ADDITIONAL TESTIMONY

ON

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

IN REGARD TO

FUEL DENSIFICATION AND ITS EFFECTS

Prepared by the

Directorate of Licensing

USAEC

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1.0 INTRODUCTION

1.1 General

The Consolidated Edison Company of New York, Inc. (the applicant) applied for an operating license for Indian Point Unit 2 by application dated October 15, 1968. The Atomic Energy Commission's Regulatory staff (the staff) subsequently completed its review and issued a Safety Evaluation on November 16, 1970.

On October 6, 1970, Consolidated Edison requested a public hearing on its application for a license to operate the facility at full power. Pursuant to a Commission Order, a public hearing before an Atomic Safety and Licensing Board (Board) commenced on December 17, 1970 in Buchanan, New York. This hearing is still in progress.

On June 18, 1971, Consolidated Edison requested the Board, pursuant to 10 CFR Section 50.57 of the Commission's regulations, to issue an order authorizing the Director of Regulation to make the necessary findings and issue a license permitting fuel loading and subcritical testing. The Board issued such an order on July 20, 1971. Following further consideration of the environmental impact of such a license the Board issued an order on October 15, 1971 affirming the authorization granted in its July 20, 1971 order.

Subsequently, the Atomic Energy Commission issued Facility Operating License No. DPR-26 on October 19, 1971, which authorized the applicant to load fuel in and perform subcritical tests on Indian Point Unit 2. Following issuance of License No. DPR-26, the original unpressurized fuel was removed for modification of the design to alleviate the effects associated with fuel densification. The modified fuel has been reloaded into the Unit 2 reactor as authorized by Change No. 2 to the Technical Specifications for License No. DPR-26.

On November 14, 1972, the Regulatory staff issued a report entitled "Technical Report on Densification of Light Water Reactor Fuels," (the Densification Report, Reference 1.1) and on November 20, 1972, the Regulatory staff requested that the applicant provide analyses and relevant bases for determining the effects of fuel densification on normal operation, transients and accidents for the Indian Point Unit No. 2 facility. On January 18, 1973, the applicant filed a response to that request (Reference 1.2).

Prior to initiating the staff review of the Indian Point Unit 2 fuel densification report, the staff completed a detailed review of fuel densification effects in connection with Point Beach Unit 2 which also has a Westinghouse reactor and nuclear steam supply system. During the Point Beach 2 review additional information was requested regarding fuel element and core performance analysis method used by Westinghouse in their analysis of the effects of fuel densification. This information was provided (References 1.3

and 1.4). As a consequence of the Point Beach 2 review additional testimony was prepared by the staff for the Point Beach 2 public hearing (Reference 1.5). In preparing the testimony for Indian Point 2 some references have been made to the Point Beach 2 testimony, mostly in connection with the staff's independent review of methods used by the applicant in its analyses. However, in order to make this testimony complete and more readily comprehensible, major portions of the staff testimony for Point Beach 2 have been repeated verbatim as deemed applicable.

In subsequent sections of this testimony, our technical review of fuel densification as it applies to Indian Point Unit 2, and our technical evaluation of the applicant's safety analyses of steady state operation, operating transients, and postulated accidents taking into account the effects of densification, are presented.

The Regulatory staff has concluded that operation of Indian Point Unit 2 as now proposed at power levels up to 2758 MWt (100% of rated power) will not present an undue risk to the health and safety of the public.

1.2 Scope of Review

The applicant stated in Reference 1.2 that the determination of reactor power capability was based on the premise that the fuel cladding in the regions of the fuel column gaps will not flatten during operation.

The staff concludes that no collapse or flattening is an appropriate assumption for Indian Point Unit 2 and used it as a basis for our evaluation. A detailed evaluation was conducted, and is described in the following chapters where calculations, made both by the applicant and the Regulatory staff are discussed.

The essential elements that must be considered in evaluating the effects of densification have been set forth in our Densification Report of November 14, 1972. The effects of fuel densification on the course of postulated plant transients and accidents were reviewed by the applicant and evaluated by the staff. Since the performance of the facility in steady state operation, and during various postulated transients and accidents had been established previously and reported in the application (FFDSAR) it was only necessary to evaluate those changes in analyses or results from the effects of densification.

1.3 References

1.1 "Technical Report on Densification of Light Water Fuels" by USAEC Regulatory Staff; Nov. 14, 1972.

1.2 "Fuel Densification - Indian Point Nuclear Generating Station Unit No. 2" (proprietary and non-proprietary versions) January 18, 1973.

1.3 Attachments A-P and 1-5 to "Fuel Densification - Point Beach Nuclear Plant Unit 2" (proprietary and non-proprietary versions) January 3, 1973 - filed January 4, 1973.

1.4 Letter and Attachments 1-15 from R. Salvatori of Westinghouse to D. F. Knuth, USAEC, January 12, 1973; filed (proprietary and non-proprietary versions) January 15, 1973.

1.5 "Additional Testimony on Point Beach - 2 Nuclear Plant in Regard to Fuel Densification and its Effects," prepared by the Directorate of Licensing USAEC, February 2, 1973.

2.0 DISCUSSION OF FUEL DENSIFICATION

2.1 General Discussion

2.1.1 Introduction

The cause of the observed fuel densification is believed to be a different phenomenon from the thermally-induced densification of fuel which occurs during fuel irradiation 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 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 in the recently observed fuel densification occurred in a relatively low temperature region (400-1000°C) in the fuel. In typical operating light water reactors almost all of the fuel operates in this temperature range. Thus most of the fuel which densified operated at temperatures where little or no thermal restructuring occurred and where thermally-activated processes were so slow as to be insigificant. Examinations of metallographic cross-sections of irradiated fuel and immersion density measurements of irradiated fuel operated at these low temperatures confirm that the pellet densification occurs by annihilation of pores.

2.1.2 Kinetics of Densification

Examinations of density changes in irradiated fuel by the Westinghouse Electric Corporation have shown that, for exposure times of less than 14 hours of power operation, no temperature-independent densification has occurred, but that after 2000 hours of reactor operation fuel densification has probably been completed. This timing is also supported by observations of in-core neutron flux distribution measurements in which indications of local power spikes appear within the first few hundred hours. If it is assumed that pore annihilation occurs by enhanced sintering or by a re-solution process, the kinetics of densification can be examined to determine qualitatively whether the observed densification times are compatible with theoretical analyses. 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 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. The staff has estimated the in-reactor diffusion coefficient for uranium to be 2.5 x 10^{-17} cm^2/sec at a fission rate of 1.8 x 10^{12} fissions/cm^3 sec. The fission rates of interest for power reactors range from 3 to 5 x 10^{13} fissions/cm^3-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 UO_2 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. For purposes of maintaining perspective, it should be noted that the evaluation model specified by the staff in the Densification Report requires the assumption of instantaneous densification.

2.1.3 Effects of Powder, Pellet, and Fabrication History

The as-fabricated properties of UO_2 pellets in reactor fuel assemblies are dependent on the many variables which exist in the complete conversion cycle for the UO_2 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.

An examination of the sintering times and temperatures for the production of UO_2 pellets produced by Westinghouse has been made. In analyzing the process data, the staff considered how the manufacturing techniques could affect densification. Westinghouse presented data regarding the pellet process parameters considered important in controlling the amount and rate of fuel densification. The staff has examined the proprietary data that form a basis for the Westinghouse conclusions that densification is correlated to process parameter and agrees that the conclusions seem to be consistent with the evidence. However, it is not clear that the pellet process parameter alleged to control densification is not also subtly dependent upon some of the other factors of the UO_2 conversion cycle. Until more definite irradiation experience is obtained from irradiation tests, the staff has concluded that the assumption that all light water reactor oxide fuels will densify to an extent consistent with present observations is required.

2.1.4 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 3.1.4a of Reference 1.1. 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 3.1.4b of Reference 1.1. 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 previously, the densification process is nearly complete in 2000 hours, whereas some of the specimens in Figure 3.1.4b of Reference 1.1 range up to 30,000 hours exposure. This is further borne out by Figure 3.1.4c of Reference 1.1 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, the staff concluded from the proposed mechanisms 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 3.1.4d of Reference 1.1 illustrates the relationship between fuel densification and internal pressure. No apparent influence of an isostatic hot pressing effect is observed.

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 3.1.4f of Reference 1.1. 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 caused by the 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 should be assumed that axial shrinkage is greater than radial shrinkage.

2.1.5 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 local power spikes.

(c) A decrease in the pellet radius increases the radial clearance 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 have been evaluated for the Indian Point Unit 2 reactor under all modes of reactor operation.

2.2 Densification Calculations by Applicant

This report is concerned with two "creep" calculations: cladding creepdown and time-to-collapse. Although similar in terminology and concept they differ both in calculational methodology and in their effect on the behavior of the fuel. Cladding creepdown is the term used to indicate the phenomenon which affects the geometry of the gap between the fuel pellets and the cladding. Time-to-collapse is the term used to indicate the calculations to determine the time required for an unsupported clad tubing to become dimensionally unstable and flatten into the axial gap volume caused by the fuel pellet densification. In this section (2.2.1) we consider the former, and in sections 2.2.3 and 2.3.1 we consider the latter.

2.2.1 Cladding Creepdown Calculation

A free-standing cylindrical tube when subjected to elevated temperature and a net external pressure undergoes a uniform change in diameter due to creep. Normal production tubes have an ovality tolerance and under high external pressure deform in an oval mode forming major and minor axes. The minor axis will touch the fuel first and, from then on, only the major axis will undergo creep. This overall behavior is what is known as clad creepdown.

The applicant's calculations for cladding creepdown rate were performed in the following sequence. With a given pressure differential across the tube, stress components were obtained based on shell theory.

Effective stress was obtained according to the Von Mises criterion. The strain, and attendant diameter changes, were calculated using empirical creep data.

The creep strain consists of three components; thermal, irradiation growth and fission-induced creep. Thermal creep constants were obtained from the out-of-pile tube tests subjected to internal pressure. Irradiation growth and fission induced creep were obtained from the measured diameter changes of about fifty rods from SAXTON, ZORITA and SAN ONOFRE. The creep relationship thus obtained was fitted to the inpile data on clad creepdown.

The correlation between measured and calculated diameter changes for irradiation growth and fission-induced creep, described in Reference 2.1 was modified by use of the lower limit line which essentially bounded the data; this limit line was used in the creep formula to minimize the creep rate and the consequent fuel pellet-clad gap closure rate.

The applicant's creep model was indexed to match the measurements of fuel rods which had been subjected to reactor operating conditions. These fuel rods had physical characteristics and environmental conditions similar to that expected for the Indian Point Unit 2. On this basis we conclude that the cladding creepdown calculation method utilized for Indian Point Unit 2 is acceptable.

2.2.2 Time to Collapse

Accident analysis studies by both staff and applicant for Indian Point Unit 2 were based on the assumption of no collapsed rods. We reviewed the model used by the applicant to calculate the cladding collapse time in support of the no-collapse assumption. The data on which the model is based were for cladding which is similar to that used in Indian Point Unit 2. On this basis the prediction of the collapse time using the applicant's model is considered by the staff to be reliable. The analytical method for predicting the collapse time is described in Reference 2.2.

A brief description of the analysis is as follows. An initial out-of-roundness of the tube introduces a bending moment as well as membrane compression stress. Time-dependent creep deformation further magnifies ovality which in turn increases the bending moment in the cladding, causing an accelerating creep rate until collapse occurs. This buckling coincides with the loss of equilibrium between the applied moment due to net external pressure and the internal resisting moment. The collapse time is taken as the time when the time derivative of the displacement in ovality approaches infinity. This behavior is calculated using incremental steps of the creep strain throughout time history, assuming a constant stress within time steps. The size of the time steps is controlled internally in the computer code by means of a percentage increase in stress from one time step to the next.

Wilson's creep law (Reference 2.2) was not used in the calculation. Instead the creep law used in the calculation was based on the model discussed above in Section 2.2.1 Further adjustments were made to the Section 2.2.1 creep law to correlate with the observed data from GINNA, NOK, ZORITA, POINT BEACH UNIT 1, and MIHAMA. This was done by modification of the creep law constants. In most cases the predicted collapse time was shorter than actual observed time to collapse.

2.2.3 Gap Conductance

The resistance to heat flow from the fuel pellet to the cladding is called gap conductance. The effect of densification is to increase the radial gap, thus decreasing the gap conductance and increasing the fuel pellet stored energy. The staff has established guidelines (Reference 1.1) for calculating the gap conductance used in analyzing the behavior of the fuel for all modes of reactor operation. The staff guidelines require the assumption of instantaneous densificiation of the pellet as given by the following formula:

$$\Delta r = \frac{(0.965 - \rho i + 2\sigma)r}{3}$$

where:

Δr = radial gap increase

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

ρi = nominal initial fraction of theoretical density.

r = nominal initial pellet radius

The staff further requires that proposed models for calculating the effect of cladding creep on fuel thermal performance be submitted for staff evaluation along with parametric studies of gap conductance as a function of burnup and linear heat rate. Such information was submitted in connection with our review of Point Beach Unit 2 (Reference 1.5). The staff evaluation of this material included making similar calculations using various versions of the GAPCON computer code (Reference 2.6) to compute gap conductances and temperature profiles (Section 2.3)

Although gap conductance is an important factor in establishing fuel pin stored energy, other factors (including flux depression factors, fuel conductivity, and surface effects) must also be considered. Westinghouse utilizes average fuel pellet temperature as the index for stored energy.

In order to establish stored energy for accident calculations, an increment of temperature is added to the average pellet temperature calculated by the Westinghouse thermal design model (Reference 1.3). This increment is equal to:

$$\Delta T = 9.375 P$$

where:

ΔT = pellet temperature increment (°F)

P = linear heat rate (kw/ft)

Westinghouse now employs a UO_2 conductivity whose integral from 0° to 2800°C is 93 watts/cm as compared with a value of 97 watts/cm used previously (Reference 1.3). This change has the effect of increasing the amount of stored energy calculated.

Gap conductance is a function of gap size, the amount of the gas (pressure, and chemical composition) in the gap, the amount of fuel densification, the surface roughness of the fuel and clad and their respective material properties and, in the case of fuel-clad contact, the contact pressure. One of the most sensitive of these is the diametral gap size. In the case of Indian Point Unit 2, the initial cold diametral gap size of 6.5 mils for region 1 and 7.5 mils for region 2 and 3 prior to densification is reduced by fuel thermal expansion, plant system pressure which elastically loads the clad thus reducing its diameter, fuel swelling, and creepdown of the cladding. The first two of the above factors occur when the reactor is brought to power, while the latter two are time and flux dependent. As explained previously the gap size is increased by the effects of densification. The gas composition also varies with time due to: release of sorbed gas at power, some solution of the helium fill gas in the fuel, and the generation and release of the fission product gases.

In addition, a small amount of thermal expansion of the clad occurs which is approximately equivalent to that produced by the elastic loading of the clad by system pressure. Both of these effects are small, and mutually canceling in effect. Finally, an important, but less quantifiable, contribution to gap closure occurs from thermal cracking of the fuel.

At beginning of core life (BOL), the staff requires the assumptions that densification has occurred, that the primary gas composition is helium (slightly modified by release of the sorbed gases from the fuel), that thermal expansion of fuel and clad occurs, that no fission gas is available, and that no creep of the clad has occurred. During reactor operation, gap conductance is strongly affected by fuel swelling, creep, and fission gas release, and it is these three processes which control its value during fuel lifetime.

As discussed in Section 2.3, methods used by the applicant to calculate gap conductance result in values of conductance consistent with those calculated by the staff using GAPON. Further analysis of the effects of the above parameters by Westinghouse show that the beginning-of-life value is the lowest value likely to be encountered during the lifetime of the fuel. The applicant has further reasoned that, considering the interaction of the above parameters, the worst later-in-life case would result for a fuel pin that has been operated at 4 kW/ft for a considerable period of time, and is then rapidly increased to a linear power density of 16 kW/ft. The calculated gap conductance for such a pin is still however not as low as for the BOL case. At 4 kW/ft the relatively low clad and fuel temperature, and fast neutron flux, would result in little or no gap closure due to fuel swelling, clad creepdown, or thermal expansion of the fuel. This is compensated for by the reduced fission gas generation and release. The power increase to 16 kW/ft would cause fuel thermal expansion. Westinghouse's fuel swelling model predicts that gas bubble swelling would occur relatively rapidly during power escalation, particularly for conditions of low restraint. In contrast, steady operation at 16 kW/ft provides relatively fast gap closure due to creepdown and fuel swelling. However, more fission gas release would occur than for the case in which the fuel pin operated at 4 kW/ft and was then raised to the higher power.

The Westinghouse fuel swelling model is based on data available in the literature and on data from specific Westinghouse irradiation tests. It was derived from profilometry measurements of cladding, fuel metallographic observations, and pycnometry measurements. Use of clad profilometry data implicitly assumes that cessation of clad creepdown is caused by contact with fuel pellets, i.e., that the gap is closed. Examination of the profilometry data shows that closure occurs prior to the time it would be calculated by simple superposition of the fuel swelling and clad creep effects. The Westinghouse model makes a comprehensive use of information in the literature to rationalize their data in mechanistic and kinetic terms, and the model permits interpolation of fuel swelling data over a large range of powers and burnup.

As the Westinghouse model is empirical and derived from observed fuel pin diametral changes, it includes the effects of densification and radial thermal ratcheting of the fuel (fuel cracking). This means that for the case of fuel swelling no consideration was given for the implicit effects of densification when the empirical model was developed and that the densification penalty involved in Section 2.2.3 is applied twice.

Gaseous fission products have a degrading effect on the conductivity of the gas in the fuel-clad gap and consequently have the tendency to reduce the gap conductance. Westinghouse has developed a model which depends on fuel temperature, time and fuel density. The model is capable of predicting release data from experimental fuel in Saxton and Chalk River. These predictions are considered best estimate; whereas for 10 typical PWR rods removed from NOK, Zorita and Yankee, the Westinghouse model overpredicts the gas release (Reference 1.3). Staff consultants at PNL reviewed the Westinghouse gas release model and found it acceptable. The staff reviewed the gas release model and concurs with acceptance of the model by our consultants.

The equations used for calculating the conductivity of the gas mixture (Reference 1.3) appeared to be acceptable based on a comparison made by ORNL. Westinghouse has performed a two-dimensional finite-difference analysis to examine the effects of pellet eccentricity and clad ovality on the fuel pellet stored energy (Reference 1.4).

This rigorous calculation satisfactorily demonstrates that the concentric case gives higher pellet average temperature than either of the eccentric cases.

Westinghouse has developed a correlation for fuel thermal expansion based on the data from four different investigations. Thermal expansion is very important in the determination of hot gap size (Reference 1.3). The thermal expansion equation developed for use in GAPCON is in excellent agreement with the Westinghouse correlation. The method of summing radial increments to determine the overall hot pellet diameter gives results comparable to those obtained using the slightly modified procedure in GAPCON.

The staff has checked the values for thermal neutron flux depression calculated by Westinghouse for beginning of core life. The values are reasonable and compare well with values calculated by the staff using GAPCON. It should be noted that the staff's GAPCON calculation provides a rough estimate and is not nearly as sophisticated as the Westinghouse calculation. Preliminary estimates based on some work by our consultants at Brookhaven indicate values of the flux depression factor about 1% higher than the Westinghouse calculations for comparable enrichments. This would amount to a stored energy equivalent to about 15°F lower average pellet temperature for the Westinghouse calculation. This is not considered significant.

Westinghouse has developed two empirical correlations for gap conductance as a function of hot diametral gap size and gas mixture conductivity (Reference 1.3). In the design calculation Westinghouse uses the larger of the two correlations. The "design" correlation is used for large hot gaps and/or low values of gas mixture conductivity, and the "annular" equation is used for the opposite conditions. The design correlation was developed from Bettis experimental data. The annular equation is of the form typical for annular gaps. A correction applied to gap size is of the form usually used for temperature jump distances. For the cases of interest, the Westinghouse correction is always larger than the temperature jump distances calculated using GAPCON. Larger corrections or jump distances always tend to decrease the gap conductance. For the linear heat rates of interest in Indian Point 2, the annular equation is used. The staff considers the annular equation to be satisfactory for these conditions (Reference 1.4).

The Westinghouse method for determining temperature drop in the fluid film is reasonable and this effect is not a significant contribution to stored energy. The values used for crud and oxide thicknesses appear reasonable.

2.3 Staff Calculations on Densification

2.3.1 Collapse Time

The computer code BUCKLE has been used to calculate time of creep collapse of an initially out-of-round tube subjected to a net external pressure under high temperature and irradiation exposure. It is used to simulate in-pile behavior of cladding.

As a result of coolant pressure the cladding tube is in a state of compressive membrane stress. A small initial ovality introduces an additional bending moment on the cladding. Under these stresses the cladding undergoes time-dependent creep deformation which increases the initial ovality. This in turn increases the bending moment in the cladding and further accelerates the creep strain until the cladding reaches a point of instability.

The code calculates circumferential stresses and both membrane and bending stresses using thick shell and curved beam models respectively. Using an empirical creep law, it calculates the creep rate, and then the creep strain, by multiplying creep rate by a preset time interval. The next iteration then calculates the change of ovality from circumferential strain and the change in moment due to increased ovality. A new stress is then recalculated and the calculated cycle is repeated until the cladding reaches the collapse criterion. The collapse criterion was taken from Reference 2.3. This criterion is basically an elastic formula whose upper bound is the elastic buckling stress with the ovality used as a parameter in the formula. The cladding is predicted to buckle when circumferential stress reaches the critical value.

The creep relationship was largely based on References (2.4) and (2.5). This relationship is an inpile correlation of uniaxial Zircaloy creep rates whose parameters are stress, temperature and fast neutron flux. The stress constant was adjusted to give a biaxial stress state result based on the theory presented by Ibrahim (Reference 2.5).

Among other things, yield stress, time increment, flux and ovality (ODmax-ODmin) are the input data. Time-dependent variations on internal rod pressure and cladding temperature are allowed in the present version.

As reported in Reference 1.5 the staff has used the BUCKLE code to independently calculate time-to-collapse for Point Beach Unit 2 for rods of similar dimensions and material operating under similar conditions of temperature pressure and neutron flux. The results showed that the staff predicted times-to-collapse were greater than those predicted by the Westinghouse code.

In summary, the staff has examined the Westinghouse Design Code's creep model and find it, and its method of use in the time-to-collapse calculation to be valid. This finding is supported by consideration of the fact that the Westinghouse time-to-collapse code when used to predict clad tubing collapses in actual operating reactors has predicted times which were less than those observed.

Westinghouse analyses predict no clad flattening for any of the Indian Point 2 fuel since the minimum time for initial clad flattening calculated for each region is significantly greater than the respective burnup time planned for that region during the first three burnup cycles. The staff will require provision in the Technical Specifications for fuel inspection and reporting at the first three refueling shutdowns. This inspection will be an audit on the time-to-collapse calculations for extended fuel exposure times.

2.3.2 Gap Conductance

The staff has under development at Pacific Northwest Laboratories (PNL) a computer code GAPCON which calculates the radial temperature distribution in the fuel and the fuel-clad gap conductance. The code (Reference 2.6), originally developed with options for both thermal reactor and fast reactor calculations, has been revised extensively to accommodate new data and to upgrade certain sub-routines. In addition, the code has the capability of being operated with certain sub-routines replaced, or added, by special calculational input. For example, the original sub-routine EXPAND calculated the increase in fuel radius using data for mixed-oxide fuel. This has been replaced with a sub-routine using the thermal expansion values for UO_2. Similarly, Zircaloy properties have been replaced with those of more recent measurements. The expansion due to fuel swelling has been calculated twice, once with the original relationship proposed by Geithoff, et al., and a second time using the Westinghouse empirically based fuel swelling values. It was found that the Geithoff model resulted in less swelling. The values of pellet flux depression are input from physics calculations or estimated from predetermined values. The expression for UO_2 thermal conductivity is now based on the data of Lyons rather than Godfrey, et al. The gas release rate, determined from the Dutt model for mixed oxide fuels, has been changed to that derived from UO_2 irradiation. Some calculational runs have also been made assuming the gas release is linear with time, for amounts of gas release calculated from the Westinghouse design model. One inadequacy (conservative in effect) is that the code does not include a sub-routine for gap closure due to cladding creepdown. In some of the staff's calculations, hot gap sizes derived from the Westinghouse design code were used as a basis for simulating Westinghouse hot gaps in GAPCON to model the effects of gap closure. The code calculates conservative values of gap conductance and fuel temperatures compared to values calculated from measured fuel temperatures by AECL, CVNA, HBWR (HPR-80) and inferred from post-irradiation examination of thermally induced structural changes of fuel tests conducted by the General Electric Company (References 2.7, 2.8, 2.9, and 2.10). Comparison of calculations employing GAPCON with the new model for fuel pellet thermal expansion vs experiments as presented in Reference 1.5 show that GAPCON effectively predicts the observed values for the experiments and that it does so in a conservative manner.

For the range of hot diametral gap sizes encountered during normal operation of fuel in the Indian Point 2 reactor, the gap conductance is most sensitive to the hot gap size. The Densification Report requirement to assume instantaneous densification to 96.5% of theoretical density from a starting pellet density based on the measured statistical distribution (lower 2 sigma value) may be considered to be a conservative input to the gap conductance calculation because: (1) the Indian Point 2 fuel is similar in manufacture to that of the Point Beach-1 region 3 fuel which has been recently examined and has only shown $\Delta L/L$ changes up to 1.4%; and (2) the entire field of data supplied by Westinghouse and General Electric on measurements of final densified fuel density, contains only one point in excess of 96.5% of theoretical density, all other points being less than 96% of theoretical density. Thus a conservative lower bounding diameter is used for the BOL hot pellet diameter. Further, the assumption of instantaneous densification is conservative, since we know from in-reactor neutron flux trace analysis that flux peaking has been observed only after 300-400 effective full power hours (EFPH) of irradiation and takes place over a finite time period which ranges up to 2000 EFPH. Thus there is good evidence to confirm that the hot pellet diameter calculated using the requirements set forth in the Densification Report is never in fact attained and that the hot gap even at BOL is smaller than that calculated using the staff methodology.

With regard to the time dependent changes in the hot gap, Westinghouse has profilometry data on a number of fuel pins for times greater than 4300 EFPH which provides information on values of the hot pellet diameters (Reference 1.4). This hot pellet diameter implicitly incorporates the effects of fuel cracking, fuel densification, and fuel swelling. It is from these measurements that Westinghouse has evolved their fuel swelling model which when used to predict the hot pellet diameter gives good agreement. When an increment for densification is added to the semi-empirical swelling model, it tends to conservatively under-predict the hot pellet diameter. This analysis of hot gap closure does not account for clad creepdown onto the pellet which also provides gap closure. The belief that all time dependent dimensional changes contribute to an improved gap conductance is only modified by the role of the time- and burnup-dependent release of fission gas. In this regard, the Westinghouse subroutine for fission gas release that is part of their design model is believed to be conservative when compared to prototypic Westinghouse fuel rods from a variety of operating reactors (NOK, Zorita, Yankee) as it over predicts the amount of gas released relative to the measured values, especially at low burnups. GAPCON predictions with more realistic but still conservative fission gas release show that the Westinghouse (calculated) beginning-of-life gap conductivities are conservative.

In summary the staff has examined the Westinghouse Design Code's thermal performance model, data, and assumptions that form a basis for their gap conductance calculation and we have found the calculated values to be comparable to those independently calculated by the staff's thermal performance code GAPCON.

The staff has confidence in GAPCON as it has used that code to predict the gap conductances and/or fuel temperatures for experimentally derived data which are available, and found that GAPCON is capable of predicting the experimentally derived values with a margin of conservatism.

2.4 Conclusions

The staff has concluded that:

(1) The direct effects of fuel densification have been adequately accounted for,

(2) The time-to-collapse calculational model used by Westinghouse has been verified by independent calculations made by the staff and is therefore acceptable.

(3) An acceptable calculational method has been used to describe the creepdown effect that tends to increase gap conductance with lifetime.

(4) The Westinghouse calculations of pellet stored energy and gap conductance for beginning-of-life in the range of 14 to 16 kW/ft are consistent with the staff calculations. In addition, the staff would not expect the values of gap conductance to be below the Westinghouse beginning-of-life value for the proposed period

of operation. Therefore we conclude that Westinghouse calculated values of gap conductance as used in the performance analysis for Indian Point 2 are acceptable.

2.5 References

2.1 Westinghouse Report E-PA-475 - "Clad Creep Model", Westinghouse Proprietary, October 1972.

2.2 Wilson, W. P., "A Method of Analysis for the Creep-Buckling of Tubes Under External Pressure, WAPD-TM-956, October 1970.

2.3 Timoshenko, S. P., Gere, J. M., "Theory of Elastic Stability", 2nd Edition 1961, McGraw-Hill Book Company.

2.4 Holmes, J. J., Williams, Myman, D. H., and Tobin, J. C., STP-380(1960) ASTM.

2.5 Ibrahim, E. F., "In-Reactor Creep of Zirconium Alloy Tubes and Its Correlation with Uniaxial Data," Application Related Phenomena in Zirconium and Its Alloys, STP-458 (1968) ASTM.

2.6 Users Guide for GAPCON; Horn, G. R. and Panisko, F. E., HEDL-TME-72-128, September 1972.

2.7 B. N. Duncan, Rabbit Capsule Irradiations of UO_2, CVTR Projects; CVNA-142, June 1962.

2.8 Bain, A. S., "Microscopic, Autoradiographic, and Fuel-Swath Heat Transfer Studies on UO_2 Fuel Elements," AECL 2588, June 1966.

2.9 Kjaerhein, G. and Rolstad, E., "In Pile Determination of UO_2 Thermal Conductivity, Density Effects and Gap Conductance" HPR-80, December 1967.

2.10 Ditmore, D. C. and Elkins, R. B., "Densification Considerations in BWR Fuel Design and Performance" NEPM-10735, December 1972.

3.0 EFFECTS OF DENSIFICATION ON STEADY-STATE AND TRANSIENT OPERATION

3.1 General

Fuel densification can affect steady-state operation by virtue of its effect on local neutron flux (due to gaps in the fuel column) and the resulting slightly shorter fuel stack length. An additional effect occurs in the transient analyses where, due to a lower gap conductance, the fuel has a higher initial stored energy and a slower heat release rate during the transient.

On the basis of calculations of the effects of fuel densification, the Indian Point Unit 2 reactor will be operated with more restrictive insertion limits on part length control rods than originally proposed, and with a reduced design total peaking factor. The changes consider the effect of local peaking caused by gaps in the fuel pellet stack and changes in the gross peaking factors, primarily axial, which can be achieved by more restrictive operation of part length control rods.

The effects of densification on power density distributions have been calculated using models in conformance with those discussed in Section 4 of the staff densification report (Reference 1.1). The primary calculations used the models and numerical data of the Westinghouse power spike model as described in Appendix E of Reference 1.1, except that the initial nominal density used was 0.936 of theoretical (the minimum of the three region densities), and the probability of gap size was changed to conform to that recommended by the staff in Section 4.2 of Reference 1.1. These calculations by the applicant take into account the peaking due to a given gap, the probability distribution of peaking due to the distribution of gaps, and the convolution of the peaking probability with the design radial power distribution. The calculations result in a peaking augmentation factor which varies almost linearly with core height and reaches a maximum value of about 1.16 at the top of the core.

Independent calculations have also been made by our Brookhaven National Laboratory consultants (Reference 3.1). Relative to the Westinghouse results, the BNL calculations have given lower peaking due to a gap of given size, similar probability distributions for multiple gaps and similar convolution results.

The augumented peaking factor function is directly combined with axial power distribution calculations to form the basis for the axial peaking - axial offset map and correlation limits. These types of peak-axial offset maps and correlations have formed the basis for axial flux distribution control for a number of Westinghouse reactors. and have been reviewed previously by the staff. If the axial offset is maintained within specified limits (but more restrictive than has been required previously) and is combined with maximum radial peaking and uncertainty factors, the total peaking factor F_Q^T would be expected to be maintained at a value less than 2.70.

This decrease in F_Q^T from the 3.00 value previously used in Indian Point Unit 2 analyses and in the previous Technical Specifications, even considering the gap peaking augumentation, is achieved primarily by the tighter restrictions on the axial distributions. The basic axial peaking - axial offset correlation without gap peaking effects has not changed. This correlation gives, for relatively small offsets, axial peaking factors less than 1.55. However, the previous Indian Point Unit 2 design axial peaking factor was about 1.8, allowing the flexibility to operate with larger offsets and less restrictive limits on control rod positions if desired.

The increased heat flux due directly to densification from the nominal steady state density, has been included based on densification from the minimum nominal fuel density of 0.936 of theoretical (Region 1) to a final value of 0.965; consideration of the "as fabricated" average fuel stack height; and axial thermal expansion of the pellet. The result is an increase of 2.0 percent in heat flux.

The effect of the variation in the initial density has been included as an additional convolution calculation as allowed for in Section 4.5 of Reference 1.1. The variation in density is convoluted with the power spike model and the calculation relating density changes to overpower temperature limits or LOCA power limits. The calculation is similar to the previous convolutions except that both radial and axial dimensional changes are included. The result of these calculations is an additional factor on heat flux of 2.0 percent for thermal overpower calculations and 0.7 percent for the LOCA calculations. Our BNL consultants have also done independent calculations using this convolution process (Reference 3.2), and have found consistent results.

The result of applying all of these factors to the average power density, for the peak power density to be considered in LOCA evaluations, and assuming full rated power and a 1.02 power uncertainty factor, is:

$$\text{Peak kW/ft} = 5.7 \times 2.70 \times 1.02 \times 1.007 \times 1.02 = 16.1 \text{ kW/ft.}$$

3.2. <u>Loss-of-Flow Transient</u>

The densification effects that could aggravate the consequences of the loss-of-flow transient are the increase in the steady state fuel temperatures (stored energy), increases in heat flux, and a decrease in gap conductance. The increase in fuel temperature provides more stored heat in the fuel which must be removed during the transient; the higher heat flux provides greater initial enthalpy in the coolant channel. The decrease in gap conductance delays the removal of heat from the fuel resulting in a higher ratio of heat flux to channel flow during the transient and thus a lower DNBR.

The Westinghouse analysis employed the FACTRAN Code (Reference 3.3) to calculate the fuel rod heat flux from the neutron flux transient. These values of heat flux were then used as input to the Westinghouse THINC Code to calculate the DNBR values as a function of time. This procedure appears to be conservative because the steady state THINC Code would overpredict the enthalpy of the coolant during that part of the transient when the ratio of heat flux to coolant flux is increasing. This would result in a lower calculated value of DNBR than would be expected in actuality.

Westinghouse has reanalyzed the transient that would result from a loss of electrical power to all four of the primary coolant pumps, taking into account the effects of fuel densification. The results show that the minimum departure from nucleate boiling ratio (DNBR) during the transient is not decreased. The previously calculated minimum DNBR during the transient was 1.42 whereas with the densification the minimum DNBR is calculated to be 1.57. The increase in minimum DNBR over that previously calculated is attributable to both the reduction in the total peaking factor and the use of actual measured flow coastdown which more than compensates for the effects of fuel densification.

The staff has also evaluated the loss of reactor coolant flow transient in connection with our review of Point Beach Unit 2. The evaluations were performed using the computer program COBRA III-C (Reference 3.4). This program calculates the heat transfer and fluid flow conditions in rod bundle nuclear fuel element subchannels during both steady state and transient conditions. It uses a mathematical model that considers both turbulent and diversion cross flow mixing between adjacent subchannels. The thermal model considers radial conduction within the fuel. Axial and circumferential conduction are ignored. The model uses circumferential averaged coolant temperature and surface heat transfer coefficients calculated from the subchannel hydraulic data at each axial node. The thermal properties of the fuel and clad are considered constant and uniform throughout a transient calculation. Gap conductance and cladding-fluid heat transfer coefficients are input to the calculations. Gap conductance is assumed to be constant throughout the transient.

The COBRA III-C Code contains the basic features required for a comparative evaluation of the applicant's computations. Details of a comparative calculation made for Point Beach Unit 2, where the same methods were used by the applicant, are reported in Reference 1.3. It is shown that there is reasonable agreement between the two sets of computations. We conclude therefore that the applicant has provided adequate assurance that the minimum DNBR is greater than 1.3 for the loss-of-flow transient.

3.3 Locked Rotor Accident Analysis

The analysis of the locked rotor accident was originally presented in Section 14.1.6 of the FFDSAR. The transient behavior was analyzed by postulating an instantaneous seizure of one reactor pump rotor. The reactor flow decreased rapidly and a reactor trip occurs as a result of a low flow signal. The temperature of the primary coolant increased, causing fluid expansion with a resultant pressure transient. The system pressure is reduced to its nominal value after about 5 seconds following the postulated locked rotor.

The thermal analysis of the hot rod in the core was performed using design conditions with respect to power, flow, and core inlet water temperature. Considering fuel densification effects, no rods are predicted to experience a departure from nucleate boiling (DNBR < 1.3).

3.4 Other Transients

The following other transients have been reviewed to determine whether the effects of densification have resulted in their consequences becoming more important:

- Uncontrolled withdrawal of a Rod Cluster Control Assembly (RCCA) from a subcritical condition.
- Malpositioning of Part-Length Rods
- Drop of a RCCA
- Malfunction of Chemical and Volume Control System (CVCS)
- Start-up of an Inactive Reactor Coolant Loop
- Reduction in Feedwater Enthalpy
- Excessive Load Increase
- Loss of External Electrical Load
- Loss of Normal Feedwater
- Loss of A.C. Power

In the applicant's FFDSAR these transients were reported to have DNB ratios in excess of 1.3, or their consequences controlled to acceptable values by limits set forth in the Technical Specifications. The staff has reviewed the consequences of these transients taking into account the effects of fuel densification and agrees with the applicant that they would not result in lower core thermal margins (i.e., DNB ratios less than 1.3).

3.5 Conclusions

The effects of fuel densification on steady-state and transient operation have been analyzed by the applicant and reviewed by the staff.

The effect on steady-state operation, mostly due to local increases in thermal neutron flux and heat generation, is to require more restrictive limits on control rod insertion so as to assure operation with the required overall and axial peaking factors. The restrictions on control rod insertion and peaking factors will be included in the Technical Specifications.

The staff has reviewed the calculations provided by the applicant and concludes on the basis of its review that the potential effects of fuel densification on the steady-state and postulated transient operation have been taken into account in an acceptable manner.

3.6 References

3.1 Peaking Factors in Pressurized Water Reactors with Fuel Densification; BNL Interim Report, December 1972.

3.2 Power Peaking Penalty with Pellet Density Fluctuations, BNL Draft Interim Report.

3.3 WCAP-7908, FACTRAN, A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod.

3.4 COBRA III: A digital Computer Program for Steady State and Transient Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements, BNWL-B-82, D. S. Rowe.

4.0 ACCIDENT ANALYSES

4.1 General

Analyses of the consequences of various postulated accidents were presented in the Indian Point Unit No. 2 Final Facility Description and Safety Analysis Report and various Amendments. The accidents evaluated were:

(1) loss-of-coolant

(2) control rod ejection

(3) steam-line rupture

(4) primary coolant pump seizure

(5) steam generator tube rupture

(6) fuel handling accident, and

(7) waste gas tank rupture.

Since effects of fuel densification will affect the consequences of the first four postulated accidents they have been reanalyzed by the applicant and reevaluated by the staff. Results of the first three accidents, loss-of-coolant, rod ejection, and steam line rupture are presented in this section. The preceding Section 3.3 presented the results of our evaluation of the fourth accident, primary coolant pump seizure. The steam generator tube rupture and fuel handling accidents will be discussed below.

Changes in the fuel pellet geometry can cause the stored energy in the fuel pellet to increase (Reference 1.1). Section 2.0 discusses the mechanisms by which the stored energy in the fuel is increased due to changes in pellet geometry. Potential increases in local power due to the indirect effects of fuel densification (formation of axial gaps) is discussed in Section 3.1. Both of these effects are accounted for in the evaluations of accidents.

The radiological consequences of accidents are independently calculated by the staff. The results of the staff's calculation of the radiological consequences of accidents for Indian Point 2 were presented in the Staff Safety Evaluation for Indian Point 2 dated November 16, 1970 (Reference 4.1). The radiological consequences would not increase as a result of fuel densification, although the transient performance of the fuel rods can change as a result of fuel densification. It is the latter factor that is discussed in the following sections.

Our evaluation of the radiological consequences of a waste gas decay tank failure was based on an assumed quantity of gas in the tank. The assumed quantity is consistent with the Technical Specification limits on maximum permitted primary system activity. Fuel densification will not affect the consequences of this accident.

The postulated refueling accident assumes the dropping of a fuel assembly in the spent fuel pool or transfer canal. The fuel rods are assumed to be at approximately 100°F during the postulated accident. Therefore, the direct effects of fuel densification will not affect the outcome of this postulated accident. The potential for mechanical failure of a flattened rod might be different from that of a normal rod; however, since our evaluation has been based on the conclusion that no clad collapse will occur (Section 1.0), this potential change in fuel rod characteristics was not considered. Furthermore, all of the rods in the dropped assembly are assumed to fail.

4.2 LOCA Analysis

The accepted Westinghouse evaluation model described in the AEC Interim Acceptance Criteria for Emergency Core Cooling Systems has been used as the basis to evaluate the Indian Point Unit 2 loss-of-coolant accident. The core blowdown and reflood analyses are virtually unaffected by the decreases in gap conductance resulting from densification and are reported in the applicant's Indian Point Unit 2 ECCS Report which was submitted to demonstrate compliance with the ECCS Interim Policy Statement. While decreased gap conductance calculated according to the staff requirements would cause the core average pellet temperature to increase, LOCTA calculations show very little change in the energy released with and without densification. Therefore, virtually the same margin of energy release as calculated by SATAN would be maintained with respect to the LOCTA calculations.

Densification has less impact on the reflood calculation. Reduced gap conductance would cause the rate of decay heat transferred across the gap to the cladding to be reduced. However, the benefit would not be significant since the gap conductance is so much larger than the calculated film coefficients during reflood.

Fuel rod heatup calculations were done with LOCTA using various assumptions related to densification and without the benefits of transition boiling after the first dryout has essentially occurred. Termination of transition boiling term results in slight increases in cladding temperature for Indian Point Unit 2. The double-ended break was previously determined to be the worst break, as was reported in the FFDSAR.

Westinghouse has performed parametric calculations of clad temperature for the double-ended break as a function of pellet average temperature and local power. The results have been presented on plots of pellet average temperature vs. local rod power as a limit line for which the clad temperature reaches 2300°F. It should be noted the limit line was derived based on calculations where the pellet average temperature was increased above the stated ordinate value by an amount determined by the formula stated in Section 2.2.3. On several plots with this limit line, Westinghouse has presented the results of many fuel design calculations wherein the average pellet temperature is calculated as a function of linear power density. Various burnups, power histories, and densification assumptions were used. The point where the particular calculation of interest intersects the 2300°F limit line determines the limiting peak linear power density required to meet the Interim Acceptance Criteria. Several combinations of power history were used for some calculations, but none resulted in higher stored energies or more limiting power densities than the beginning of life cases. Using the staff requirements for initial pellet density, assumptions, the limiting linear power density was determined to be 17.35 kW/ft. This value is greater than the maximum linear power density of 16.1 kW/ft. obtained using reactor operating limits as described in Section 3.1.

4.3 Rod Ejection Accident

Taking into account changes in the fuel from the densification process and proposed changes in the operational limitations on control rod patterns and motion to provide more restrictive control of power distribution, certain aspects of the analyses of the control rod ejection transient have changed. Since there are few, if any, significant changes in the important reactor kinetics parameters, there would be little change in the core transient power history for ejection of a control rod of given reactivity worth. However, since the limits on control rod bank insertion have changed, there has generally been a decrease in the rod worth available in a given reactor state for a rod ejection transient, and a corresponding decrease in the transient energy release from this accident. For the hot spot calculation there are changes in the peaking factors, and thus power density, for both initial and transient states as a result of gap peaking and control rod position limitations, and there are changes in initial fuel temperature for power cases due to changes in assumed fuel-clad gap size and gap thermal conductivity.

The applicant has redone calculations for the extremes of power level and time-in-cycle, using parameters now considered to be appropriate. The methods used have been developed since the submittal of the FFDSAR, and they are described and compared with more exact calculations in a topical report (WCAP-7588, Reference 4.2). As with the CHIC-KIN calculations that were presented in the FFDSAR, the analysis is done in two stages; first a transient calculation using the TWINKLE spatial kinetics code in a one-dimensional version, then a hot spot heat transfer calculation with the FACTRAN code using appropriate peaking factors and power histories from the TWINKLE calculation. These codes, when using similar parameters, give similar results to the previous CHIC-KIN calculations.

The Regulatory staff and our consultants at Brookhaven National Laboratory (BNL) have examined the calculational models and data, and BNL has made some check calculations in connection with our review of Point Beach 2 (Reference 4.3) using appropriate input data and their own developed codes. On the basis of this review we have concluded that the methods and input are generally satisfactory and conservative. The two stage method of calculation, using a one-dimensional, space-time calculation followed by a hot spot calculation gives, with the input data used, conservative hot spot peak energies. This is especially true for the full-power cases where maximum transient peaking factors are added to the initial condition maximum peaking factors of 2.75, with the densification factor included along with a 2-sigma initial density variation allowance. The method and codes, especially TWINKLE in its one-dimensional version, are very similar to those developed and used at BNL. The nuclear input data, including the reactivity coefficients, weighting factors, control rod reactivity worths, and peaking factors, are generally conservative.

The calculated transients are generally insensitive to heat transfer parameters and in particular to gap conductances. In fact an artificially large gap conductance is used in the hot spot calculations to increase the calculated clad temperature. The primary effect of fuel densification and change in gap size is on the initial conditions for the hot spot fuel calculations in the full power calculations. The assumption of a larger gap, and the very conservative assumption that design peaking conditions occur at the same spot as the maximum transient peaking, give a higher initial heat content of the hot fuel pellet as a base for the hot spot calculation. Our review of the initial fuel temperature for the BOL full power case indicated that reasonable temperatures were used for the assumed conditions. The gap conductance would in general be expected to increase, and thus decrease temperatures, for the EOL case. But rather than fully evaluate this change at this time, the EOL power case was evaluated on the basis of BOL initial fuel temperatures. By taking the control bank limit to be 30 percent inserted at full power, at BOL, as is actually the case rather than the arbitrarily fully inserted case analyzed by the applicant, the conditions become similar to BOL. The decreased delayed neutron fraction at EOL would have almost no effect for a small ejected rod worth and all nuclear parameters such as reactivity coefficients and peaking would be improved. Thus the accident at EOL would have lower energy release than at the BOL.

The calculated peak average fuel pellet energy content is less than 170 cal/gm, and pellet centerline melting temperatures are not reached. The peak clad temperature for the worst case rod ejection is 2245°F.

4.4 Steam Line Break

The applicant stated, in reference 1.2, that the minimum DNBR would be reduced from the value in the FFDSAR (>2.00). The reactivity transient is unchanged from that presented in the FFDSAR. The acceptance criteria in the FFDSAR were:

(1) With a stuck rod and minimum engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact.

(2) With no stuck rod and all equipment operating at design capacity insignificant cladding rupture occurs.

In these criteria the applicant did not exclude the occurrence of DNB or clad perforation, although the analysis did not indicate that such events occurred. The analysis includes the assumption of proper operation of the Safety Injection System (SIS), reactor trips (nuclear flux, ΔT, and SIS), feedwater isolation, and closure of the main steam isolation valves. In addition steam flow nozzles (16" i.d.) in each steam pipe (28" i.d.) serve as steam flow limiters.

A complete description of the methods used and the results of the steam line break calculation is found in Section 14.2.5 of the FFDSAR. The staff concludes that densification would have only an insignificant effect on the consequences of a steam line break.

4.5 Conclusions

The effects of fuel densification in the analysis of postulated accidents has been considered by the applicant and reviewed by the staff. The assumed occurrence of densification affected the analysis of the LOCA, the control rod ejection, the steam-line rupture, and locked-rotor accidents.

Consideration of the LOCA analysis by the applicant resulted in a limitation on a total peaking factor of 2.70. At this peaking factor the calculated peak cladding temperature is less than 2300°F.

The densification criterion also caused changes in the results of the analysis of the control rod ejection accident. Center-melt of the fuel is not reached, and the embrittlement of the cladding is insignificant. Less than 15% of the fuel pins go into DNB.

An insignificant effect is seen on the steam line break analyses due to the postulated densification. The minimum DNBR is above 2.0 and no failed fuel is predicted.

The applicant's analyses of the effects of densification on accidents were subjected to audit calculations by the staff. On the basis of its review the staff concludes that the applicant has considered the effects of densification on accident analysis in an acceptable manner.

4.6 References for Chapter IV

4.1 Safety Evaluation by the Division of Reactor Licensing, U. S. Atomic Energy Commission, "In the Matter of Consolidated Edison Company of New York, Incorporated Indian Point Nuclear Generating Unit No. 2, Docket No. 50-247" dated November 16, 1970.

4.2 WCAP-7588- An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods.

4.3 Rod Ejection Transient at Power in Point Beach Unit 2, BNL Draft Interim Report.

5.0 SUMMARY AND CONCLUSIONS

The effects of fuel densification have been considered in analyses of normal operation, operation during transient conditions, and postulated accident conditions. On the basis of the staff review of the applicant's calculations, and independent calculations performed by the staff and its consultants, the staff concludes that:

(1) The effects of densification during steady state and transient operation of Indian Point Unit 2 will not cause the limits on DNBR, or cladding strain, or centerline temperatures, to become less conservative than values previously established in the FFDSAR.

(2) The effects of densification were included in the calculation of fuel rod behavior during postulated loss-of-coolant accidents. The LOCA analysis is acceptable and complies with the June 1971 Interim Policy Statement, even after modifying the use of the transition boiling heat transfer coefficient as described in Section 4.2.

(3) An acceptable calculational method has been used to describe the creepdown effect that tends to increase gap conductance with lifetime.

(4) Operating restrictions regarding control rod and power distribution resulting in operation of the reactor with a total

peaking factor of less than 2.70 are necessary to assure compliance with paragraphs 1-3 and will be incorporated into the Technical Specifications.

(5) Fuel inspection and reporting requirements during the first three refueling shutdowns will be included in the Technical Specifications as an audit on the predicted result of no fuel collapse for Regions 1, 2, or 3 during the first three fuel cycles.

On the basis of the above five conclusions the applicant is in compliance with the staff densification report; on this basis the staff concludes that Indian Point Unit 2 can be operated at 2758 MWt (100% of rated power) with no undue risk to the health and safety of the public.