UNITED STATES OF AMERICA
ATOMIC ENERGY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
WISCONSIN ELECTRIC POWER COMPANY	)	DOCKET NO. 50-301
WISCONSIN-MICHIGAN POWER COMPANY	)	
	)	
(Point Beach Nuclear Plant, Unit 2)	)	

MOTION FOR SUMMARY DISPOSITION

Pursuant to 10 CFR §2.749, the regulatory staff (staff) moves the Atomic Safety and Licensing Board (ASLB) for summary disposition in the above-captioned proceeding on the ground that there is no genuine issue to be heard with respect to any material fact. In support of this motion, are attached a Statement of Material Facts As To Which There is no Genuine Issue to be Heard; the joint affidavit of Donald F. Knuth and Victor Stello, Jr.; the joint affidavit of Donald F. Knuth, Victor Stello, Jr. and Lester S. Rubenstein; and the Technical Report on Densification of Light Water Reactor Fuels, dated November 14, 1972. Accordingly, the ASLB should issue a supplemental initial decision that favorably disposes of the fuel densification phenomenon question with respect to the Point Beach Unit 2 Nuclear Facility.

In support of this motion the Staff states:

1. The affiants are members of the regulatory staff having

technical responsibility for the review and evaluation of the fuel densification phenomenon concerning the Point Beach Unit 2 Nuclear Facility.

2. On December 15, 1972 the Atomic Safety and Licensing Appeal Board in ALAB-86 remanded to the ASLB for further proceedings on the questions of (1) the effect of the fuel densification phenomenon on the Point Beach Unit 2 Nuclear Facility; (2) whether the facility can be permitted to operate at levels in excess at 300 Mwt; (3) and whether any limiting conditions or technical specifications need be imposed on operation at levels in excess of 300 Mwt as a result of the fuel densification phenomenon.

3. Affiants have reviewed and evaluated the information concerning the fuel densification phenomenon concerning the Point Beach Unit 2 Nuclear Facility.

4. Affiants have concluded that Point Beach Unit 2 Nuclear Facility can be operated at power levels up to 1518 Mwt. (100% power) without endangering the health and safety of the public and that the issuance of a full power license will not be inimical to the health and safety of the public.

WHEREFORE, the Staff requests:

1. That the ASLB issue a supplemental initial decision under §2.749 determining that a full-term operating license for the Point Beach Unit 2 Nuclear Facility should be authorized; or

2. That, in the alternative, if the ASLB finds it cannot grant the motion in whole or in part, that the ASLB should specify which material facts are in issue and which are not.

Respectfully submitted,

Joseph Gallo

Joseph Gallo
Assistant Chief Hearing Counsel

Dated at Bethesda, Maryland
this 28th day of January, 1973.

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ATOMIC ENERGY COMMISSION

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CERTIFICATE OF SERVICE

I hereby certify that copies of "Motion for Summary Disposition and Supporting Documents, namely Statement of Material Facts as to which there is no Genuine Issue to be Heard, Joint Affidavit of Donald F. Knuth and Victor Stello, Jr., Joint Affidavit of Donald F. Knuth, Victor Stello, Jr., and Lester Rubenstein, and the Technical Report on Densification of Light Water Reactor Fuels, dated November 14, 1972", in the captioned matter, have been served on the following by deposit in the United States mail, first class or air mail, this 28th day of January, 1973:

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Washington, D. C. 20545

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Washington, D. C. 20545

Joseph Gallo

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Assistant Chief Hearing Counsel

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MEMORANDUM IN SUPPORT OF STAFF'S MOTION FOR SUMMARY DISPOSITION

I. Statement of the Proceedings

On May 17, 1972, the Atomic Safety and Licensing Board (Board) issued an initial decision resolving each of the radiological health and safety issues which had been raised concerning operation of the Point Beach 2 nuclear power plant. This finding was affirmed by the Atomic Safety and Licensing Appeal Board on November 14, 1972. While the record was still open for consideration of environmental issues the intervenors, on August 21, 1972, filed a motion to reopen the radiological phase of the proceedings. The motion sought specifically to require the regulatory staff (Staff) to evaluate the "Ginna fuel problem" and to submit their findings to this Board for review on the record.

At the hearing on the Intervenor's motion the Staff asked the Board to defer action on the motion until the Staff completed its evaluation of the information being received from the R. E. Ginna Nuclear Facility and supplied the Board with this information. (Tr. 5331-32). The Board agreed to defer action.

On November 20, 1972 the Staff advised the Board that a report dated November 14, 1972 on the general fuel densification phenomenon had been prepared [Technical Report on Densification of Light Water Reactor Fuels (Report)]. The Report requested that the Applicant investigate the "Ginna fuel problem" in accordance with recommended evaluation techniques. The Staff also again requested the Board to defer ruling on the Intervenor's motion until after Applicant had submitted this information and the Staff had made a proper evaluation.

The Applicant supplied the requested information on December 13, 1972. On December 22, 1972, the Staff asked the Applicant for additional information. This has also been supplied by the Applicant.

On December 8, 1972 the Board denied Intervenor's Motion to Reopen and issued an Initial Decision authorizing operation of the reactor at full power. (See: Memorandum and Order with Respect to Intervenor's Motion to Reopen). On December 15, 1972, in ALAB-86, the Atomic Safety and Licensing Appeal Board (ASLAB) reversed this decision and instructed the Licensing Board to augment the record and issue a supplemental initial decision on the fuel densification phenomenon as it affects the Point Beach Unit 2 facility.

The Staff has now completed its evaluation of the fuel densification problem. The staff has concluded, after examination of information supplied by the Applicant, and after conducting an independent evaluation, that the Point

Beach Unit 2 Nuclear Facility can be operated at power levels up to 1518 Mwt. (100% power) without endangering the health and safety of the public and that the issuance of a full power license will not be inimical to the health and safety of the public.

II. These Proceedings are Properly Disposable on a Motion for Summary Disposition

This case is properly disposable by a motion for summary disposition as provided by 10 CFR § 2.749. 10 CFR § 2.749 is based upon Rule 56 of the Federal Rules of Civil Procedure and the Model Summary Decision Rule drafted by the Administrative Conference of the United States for use by administrative agencies. See, Gellhorn & Robinson, Summary Judgment in Administrative Adjudication, 84 Harv. L. Rev. 612, 628 (1971). The purpose of such a motion is to avoid useless and dilatory proceedings where there are no material issues to be tried.

The inclusion of this procedure in the regulations vests the Board with the power to determine if material factual issues exist, so as to require an evidentiary hearing, or to siphon out those issues which are clearly not in dispute or upon which one party cannot proffer any evidence. (10 CFR § 2.749(a); and F.R. Civ. P. 56(d).

Under the Federal Rules the motion is designed to pierce the general allegations in the pleadings, separating the substantial from the insubstantial, by requiring the parties to support their allegations with either affidavits, depositions, interrogatories or oral testimony. 6 Moore § 56.04[1]. Mere allegations in the pleadings will not create an issue as against a motion for summary disposition supported by affidavits. 10 CFR § 2.749(b); F.R. Cir. P. 56(c). Without this requirement, summary disposition would be a nullity - - one capable of being defeated solely by a general allegation.

The intervenors clearly have been and are being provided with the access to an evidentiary hearing on the "Ginna Fuel Problem." (ALAB-86) All the parties to the action were allowed discovery, permitted, if they pleased, to take depositions or interrogatories. The time for discovery closed on January 16, 1973.[1/] Intervenors were aware of this date and the need to prepare their case in an expedited manner. The parties were warned in ALAB - 89 (at 2) that the ASLAB would not allow further time extensions, and that a schedule should be established by the licensing board for both the completion

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1/ By Order of the Board, January 17, 1973, the intervenors were ordered to answer the Applicant's interrogatories by January 27, 1973.

of discovery and the filing of specific contentions. (p.2) The time for both has passed. Intervenors have failed to file any specific contentions.[1/]

At no point have the Intervenors thus far adequately particularized their contentions. At no point have they identified a single material factual issue notwithstanding the opportunity provided them.

It is imperative to the orderly process of this proceeding that the intervenors either present their counter-affidavits, or that the Board rule favorably on this motion. Otherwise unwarranted further delay will be added to these proceedings, (*See* ASLB, Initial Decision, 12-18-72, at 13) and the intervenors will be allowed to indulge in a fishing expedition at the expense of the public. A "public hearing is not an opportunity for the commencement of a *de novo* review of an application for a license which will duplicate the review conducted by the AEC Regulatory Staff, which would permit the intervenors to ultimately determine whether or not, in fact, there are matters they wish to controvert, and which would automatically delay the proceedings for a considerable length of time." (Initial Decision, 12-18-72).

[1/] "Definition of the matters in controversy is widely recognized as the keystone to the efficient progress of a contested proceeding. In order to put a matter in issue, it will not be sufficient merely to make an unsupported allegation." Statement of Considerations published with the Commission's revised Rules of Practice (10 CFR Part 2) 37 Fed. Reg. 15128 (July 28, 1972).

It should also be emphasized that the intervenors are not entitled to have the motion denied on the mere hope that at the hearing they will be able to discredit the movant's evidence. The intervenors are required to point out something that indicates the existence of a triable issue of material fact if they are to defeat this motion. Radio City Music Hall v. United States, 135 F. 2d 715 (2d Cir. 1943); Orvis v. Brickman, 95 F. Supp. 605 (D.C.D.C. 1951). In granting the defendant's motion for summary judgment under the Federal rules, the Court in Orvis said:

> All the plaintiff has in this case is the hope that on cross-examination ... the defendants ... will contradict their respective affidavits. This is purely speculative, and to permit trial on such basis would nullify the purpose of Rule 56, which provides summary judgment as a means of putting an end to useless and expensive litigation and permitting expeditious disposal of cases in which there is no genuine issue as to any material fact.

Allowing the intervenors to participate in an evidentiary hearing on the basis of their general statements would be a negation of the purpose of Section 2.749. Once the intervenors proffer sufficient basis to raise a factual issue then a hearing will be required. If the intervenors cannot do so, nor show how credibility is at issue, nor in any way show how a hearing on the merits is necessary, there is no sound basis for and,

intervenors have no right to a hearing. <u>See</u> Baumler v. Ford Motor Co., 89 F. Supp. 218 (D. Minn. 1951).

Respectfully submitted,

Joseph Gallo

Joseph Gallo
Assistant Chief Hearing Counsel

Dated at Bethesda, Maryland
this 28th day of January, 1973

UNITED STATES OF AMERICA
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(Point Beach Nuclear Plant, Unit 2)	)	

STATEMENT OF MATERIAL FACTS AS TO WHICH THERE IS NO GENUINE ISSUE TO BE HEARD

1. The problem of fuel densification phenomenon was first noted at the R. E. Ginna Nuclear Power Plant. As a result of the regulatory staff's (staff) evaluation of the Ginna data, the Wisconsin Electric Power Company and Wisconsin-Michigan Power Company (Applicant) was asked to submit data with respect to the fuel densification phenomenon asit related specifically to the Point Beach Unit 2 facility. (Affidavit pp. 2-3). The Applicant's evaluation was required by the staff to be in compliance with the techniques specified in a report prepared by the staff entitled "Technical Report on Densification of Light Water Reactor Fuels" dated November 14, 1972 (Report). The Applicant submitted the required data.

2. The Report was written by members of the staff and it sets forth the essential elements that must be used in an evaluation of the effects of fuel densification in a nuclear facility.

3. The Applicant has supplied all necessary documentation to the staff for its review and analysis of how fuel densification can effect the operation of the Point Beach Unit 2 nuclear facility (hereafter cited as "Facility")

4. The staff concluded:

   a. That the Applicant's analysis of fuel densification effects on the Facility were in compliance with the requirements set forth in the Report.

   b. That based on the staff's review of the information provided by the Applicant, the plant could be operated at power levels up to 1518 Mwt (100%) power without endangering the health and safety of the public and that the issuance of a full power license will not be inimical to the health and safety of the public.

5. The assumption that densification to a value of 96.5% occurs immediately upon the operation of the facility at substantial power is an acceptable and conservative assumption. (Affidavit, at 4.)

6. Densification causes a decrease in the pellet length which causes the linear heat generation rate to increase. (Affidavit, at 4.)

7. For purposes of calculating local conditions in a fuel rod, the assumption that the pellet densifies from an initial density that is lower than the average by an amount equivalent to a 2-σ variation in pellet density is an acceptable and conservative assumption. (Affidavit, at 4; Report, at 67.)

8. Decrease in the pellet radius increases the radial clearance gap between the fuel pellet and fuel cladding. This causes a decrease in thermal conductance which in turn causes an increase of stored energy in the fuel pellet. (Affidavit, at 5.) See paragraph 21 infra.

9. Decrease in pellet length can also lead to the generation of axial gaps within the fuel column. These gaps will result in increased local neutron flux (i.e., generation of a local power spike). (Affidavit, at 5.) See paragraph 21 infra.

10. The Applicant included in his analysis of fuel densification effects, the effects of increased linear heat generation rate and increased radial gap size. (Affidavit, at 5.)

11. The Applicant has included the effects of cladding creepdown in calculating the gap conductance. (Affidavit, at 5.)

12. Cladding creepdown results when a free standing tube is subjected to a high temperature coolant, fast neutron flux and net external pressure for a prolonged period of time. The stresses caused by the external pressure cause the cladding to creep. Creep affects can cause the tube to become oval forming major and minor axes. The minor axis will touch the fuel pellet first and after a longer period of time the major axis will eventually creep onto the fuel pellet. This process is what is known as cladding creepdown. (Affidavit, at 6.)

13. The model formulated from data taken from Saxton, Zorita and San Onofre operating reactors will acceptably predict cladding creepdown for Point Beach Unit 2 fuel rods. (Affidavit, at 6.)

14. The calculated time to collapse for Point Beach Unit 2 facility is greater than 12,000 hours. (Affidavit, at 7.)

15. The Point Beach Unit 1 facility operated over 13,000 hours without observing collapsed rods in the pressurized fuel rods. (Affidavit, at 7.)

16. The Point Beach Unit 2 facility fuel rods are all prepressurized. (Affidavit, at 7.)

17. The Applicant's method for calculating gap conductance is an acceptable and conservative procedure. (Affidavit, at 8.)

18. Fuel densification will affect steady-state operation. (Affidavit, at 9.) See paragraph 21 infra.

19. Fuel densification affects local neutron flux and results in a slightly-shorter fuel stack length, compared to pre-operational length. (Affidavit, at 9.) See paragraph 21 infra.

20. Fuel densification causes a higher initial stored energy and a slower heat release rate during the transient. (Affidavit, at 9.) See paragraph 21 infra.

21. Operating the reactor with the restrictions proposed by the Applicant on control rods and the total peaking factor will properly account for the effects of densification noted in paragraphs 8, 9, 18, 19, and 20. (Affidavit, at 11.)

22. 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 power and a 1.02 power uncertainty factor, would be a peak linear heat generation rate of 15.6 kW/ft. (Affidavit, at 12.)

23. The calculated minimun DNBR for the Point Beach Unit 2 facility for the loss of flow transient will be 1.31. (Affidavit, at 13.)

24. The effect of fuel densification on a locked rotor accident analysis is an increase in cladding temperature to a value of 2285 degrees F. and an amount of $Zr-H_2O$ reaction of less than 2% by weight. (Affidavit, at 13-14.) The staff has concluded that the consequences of this accident have been analyzed and found not to present an undue risk to the health and safety of the public.

24. The following accidents are not affected in any manner by densification of fuel:

 a. steam generator tube rupture

 b. fuel handling accident

 c. waste gas tank failure

 (Affidavit, at 14-15.)

25. The staff has concluded that the Interim Acceptance Criteria for a LOCA have been met by Point Beach Unit 2. (Affidavit, at 15-16.).

26. The staff has concluded that the effects of fuel densification on a postulated control rod ejection accident have been analyzed and found not to present an undue risk to the health and safety of the public. (Affidavit, at 16-20.)

27. The staff has concluded that the effects of a postulated steam line break have been analyzed and found not to present an undue risk to the health and safety of the public. (Affidavit, at 20-21.)

Respectfully submitted,

Joseph Gallo

Joseph Gallo
Assistant Chief Hearing Counsel

Dated at Bethesda, Maryland
this 28th day of January, 1973.

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AFFIDAVIT IN SUPPORT OF
MOTION FOR SUMMARY DISPOSITION

We, the undersigned Donald F. Knuth, Victor Stello, Jr., and Lester S. Rubenstein, being each severally and duly sworn, each for himself deposes and says:

1. We are the co-authors of a document entitled "Technical Report on Densification of Light Water Reactor Fuels," dated November 14, 1972.

2. To the best of our knowledge and belief the Report is accurate and complete.

3. The Professional Qualifications of Messrs. Knuth and Stello are attached to their prior joint affidavit. The Professional Qualifications of Mr. Rubenstein are attached to and made a part of this affidavit.

Donald F. Knuth

Victor Stello Jr.

Lester S. Rubenstein

Subscribed to and sworn before

me this 26 day of Jan., 1973.

Muriel M. Graff

My Commission expires July 1, ~~1973.~~ 1974

LESTER S. RUBENSTEIN

PROFESSIONAL QUALIFICATIONS

My name is Lester S. Rubenstein. I am a metallurgist employed with the U.S. Atomic Energy Commission, Germantown, Maryland. Currently, I am in the Fuels and Materials Branch of the Division of Reactor Development. I am responsible for the technical management of a number of fuel and material development programs concerning nuclear reactors. These programs have dealt with a broad spectrum of material development problems, including light water, fast breeder, tests in space nuclear reactors, as well as isotropic power sources. I have been employed with the Commission since 1967.

My formal eduction consists of a B.S. in Metallurgical Engineering from the University of Arizona in 1953 and an M.S. from the Carnegie Institute of Technology in 1962. While at Carnegie I was a teaching assistant in Mechanical Metallurgy, X-Ray Metallography, and Ferrous and Non-Ferrous Metallography. I was also a research assistant in Mechanical Metallurgy. Graduate Metallurgy courses included: Elasticity, Dislocations, Diffusions, Kinetics, Theory of the Properties of Solids [1,2], Thermodynamics in Solids, Science of Process Metallurgy, Physical Chemistry of Metallurgical Reactions, Alloy Steels [1,2], Recrystallization and Precipitation. I also have graduate courses in Chemistry and Physics. I was also recipient of a Faculty Ford Foundation Followship.

From 1957 to 1959 I was a metallurgical engineer employed with the Westinghouse Electric Corporation, Bettis Atomic Power Plant, Pittsburgh, Pa. I was responsible for the design, completion and publication of results of experimentation dealing with alloys for nuclear application. I determined physical, mechanical and corrosion properties and the effects of heat-treatment impurities and process variables on these properties. During this period I was a technical editor of Zirconium Highlights, an AEC monthly publication on zirconium and zirconium alloys. While employed with Westinghouse I took a course in reactor plant technology (1957-58).

From 1962 to 1963 I was a research metallurgist with TRW, Corporation, Cleveland, Ohio, I investigated the effects of thermal and mechanical treatment on the physical and mechanical properties of refractory metals in alloys. I also studied recovery recrystallization and precipitation kinetics of dispersion and solid solution strengthened alloys.

I was employed as a research metallurgist and group leader with the Lewis Research Center, NASA, Cleveland, Ohio from 1963 to 1967. I was engaged in basic mechanical metallurgical studies which concerned the fiber reinforcement of metallic composite materials from the viewpoint of their elastic and plastic properties, and of their constituent thermodynamic stabilities. I was a group leader in refractory metal research, primarily in tungsten and chromiun base alloys. As a result of my research in this area I was an IR-100 winner, awarded by Industrial Researach Magazine.

I am the author of approximately twenty technical publications, and the holder of several zirconium alloy patents.

I am listed in the American Men of Science.

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AFFIDAVIT IN SUPPORT OF
MOTION FOR SUMMARY DISPOSITION

We, the undersigned Donald F. Knuth and Victor Stello, Jr., being each severally and duly sworn, each for himself, deposes and says:

1. I, Donald F. Knuth, am employed with the United States Atomic Energy Commission, Bethesda, Maryland. My title is Assistant Director for Reactor Safety, Directorate of Licensing. My duties and responsibilities are the planning, directing and supervising of the program activities of the Reactor Systems Branch, Electrical Instrumentation and Control Branch and Operational Safety Branch. My professional qualifications are attached herewith and made a part of this affidavit.

2. I, Victor Stello, am employed with the United Stated Atomic Energy Commission, Bethesda, Maryland. My title is Chief of the Reactor

Systems Branch. My duties and responsibilities are the review and evaluation of reactor thermal hydraulic design, reactor coolant system design and auxiliary system design, and emergency core coolant systems design. My professional qualifications are attached and hereby made a part of this affidavit.

3. During the course of our employment we were called upon to review the information, data and analyses submitted by Wisconsin Electric Power Company and Wisconsin-Michigan Power Company (Applicant) regarding the fuel densification phenomenon as it relates to the operation of the Point Beach, Unit 2 facility (Facility).

5. We have both personally reviewed the data and analyses submitted to the Atomic Energy Commission by the Applicant, and through our general supervision, direction and participation we have been involved in the regulatory staff's (Staff) independent analyses and evaluation of the fuel densification phenomenon, and review of the Applicant's data.

6. The matter of fuel densification and cladding collapse was first evidenced at the R. E. Ginna Nuclear Power Plant during a refueling outage of that reactor in April and May of 1972.

7. The results of an examination of the Ginna reactor and other nuclear facilities with respect to fuel densification and the potential

of cladding collapse are provided in the Technical Report on Densification of Light Water Reactor Fuels, dated November 14, 1972 (hereafter cited as "Report"). The Report is attached hereto as an exhibit.

8. The Report provides the essential elements that are to be used for evaluating effects of fuel densification in a nuclear facility. The essential elements will be discussed further in the following items.

9. The Applicant has provided documentation to demonstrate that he has evaluated the Point Beach Unit 2 facility and that said facility is in compliance with the requirements set forth in the Report.

10. We have concluded that the Applicant has analyzed the effects of fuel densification with respect to the facility in compliance with the requirements set forth in the Report. We have also concluded that based on our review of the information provided by the applicant, and our independent analyses and evaluation of the effects of densification, the plant can be operated at power levels up to 1518 Mwt (100% power) without endangering the health and safety of the public and that the issuance of full power license will not be inimical to the health and safety of the public.

11. The Report was used as a baseline for the Facility evaluation and requires the use of the following assumptions and recognition of the following densification effects. The basis for these procedures and findings are set forth in the Report.

A. The Report specifies the assumption that fuel densification to a value of 96.5% of theoretical occurs immediately when the reactor is operated at substantial power. As indicated in the Report this assumption results in a conservative assessment of densification of fuel. Both the Applicant and Staff have used this assumption in their evaluation.

B. Densification causes a decrease in the pellet length which causes the linear heat generation rate to increase. The Report requires that the increase in linear heat generation rate be calculated upon the assumption that the pellet densifies from an initial density that is lower than the average. The procedures for this calculation is set forth on page 67 of the Report.

C. Decrease in the pellet radius increases the radial clearance gap between the fuel pellet and fuel cladding. This causes a decrease in the thermal conductance which in turn causes an increase of stored energy in the fuel pellet. The Report (page 65) sets forth

the specific procedures for calculating the increase in the size of the radial gap.

D. Decrease in pellet length can also lead to the generation of axial gaps within the fuel column. These gaps will result in increased local neutron flux (i.e., generation of a local power spike). The specific procedures for evaluation of the effect are provided on page 64 of the Report.

E. Page 67 of the Report addresses the effects of increased linear heat generation rate and increased radial gap size. These must be accounted for in the probability analysis used for the power spike calculation. This procedure was adopted by the Applicant.

12. Based on the requirements described in paragraph 11, the Applicant performed Fuel Densification Calculations concerning the facility.

A. The Applicant has included cladding creepdown in calculating the gap conductance. The Report (page 66) allows for including creepdown of the cladding provided that the Staff reviews and accepts the proposed model.

Cladding creepdown results when a free standing tube is subjected to a high temperature coolant, best neutron flux and net external pressure for a prolonged period of time. The stresses caused by

the external pressure cause the cladding to creep. Creep affects cause the tube to become oval forming major and minor axes. The minor axis will touch the fuel pellet first and after a longer period of time the major axis will eventually creep onto the fuel pellet. This process is what is known as cladding creepdown. There are three types of creep that need to be considered in performing this calculation. There are:

i. Thermal
ii. Radiation growth
iii. Fission induced.

Data derived from both out of pile and in pile tests were used to define the three components listed above. The overall formulation of the model was checked against data taken from several operating reactors (i.e., Saxton, Zorita and San Onofre). After a detailed review, the staff concluded that the model will predict the cladding creepdown for Point Beach Unit 2 fuel rods in an acceptable manner.

B. Time to Collapse.

The Applicant performed an analysis which showed that the fuel cladding in the Point Beach Unit 2 Reactor will not flatten in the initial cycle of operation. The procedure for calculating the flattening of fuel rod is similar to the procedure described in

paragraph 12A above. These calculations were performed as required on page 7 of the Report.

Data from operating reactors, (i.e., Ginna, Zorita, NOK and Point Beach, Unit 1) were used in formulating the analytical procedure used to calculate the collapse time for the fuel rod. The model was compared to these data and found to predict a shorter time to collapse than were observed in these operating reactors.

The staff also made calculations of the time for collapse using a computer code entitled BUCKLE. Based upon our evaluation of the Westinghouse analyses and our independent analyses, we have determined that the calculated time to collapse for the Point Beach Unit 2 facility is greater than 12,000 hours. As a point of reference, the Point Beach Unit 1 facility operated over 13,000 hours without observing collapsed rods in the pressurized fuel rods. Hence, our Technical Specification will limit the fuel residence time to 12,000 hours to assure no fuel cladding collapse. If residence times will exceed 12,000 hours, under present operating conditions, the assumption of cladding collapse will be required.

C. Gap conductance.

The applicant has performed calculations to evaluate the effect of fuel densification on the gap conductance. The model that was used by the applicant included consideration of: fuel density, burnup, cold gap dimensions, UO2 thermal expansion and conductivity, the linear heat rating of the fuel and the cold fill gas pressure. The model has been substantiated by comparing the results of the calculations to measurements reported in the literature. Included in the model was the requirement as set forth in the Report for assuming that the fuel densifies immediately. The staff has used an independent evaluation model thru use of a computer program GAPCON. The GAPCON code was verfied by comparison to available experiments. Results of this comparison indicates that the GAPCON calculations predict higher stored energy than was observed in experiments. Based upon our evaluation of the Westinghouse analyses and our independent analyses we have concluded that the values of heat transfer and fuel stored energy for the cases of interest were conservatively calculated.

13. The Effect of Densification of Steady State and Transient Operation.

A. 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 a concomitant result of a slightly - shorter fuel stack length. An additional effect is seen 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.

As a result of calculations involving the effects of fuel densification, the Point Beach Unit 2 reactor will be operated with more restricted control rod limits and with a reduced design total peaking factor than would be permitted without consideration of the effects of densification. 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 control rods.

The effects of densification on power density distributions have been calculated with models in conformance with those discussed in Section 4 of the Report. The primary calculations used the models and numerical data of the Westinghouse power spike model

as described in Appendix E of the Report except that the initial nominal density used was 0.918 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 the Report. These calculations by the applicant included 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 is a not-quite-linear function of core height and reaches a maximum of about 1.18 at the top of the core.

Parallel and independent calculations have also been done by our Brookhaven National Laboratory consultants. 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. We have reviewed these BNL calculations and their bases and we have adopted them for use in our evaluation.

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 traditionally formed the

basis for axial flux distribution control and have been reviewed for a number of Westinghouse reactors. By maintaining the axial offset within reasonable limits (but more restrictive than has previously been required) and then combining with maximum radial peaking and uncertainly factors, the total peaking factor F_Q^T would be expected to be maintained under 2.62.

This decrease in F_Q^T from the 2.80 value previously used in Point Beach analyses and in the Technical Specifications, even considering the gap peaking augmentation, 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 less than 1.55. However, the previous Point Beach design axial peaking factor was about 1.7, 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 .918 of theoretical to a final value of .965, a measured average stack height of 144.5

inches; and thermal expansion of the pellet. The result is an increase of 1.8 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 the Report. The variation in density is convoluted with the power spike model and the calculated density changes to over-power 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 1.5 percent for thermal overpower calculations and 0.7 percent for the LOCA calculations. Our BNL consultants have also done parallel calculations on this convolution process and have found consistent results. We have reviewed these BNL calculations and their bases and we have adopted them for use in our evaluation.

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 power and a 1.02 power uncertainty factor, would be:

$$\text{Peak kW/ft} = 5.7 \times 2.62 \times 1.018 \times 1.007 \times 1.02 = 15.6 \text{ kW/ft.}$$

B. Loss of Flow Transient

The applicant has reanalyzed the transient that would result from a loss of electrical power to the primary coolant pumps. The re-analysis was performed to account for the effects of fuel densification. The previously calculated (FSAR) minimum DNBR during the transient was 1.55. The applicant proposed to operate the Point Beach Unit 2 facility with a decreased inlet temperature of 2 degrees F so that the calculated minimum DNBR with densification will be 1.31. The staff has concluded that the loss of flow transient is acceptable since the minimum DNBR exceeds our acceptance criteria of 1.30. The staff has also performed an independent calculation of this transient using a version of the COBRA code (i.e., COBRA III-c). Based upon these calculations, we agree that the 2 degree reduction in inlet temperature assures a minimum DNBR greater than 1.3.

C. Locked Rotor Accident Analysis.

The postulated accident is initiated by assuming an instantaneous seizure of one reactor coolant pump rotor. As a consequence of this assumption the reactor coolant flow rate is decreased rapidly so that core flow is reduced to less than one-half of its normal

full flow value within a few seconds. The applicant has reanalyzed this accident to account for the effects of fuel densification. The results of the reanalysis indicate that the peak clad temperature is increased to a value of 2285 degrees F. with the amount of ZR-H_2O reaction less than 2 percent by weight. The staff has performed independent calculations of this accident using the COBRA-III-c code. These calculations showed lower temperatures than the Applicant's calculations and indicate that the Applicant's analyses are conservative. In view of the fact that the external coolant pressure is high and that the high temperatures occur for short times, we have concluded that the consequence of this transient is acceptable.

14. Accident Analysis.

A. The following accidents were reevaluated

1. Loss of Coolant Accident
2. Control Rod Ejection
3. Steam-line rupture
4. Primary coolant pump seizure (see paragraph 13 above)
5. Steam generator tube rupture
6. Fuel handling accident
7. Waste gas tank failure

Results of the first three accidents listed above will be discussed below. The 4th accident as listed has already been discussed in

paragraph 13 above. The remaining 3 accidents will not be affected by fuel densification. The reason the consequences of fuel densification do not impact on these accidents is mainly that the consequences of these accidents are limited by Technical Specification limits on radioactive concentrations. Specifically, the gas decay tank rupture accident and the steam generator tube rupture are based upon the Technical Specification limit of coolant activity. The fuel handling accident is based on the assumption that all of the rods in a dropped assembly are mechanically failed upon dropping. Hence, as noted previously, this accident is not affected by fuel densification.

B. LOCA

The loss of coolant accident was reevaluated by the Applicant using the analyses techniques specified in the Interim Acceptance Criteria. The effects noted previously in paragraph 2 were incorporated in the detailed calculations. As noted previously the staff is in agreement with the input parameters as recalculated by the Applicant. The results of the Applicant's revised LOCA analyses show that the Interim Acceptance Criteria values are

met provided that the peak linear heat generation rate is below 15.9 Kw/ft. with an initial density of .918, and 15.6 Kw/ft with an initial density of .908. Appropriate Technical Specification limits will be used to ensure that reactor operation will be maintained within the envelope of parameters used in the analyses. The 1800 F limit for collpased rods were not applied in this analysis since the Applicant has demonstrated that the fuel rods in the Point Beach Unit 2 facility will not be collapsed during the proposed period of operation (12000 effective full power hours). See 12.B. Supra.

C. Control Rod Ejection

As a result of changes in the fuel from the densification process and changes in the operational limits of control rods as part of the more restrictive control of power distribution, certain aspects of the analyses of the control rod ejection transient have changed. One would expect relatively little or no changes in the important kinetics parameters so that for a given control rod reactivity worth there would be little change in the core transient power history. However, since limits on control bank insertion have changed, there has generally been a decrease in rod worth in

a given reactor state available for an ejection transient, thus tending to decrease transient energy production. For the hot spot calculations 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 are those developed since the CHIC-KIN code calculations presented in the FSAR, and are described and compared with more exact calculations in the rod ejection with the TWINKLE Code described in Westinghouse Tropical Report, WCAP-7588. As with the CHIC-KIN calculations, 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 (WCAP-7908) using appropriate peaking factors and power histories from the TWINKLE calculation. These codes,

when using similar parameters, give similar results to CHIC-KIN calculations.

The regulatory staff and our consultants at Brookhaven National Laboratory have examined the calculation models and data, and BNL has made some check calculations using what they believe to be appropriate input data and their own developed codes. We have reviewed these BNL calculations and their bases and we have adopted them for use in our evaluation. Based on this review of the Applicant's calculations, we have concluded that the Applicant's methods and input are generally satisfactory and conservative. The two stage method of calculating, 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 placed on top of initial condition maximum peaking factors of 2.70, with the densification factor included along with a full 2 sigma initial density variation allowance. The method and codes, especially TWINKLE is its one dimensional version, are very similar to those developed and used at BNL. The nuclear input date, including the reactivity coefficients, weighting factors, control rod reactivity worths, and peaking factors, are generally conservative. In particular, the rod

worths and peaking factors for the configurations of interest have been checked in the Point Beach 1 startup tests.

The transients are generally insensitive to heat transfer parameters and in particular to gap conductances. 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 larger gap, and assuming very conservatively that design peaking conditions occur at the same spot as the maximum transient peaking, gives a higher initial heat content of the hot fuel pellet as a base for the hot spot calculation. The 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 EOL case would have lower energies than the BOL case.

The results of the transient analyses are within reasonable limits. The peak average fuel pellet energy content is under 170 cal/gm, and in fact center line melting temperatures are not reached. The peak clad temperatures are below 2400°F (above 2300°F for only a few seconds at the hot spot). The total number of fuel pins calculated to be in DNB remains under the 15 percent presented in the FSAR.

D. Steam Line Break

The applicant stated, that the minimum DNBR for the steam line break accident could be reduced from the >1.80 value reported in the FSAR to a value > 1.60 by including densification effects. The reactivity transient is unchanged from that presented in the FSAR. The acceptance criteria in the FSAR were:

(1) With a stuck rod and minimum engineered safety

features the core remains in place and essentially 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, Δ T, 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 and results of the steam line break is found in Sec. 14.2.5 of the FSAR. The staff concludes that densification would have only a minimal impact on the consequences of a steam line break.

Donald F. Knuth

Victor Stello Jr.

Subscribed to and sworn before me this 16th day of January, 1973.

Helen Hershey

My Commission expires July 1, 1974, ~~1973.~~