinkage.

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FIG. 3.1.4 a

Trend of data on fuel pellet stack length change during irradiation as a function of initial fuel density. From proprietary data of the Westinghouse Corp., with permission.

Trend of data on change in fuel pellet density as a function of peak linear power density, for irradiation times from 1500 to 30,000 hours. From proprietary data of the Westinghouse Corp., with permission.

Trend of data on change in fuel pellet density as a function of burnup. From proprietary data of the Westinghouse Corp., with permission.

FIG. 3.1.4 D

Trend of data on porosity distribution in fuel pellets of initial density of about 90% of theoretical density, after irradiation for 4000 hours. From proprietary data of the Westinghouse Corp., with permission.

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FIG. 3.1.4 e

Data on fuel pellet volume change as a function of initial fuel rod gas pressure for initial fuel densities of 90 to 95% of theoretical density and irradiation times of 3000 to 30,000 hours. From proprietary data of the Westinghouse Corp., with permission.

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Fig. 3.1.4 f

Trend of fuel pellet volume change as a function of pellet length change for irradiation times of 3000 to 10,000 hours.
From proprietary data of the Westinghouse Corp., with permission.

3.1.5 Densification Effects:

Densification of fuel causes a decrease in the volume of the fuel pellet with corresponding changes in the pellet radius and length. There are three principal effects associated with fuel densification:

(a) A decrease in the pellet length will cause the linear heat generation rate to increase by an amount in direct proportion to the percentage decrease in pellet length.

(b) A decrease in the pellet length can lead to generation of axial gaps within the fuel column, resulting in increased local neutron flux and the generation of a local power spike.

(c) A decrease in the pellet radius increases the radial clearance gap between the fuel pellet and fuel rod cladding causing a decrease in the gap thermal conductance, and consequently in the capability to transfer heat across the radial gap. This decrease in heat transfer capability will cause the stored energy in the fuel pellet to increase. A decrease in radial gap conductance also will degrade the heat transfer capability of the fuel rod during various transient and accident conditions.

In summary, the effects of fuel densification cause the fuel rod to contain more stored energy, increase the linear heat generation rate of the pellet, decrease the heat transfer capability of the fuel rod and create the potential for a local power spike in any fuel rod. To assess the safety implications of fuel densification, all of these effects must be evaluated for each reactor under all modes of reactor operation.

3.2 Associated Phenomena

3.2.1 Cladding Creep and Collapse

In normal reactor operation, fuel rod cladding creeps inward when subjected to the hydrostatic compressive stresses of the pressurized water coolant. As a consequence, the cladding ultimately will contact the fuel pellets or will continue to creep inward in regions where axial gaps have been generated. The time required to reach cladding-pellet contact or cladding collapse depends on many factors, some of which are

- (1) cladding diameter,
- (2) cladding wall thickness,
- (3) initial cladding ovality,
- (4) fabrication history,
- (5) cladding temperature,
- (6) reactor coolant pressure,
- (7) initial fuel-rod internal gas pressure,
- (8) additional fuel-rod internal gas pressure from released fission gases and temperature changes, and
- (9) neutron flux level

The driving forces promoting cladding collapse are stresses caused by the pressure differential across the cladding and the bending stresses associated with the out-of-roundness of the cladding tube. At these stress levels, deformation of the cladding takes place primarily by creep. Stress and creep are interrelated because creep enhances the ovality of the tube, which raises the stress level, which in turn enhances the creep rate. The

creep rate of Zircaloy is a function of stress, temperature, neutron flux level, and fabrication history of the cladding.* The creep rate is quite sensitive to changes in the differential pressure across the cladding because of the dependence on stress (a power-law). The creep rate varies with the exponential of the temperature and is approximately proportional to the neutron flux. Thus, although both neutron flux and temperature are important factors influencing the creep of Zircaloy, the variation with temperature becomes relatively more important at temperatures above 650°F (and dominates at temperatures above 700°F) while the dependence on neutron flux is relatively more important at lower temperatures. One must be cautious about extrapolating data obtained at high temperatures (e.g., 700°F) to a lower temperature regime because the creep rate could be grossly underestimated. Conversely, one must be wary of extrapolating low temperature data obtained in-pile to higher temperatures because the creep rates could again be underestimated.

* See for example:

1. F. A. Nichols, "On the Mechanisms of Irradiation Creep in Zirconium Base Alloys," J. Nucl. Mater., 37 (1970), 59.
2. G. R. Piercy, "Mechanisms for the In-Reactor Creep of Zirconium Alloys," J. Nucl. Mater., 26 (1968), 18.
3. V. Fidleris, "Uniaxial In-Reactor Creep of Zirconium Alloys," J. Nucl. Mater., 26 (1968), 51.

In predicting the time at which the inward creep of the cladding will result in collapse of the cladding into contact with the pellets or into an axial gap in the pellet column, consideration has to be given to the anticipated operating mode because of the importance of the time-temperature-pressure history. The potential for creep-collapse is enhanced by increasing the stress level, temperature, neutron flux, and residence time. The trend in reactor design has been toward higher powers and higher coolant pressures, and consequently higher neutron flux levels, cladding temperatures, and stress levels.

One approach used to lower stress levels is to reduce the pressure differential across the cladding by prepressurizing the fuel rod. The exact amount of prepressurization that is required to preclude cladding collapse depends on the specific fuel rod geometry and operating conditions. Thus, pressurization of fuel may not preclude collapse unless the amount of pressurization is determined from an analysis of operating conditions in the plant that will use the fuel.

Typical conditions in PWR plants include system pressures of 2000-2250 psi, cladding midwall temperatures of 570-700°F, neutron flux levels ranging from 1×10^{13} - 1.2×10^{14} n/cm²-sec, and residence times of the fuel in the reactor of approximately three years. Staff estimates of the behavior of fuel cladding under the above conditions generally predict that under high pressure differentials (~ 2000 psi) and fast neutron flux levels of 5×10^{13} n/cm²-sec, clad collapse would occur during the first fuel cycle (~ 8000 hours). If the cladding temperatures were 600°F, reducing the fast neutron flux to

1×10^{13} n/cm²-sec would extend the time to collapse until sometime in the third fuel cycle. Reducing the pressure differential to 1000 psi would extend the collapse time, for a fast flux of 5×10^{13} n/cm²-sec, to beyond the end of the third cycle. For a fast flux level of 5×10^{13} n/cm²-sec, and with no credit taken for the effects of fission gas buildup, the Staff's estimates of the general dependence of collapse time on temperature and pressure differential are given in Table 3.2.1.

Table 3.2.1

TIME-TO-COLLAPSE ESTIMATES

Cladding Midwall Temperature (°F)	Pressure Differential (psi)	Time-to-Collapse
570	2500	During Cycle 1
	1650	Beyond Cycle 3
	1000	Beyond Cycle 3
600	2000	During Cycle 1
	1650	During Cycle 3
	1000	Beyond Cycle 3
635	2000	During Cycle 1
	1650	During Cycle 3
	1000	Beyond Cycle 3
675	2000	During Cycle 1
	1650	During Cycle 2
	1000	Beyond Cycle 3
700	2000	During Cycle 1
	1650	During Cycle 2
	1000	During Cycle 3

Three consequences of cladding creepdown are of interest. The first of these is primarily beneficial since it causes closure of the radial clearance gap and a consequent increase in radial gap thermal conductance. On the other hand, creepdown on the pellets early in the life of the fuel contributes to fuel column axial gap generation by preventing fuel column settling, and creepdown later in life can lead to collapse of the cladding into the axial gap. Collapses lead to increased local power peaking, and an increased probability of leaking of individual fuel rods.

The fuel manufacturers have developed a significant body of both analytical and experimental information relative to creepdown of the cladding onto the pellets as a result of efforts to understand cladding creepdown onto the fuel during normal operation. This information was considered by the Staff in the present study.

There exist two sets of experimental data on the times and conditions for creep-collapse. One set is derived from the observations on the Ginna, Beznau-1 and Point Beach Unit 1 reactors. The second set is drawn from in-reactor and out-of-reactor tests of tubing with various internal and external pressures, usually prototypical of production tubing, and exposed at a variety of temperatures. For the out-of-pile tests of tubing, appropriate creep rate modifications are made to accommodate enhanced creep due to irradiation, this component of the creep rate being important at the cladding temperatures of interest.

In general, the calculations performed by the fuel manufacturers predict shorter times to collapse for equivalent conditions than are observed in the

experimental tests probably due to the conservative assumption used in most of the calculational models that the rod internal pressure does not change as a result of fission gas buildup and to selection of the upper bound of the initial ovality data taken from the control charts of the manufactured tubing. This difference between calculation and experiment has led to adjustment of the calculational models so that the results match the experimental data.

The various creep-collapse models of the fuel vendors appear to be consistent in methodology and approach, and it is the opinion of the Staff that such models should be able to predict conservatively the time of creep-collapse over a wide range of operating conditions. The Staff is currently developing an analytical code to calculate the time-dependent ovality changes of a fuel rod for use in independently assessing fuel vendors' calculations. The Staff will compare the vendors' calculational models with the staff model by reviewing the results of sample calculations of standard problems. In addition, parametric studies will be performed to determine the sensitivity of creep-collapse time to differential pressure, cladding temperature, initial flux, cladding thickness, and fission gas production. Corrections will be made for second order effects, such as internal pressure changes due to sorbed gases, internal void volume changes, and solubility of the fission gases in the fuel. The capability to account for these parametric variations will be important in evaluating the various ways in which a reactor vendor may propose to alter the operational mode of a given plant to delay the occurrence of clad collapse. These operational changes may involve pressurization of fuel rods,

lowering of system coolant pressure, changes in reactor coolant temperature, and shifts in the core axial power profile.

3.2.2 Effects of Fuel Densification and Cladding Creep on Radial Heat Transfer

The suitability of current steady state models for evaluating the effects of fuel densification on radial gap conductance has been reviewed by the Staff. The Staff's consultant, the Pacific Northwest Laboratories of the Battelle Memorial Institute (PNL), prepared a study of radial gap conductance.* The GAPCON code was used by PNL to calculate radial gap conductance as a function of linear power density, burnup, fuel density, cold gap dimension, UO_2 thermal conductance and cold fill gas pressure. Although the effects of fuel densification are not specifically included in the parameters considered in the PNL evaluation, changes in the cold gap dimensions were used to simulate the effects of densification on the radial gap. The staff will use the GAPCON code to evaluate values proposed by applicants and licensees for the radial gap conductance to be used for densified fuel.

The radial gap conductance can vary as a function of time because of burnup. The significant factors contributing to the change in gap conductance are; cracking of the fuel pellet, fission gas release, swelling of the fuel pellet, cladding creep onto the fuel pellet, and release of the sorbed gases.

* C. R. Hann, "Fuel Rod Gap Conductance and UO_2 Thermal Conductivity" Study for the Reactor Systems Branch, Directorate of Licensing, USAEC, July 1972.

GAPCON includes considerations of all of these effects except cladding creep. Cladding creep will decrease the gap causing an increase in the gap conductance. As described in Section 3.2.1 the Staff is developing an analytical model to account for cladding creepdown and determine the time-to-collapse. When this analytical model is developed, it will be incorporated in the GAPCON code so that time-dependent gap changes due to the cladding creep can be calculated.

3.2.3 Effects of Collapse on Power Change Rates

Rapid power increases in rods with collapsed sections could cause stresses at the collapsed cladding section which are in excess of the yield stress as a result of the differential axial thermal expansion between the cladding and the column of fuel pellets that is constrained by cladding which has crept down. The thermal expansion of uranium oxide is greater than that of Zircaloy and the thermal response time for the fuel during a power change is much shorter than that of the clad. Thus, a power change in the fuel will cause the temperature to change in the pellets more rapidly than the radial heat flow can adjust the temperature of the fuel cladding. Since irradiated cladding has little residual ductility, cracking and possibly a breach of the cladding could occur if the yield strength is exceeded.

The Staff and its consultants have checked the Westinghouse calculations for Ginna by calculating the average unit axial stresses in the cladding arising from the differential expansion of the outer ring of the fuel pellet and the cladding tube. The outer ring of fuel is controlling because, for dished pellets, only this outer annular area will contact adjacent pellets.

The unit strain in the cladding is calculated by the following equation:

$$\xi_{\text{clad}} = (\alpha \Delta T)_{\text{fuel}} - (\alpha \Delta T)_{\text{clad}}$$

where:

α = coefficient of thermal expansion of the fuel and cladding and

ΔT = the temperature change in the fuel pellet at the dish radius, and the change in the average cladding temperature for a specified change in average linear heat rate of the fuel rod.

From this formulation a curve of unit cladding strain as a function of change in rod average linear heat rate was generated (See Figure 3.2.2). Using the elastic modulus of Zircaloy at operating temperature (10^7 psi), the cladding stress associated with a fuel rod power increase can be determined. Based on tensile stress-strain data for Zircaloy at the temperature of interest, a power change rate in terms of kW/ft per hour can be selected which would allow time for relaxation of the cladding stresses so that the maximum value would remain well below the yield strength. Such calculations can be used to select appropriate limits on allowable rates-of-change of power level in operating reactors to minimize the strain on collapsed cladding regions.

The use of a two-dimensional finite element stress analysis method would permit estimation of the local stress changes at the pellet collapsed section interface due to power changes and estimation of the effect of stress concentrations in the highly deformed cladding. Such an analytical capability is currently under development and will be of use, particularly for considerations of the behavior of fuel with collapsed sections during over-power transients.

FIGURE 3.2.2 Unit Fuel Clad Strain as a Function of a Change in Fuel Rod Linear Heat Rate from a Steady State Linear Heat Rate of 5.0 KW/ft.

3.2.4 Irradiation Growth and Fuel Rod Bowing

Kreyns has shown that the irradiation growth of Zircaloy is proportional to the square root of neutron fluence.* Hasketh has suggested that this dependence is explainable by postulating a constant number of interstitial loops of appreciable size and their growth by the long range diffusion of mobile defects.** This mechanism, in turn, would be affected by the rates of recombination of the point defects generated by the neutrons, a process that is temperature-dependent and that would tend to limit irradiation growth at high temperatures.

Prior to the first reports of Kreyns' work in about 1966, fuel rod growth was not anticipated by designers. Subsequently, the growth effect was observed in reactor fuel rods, and changes in fuel assembly design were made to accommodate the elongation. However, it is now apparent that there has been some difficulty in accurately predicting the magnitude of the growth. For example, in Ginna fuel assemblies, some fuel rods have elongated enough so that at room temperature the rods contact the top plate of the fuel assembly, resulting in bowing of those fuel rods. The rods are not bowed at operating temperatures, because the thermal expansion of the stainless steel control rod thimbles that tie the upper and lower assembly plates rigidly together is sufficiently greater than that of the Zircaloy rods at operating temperature ($\sim 577^{\circ}\text{F}$) to reestablish a clearance gap between the fuel rods and the top plate.

* P. H. Kreyns, WAPD-TM-583 (1966) 35

** R. V. Hesketh, "Non-Linear Growth in Zircaloy-4", J. Nucl. Mater. 30 (1969) pp. 219-221.

The Staff has examined the current Westinghouse design model for the irradiation growth of Zircaloy and has also calculated the axial buckling stresses for Zircaloy clad tubing in normal reactor operation and under loss-of-coolant accident conditions, assuming that the fuel rod length increase as a result of irradiation growth has resulted in interference between the fuel rod and top tie-plate as a result of inadequate clearance. For normal operation, this assumption means that it is possible to produce a large enough interference force to cause axial buckling of the clad tubes. However, to achieve this force and get axial buckling would require a lateral deflection greater than could occur in a real subassembly. To show this, Westinghouse performed an analysis assuming that the fuel rods were in contact with the tie-plate at cold shutdown conditions, with the fuel rods bowing to the point of rod-to-rod contact. They also assumed that the fuel rods do not straighten during startup and operation. The results of this analysis show that, even with the conservatively postulated maximum rod interference at cold shutdown (i.e., bowing until adjacent rods contact each other), the calculated clad stresses at operating conditions are less than the proportional limit. On this basis, and on the basis of out-of-pile tests on a fuel assembly in which varying degrees of fuel rod-adaptor plate interference were simulated, Westinghouse concludes that the lateral deflection is totally elastic and below the buckling stress.

Current Westinghouse fuel designs provide an end clearance gap that is based on correlation derived from a large number of measurements of irradiated fuel rods, and which is sufficient to eliminate the bowing problem.

Staff analyses have shown that the limiting case in determining an adequate end clearance gap arises in shutting down from normal operation. In this case, the temperature of the vertical tie rods of the fuel assembly (the control rod thimbles in most designs) decreases more rapidly than the fuel rod cladding temperature. The resulting differential thermal elongations are then maximized. Provided that the cladding temperature does not lag the tie rod temperature by more than 300°F, end clearance gaps in current fuel assembly designs are adequate.

4.0 Analysis of Densification Effects

As noted in Section 3.1.5, the occurrence of fuel densification in reactor fuel rods causes the fuel rods to contain more stored energy, increases the linear heat generation rates in the fuel rods, decreases the heat transfer capability of the fuel rods, and creates the potential (if axial gaps are formed in the fuel column) for local power spikes and for cladding collapse. In safety evaluations of power reactors it is necessary to consider these effects of fuel densification in the analyses made for all modes of reactor operation. These include normal operation, operation during various operational transient conditions, and postulated accident situations of overpower transients and postulated loss-of-coolant accidents (LOCA).

Analyses are performed to provide assurance that the fuel cladding integrity is maintained in the event of overpower operating transients. Certain limits relating to fuel performance must not be exceeded. These limits include limitations on the allowable cladding strain, on the heat transfer regime (i.e., MCHFR greater than 1.0 for a BWR and DNBR greater than 1.3 for a PWR*), and on the allowable center-line temperature of the fuel. The safety analysis for a reactor facility should contain an evaluation showing that these limits are not exceeded in the event of overpower transients, taking into account the effects of fuel densification.

* MCHFR and DNBR are minimum critical heat flux ratio and departure from nucleate boiling ratio. These ratios are measures of the clad-to-coolant heat transfer regime.

The current criteria for acceptable performance of emergency core cooling systems (ECCS) and evaluation models for calculating the course of a LOCA are set forth in the AEC Interim Policy Statement issued in June 1971. The LOCA and the performance of the ECCS must be analyzed using the acceptable evaluation models and such other requirements as may evolve as a result of the rulemaking hearing now in progress (Docket No. RM 50-1). These evaluations should be performed taking into account the effects of fuel densification as they may influence the behavior of fuel rods during the LOCA. The calculated peak cladding temperature in a LOCA for any fuel rod should not exceed the limit set forth in the Interim Policy Statement (2300°F), or any new limits that may evolve from the rulemaking hearing now in progress. (As of this writing, the Staff has filed supplementary testimony in the rulemaking hearing noting that the Interim Policy Statement criteria would be improved if the peak cladding temperature limit were reduced to 2200°F.)

Calculations should be performed to determine whether any fuel rods would be expected to experience cladding collapse during their residence in the reactor. For any fuel rods that could collapse, the calculated peak cladding temperature in a LOCA should be further limited to 1800°F, to preclude any significant oxidation of the cladding in the collapsed region that might result in embrittlement of the cladding.

The following subsections describe certain essential assumptions and approaches that should be included in calculational models used to evaluate the effects of fuel densification in safety analyses for power reactors.

4.1 Calculation of Fuel Column Gap Size

As previously noted (See Section 2.5), the Staff has concluded that the axial fuel column gaps observed in operating reactors arise from the densification of the fuel pellets and the resulting axial shrinkage of the pellets. The Staff has concluded, in view of the present incomplete state of knowledge about the time dependence of the densification phenomenon, that the evaluation models used for predicting the behavior of densified fuel should assume that fuel pellet shrinkage occurs as soon as the reactor begins power-range operation. The occurrence of fuel pellet-cladding interaction and the formation of axial fuel column gaps should be assumed to occur at the same time unless a clear basis from operating data can be presented to show that significant fuel column gapping does not occur.

The maximum theoretical gap size at a given core elevation should be determined by consideration of the difference between the final and initial pellet densities. The final density should be taken as 96.5% of theoretical. The nominal value for initial density should be based on data from pellet manufacture for the specific fuel loading considered. The fractional change in pellet length should be assumed to be $1/2$ of the fractional volume change and the fractional change in radius to be $1/3$ of the fractional volume change (See Section 3.1.4). This assumption maximizes the length change and results in a more conservative evaluation of the power spike that could result from the densification process. In addition, an appropriate increase in the maximum gap size from densification must be made to account for irradiation-induced growth of the fuel rod cladding.

4.2 Calculation of Probability of Gap Size and Location

A method, acceptable to the Staff, should be developed for calculating the probability distribution of gap sizes and locations. For the present, the calculated distribution of gap sizes should be based upon the measured collapse data from the Ginna Plant. Westinghouse has developed a method for calculation of the probability of a given gap size occurring at a given location (See Appendix E). The Staff is in general agreement with the theoretical basis for the design probability trend proposed by Westinghouse, but does not believe that the Ginna data support the values used by Westinghouse in determining the fraction of gaps (F_k) having a size within a gap size interval k (as discussed on page 3 of Appendix E). If the Westinghouse computational method is used for calculations of the probability of gap size and location, the values shown in Table 4.2.A for the gap size probability (labelled "AEC Probability Values") should be used.

As additional data on the frequency of distribution of gap sizes as a function of distance along the fuel rod from the bottom of the rod become available from gamma scanning, x-ray, or neutron radiographic examination of power reactor irradiated fuel, appropriate revisions may be made in probability models.

Table 4.2.A

PROBABILITY OF GAP SIZE

<u>Gap Size Interval (k)</u>	<u>Fraction of Maximum Gap Size</u>	<u>Westinghouse Probability Values Fk</u>	<u>AEC Probability Values Fk</u>
1	0 - 0.143	0.273	0.015
2	0.143 - 0.286	0.227	0.106
3	0.286 - 0.429	0.182	0.136
4	0.429 - 0.572	0.130	0.303
5	0.572 - 0.715	0.104	0.227
6	0.715 - 0.858	0.055	0.136
7	0.858 - 1.00	0.029	0.076

4.3 Calculation of Power Spikes

Calculations should be performed to determine the maximum local power peaking that can result from axial gaps in densified fuel rods. These calculations should be based on the calculated probability distributions of gap sizes and locations (See Section 4.2), combined with the normal core power distribution. The methods used in these calculations should be selected to provide a confidence level of at least 95% that the actual power peaking in the reactor core will not exceed the maximum calculated peaking for more than one rod in the core. Use of a particular model is not required, provided that the prescribed confidence level is shown to be achieved.

To establish the power peaking, the power spike should be calculated as a function of gap length. As the gap length is increased, the power peaking in an adjacent rod is increased, approaching an asymptotic value as the gap length approaches about two inches. It is also necessary to calculate the amount of attenuation in the power peaking for rods located one, two, and three rows distant from the rod of interest.

For the case of fuel rods with collapsed cladding, the calculations must be performed taking into account the increased moderation which occurs in the region of a collapsed section due to the local increase in the water-metal ratio. This assumption will result in greater peaking factors for rods with collapsed cladding sections.

An evaluation model for power spike calculations has been developed by Westinghouse based on the Ginna data. The Staff has had similar calculations performed independently by Brookhaven National Laboratory (BNL). Comparison

of the results indicates that the Westinghouse calculations are in general agreement with the magnitude of power spikes calculated by BNL. The Staff has concluded that use of the Westinghouse model is acceptable. The Westinghouse model (described in Appendix E) has been distributed previously to other reactor vendors and fuel element suppliers for information.

4.4 Calculations of Radial Gap Conductance

Calculations should be performed to evaluate the effects of densification on the radial gap thermal conductance of fuel rods, as a function of linear power density, burnup, fuel density, cold gap dimension, UO_2 thermal conductance, and cold-fill gas pressure. Although the effects of fuel densification are not specifically included in the parameters to be considered, changes in the cold gap dimensions can be used to simulate the effects of densification on the radial gap.

As noted previously, the assumption should be made that the fuel densifies immediately upon the start of power-range operation. The radial gap conductance should be calculated assuming that the radial gap increases according to the following expression:

$$\Delta r = \left(\frac{0.965 - \rho_a + 2\sigma}{3} \right) r$$

Where:

Δr = radial gap increase

σ = standard deviation in the measured probability distribution for the pellet fraction of theoretical density

ρ_a = nominal initial fraction of theoretical density

r = nominal initial pellet radius

The value for σ in the expression above is used to account for the variation in the pellet density that occurs in the fuel fabrication process. The standard deviation should be obtained from data generated from the fuel process used for fabrication.

The use of the measured statistical variation in initial pellet density is required because of the possibility that the region of peak power generation could extend over one to two feet of the fuel rod, and that a number of pellets (20 to 40) could be in the region of peak power generation. This requirement differs from the use of nominal initial density in calculating the total shrinkage of the fuel column (described in Section 4.5) because the generation of axial gaps depends on the behavior of all of the pellets in the fuel rod (about 240) which can be characterized adequately by use of nominal fuel parameters.

Until a suitable model for accounting for creepdown of the cladding is developed and verified, the gap conductance should be evaluated assuming that cladding creep does not contribute to gap closure. Proposed models should be submitted for Staff review before including any benefits of gap closure in the densification model. The models should be verified by comparison with appropriate experimental data.

Since it is not possible to define the point in operating lifetime when the gap conductance will be at a minimum, parametric studies should be performed to estimate the point in lifetime when the gap conductance attains its minimum value. Further, the change in gap conductance as a function of fuel burnup must be analyzed for a range of linear heat generation rates. The

minimum gap conductance established by such analyses (without consideration of cladding creep) should be used to analyze the behavior of the fuel for all modes of reactor operation.

4.5 Calculation of Linear Heat Generation Rate

As noted previously, the reduction in length of the fuel pellet column as a result of fuel densification will cause an increase in linear heat generation rate in direct proportion to the length decrease. In determining the linear heat generation rate, the calculated changes in the radial gap conductance and length of the pellet should be combined with the results of the power spike calculation (Section 4.3). Since the location of the power spike at a given elevation is given by a probability distribution, the effect of the statistical variation of pellet density on the calculated pellet dimensions may be accounted for in the probability analysis used for the power spike calculations, rather than in the calculation of radial and length changes. If this is not done, the expression given in Section 4.4 should be used for radius changes and the following expression should be used to calculate the decrease in pellet length in determinations of the linear heat generation rate:

$$\Delta L = \left(\frac{0.965 - \rho_i + 2\sigma}{2} \right) L$$

where:

ΔL = decrease in pellet length

L = length of pellet

and ρ_i, σ as defined in Section 4.4

4.6 Calculation of Cladding Collapse

As noted in Section 3.2.1, the various creep-collapse models developed by the fuel manufacturers appear to be consistent in method and approach. Such models, with appropriately conservative parameter values, should be used to predict the time of creep-collapse for fuel rods at the specific operating conditions for those rods. The acceptability of a given model to the Staff will be based upon comparisons of the results of calculations with existing experimental data and with the results of independent calculations made by the Staff and its consultants.

5.0 Staff Findings and Conclusions

Irradiation-induced densification of uranium oxide fuel has been observed. The observed densification has been shown to be relatively independent of linear heat generation rate (and consequently fuel temperature), has been observed to occur in both pressurized and unpressurized fuel, and is known to have occurred early in the fuel operating cycle (less than 2000 hours of operation in the power range).

Examinations of fuel assemblies after irradiation in some operating reactors, and in-core neutron flux measurements during operation have indicated the presence of axial gaps in the fuel pellet columns of some fuel rods, and the occurrence of short flattened or "collapsed" sections of cladding in some rods. Based on evaluation of experimental information, operating experience, and methods of fuel assembly fabrication, the Staff has concluded that the axial gaps arise from the densification of the fuel, the resulting axial shrinkage of the pellets, and the hangup of a pellet (due primarily to creep-down of the cladding) which prevents settling of the pellet column as a unit, rather than as a result of faulty assembly or of Zircaloy growth and fuel-clad mechanical ratcheting.

Sufficient experimental data are not available from all the manufacturers of light water reactor fuel to define clearly the degree of densification that might occur with fuel manufactured by each particular vendor. Fuel manufactured by the Westinghouse Corporation is known to densify. Fuels data accumulated by the General Electric Company have also indicated that densification had occurred. Until sufficient experimental information is available to permit

an unambiguous understanding of the densification mechanism (and of possible methods to control or eliminate densification), the Regulatory Staff will assume that irradiation-induced densification occurs in all water-reactor fuels.

Completion of the densification process is believed to require less than 2000 hours of reactor operation in the power range, but a full understanding of the time dependence of the process is lacking at present. The Regulatory Staff has concluded that until such an understanding is obtained, it should be assumed that densification occurs immediately when the reactor is operated at substantial power.

Similarly, because the time dependence of the creepdown of the cladding is not well established, the Staff has concluded that it should be assumed that axial gaps occur immediately when the reactor is operated at substantial power unless convincing operating data are available to show that gaps do not occur. The Staff has concluded, however, that available calculational models used to describe the creepdown and collapse of the cladding are sufficiently accurate for use in determining whether cladding collapses will occur in a given reactor core during a particular operating cycle. Surveillance programs should be developed for examination of fuel rods in operating reactors to provide additional assurance of the adequacy of the calculational models.

The effects of fuel densification cause the fuel rod to contain more stored energy, increase the linear heat generation rate of the fuel pellet, decrease the heat transfer capability between the pellet and cladding, and create the potential for occurrence of a power spike or a collapsed section

in any fuel rod. Methods to evaluate the magnitude of these effects are discussed in Section 4.0. The Staff has concluded that these methods should be used to include consideration of the possible effects of fuel densification in safety evaluations of nuclear facilities for normal operation, operation during various transient conditions, and postulated accident situations.

Specifically:

(1) It should be shown that the effects of densification during steady-state and transient operation of the reactor do not cause the limits on cladding strain, critical heat flux, and centerline temperatures (previously established in the facility safety evaluation) to be exceeded.

(2) The effects of densification should be included in calculating the behavior of the fuel rods during postulated loss-of-coolant accidents. The criteria for evaluating the performance of emergency core cooling systems are presently set forth in the AEC Interim Policy Statement issued in June 1971 which is currently the subject of a rulemaking hearing (Docket No. RM 50-1). These criteria or such other requirements that may evolve as a result of the hearing now in progress are to be used.

(3) If calculations indicate that the cladding of some fuel rods can be expected to collapse during an operating cycle:

(a) limits should be established on the rate of change of power level to minimize the strain on the collapsed cladding regions,

(b) limits should be imposed on permissible steam generator leakage to assure that potential releases of radioactive material from leaking fuel rods meet applicable criteria, and

(c) analyses should be provided to show that the calculated peak cladding temperature of any collapsed fuel rod during a postulated LOCA will not exceed 1800°F.

(4) Any operating restrictions necessary to assure that the above conditions are satisfied should be developed and incorporated into the technical specifications for each facility. Additional operating experience and results of experimental programs may indicate that changes in these restrictions or the methods used to derive them are warranted. Flexibility to consider this information has been identified when appropriate (for example, see Section 4.0).

In summary, the Regulatory Staff has concluded that adequate data are available to account for the effects of densification in the safety evaluations of nuclear power reactors. Considerable reliance was placed on direct application of data obtained from operating reactors. These reactors have been operated safely with densified fuel and collapsed cladding, and the Regulatory Staff has concluded that implementation of the recommendations of this report will provide increased assurance of continued safe operation of these reactors and other reactors in which fuel densification is expected to occur.

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 R R KOPIROWSKI
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June 23, 1972

Mr. Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Skovholt:

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Enclosed find six (6) copies of the non-proprietary version of
a report submitted to you with our letter of June 12, 1972. An addi-
tional thirty-four (34) copies will be transmitted to you on
June 26, 1972.

Very truly yours,

R. R. Koprowski
Vice President and
Chief Engineer

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ATTACHMENT TO RGE LETTER

DATED JUNE 12, 1972

R.E. GINNA UNIT NO. 1

CYCLE 2 EVALUATION

(Non-Restricted Version)

This document is a non-restricted version of the proprietary attachment to the Rochester Gas & Electric Company letter of June 12, 1972. Selected material in the proprietary attachment has been deleted in accordance with 10CFR Part 9, Section 9.5

1.0 BACKGROUND INFORMATION

The cycle 1B fuel loading pattern of Figure 1 was determined by replacing 12 leaking fuel assemblies from Region 3 by 12 fresh Region 4A assemblies during a scheduled turbine inspection in the spring of 1971. The plant operated normally at the nominal 1300 MWt full power rating until the March 1972 scheduled shutdown in preparation for the 1520 MWt uprating. During a six week period, the plant operated at a maximum power of 1455 MWt except for 6 hours of operation at 1520 MWt. During this time period, primary coolant activities increased which indicated additional fuel leakers. Examination of the cycle 1B fuel (see Section 2) revealed fuel leakers, bowed fuel rods, and an appreciable number of rods with collapsed sections - especially in Region 2.

As a result of the fuel examination, the cycle 2 fuel loading pattern was changed to that shown in Figure 2 and Table 1. Essentially, 61 fresh Region 4 and 4A assemblies replace all of the cycle 1B, Region 3, most of the Region 2 and only 3 of Region 1 assemblies.

2.0 RGE CYCLE 1B VISUAL EXAMINATION SUMMARY

An underwater television viewing system was used to conduct visual examinations on 54 RGE fuel assemblies. Twenty Region 1 assemblies, twenty-two Region 2 assemblies, eleven Region 3 assemblies and one Region 4A assembly were included in the visual examination sampling plan. Each examined fuel assembly was visually scanned on all four faces. The majority of the peripheral fuel rods examined (90%) showed no abnormalities. The remaining 10% of peripheral rods showed a series of abnormal conditions including failures, bowed rods, and collapsed cladding. Table 2 summarizes the total number of peripheral fuel rods examined, the type of anomalies observed and the frequency with which these anomalies were observed in each region of fuel. Each type of observed anomaly, and the frequency of occurrence are discussed in the following paragraphs.

Failed Fuel Rods (Visually confirmed breached cladding)

A total of 4 visually confirmed failed fuel rods were observed. One failure (a split in the cladding) occurred in a Region 1 fuel assembly. The remaining three failures (apparent holes in the cladding) were observed in 3 different Region 3 fuel assemblies. No failed fuel rods were observed on the periphery of Region 2 or Region 4A assemblies. The failed rods constitute .14% of the peripheral rods viewed.

Collapsed Cladding

A collapse is a short segment of unsupported cladding, which due to high differential pressure has crept down to the point of appearing partially or totally flat in profile. Collapsed rods were the most frequently observed anomaly. [*]

* Westinghouse material customarily held in confidence, deals with the results.

Collapsed rods were observed in Regions 1, 2, and 3 fuel. [*]
Observed collapsed areas were distributed over the upper 40% of the fuel rod length with the highest frequency of collapse occurring between 120 and 140 inches from the bottom of the rod. [*]
Only one instance of collapse in adjacent rods at the same axial location was observed. The lengths of collapse vary from .5 to 4.0 inches. [*] In general the length of collapse increases as the observed axial location approaches the top of the fuel rod.

Fuel Rod to Top Nozzle Gaps

Rod to nozzle gaps were estimated for the peripheral rods in the visually examined assemblies. [*] No instances of rod interference were observed in the Region 3 or Region 4A assemblies.

Rod to nozzle gaps are on the average greater for collapsed rods than for non-collapsed rods.

* Westinghouse material customarily held in confidence - deleted material describes frequencies of observed anomalies.

TABLE 2

SUMMARY OF RGE CYCLE 1B VISUAL EXAMINATIONS

<u>REGION</u>	<u>NUMBER OF EXAMINATIONS</u>		<u>NUMBER OF PERIPHERAL RODS COLLAPSED</u>	<u>NUMBER OF PERIPHERAL RODS TOUCHING END PLATE</u>
	<u>ASSEM.</u>	<u>PERIPHERAL RODS</u>		
1	20	1040	21	*
2	22	1144	83	*
3	11	572	20	0
4A	1	52	0	0

* Westinghouse material customarily held in confidence, values are approximately 5%.

3.0 EVALUATION

Collapsed rod sections and axial pellet gaps due to fuel densification causes a local power spike in the neighboring rods and pellet edges adjacent to the collapsed sections and/or pellet gaps. A number of adjacent pellet gaps and/or collapsed rod sections will cause a power spike of greater magnitude than one occurrence. Section 3.1 examines the effect of these occurrences on power spiking and core power capability.

3.1 Effect of Fuel Densification on Power Capability

Observations in several operating PWRs indicate that pellet densification occurs in pile. [1] This has been observed in post-irradiation hot cell measurements [2] and flattening of fuel cladding where there is no fuel pellet to provide support. [3]

The above phenomena can affect power capability in the following ways:

- 1) The core average kw/ft and heat flux is raised due to shortening of the pellet stack;
- 2) Local power peaks will exist in the vicinity of a gap in the pellet stack and the adjacent rod(s)
- 3) Additional local power peaking occurs when the fuel rod is flattened thereby providing more local neutron moderation;
- 4) Lattice geometry is distorted by the flattened rod sections with a resultant effect on DNB ratios.

¹ Westinghouse material customarily held in confidence - The essence of the deleted material is "a density a few percent above the as-loaded condition"

² Westinghouse material customarily held in confidence.

³ Westinghouse material customarily held in confidence - This material describes the timing of the densification and collapse phenomena.

- 5) Power increases in rods with flattened sections may result in exceeding yield stresses at the lower collapsed cladding section due to the differential axial expansion between fuel and clad.

With respect to item (5), operation of the Ginna unit as described in the cover letter will assure that rods with flattened sections operate at a power level less than previously experienced. Item (1) raises the core average specific power by [1] , when the specific cycle 2 loading is considered. [2]

As determined by the W3 DNB correlation, the effect of local power spikes on DNBR ratios is approximately half the magnitude of the power perturbation (this is determined by an "F" factor which relates uniform and non-uniform DNB heat flux). A power level reduction sufficient to offset the power spike effect on Kw/ft is more than sufficient to maintain DNB ratios above previous values, even allowing for local geometry changes.

Local power spikes resulting from air gaps and water have been analyzed with [3] transport codes [3] and confirmed by experiments in the Saxton reactor. Conclusions are shown in Figure 3. The effect is highly localized and can be represented conservatively by [3]. The perturbation is rapidly attenuated as follows. [3]

On the basis that gaps are [3] more probable at EOL Cycle 2 than observed collapses at EOL Cycle 1B, the curve of Figure 4 results.

[4] It can be seen that the model is quite conservative with respect to the data. [4]

¹ Westinghouse material customarily held in confidence - Less than 2%.

² Westinghouse material customarily held in confidence

³ Westinghouse material customarily held in confidence - This material describes the computational models used and the analytical results.

⁴ Westinghouse material customarily held in confidence - The deleted material deals with specific analytical methods, results and experimental confirmation.

When the above model is applied in the Ginna Cycle 2 power distribution analysis, [1], it is concluded that gaps in the pellet stack in the vicinity of high power channels represent a more stringent power capability limit than collapses. [1]
The high power channel is located in Region 4A, which contains pressurized rods with no observed or predicted collapses.

Using the core wide census from Figure 5, it can be stated that no rod in the core will exceed the local power generated in the hot channel ($F_H^N = 1.66$) while allowing for a localized 17% power spike; consequently this would be appropriate design basis for cycle 2. Recalling the axial stack height change of [2] this would require that the product $F_Q \times \text{Power Level}$ be reduced by [2] to maintain Kw/ft margins.

Continued surveillance of the in-core flux peaking is planned, and comparison of new data with Figure 4 provides a quick confirmation of the continued conservatism in the models used to arrive at the Cycle 2 power rating. It is felt that the model is unduly conservative at this point.

¹Westinghouse material customarily held in confidence - The deleted material deals with specific analytical methods, results and experimental confirmation.

²Westinghouse material customarily held in confidence - The deleted values are less than 2% and less than 20% respectively.

3.2 Effects of Axial Fuel Rod Growth

Due to the different expansion characteristics of the zircaloy fuel rod cladding and the stainless steel thimbles, and the non-reversible irradiation zircaloy growth of the cladding, thermal differential contraction during reactor cooldown could result in rod interference with the top adapter plate at cold shutdown. Yielding of the guide thimbles or failure of guide thimble-adapter plate joints could result if excessive rod interference were to occur. The rod interference distribution, bowing of the fuel rods, and the reaction forces in the guide thimbles of the fuel assembly due to temperature change and irradiation growth are being evaluated. The growth equation is based on data from RGE cycle 1A and cycle 1B. Any problems due to this effect would occur at the end of life and will be addressed in a subsequent report.

FIGURE 1 Cycle 1B Core Loading Pattern

RGF CYCLE 2 FIGURE 2

LOADING PATTERN FOR CYCLE 2

XX - Position in Cycle 1 B
R - Region Number

TABLE 1

GINNA CORE LOADINGS

FUEL DESCRIPTION				ASSEMBLIES LOADED		
FUEL BATCH	ENRICH (w/o)	DENSITY (%)	PRE- PRESSURIZED	CYCLE 1A	CYCLE 1B	CYCLE 2
Region 1	2.44	94	No	41	41	38
Region 2	2.78	92	No	40	40	10
Region 3	3.48	90	No	40	28	
Region 4A	3.20	92	Yes		12	12
Region 4	3.14	94	Yes			40
Region 4B	2.90	94	Yes			21
			TOTAL	121	121	121

*
SCALES, CURVES AND DATA DELETED AS WESTINGHOUSE MATERIAL CUSTOMARILY HELD IN CONFIDE
←

FIGURE 4
POWER SPIKE LIMITS

CURVES AND DATA DELETED AS WESTINGHOUSE MATERIAL CUSTOMARILY HELD IN CONFIDENCE

FIGURE 5
RELATIVE POWER CENSUS

* CURVES AND DATA DELETED AS WESTINGHOUSE MATERIAL CUSTOMARILY HELD IN CONFIDENCE

Regulatory

File Cy.

50-244

Westinghouse Electric Corporation

Power Systems

Nuclear Fuel Division

Box 355
Pittsburgh Pennsylvania 15230

NFD-NE-584

June 30, 1972

Mr. Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing
U. S. Atomic Energy Commission
Washington, D.C. 20545

Subject: Fuel Abnormalities at the R. E. Ginna Plant
and Other Operating PWRs

Dear Mr. Skovholt:

The attached material is in response to questions posed by members of the Regulatory staff during our June 23, 1972, meeting. When the questions were formulated you had apparently not yet had an opportunity to review the submittal made by Rochester Gas and Electric Corporation, dated June 22, 1972. For the sake of completeness, some of the information in that submittal is repeated here.

Section 3 of the attached material provides a description of the fuel elements, including UO_2 density, dimensions, etc., as well as an operating history of 2 defective fuel. A description of local flux peaks observed on flux traverses, and information on the expected growth rate of zirconium, are provided in Section 5. Fuel rod crimping as observed at the Ginna and NOK plants, is described in Section 4; these are the only instances of observed crimping, and no pre-pressurized fuel has evidenced this phenomena. The effect of the fuel abnormalities on core and plant performance during normal operation and anticipated transients is discussed in Section 6.

This material is non-proprietary and may be placed in the public document room.

Sincerely,

W. D. Leggett, III
Manager, Nuclear Engineering
Nuclear Fuel Division
Westinghouse Electric Corporation

CC: R. R. Koprowski, RGE

3636

~~Received w/lt dated~~ 6-30-72

FUEL ABNORMALITIES
AT THE
R. E. GINNA PLANT
AND
OTHER OPERATING PWRs

JUNE 30, 1972

1.0 INTRODUCTION

Cycle 1B of the Ginna Nuclear Plant was initiated in the spring of 1971. Twelve Region 4A assemblies replaced twelve leaking Region 3 assemblies during the cycle 1A-1B outage. The plant operated normally during cycle 1B at the nominal 1300 MWt full power rating until the March 1972 scheduled shutdown in preparation for the 1520 MWt uprating. The plant then operated at a maximum power of 1455 MWt except for 6 hours of operation at 1520 MWt. Primary coolant activity increased during this time period which indicated additional fuel leakers. Ginna was removed from service on April 14, 1972, for inspection, maintenance, and cycle 1B-2 refueling. Examination of the cycle 1B fuel (Section 4.0) revealed a number of anomalies including fuel leakers, bowed fuel rods, and an appreciable number of fuel rods with collapsed sections. Ginna cycle 2 fuel performance is discussed in Section 6.0 based on the observed anomalies, the predictability of the anomalies (Section 5.0) and the administrative controls to be applied to cycle 2 (Section 2.0). Section 3.0 contains pertinent fuel rod design and operation information.

2.0 ADMINISTRATIVE CONTROLS

The administrative controls given in Table 1 have been instituted to either minimize further fuel damage or to mitigate the radiation release to the environment should fuel damage occur in spite of the steps taken for prevention.

3.0 DESIGN AND OPERATIONS SUMMARY

The maximum operating power level has been reduced, on an interim basis, from the uprated 1520 MWt to 1266 MWt (1520/1.20). This power level is considered 100% power when interpreting safety limits, limiting safety system settings, and limiting conditions of operation. The reduced power level is necessary, utilizing the very conservative information and analysis available on a short term basis, to maintain thermal-hydraulic margins and to minimize fuel clad interaction.

The Ginna core consists of 121 fuel assemblies. The number of assemblies in the core by fuel region is given in Table 2 along with the enrichment, density, and pressurization of each region. Figure 2 gives the cycle 1B fuel loading pattern and Figure 3 gives the cycle 2 fuel loading pattern.

Table 3 gives pertinent fuel rod mechanical design parameters.

The core average burnup for cycles 1A and 1B is given in Table 4 along with the end of cycle region average burnup. Also given in Table 4 is the time and volume averaged linear power for the entire core and by region for cycles 1A and 1B.

4.0 FUEL EXAMINATION

An underwater television viewing system was used to conduct visual examinations on 54 Ginna fuel assemblies. Twenty Region 1 assemblies, twenty-two Region 2 assemblies, eleven Region 3 assemblies and one Region 4A assembly were included in the visual examination sampling plan. Each examined fuel assembly was visually scanned on all four faces. The majority of the peripheral fuel rods examined (90%) showed no abnormalities. The remaining 10% of peripheral rods showed a series of abnormal conditions including failures, bowed rods, and collapsed cladding. Table 5 summarizes the total number of peripheral fuel rods examined, the type of anomalies observed and the frequency with which these anomalies were observed in each region of fuel. Each type of observed anomaly, and the frequency of occurrence are discussed in the following paragraphs.

Leaking Fuel Rods

All remaining Region 3 fuel assemblies were replaced during the cycle 1B-2 refueling. Region 3 was originally intended to remain in the core for 3 cycles. Fuel examination in 1971 indicated that the majority of Region 3 fuel assemblies contained leaking fuel rods. Local hydriding with subsequent cladding breach is believed to be the cause of the majority of the leaking fuel rods. Fuel rods either leaking or suspected of leaking were also found in Ginna Regions 1 and 2 but to a much lesser extent than in Region 3.

Failed Fuel Rods (Visually confirmed breached cladding)

A total of 4 visually confirmed failed fuel rods were observed. One failure (a split in the cladding) occurred in a Region 1 fuel assembly. The remaining three failures (apparent holes in the cladding) were observed in 3 different Region 3 fuel assemblies. No failed fuel rods were observed on the periphery of Region 2 or Region 4A assemblies. The failed rods constitute .14% of the peripheral rods viewed.

Collapsed Cladding

A collapse is a short segment of unsupported cladding, which due to high differential pressure has crept down to the point of appearing partially or totally flat in profile. Collapsed rods were the most frequently observed anomaly.

Collapsed rods were observed in Regions 1, 2, and 3 fuel. Observed collapsed areas were distributed over the upper 40% of the fuel rod length with the highest frequency of collapse occurring between 120 and 140 inches from the bottom of the rod. The lengths of collapse vary from .5 to 4.0 inches. In general, the length of collapse increases as the observed axial location approaches the top of the fuel rod.

Collapsed cladding was also observed in Beznau Unit No. 1 Region 2 (approximately 2%) but not in Regions 1 and 3. No collapsed rods have been observed at Zorita which is in its third cycle and no collapsed rods have been observed in pre-pressurized fuel.

Fuel Pellet Gaps

Fuel densification will occur (discussed in Section 5) irrespective of whether the cladding collapses. This has been verified by hot cell examination (x-ray and gamma scanning) of Beznau Unit No. 1 fuel rods discharged after cycle 1. As a result, since the fuel cladding does not shrink axially, either the fuel pellet stack is shortened or gaps occur between the fuel pellets. These gaps exist where the fuel cladding collapses but they may also exist in uncollapsed fuel.

Fuel Rod to Top Nozzle Gaps

Rod to nozzle gaps were estimated for the peripheral rods in the visually examined assemblies. No instances of rod interference were observed in the Region 3 or Region 4A assemblies. Rod to nozzle gaps are on the average greater for collapsed rods than for non-collapsed rods.

Two rods were observed in contact at Ginna 1A cycle shutdown. This assembly was visually examined at the recent shutdown. Additional rods are now in contact but the assembly continues to perform satisfactorily.

5.0 PREDICTABILITY OF ANOMALIES

Fuel Rod Collapse

Creepdown of fuel cladding onto the fuel is a predictable result of long-term operation in a LWR. Recognition of this phenomenon has led Westinghouse to pre-pressurize fuel regions starting in 1969.

In Zorita, Beznau No. 1 and Ginna, the initial core loading of Zircaloy-clad fuel was not pre-pressurized.* Given a significant length of unsupported (by fuel pellets) clad, inward creep results in increased ovality and eventual instability leading to collapse of the clad into the axial gap. This phenomenon was observed at Beznau No. 1 during the first refueling and at Ginna during the recent cycle 1B refueling.

On-site and post-irradiation evaluations of Beznau No. 1 fuel have indicated that the axial fuel gap is not the result of missing fuel pellets. Rather, the fuel stack has densified and incomplete settlement has resulted in local and distributed gaps along the fuel rod length. This description has been verified by gamma-scanning, x-ray, rod weighing and density determinations made on Beznau No. 1 fuel, Westinghouse test rods irradiated in Saxton and other Westinghouse post-irradiation studies.

Observations of fuel following Beznau No. 1 cycle 1 and Ginna cycles 1A and 1B have been evaluated and provide the basis for a consistent model of creep-induced clad collapse. Using the model and other analytical studies of tube collapse, an estimate has been made of the extent of additional collapses anticipated during subsequent operation. Region 1 rods exhibit significantly fewer collapses, generally of smaller size. These observations are consistent - small gaps lead to long collapse times. Region 1 rod collapse frequency will also increase during cycle 2, however, the already observed tendency for fewer and smaller collapses than in Region 2 is expected to continue.

* Cores fabricated since 1970 are pre-pressurized in order to inhibit clad creep.

Local Abnormal Flux Peaking

Local power spikes resulting from air gaps and water gaps have been analyzed with transport codes and confirmed by experiments in the Saxton reactor. A model has been developed for combining multiple gaps and/or collapses in order to predict the maximum power spike which could occur. The model produces the probability of not exceeding a given power spike.

Nearly all flux traces taken at Westinghouse plants utilizing the in-core movable detector system show local flux peaks, the vast majority of the peaks being less than 1%. Flux peaks in the range of 4 to 9% have been detected on four traces (out of several thousand). This local flux peaking can be noted on the flux traces after a few months of operation. No collapsed rods were noted during the Ginna cycle 1A-1B refueling several months after the start of operation. This provides confirmation that gaps exist between fuel pellets even though the cladding has not collapsed. The models developed to predict local flux peaking is very conservative when compared to the flux peaks noted on flux traces. Continued surveillance of the in-core flux peaking is planned and the comparison of the new data with the models will provide confirmation of the conservatism.

Zircaloy Growth

Westinghouse has developed a Zircaloy fuel rod growth correlation based on observations and measurements made at Saxton, Beznau No. 1, Mihama 1, Zorita and Ginna. A total of approximately 2000 measurements are included in the available Westinghouse data. At Zorita and at Ginna, data have been obtained after each of two cycles providing a good test of the predictive ability of the design models. The growth observed after two cycle exposure has in all cases been within the design uncertainty band and in most cases has been below the mean prediction. In addition, as a measure of overall system predictions, the number of rods touching the top nozzle in the cold condition correlates well.

Based on these data, design evaluations have been performed to provide assurance of satisfactory performance during cycle 2 operation and the subsequent cold shutdown. Since the stainless thimble thermally expands to a greater extent than the Zircaloy-clad fuel rods, the clearance between

the top of the fuel rod and the top nozzle is a minimum in the cold shutdown condition. During operation, the rod exhibiting the highest growth will not contact the top nozzle until approximately 2/3 of the full second cycle exposure has been accumulated. However, contact will not result in rod bowing until a substantial additional interference is experienced. This level of interference will not occur by the end of cycle 2. Some rods are currently in contact at the cold shutdown condition. Upon return to power, the differential thermal expansion between rods and thimbles will provide an additional 0.4 inch relative expansion of the assembly and provide a gap for all rods.

At subsequent cold shutdowns additional rods are expected to contact the top nozzle. Analyses have been performed which demonstrate that loads imposed on the fuel rods during cold shutdowns remain in the elastic range for all rods which do not exhibit collapse. Some number of collapsed rods are expected to be in significant interference cold. These rods may be stressed beyond yield in the cold condition and some could crack and release their fission gas inventory as a result. Use of a controlled approach to the cold shutdown condition should cause the few failures so anticipated to occur over a sufficient time span to limit the activity released.

The rods impose a load on the assembly in the cooled down condition which has been analyzed and found to be acceptable. A conservative assessment of tolerance factors was made to quantify the risk of parting of the bottom nozzle. This risk has been determined to be considerably less than one chance in one hundred for the lead growth assembly. Should such parting occur, the thimbles would be retained in their current positions by guide pins, scram action would not be compromised and the assembly could sustain additional subsequent irradiation.

6.0 PERFORMANCE EVALUATION

Fuel Clad Interaction

When the fuel pellets are tightly constrained by the fuel cladding, and a power increase takes place, the greater thermal expansion of the uranium oxide can potentially result in clad stresses above the Zircaloy yield strength. Although some local clad ductility may exist, it is difficult to take quantitative advantage of it. Hence, clad yield stress is used as the design limit - exceeding this limit is then defined as clad failure. Such failure is not however to be characterized as an abrupt or massive blowout or loss of integrity. Failure here will likely take the form of local cracking, with some being through wall cracks.

Two conditions exist in this fuel which give rise to this constraint:

- (1) clad creeps down on the pellet, providing radial constraint; and
- (2) clad collapses down into a pellet axial gap, providing axial constraint.

The potential for clad failure in both cases is considered low because of the reduced power operation and considering the revised, more restrictive, control rod insertion limits. The exception to this may be certain overpower transients which would cause an increment in rod average power. The administrative restrictions listed in Table 1 are intended to minimize fuel clad interaction and these modes of failure. In addition, technical specification limit coolant activity and potential releases. Finally, the coolant activity monitor provides a means to continuously evaluate clad failures and enables action to be taken in a timely manner should a large number of failures occur.

DNBR Evaluation

With respect to DNB under normal operation and expected transients in Ginna Cycle 2, the following offsetting considerations exist:

- 1) Core power will be reduced by a factor of $1/1.20$; this will increase DNB ratios by 34%.
- 2) If fuel stack height and local power spikes were to increase the local heat flux by 20%, the DNB ratio is calculated to be reduced by 10%. Halving the power spike effect is a consequence of the integrating "F" factor in the W-3 correlation, which relates uniform and non-uniform DNB heat fluxes.
- 3) Flow patterns will be altered in the vicinity of a collapsed section; however, sudden contraction and expansion effects are not detrimental. In fact, increased turbulence and mixing effects tend to improve heat transfer.
- 4) Rod spacing may be altered slightly as a consequence of the collapsed section on some fuel rods; this effect will reduce DNB ratios by less than 10%.
- 5) The fuel in the vicinity of collapsed rods will run at power level more than 15% below the core hot channel; this leads to DNB ratios increased by more than 25%.
- 6) If a collapsed section is in contact with an adjacent rod, the DNB ratio is not necessarily reduced (as evidenced by the contact of the support grids and fuel rods). Electrically heated DNB tests show a penalty for full length contact of 7%, which is clearly an upper limit.

It can be seen that those effects increasing DNB ratios for Cycle 2 are far in excess of the detrimental effects. Thus, it is concluded that for Cycle 2 operation, DNB ratios during normal operation and in expected transients are at least as high as those reported during previous safety analyses.

Fuel Temperature

Fuel densification increases the beginning of life fuel pellet/cladding gap which in turn leads to an increase in fuel temperature. This temperature increase is less than 200°F. The maximum fuel central temperature at overpower remains less than the minimum UO_2 melting point of about 4800°F (45,000 MWD/MTU burnup). Unirradiated UO_2 melting point is about 5100°F.

The amount of energy stored in the fuel is larger, for a given linear power rating, due to the increased fuel temperature. This stored energy affects the loss of flow (LOF) and loss of coolant (LOCA) accidents. Analysis of a 3 ft.² LOCA showed a 175°F peak cladding temperature increase to 2210°F assuming the same maximum linear power (core power decrease offsets power spike) and no reduction in total core power. Decreasing the total core power enhances cooling during blow-down. The densification and higher fuel temperatures result in a slower LOF heat flux decrease following reactor trip. The slower heat flux decrease is expected to have less effect than the reduced operating power level.

TABLE 2

GINNA CORE LOADINGS

FUEL DESCRIPTION				ASSEMBLIES LOADED		
FUEL BATCH	ENRICH (w/o)	DENSITY (%)	PRE- PRESSURIZED	CYCLE 1A	CYCLE 1B	CYCLE 2
Region 1	2.44	94	No	41	41	38
Region 2	2.78	92	No	40	40	10
Region 3	3.48	90	No	40	28	
Region 4A	3.20	92	Yes		12	12
Region 4	3.14	94	Yes			40
Region 4B	2.90	94	Yes			21
				—	—	—
			TOTAL	121	121	121

TABLE 1

ADMINISTRATIVE CONTROLS TO BE APPLIED TO GINNA CYCLE 2

1. Control Rod Insertion Limits as given in Figure 1 under normal operating conditions (except for physics tests). During load losses or automatic turbine runbacks, this limit may be exceeded by automatic (or manual) rod control but the limits will be re-established once power has stabilized.
2. No part-length control rods in the core when reactor is at power, (except for special tests).
3. Cycle 2 initial startup:
 - a) 3% power/hour from 0% to 75% power
 - b) 3% power/day from 75% to 100% power
 - c) continuous chemistry follow.
4. Cycle 2 operation other than initial startup:
 - a) 10% power/hour increase
 - b) revert to initial startup procedures should low power (less than 95%) be maintained continuously for more than 24 days.
5. Cycle 2 cold shutdown (anytime): Cool down slowly and monitor primary coolant chemistry continuously. Hold at a constant temperature if a greater than expected increase in activity occurs until the situation can be evaluated.
6. Shutdown if overpower transient occurs and monitor coolant.

TABLE 3

FUEL ROD MECHANICAL DESCRIPTION
(Beginning of Life)

Fuel Rod O.D. = 0.422 in.

Cladding Thickness = 0.0243 in.

Cladding Material = Zircaloy-4

Fuel Pellet O.D. (Regions 1, 2, 3) = 0.3669 in.

Fuel Pellet O.D. (Region 4A) = 0.3649 in.

Fuel Pellet O.D. (Region 4, 4B) = 0.3659 in.

Fuel Pellet Height = 0.600 in.

Fuel Pellet Stack Height (Regions 1, 2, 3) = 144 in.

Fuel Pellet Stack Height (Region 4A) = 142.8 in.

Fuel Pellet Stack Height (Regions 4, 4B) = 142 in.

Guide Thimble Material = 304 SS

TABLE 4

GINNA FUEL OPERATING HISTORY

	<u>CORE</u>	<u>REGION 1</u>	<u>REGION 2</u>	<u>REGION 3</u>	<u>REGION 4A</u>
Average Burnup-End of Cycle 1A (MWD/MTU)	7700	9,200	8,300	5,600	-----
Average Burnup-End of Cycle 1B (MWD/MTU) (Cy. 1B)	8830	18,800	18,000	10,000	9,500
Time/Volume Averaged Linear Power - Cycle 1A (kw/ft)	4.2	4.8	4.4	3.2	-----
Time/Volume Averaged Linear Power - Cycle 1B (kw/ft)	4.8	5.3	5.2	3.4	5.1

TABLE 5

SUMMARY OF RGE CYCLE 1B VISUAL EXAMINATIONS

<u>REGION</u>	<u>NUMBER OF EXAMINATIONS</u>		<u>NUMBER OF PERIPHERAL RODS COLLAPSED</u>
	<u>ASSEM.</u>	<u>PERIPHERAL RODS</u>	
1	20	1040	21
2	22	1144	83
3	11	572	20
4A	1	52	0

A B A - CYCLE 1A POSITION
B B - REGION NUMBER

FIGURE 2

Cycle 1B Core Loading Pattern

RGE CYCLE 2 FIGURE 3

LOADING PATTERN FOR CYCLE 2

XX - Position in Cycle 1 B
R - Region Number

APPENDIX C

Regulatory

File Cy.

ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14604

R. R. KOPROWSKI
VICE PRESIDENT

TELEPHONE
AREA CODE 716 546-2700

July 28, 1972

Mr. Donald J. Skovholt
Assistant Director for
Operating Reactors
Directorate of Licensing
U. S. Atomic Energy Commission
7920 Norfolk Avenue
Bethesda, Maryland 20014

Subject: Non-Proprietary Documentation for the
Submittals of June 20 and 22, 1972,
R. E. Ginna Nuclear Power Plant Unit No. 1,
Docket No. 50-244

Dear Mr. Skovholt:

Enclosed are forty (40) copies of the non-proprietary document titled "Ginna Cycle 2 Accident Analyses Evaluation".

This document, together with the non-proprietary report "Fuel Abnormalities at the R. E. Ginna Plant and Other Operating PWR's" submitted on June 30, 1972, completes the non-proprietary documentation corresponding to the submittals of June 20 and 22, 1972 titled "Responses to Questions Asked of Rochester Gas and Electric Corporation on Ginna" and "Ginna Cycle 2 Performance Evaluation".

Very truly yours,

R. R. Koprowski
Vice President and
Chief Engineer

Enclosure

4403
W

GINNA CYCLE 2

ACCIDENT ANALYSES EVALUATION

Submitted by

Rochester Gas and Electric Corporation

July 28, 1972

ACCIDENT EVALUATION

Each FSAR accident related to fuel performance has been reviewed and is summarized below. Two considerations, in addition to those noted in the FSAR, are included in the evaluation. These considerations are an increase in the maximum fuel temperature and mechanical failure of the fuel rod cladding.

Fuel densification increases the beginning of life fuel pellet/cladding gap which in turn leads to an increase in fuel temperature. This temperature increase is less than 200°F. The maximum fuel central temperature at 118% overpower remains less than 4700°F while the minimum UO_2 melting point is about 4800°F (45,000 MWD/MTU burnup). Unirradiated UO_2 melting point is about 5100°F.

The moderator temperature coefficient is important in most of the accidents discussed below. The FSAR uses +3 pcm/°F as the most positive and -35 pcm/°F as the most negative moderator temperature coefficients. The corresponding coefficients for cycle 2 are -3 pcm/°F as the most positive and -32 pcm/°F as the most negative. Cycle 2 values are, therefore, less severe than the FSAR analysis.

Uncontrolled RCCA withdrawal from a subcritical condition is less severe in cycle 2 than the accident analyzed in the FSAR. This accident is terminated by the 25% power (nuclear) trip which is lower than the FSAR setting due to 100% power now being 1266 MWt. The maximum cycle 2 one bank reactivity insertion rate is $1.4 \times 10^{-4} \Delta k/\text{sec}$. compared with the maximum FSAR analyzed rate of $8 \times 10^{-4} \Delta k/\text{sec}$. Doppler feedback for cycle 2 is equal to that analyzed in the FSAR even without considering the higher fuel temperatures possible due to the fuel densification. The mechanical integrity of the fuel is not compromised during this transient.

Uncontrolled RCCA withdrawal at power, excessive heat removal due to feedwater temperature decrease, excessive load increase, and chemical and volume control system (CVCS) malfunction are no more severe than shown in the FSAR. All of these incidents lead to power excursions which are terminated at a power level equal to or less than the overpower (nuclear) trip. The maximum power level achievable during any of these transients is 1494 MWt (118% of 1266 MWt) since the overpower (nuclear) trip will be set at 1367 MWt (108% of 1266 MWt). Excessive heat removal due to feedwater decrease, excessive load increase, and the CVCS malfunction increase power by adding reactivity but at a slower rate than the maximum RCCA withdrawal rate analyzed. Uncontrolled RCCA withdrawal at power is analyzed at reactivity insertion rates up to $8 \times 10^{-4} \Delta k/\text{sec}$. while the maximum insertion rate by one bank during cycle 2 is about $1.4 \times 10^{-4} \Delta k/\text{sec}$. As previously noted, the cycle 2 moderator temperature coefficients are less severe than used in the FSAR analysis. DNBR, based on the previous DNBR evaluation, is no lower than predicted by the FSAR. The mechanical integrity of the fuel rod cladding is not compromised during any of these transients unless cracks already exist at the collapse locations.

The control misalignment remains less severe than discussed in the FSAR from a DNBR standpoint. Factors which make this incident much less severe are the control rod insertion limits given in Figure 1 and the administrative control which says part-length control rods will not be in the core when the reactor is at power.

Drop of a single RCCA, either full-length or part-length, is incapable of increasing the hot channel factors to a point at which DNB occurs. This remains unchanged from previous analysis. The automatic turbine runback and power reduction to 75% power is more than sufficient to maintain DNBR above 1.3 even for the drop of a full RCCA bank. The automatic turbine runback is discussed in section 14.1.4 of the FSAR. The cladding mechanical integrity of uncracked collapsed rods is maintained for all cases. Fuel rods with cracked collapsed sections may be further damaged for a single full-length control drop if the reactor returns to full power and for the drop of a full-length bank even with turbine runback. The latter case damage is due to the particular power profile followed in the turbine runback. Power initially drops and then returns to about 90% after the dropped bank is inserted and the high hot channel factors are established before finally settling to 75% power.

Startup of an inactive reactor coolant loop, loss of all A.C. power to the station auxiliaries, and steam generator tube rupture remain unchanged from the FSAR analysis. These transients are not dependent on new safety system trip points or, per se, the full power operating level. Startup of an inactive coolant loop, which does not trip the reactor, is dependent on the feedback mechanisms. The FSAR, as already noted, uses conservative values for feedback relative to cycle 2 feedback. The dependence of loss of A.C. power on reactor power level is in the amount of recirculation flow. The cycle 2 situation relative to the FSAR analysis is, effectively, loss of all A.C. power with the reactor at partial power. Steam generator tube rupture is dependent on coolant activity and the accident consequences remain unchanged since the Technical Specification limit on coolant activity remains unchanged. Mechanical integrity of the cladding is not a concern for these incidents since power never exceeds 100%.

The FSAR analysis remains applicable for loss of external electrical load and loss of normal feedwater. More conservative moderator coefficients were used in the FSAR analysis than are predicted for cycle 2. The lower operating power level also makes the accidents less severe than the FSAR analysis. Neither incident results in cladding mechanical damage even though a slight ($\sim 1\%$) power increase is possible for loss of external electrical load.

The fuel handling accident consequences remain unchanged from those presented in WCAP-7518-L.

The FSAR analysis of the steam pipe rupture is applicable to cycle 2. Cycle 2, with an EOL shutdown margin of 2.6%, is less severe, from a DNBR standpoint, than the FSAR analysis (1.9% shutdown margin) even considering a 20% increase in the hot channel factor due to collapsed rods. Cycle 2 peak power for the main steam line rupture is about 24% compared with the FSAR value of about 49%. The gross disparity in average heat flux more than compensates for the increase in the cycle 2 hot channel factor due to collapsed rods. The mechanical integrity of the peak power rod is marginal should that rod contain a collapsed section. Further damage may occur to fuel rods with cracks at the collapse location should those fuel rods be located in the vicinity of the stuck control rod.

For the loss of flow accident, pellet shrinkage and resultant higher fuel temperatures will result in a slightly slower heat flux decrease following trip. The corresponding decrease, in DNB ratio, however, is expected to be quite small compared to the gain in DNBR margin due to the maximum power level being lowered from 1520 MWt to 1266 MWt.

Ejection of an RCCA is less severe for cycle 2 than the FSAR analysis. The control rod insertion limits in Figure 1 limit the potential ejected rod worth to much less than those analyzed in the FSAR. The limitations or possible control rod configuration also limit the hot-channel factor to much less than the FSAR values. The beginning of life hot full power comparison is:

	<u>FSAR</u>	<u>Cycle 2</u>
At full power:		
Ejected rod worth (%)	.36	.07
F_Q^N (including any collapsed rod penalty)	4.9	3.3
At zero power:		
Ejected rod worth (%)	.675	.44
F_Q^N (including any collapsed rod penalty)	8.0	5.7

The FSAR has already considered possible fuel rod cladding failure for the RCCA ejection accident.

The pellet densification affects the loss of coolant accident analysis in two particular areas:

1. The maximum linear power for a given average linear power becomes larger because of local flux peaking.
2. At the maximum linear power, the amount of energy stored in the fuel becomes larger as a result of deterioration of gap heat transfer.

For the Ginna plant, the average power will be reduced such that the maximum linear power does not exceed the value considered in the ECCS analysis performed previously. Hence the present analysis is done at the same peak linear power. In our experience, the reduction in total power has the effect of enhancing cooling during the calculated blowdown, mainly through reduced core quality. This effect is not considered here; the previous blowdown transient is conservatively retained.

The present gap conductance used in the core thermal model (LOCTA code) includes a margin for model and fabrication uncertainty in the pellet temperature calculation and a variation with linear power. Fuel rod distortion, based on FLECHT tests⁽¹⁾ does not significantly affect the LOCA results and, hence, is not considered in this analysis. This more conservative model yields an increase of about 175F° in the calculated peak clad temperature for the 3 ft.² break. In addition to this effect, the pellet shrinkage results in an increase in initial pellet temperature of about 270F°. The ECCS Interim Criteria limitations are satisfied since the clad temperature peak is 2210°F and the total core metal water reaction is well below 1%.

(1) ANS 16th Annual Meeting, Special Symposium, Los Angeles, Calif. June 1970. "Heat Transfer Behavior for Flow Blockage of a Fuel Rod", by J. O. Cermak, A. S. Kitzes, L. S. Tong

November 3, 1972

APPENDIX D

SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U. S. ATOMIC ENERGY COMMISSION

CONCERNING

OPERATION WITH FUEL DENSIFICATION

R. E. GINNA UNIT NO. I

POINT BEACH UNIT NO. I

H. B. ROBINSON UNIT NO. II

AND

PALISADES

NUCLEAR POWER PLANTS

DOCKET NOS. 50-244, 50-266, 50-261 & 50-255

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1.0 Introduction

The safety implications of the collapsed cladding discovered at the R. E. Ginna plant were evaluated for all operating reactors. The causes and effects of the shrinkage of fuel and the creep of cladding are under investigation by the Regulatory Staff and by fuel manufacturers.

Conservative restrictions were placed on the operation of power reactors based on the knowledge available at the time. It is expected that as better quantitative analyses of fuel rod performance become available, it will be possible to remove some of the restrictions.

The purpose of this report is to summarize the safety considerations involved and give the general basis for the restrictions which were placed on operating reactors.

2.0 Fuel Rod Behavior

Several physical changes take place in the dimensional configuration of the fuel assemblies during their core life. The critical dimensions that are affected are: fuel pellet length, fuel pellet diameter, cladding length, cladding diameter, and control rod guide thimble length.

During normal operation, as the fuel generates heat, thermal expansion of the fuel pellets, cladding, and guide thimble occurs. For a typical 5 kW/ft rod, the net effects of thermal expansion are: the diametrical gap from fuel pellet to cladding reduces by 2 mils, the height of the pellet stack increases 0.7 inch and the cladding height increases 0.2 inch, resulting in a plenum length decrease of 0.5 inch. In the case of assemblies with stainless steel control rod

during normal operation and anticipated transients include the effect of power peaking on the DNBR (departure from nucleate boiling ratio) and the added stresses in crimped fuel due to differential thermal expansion during normal startup and load swings.

Attention has been given also to the possible safety implications of the bowed fuel rods observed in some Ginna fuel assemblies. The phenomenon of bowing due to lack of end clearance for the fuel rods was found to be characteristic of certain groups of fuel assemblies in the Ginna reactor. All fuel assemblies with insufficient end clearance were removed from the Ginna reactor in October 1972 and this design deficiency does not occur in fuel assemblies in later plants.

3.2 Loss-of-Coolant Accident

Flux spikes caused by air gaps or water gaps cause local power peaking that results in higher fuel temperatures and stored energy during normal operation. This higher stored energy would result in higher cladding temperatures in the event of a loss-of-coolant accident. A power peaking factor of 1.2 has been applied to account conservatively for this effect in this initial evaluation for operating reactors.

Fuel densification results in a larger radial gap between the fuel pellet and the cladding, thereby reducing the heat conductance of the gap. This decreased gap conductance causes increased fuel

temperature and stored energy at any given linear power in the fuel rod and therefore results in higher cladding temperatures under accident conditions. In considering this effect in new fuel, it is assumed that the cladding has undergone no creep-down but that the fuel has shrunk to its maximum density. The radial shrinkage is taken to be $1/3$ of the volumetric shrinkage even though some evidence indicates that the shrinkage may be greater axially rather than being isotropic. In significantly exposed fuel, the creepdown of the cladding is accounted for by increasing the gap conductance.

Axial shrinkage causes an overall reduction in the height of the fuel stack and thus the height of the active core of the reactor. Although isotropic shrinkage would dictate that $1/3$ of the volumetric shrinkage should be taken in the axial direction, it is assumed that $1/2$ of the volumetric shrinkage is axial. Thus, the total assumed shrinkage will be greater than the actual shrinkage. The effect of axial shrinkage is to increase slightly the linear heat generation rate corresponding to any particular power level.

The methods of reanalysis of the LOCA are the same as those required by the Interim Policy Statement on ECCS. However, the cladding temperature criterion was made more stringent for fuel rods with crimped cladding. The criterion of 2300°F was applied

to fuel rods which were not expected to be crimped, that is, fuel rods which had been prepressurized. This internal pressure is expected to reduce the rate of creepdown of the cladding and avoid or at least delay the crimping.

To provide an appropriate margin for the degraded condition of the cladding material in the unpressurized fuel, assumed to be already crimped, a significantly lower peak clad temperature criterion of 1800°F was adopted to preclude significant oxidation and embrittlement of the cladding.

3.3 Overpower Transients

As shown in the safety analysis reports, the most severe overpower transients result in the reactor power reaching 120% of the thermal power rating of the plant. It is assumed that this transient occurs in a reactor core which contains crimped fuel rods. The result is that the fuel will expand approximately 20% more than during normal operation, thereby expanding into the crimped region of cladding. Additional stresses are thereby imposed on the cladding, and may cause leaks in the cladding.

In the analysis, it is assumed that all rods which are crimped will fail. The number of rods which are assumed to be crimped is based on the experience at the Ginna reactor.

The resultant 2-hour dose at the site boundary was calculated for 100 failed rods and for failure of 1% of all the rods in the core. The results are shown in Table I. It was found that the limiting condition was the iodine dose to the thyroid. In these calculations, it was assumed that: 10% of the iodines were released from the failed rods; 10% of the iodines which reached the secondary coolant would be released to the environment; the leakage from primary to secondary system is 1 gpm; and that the iodines would continue to be released to the environment for 2 hours after the transient. Calculations were performed based on accident meteorology and on annual average meteorology for each site.

It was recognized that as the fuel cladding barrier was jeopardized by crimping, the primary system barrier should be more closely guarded. Therefore, tighter limits were placed on allowable steam generator leakage. These tighter limits would give greater assurance of steam generator tube integrity. Also these tighter limits, with routine steam generator leakage, reduce the dose at the site boundary in the event of an overpower transient. The criterion established for these leakage limits was that the 2-hour dose at the site boundary, calculated as described, and using accident meteorology, should not exceed the yearly allowable dose for an unrestricted area, that is, 0.5 Rem whole body or 1.5 Rem to the thyroid.

3.4 Operational Transients

Local power peaks resulting from water gaps (crimped fuel) and air gaps (gaps between pellets but without cladding collapse) increase the local heat flux and thus decrease the DNBR at the hot spot. As determined by the W3 DNB correlation, the effect of local power peaks on DNB ratios is approximately half the magnitude of the power perturbation. This was found to be less limiting than the linear power restrictions imposed by LOCA considerations and therefore limits based on DNBR considerations have not appeared explicitly.

Limits have been placed on the rate of change of power. The purpose of these limits is to reduce the probability of causing crimped fuel rods to leak. This would come about if the crimped rods are at low power or hot standby conditions for a long time during which the crimped cladding would creep down, further reducing the space needed for axial differential expansion. A rapid return to power and thus rapid expansion of the fuel can cause rupture of the cladding, whereas a slow expansion allows plastic deformation to take place. The actual rates of change are based on recommendations from Westinghouse to the individual utilities. The safety significance of these limits is a reduction in the possible number of fuel failures during normal operation.

3.5 Bowed Fuel Rods

The bowing of fuel rods in the R. E. Ginna reactor was caused by a lack of sufficient clearance between the ends of the fuel rods and the fuel assembly nozzles. The end clearance is necessary to allow for radiation induced growth of the Zircaloy cladding. The clearances provided for the cold condition of the Ginna fuel assemblies in the first core was 0.435 inch in region 1, 0.686 inch in region 2 and 0.836 inch in region 3. Subsequent Westinghouse fuel assembly designs provided end clearance of at least one inch. The Ginna fuel assemblies with less than one inch clearance were removed from the reactor in October. This type of bowing is not expected to occur in the fuel assemblies of any other licensed pressurized water reactor.

Operation of the Ginna reactor with fuel assemblies having less than one inch clearance was permissible for the few months from June through October. Although the lack of end clearance caused bowing of the fuel rods in the cold condition, it is unlikely that there was bowing during reactor operation when the fuel rods were hot. This is so because the end clearance increases as the fuel assembly temperature increases. The stainless steel control rod thimbles, which also function as the tie rods between the top and bottom fuel assembly nozzles, expand during heat up more than Zircaloy fuel rods. This differential expansion provides 0.4 inch

more end clearance in the hot condition than in the cold condition. It was found that the bowing which was observed in the cold condition did not strain the Zircaloy cladding beyond the elastic limit and, therefore, when the added 0.4 inch of clearance occurred during heat up, the fuel rods would return to a straight condition. The added 0.4 inch of clearance was more than adequate to accommodate Zircaloy growth during the remaining few months of operation with those fuel assemblies.

However, the bowing in the cold condition did place additional stresses on the control rod thimbles and weldments and it was found that this bowing condition could occur during a loss-of-coolant accident. This condition was evaluated considering such effects as the Zircaloy growth during the months of operation, differential thermal expansion, and hydraulic forces during blowdown. It was concluded that there was a very low probability of failure of fuel assembly nozzle plate welds and that the integrity of the core was not jeopardized.

The Palisades Nuclear Reactor is operating with fuel assemblies having end clearance less than one inch. However, the fuel assemblies in this reactor do not have stainless steel control rod thimbles acting as tie rods, but instead have Zircaloy guide bars. These guide bars are expected to undergo the same zirconium growth phenomena as the Zircaloy cladding of the fuel rods and therefore

a loss of end clearance is not expected to occur. However, there is some uncertainty as to the rate of growth of these different parts of the fuel assembly and careful attention will be given, therefore, to the inspection of the Palisades fuel assemblies during the first refueling.

4.0 Interim Conditions for Operation

4.1 Objectives

In view of the safety considerations arising from operation with crimped fuel, operating limits were developed for each reactor. A limit was placed on linear power density (kW/ft) to prevent excessive cladding temperatures in case of a LOCA. A limit on steam generator leakage was imposed to prevent excessive radioactive release to the environment in case of an overpower transient. A limit was placed on the rate of change of power to reduce the possible number of fuel rod failures and thus reduce the release of radioactivity to the environment during normal operation. Application of these limits does not necessarily require a reduction in reactor power level but in most cases requires more careful management of the power distribution in the reactor core.

4.2 Linear Power Density Limits

Based on the results of recalculation of the LOCA and application of the 2300°F criterion for prepressurized fuel and 1800°F for unpressurized fuel, limits were placed on the allowable linear

power density for a number of reactors. The H. B. Robinson plant was limited to 13.2 kW/ft for unpressurized fuel and 15.2 kW/ft for prepressurized fuel. R. E. Ginna limits were set at 16.0 kW/ft for unpressurized fuel and 16.3 kW/ft for prepressurized fuel. Point Beach was limited to 16.0 kW/ft for unpressurized fuel. In the Palisades plant, authorized only 60% of full power, the power level and permissible power distributions are sufficiently limiting to preclude exceeding the criteria.

4.3 Steam Generator Leakage

Based on the calculation of the 2-hour dose at the site boundary resulting from rupture of all crimped fuel rods, limits were placed on steam generator leakage in operating nuclear power plants. The bottom row of Table 1 shows the calculated steam generator leakage corresponding to 1.5 Rem/2 hrs to the thyroid resulting from failure of all crimped rods. The estimate of the number of crimped rods is based on the Ginna experience and varies with the initial density of the fuel pellets and the anticipated exposure of the fuel. Limits compatible with Table 1 were placed on steam generator leakage at the following plants: H. B. Robinson, 0.3 gpm; Point Beach 1, 0.7 gpm; R. E. Ginna, 0.1 gpm; and Palisades, 0.3 gpm.

4.4 Power Rates of Change

In order to reduce the number of fuel failures during normal operation, limits on the rate of change of power were adopted by several operating organizations. These include R. E. Ginna, Point Beach 1, and H. B. Robinson. These are prudent measures taken to avoid a forced shutdown due to high levels of radioactivity in the primary system or possible excessive releases to the environment. Such measures were not considered worthwhile in the Palisades plant because there was reasonable assurance that fuel rod collapse would not occur during operation at 60% of full power. Adequate protection for the public is provided in the technical specifications of all nuclear plants by limiting primary system radioactivity and rates of release of radioactivity to the environment.

5.0 Conclusion

The problems exhibited by the behavior of the fuel rods in the Ginna reactor have brought to light a number of new safety considerations applicable to all power reactors. The Regulatory Staff will continue to pursue the complete resolution of all considerations. However, the evaluation performed to date provides reasonable assurance that the health and safety of the public will not be endangered by continued operation of power plants with unpressurized fuel rods under the restrictions imposed.

TABLE I

CONSEQUENCES OF FUEL FAILURES WITH
CONCURRENT UNISOLATED TWO HOUR
STEAM GENERATOR TUBE LEAK

	<u>H. B.</u> <u>ROBINSON</u>	<u>POINT</u> <u>BEACH-1</u>	<u>GINNA</u>	<u>PALISADES</u>
Power Level, Mwt	2200	1518	1266	1320 (60% power)
Primary Coolant Volume, ft ³	9343	6040	6297	10900
Total No. of Rods in Core	32028	21659	21659	43168
Accident X/Q, Sec/m ³	8.9x10 ⁻⁴	3.0x10 ⁻⁴	5.3x10 ⁻⁴	2.6x10 ⁻⁴
Annual Avg. X/Q, Sec/m ³	2.0x10 ⁻⁵	1.5x10 ⁻⁶	5x10 ⁻⁶	2.x10 ⁻⁶
Core Fraction Failed:				
a) 100 Rods	0.00312	0.00462	0.00462	0.00232
b) 1%	0.01	0.01	0.01	0.01
Steam Generator Tube Leak, gpm	1.0	1.0	1.0	1.0
Fraction of Primary Coolant Lost in two hours	0.00172	0.00266	0.00255	0.00147
Fraction of Iodines Released	0.1	0.1	0.1	0.1
Iodine DF in Secondary Coolant	0.1	0.1	0.1	0.1
Accident Met. Thyroid Dose, Rem				
Condition a) 100 Rods	2.5	1.4	1.9	0.28
Condition b) 1% Core	8.	3.	4.	1.2
Annual Average Met. Thyroid Dose, Rem				
Condition a) 100 Rods	0.056	0.0068	0.018	0.0022
Condition b) 1% Core	0.18	0.015	0.039	0.0095
Steam Generator Leakage, gpm Corresponding to 1.5 Rem Thyroid	0.28	0.75	0.18	0.25

REFERENCES

1. Letter and proprietary attachment dated June 12, 1972, from Rochester Gas and Electric Corporation (RG&E).
2. Letter and proprietary attachment dated June 20, 1972, from RG&E.
3. Letter and proprietary attachment dated June 22, 1972, from RG&E.
4. Letter dated June 23, 1972, from RG&E transmitting non-proprietary version of June 12 letter's attachment.
5. Letter and non-proprietary attachment dated June 30, 1972, from Westinghouse.
7. Letter and non-proprietary attachment dated July 28, 1972, from RG&E.
8. Letter and non-proprietary attachment dated August 21, 1972, from RG&E.

1. All gaps located in a given axial interval have their centers coincident with the midpoint of that interval;
2. Gaps located in a given axial interval do not effect the power peaking in an adjacent interval.

For each axial interval to be analyzed, a set of events is input to the DRAW computer code. An event is characterized by a probability of occurrence (e.g., $p = .0014$ that a fuel rod has a gap in axial interval 30 with a gap size in the range 1.5" to 2.0") and the resulting power spikes on neighboring fuel rods (e.g., a 2" gap results in a 6% power spike on a rod 1 pitch away, a 4% spike on a rod $\sqrt{2}$ pitches away, etc.). The DRAW code then uses random numbers to determine which events have occurred in a square array of fuel rods (typically 7x7) and superimposes the individual power spikes resulting in a total power spike on the center rod. For each axial interval DRAW recomputes a total power spike on the center rod 10,000 times to develop a good statistical sample. Based on this sample, a curve of power spike vs. probability of not exceeding power spike can then be generated.

Table 3.1 is an example of the input to DRAW. Notice that the power spike on the center rod due to a single event can be 1 of 6 values. If a particular event occurs in a rod 1 pitch away from the center rod, then the value in column A is the power spike on the center rod due to this event. If, however, the same event occurs in a rod $\sqrt{2}$ pitches away, then the value in column B is the power spike on the center rod. The positions corresponding to columns C, D and E are illustrated graphically in the drawing below Table 3.1. Column X accounts for the possibility that the center rod may also have a gap which contributes to the total power spike on the center rod. Details concerning the probability of occurrence values and power spikes due to a single even can be found in sections 3.1.2 and 3.1.3, respectively.

The results of the DRAW analysis are presented in Figures 3-1 and 3-2. Notice that the curves in Figure 3-1 correspond to fuel with an initial density equal to 94% of theoretical density while the curves in Figure 3-2 correspond to fuel with a 92% initial value. This density dependence is explained fully in section 3.1.2.

3.1.2 PROBABILITY OF OCCURRENCE

The probability that a fuel rod has a gap of a particular size at a particular axial location is calculated using the equation below:

$$P_{gjk} = F_g \times F_j \times F_k$$

where:

P_{gjk} = Probability that a fuel rod has a gap in axial interval j with a size in gap size interval k .

F_g = Fraction of fuel rods having a gap in the fuel stack.

F_j = Fraction of gaps occurring in axial interval j
($j = 1, 2, \dots, 36$).

F_k = Fraction of gaps having a size within gap size interval k
($k = 1, 2, \dots, 7$).

The fraction F_k and the index k deserve more attention. There is one and only one value of F_k for each k . The index k , however, does not uniquely define the range of gap sizes (in inches) within each interval. It only defines the range of gap sizes within each interval relative to the maximum gap size. For example, the interval $k = 1$ includes all gaps having a size between 0% and 14.3% of the maximum size while $k = 2$ includes those gaps with a size between 14.3% and 28.6% of the maximum, etc. The maximum gap size (in inches) is a function of axial location and initial density of the fuel.

3.1.2.1 Determining F_g

$F_g = 1.0$ has been assumed in the analysis. In the following discussion it is demonstrated that this value is conservative. Figure 3-3 displays a plot of flux blip frequency vs. axial interval. This data has been compiled from the latest WEP* and CPL** flux maps. Notice that the flux blip frequency drops off very rapidly in the neighborhood of a grid and near the top of the fuel. It is very difficult to detect flux blips at these positions since the flux gradients are so large. Thus it is concluded that the gap frequency vs. axial interval curve more closely resembles the straight line on the same figure. The steps below demonstrate how an estimate for F_g based on this data can be attained.

1. Integral under straight line = 440 gaps.
2. Assume each gap occurs on a different rod.
3. Measured instrumentation thimbles = 65.
4. Each thimble can detect gaps on at least the 8 surrounding fuel rods.
5. Monitored fuel rods = $8 \times 65 = 520$.
6. Estimate for $F_g = \frac{440}{520} = 0.85$.

3.1.2.2 Determining F_j

If the number of gaps expected in each axial interval (straight line curve in Figure 3-3) is divided by the total number of gaps (440), the parameter F_j as shown in Figure 3-4 is generated.

3.1.2.3 Determining F_k

Curve 1 in Figure 3-5 shows the frequency of collapse vs. collapse size at elevation 130 ± 10 " based on observations from region 2 of RGE. Curve

* Wisconsin Michigan Power Co/Wisconsin Elec. Power Co. - Point Beach No. 1.

** Carolina Power and Light Co. - H. B. Robinson No. 2.

2 represents the expected frequency of gaps vs. gap size at 130 ± 10 ". The distribution of collapse sizes represents the distribution of gap sizes for sizes > 1.75 " since gaps greater than this value have already collapsed at the time of the RGE observations.

The remaining portion of curve 2 is an extrapolation of the collapsed rod data and is based on the fact that small gaps had not yet collapsed. Support of curve 2 is demonstrated below.

1. Integral under curve 2 is 154 gaps while integral under curve 1 is 66 collapses.
2. There were 1144 region 2 fuel rods visually examined.
3. Using the random distribution curve in Figure 3-4 and assuming $F_g = 1.0$, we would expect that 13.9% of all the region 2 fuel rods examined or 159 fuel rods have gaps between 120" and 140".
4. Thus we conclude that the distribution of gaps more closely resembles curve 2 than curve 1.

To determine the F_k vs. k curve from curve 2 of Figure 3-5, simply divide the number of gaps in each interval by the integral (154) and divide the gap size scale by the maximum gap size (3.5"). The result is shown in Figure 3-6.

Probability of occurrence calculations are illustrated in Table 3-2 for axial intervals 30 and 25. The F_j values are taken from Figure 3-4 while the F_k values are taken from Figure 3-6.

The next step is to determine the range of gap sizes within each gap size interval for a given axial interval. To do this it is necessary to know the maximum gap size as a function of fuel rod height. The empirical equation used in evaluating maximum gap sizes is shown below and is illustrated graphically in Figure 3-7.

$$\Delta L = \left(\frac{.965-f}{2} + .004 \right) L$$

where:

- ΔL = maximum gap size
- L = axial position
- f = initial fraction of theoretical density.

The first term represents the maximum gap size formation due to pellet densification. Here it is assumed that all fuel in the cold shoulder densifies to .965 geometric (.970 true) and that half of the volumetric change is in the axial direction. The second term is the zircaloy growth contribution to gap sizes. A pellet which hangs up will move up with zircaloy growth and, thereby increase the size of the gap. The $.004 \frac{\Delta L}{L}$ value corresponds to an end of cycle 2 growth,

Support of this empirical equation is shown in Figure 3-8 and 3-9. These curves represent the maximum collapse size vs. fuel rod height for regions 1 and 2 of RGE respectively. In comparing our empirical equation with the data, the actual growth values for regions 1 and 2 was used.

The range of gap sizes (in inches) included in each gap size interval can now be evaluated. Using Figure 3-7, first determine the maximum gap size associated with a particular axial interval (e.g., $\Delta L = 1.95''$ at $j = 30$ and $f = .94$). Then simply multiply the normalized gap size scale in Figure 3-6 by this value to determine the range of gap sizes for each k . An example of this calculation is illustrated in Table 3.3 at $j = 30$ ($L = 118''$) and $j = 25$ ($L = 98''$) for $f = .94$.

3.1.3 POWER SPIKES DUE TO A SINGLE EVENT (GAP)

Figure 3-10 presents a plot of power spike (%) in the fuel rod which is one fuel rod pitch away from the gap vs. gap size. The drawing below the plot represents the attenuation of the power spike on the center rod as the x-y distance from the gap to the center rod increases. For example,

the power spike on the center rod is 6% due to a 1.8" gap on a rod 1 pitch away while the spike on the center rod is 0.6x6% or 3.6% due to the same size gap on a rod $\sqrt{2}$ pitches away, etc. The plot of power spike vs. gap size has been generated using the DOT transport code in R-Z geometry while the x-y attenuation factors are the values calculated at BNL.

The information in Figure 3-10 is used in the following manner. After a range of gap sizes has been associated with each gap size interval (see Table 3.3), a power spike value corresponding to the maximum gap size in each interval k is to be determined from the curve in Figure 3-10. The attenuation factors are then applied to each of these power spikes as illustrated in Table 3.4.

The possibility of a power spike on the center rod due to the presence of a gap on the center rod must also be considered. Figure 3-11 shows how the power spike on the fuel directly above and below the gap varies as a function of axial distance from the center of the gap. Although the edge of the fuel adjacent to the gap may experience a 12% power spike, the effective power spike is only 3% taking advantage of axial heat transfer. A 3% power spike to be independent of gap size has been assumed. The effective power spike is derived in Section 4.6.

3.1.4 INCORPORATION OF THE POWER SPIKE INTO F_Q^N

It remains to select a power spike penalty. For simplicity in the discussion it is assumed that all fuel in the core has the same nominal density. The criterion for determining the power spike penalty is that less than 1 rod will exceed F_Q^N at a 95% confidence level where

$$F_Q^N = F_{xy}^N(Z_0) \cdot F_Z^N(Z_0) \cdot S(Z_0) \cdot F_U \text{ and}$$

$$F_{xy}^N(Z) = \frac{\text{peak power density in horizontal plane at elevation } Z}{\text{avg. power density in horizontal plane at elevation } Z}$$

$$F_Z^N(Z) = \frac{\text{avg. power density in horizontal plane at elevation } Z}{\text{core average power density}}$$

$S(Z)$ = Power spike factor at elevation Z resulting in this 95% confidence level.

F_U = Nuclear uncertainty factor.

Z_0 = Elevation which maximizes this product.

$S(Z)$ can be arrived at by convoluting the X-Y power census with the curves in Figure 3-1 and 3-2. The X-Y power census used in the convolution is presented in Table 3.5, and an example of the convolution procedures is presented in Table 3.6. Columns 1 and 2 of Table 3.6 repeat the X-Y power census. The values in column 3 are power spike factors each of which results in the product 1.435×1.14 when multiplied by the corresponding relative power density (column 1). Each value in column 4 is the probability of exceeding the corresponding power spike factor and has been taken from the $J = 30$ curve in Figure 3-2. Finally, column 5 represents the expected number of fuel rods exceeding 1.435×1.14 in each power density range. The sum of these numbers is the expected number of fuel rods exceeding 1.435×1.14 in the entire core and is equal to 0.2153.

Notice from Table 3.6 that the 1.14 power spike factor corresponds to a 0.2% probability of exceeding the power spike in interval 30 (118'). If the location of peak heat flux is in axial interval 30, then F_Q^{II} would be calculated as

$$F_Q^N = 1.435 \times F_Z^N \times 1.14 \times F_U$$

for $f = .92$ and less than 1 rod would exceed this value as demonstrated in Table 3.6. To demonstrate a 95% confidence level, we must limit the expected number of rods exceeding F_Q^N to the value which satisfies the equality $1 = \overline{pn} + 1.645 \sqrt{\overline{pn}}$ or $\overline{pn} = 0.223$

The result of the convolution is that $\overline{pn}$ is less than 0.223 when applying a power spike factor corresponding to a 99.8% probability level F_{xy}^N . Thus

it is conclude that less than 1 rod will exceed $1\frac{N}{Q}$ at a 95% confidence level. Curves of power spike factor $S(Z)$ vs. elevation are presented in Figure 3-12 and have been generated by performing this convolution at specific axial intervals.

3.2 EXPERIMENTAL VERIFICATION OF THE POWER SPIKE MODEL

Figures 3-13 and 3-14 present a comparison of the flux blip data and the power spike curves calculated with the model. Only the largest flux blips have been plotted. (Actually there are hundreds of flux blips in the 1%-2% range). These figures clearly demonstrate the conservatism in the power spike model.

3.3 APPLICATION OF "FLUX SPIKE" PENALTY TO F_Q

Since the magnitude of the flux spike penalty varies as a function of height in the core it has to be applied to the basic power shape data which determines the design value of F_Q . The basis for protection against exceeding local power density limits is given in topical reports⁽¹⁾ which describe the use of excor detector signals (axial offset) in the over-power protection logic.

Briefly an upper bound on F_Q is set as a function of axial offset by consideration of all allowable operating situations. When F_Q is increased locally by the height dependent flux spike penalty, the individual points are increased in F_Q by different amounts.

The result is a revised plot of F_Q vs. axial offset which requires a revised upper bound different in shape and magnitude from the previous upper bound. This is illustrated by Figure 3-15 which gives a revised upper bound (with flux penalties for 92% dense fuel) and the previous upper bound. It should be noted that this example used data for a reload - Cycle 2 core.

3.4 REFERENCES

- 1.0 McFarlane, A.-F., "Power Peaking Factors", WCAP-7912 (1972).

TABLE 3.1

EXAMPLE INPUT TO DRAW

Initial Fraction of Theoretical Density = 0.94

Axial Interval = 30 (118 ± 2)

EVENT (GAP SIZE INTERVAL)	PROBABILITY OF OCCURRENCE	POWER SPIKES (%)					
		A	B	C	D	E	X
0-.28"	.01270	1.3	0.8	0.4	0.3	0.2	3.0
.28-.56	.01050	2.4	1.4	0.7	0.5	0.3	3.0
.56-.84	.00844	3.5	2.1	1.1	0.7	0.5	3.0
.84-1.12	.00603	4.4	2.6	1.3	0.9	0.6	3.0
1.12-1.40	.00483	5.2	3.1	1.6	1.0	0.7	3.0
1.40-1.68	.00255	5.8	3.5	1.7	1.2	0.8	3.0
1.68-1.95	.00135	6.2	3.7	1.9	1.2	0.8	3.0

Axial Interval = 25 (98 ± 2)

0-.23	.01050	1.1	0.7	0.3	0.2	0.1	3.0
.23-.46	.00874	2.0	1.2	0.6	0.4	0.3	3.0
.46-.69	.00701	2.9	1.7	0.9	0.6	0.4	3.0
.69-.92	.00501	3.7	2.2	1.1	0.7	0.5	3.0
.92-1.15	.00400	4.5	2.7	1.4	0.9	0.6	3.0
1.15-1.38	.00212	5.1	3.1	1.5	1.0	0.7	3.0
1.38-1.62	.00112	5.6	3.4	1.7	1.1	0.7	3.0

TABLE 3,2

PROBABILITY OF OCCURRENCE VALUES FOR AXIAL INTERVAL 30

k	F_g	F_{30}	F_k	P_{g30k}
1	1.0	.0464	.273	.01270
2	↓	↓	.227	.01050
3	↓	↓	.182	.00844
4	↓	↓	.130	.00603
5	↓	↓	.104	.00483
6	↓	↓	.055	.00255
7	↓	↓	.029	.00135

PROBABILITY OF OCCURRENCE VALUES FOR AXIAL INTERVAL 25

k	F_g	F_{25}	F_k	P_{g25k}
1	1.0	.0386	.273	.01050
2	↓	↓	.227	.00874
3	↓	↓	.182	.00701
4	↓	↓	.130	.00501
5	↓	↓	.104	.00400
6	↓	↓	.055	.00212
7	↓	↓	.029	.00112

TABLE 3.3

RANGE OF GAP SIZES FOR EACH k

$j = 30$ ($L = 118"$), $f = .94$, $\Delta L = 1.95$

k	Normalized Gap Size	Gap Size (Inches)
1	0-.143	0-.28
2	.143-.286	.28-.56
3	.286-.429	.56-.84
4	.429-.572	.84-1.12
5	.572-.715	1.12-1.40
6	.715-.858	1.40-1.68
7	.858-1.000	1.68-1.95

$j = 25$ ($L = 98"$), $f = .94$, $\Delta L = 1.62$

K	NORMALIZED GAP SIZE	GAP SIZE (INCHES)
1	0-.143	0-.23
2	.143-.286	.23-.46
3	.286-.429	.46-.69
4	.429-.572	.69-.92
5	.572-.715	.92-1.15
6	.715-.858	1.15-1.38
7	.858-1.000	1.38-1.62

TABLE 3.4

CALCULATION OF POWER SPIKES DUE TO A
SINGLE EVENT GAP

$j = 30, f = 0.94$

k	Gap Size (Inches)	A	B (.6xA)	C (.3xA)	D (.2xA)	E (.13xA)
1	0-.28	1.3	0.8	0.4	0.3	0.2
2	.28-.56	2.4	1.4	0.7	0.5	0.3
3	.56-.84	3.5	2.1	1.1	0.7	0.5
4	.84-1.12	4.4	2.6	1.3	0.9	0.6
5	1.12-1.40	5.2	3.1	1.6	1.0	0.7
6	1.40-1.68	5.8	3.5	1.7	1.2	0.8
7	1.68-1.95	6.2	3.7	1.9	1.2	0.8

TABLE 3.5
X-Y POWER CENSUS

<u>% OF CORE</u>	<u>NUMBER OF RODS*</u>	<u>RELATIVE POWER DENSITY IN X-Y PLANE</u>
.135	53	1.417 - 1.435
.065	26	1.405 - 1.417
.100	39	1.398 - 1.405
.100	39	1.396 - 1.398
.100	39	1.390 - 1.396
.100	39	1.386 - 1.390
.100	39	1.382 - 1.386
.100	39	1.379 - 1.382
.100	39	1.376 - 1.379
.100	39	1.374 - 1.376

* Calculated for a four loop reactor.

TABLE 3.6

EXAMPLE OF CONVOLUTION

$j = 30, f = .92$

(1) RELATIVE POWER DENSITY IN X-Y PLANE	(2) n	(3) FACTOR NEEDED TO EQUAL 1.435×1.14	(4) p	(5) pn
1.417 - 1.435	53	1.140	$2.0E-3$	.1060
1.405 - 1.417	26	1.155	$9.0E-4$	.0234
1.398 - 1.405	39	1.164	$5.0E-4$	.0195
1.396 - 1.398	39	1.170	$4.0E-4$	.0156
1.390 - 1.396	39	1.172	$4.0E-4$	.0156
1.386 - 1.390	39	1.177	$3.0E-4$	.0117
1.382 - 1.386	39	1.180	$2.0E-4$	.0078
1.379 - 1.382	39	1.184	$1.5E-4$	.0059
1.376 - 1.379	39	1.186	$1.5E-4$	.0059
1.374 - 1.376	39	1.189	$1.0E-4$	.0039
				$\overline{pn} = .2153$

n = number of fuel rods in each range.

p = probability of exceeding power spike factor.

$\overline{pn}$ = expected number of rods exceeding 1.435×1.14 .

POWER SPIKE (%) VS. PROBABILITY OF NOT EXCEEDING POWER SPIKE

Initial Fraction of Theoretical Density = .94

POWER SPIKE (%) VS. PROBABILITY OF NOT EXCEEDING POWER SPIKE

Initial Fraction of Theoretical Density = .92

FLUX BLIPS VS. AXIAL INTERVAL

FIGURE 3-3

FIGURE 3-4

F_k vs. GAP SIZE INTERVAL

COLLAPSE SIZE VS. FUEL ROD HEIGHT
REGION 1, R.E. GINNA UNIT I,
CYCLE 1B

3-24

FIGURE 3-8

COLLAPSE SIZE VS. FUEL ROD HEIGHT
REGION 2, R.E. GINNA UNIT 1,
CYCLE 1B

3-25

FIGURE 3-9

POWER SPIKES DUE TO A SINGLE EVENT

POWER CHANGE AS A FUNCTION OF AXIAL DISTANCE
FROM THE GAP CENTER FOR THE ROD WITH THE GAP

3-27

FIGURE 3-11

POWER SPIKE FACTOR VS. FUEL STACK HEIGHT

Curves used in the calculation of F_Q^N
such that less than 1 rod exceeds
 F_Q^N at a 95% confidence level.

3-28

FIGURE 3-12

BLIP SIZE (%) vs. FUEL ROD HEIGHT

FIGURE 3-13

FIGURE 3-14

F_Q^N vs. AXIAL OFFSET

Typical Results For
Demonstration Only

FIGURE 3-15