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

MAR 27 1980

Mr. Thomas M. Anderson, Manager
Nuclear Safety Department
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Anderson:

SUBJECT: REVIEW OF WCAP-8720, "IMPROVED ANALYTICAL MODELS USED IN WESTINGHOUSE FUEL ROD DESIGN COMPUTATIONS"

Enclosed is a copy of our March 13, 1980 letter to Commonwealth Edison Company that transmits license Amendment Nos. 53 and 50 for Zion Station Unit Nos. 1 & 2, respectively, and our safety evaluation of these amendments. These amendments allow an increase in the LOCA peaking factor limit, based on removal of some of the conservatism from the PAD computer code as described in WCAP-8720. The enclosure includes the NRC Core Performance Branch safety evaluation of the reduction of conservatism in the Westinghouse PAD computer code.

Based upon our review, we have concluded that the changes described in our enclosed safety evaluation are acceptable for use with the Westinghouse PAD-3.3 code in plant safety analyses. This acceptance is limited to the current version of the code as approved by the staff in its February 9, 1979 letter to you for application in LOCA analyses.

Although our evaluation has been made as the basis for license amendments for Zion Units 1 and 2, it has been used also as the basis for similar license amendments for Turkey Point Units 3 and 4, and is also applicable to any plant LOCA analysis which uses the Westinghouse PAD code. You should incorporate these changes and the enclosed Core Performance Branch safety evaluation report in the revised approved issue of WCAP-8720. In the meantime should you incorporate these changes in future safety analyses, please reference the enclosed license amendments for the Zion Station including the Core Performance Branch safety evaluation report.

Sincerely,

John F. Stolz

John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management

Enclosure:
March 13, 1980 Letter from A. Schwencer (NRC) to D. Peoples (Commonwealth Edison Co.), transmitting license amendments and safety evaluation for Zion Station

RD-8-2
W

cc w/o enclosure: See next page

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Mr. A. Ball
Westinghouse Electric Corporation
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 13, 1980

Docket Nos. 50-295
and 50-304

Mr. D. Louis Peoples
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Peoples:

The Commission has issued the enclosed Amendment No. 53 to Facility Operating License No. DPR-39 and Amendment No. 50 to Facility Operating License No. DPR-48 for the Zion Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated March 22, 1979, as supplemented May 3, 1979, and January 25, 1980.

These amendments modify the Technical Specifications, Appendix A to the licenses, to increase the allowable LOCA peaking factor from 1.86 to 1.93 based on an ECCS reanalysis.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 53 to DPR-39
2. Amendment No. 50 to DPR-48
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

Dupe approx. 25 pages

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-295

ZION STATION UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 53
License No. DPR-39

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated March 22, 1979 as supplemented on May 3, 1979 and January 25, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-39 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 53 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

W. P. Gammill

William P. Gammill, Acting Assistant
Director for Operating Reactor Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 13, 1980

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-304

ZION STATION UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
License No. DPR-48

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated March 22, 1979 as supplemented on May 3, 1979 and January 25, 1980, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-48 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 50 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

W. P. Gammill

William P. Gammill, Acting Assistant
Director for Operating Reactor Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 13, 1980

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. DPR-39

AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. DPR-48

DOCKET NOS. 50-295 AND 50-304

Revise Appendix A as follows:

Remove Pages	Insert Pages
45	45
63a	63a

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.2.2 Power Distribution Limits

A. Hot Channel Factor Limits*

1.1 At all times, except during physics tests at $\leq$75% rated power**, the hot channel factors defined in the bases must meet the following limits:

Units 1 and 2

$$F_Q(Z) \leq \left[F_Q(Z)\right]_L = \begin{Bmatrix} 1.93/P \times K_1(Z), \text{ for } P > .5 \\ 3.86 \times K_1(Z), \text{ for } P \leq .5 \end{Bmatrix}$$

and $F^N_{\Delta H} \leq 1.55 \left[1+0.2(1-P)\right] \times RBP$,

where:

$\left[F_Q(Z)\right]_L = F_Q(Z)$ limit;

1.93 = F_Q constant (LOCA limiting value);

P = fraction of rated power at which the core operated during F_Q and $F^N_{\Delta H}$ measurement;

$K_1(Z)$ = factor from Figure 3.2-9 selected at the core elevation, Z, of the measured F_Q;

4.2.2 Power Distribution

A. Hot Channel Factor Limits

1.1 Following initial core loading and at a minimum of regular effective full power monthly intervals thereafter, power distribution maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of this specification are satisfied

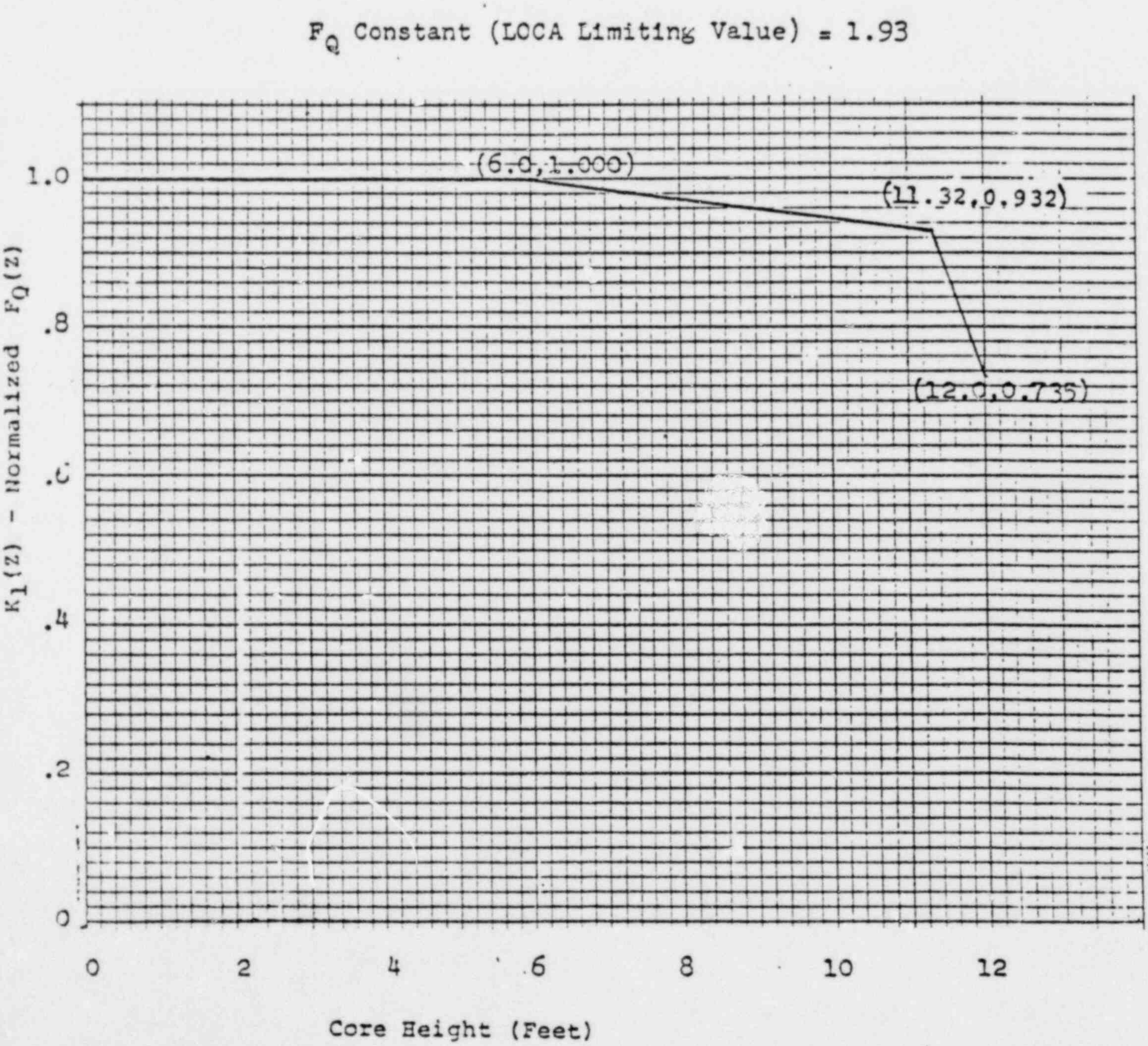
Following initial loading and each subsequent reloading, a power distribution map using the Movable Detector System, shall be made to confirm that power distribution limits are met, in the full power configuration before a unit is operated above 75% of rating.

*The hot channel factors above are defined for a period not to exceed the predicted minimum time to collapse exposure levels for each fuel region as referenced in the bases.

**During physics tests which may exceed these hot channel factor limits, the reactor may be in this condition for a period of time not to exceed eight hours continuously.

Amendment No. 53, Unit 1
Amendment No. 50, Unit 2

Figure 3.2-9 Hot Channel Factor Normalized Operating Envelope for Units 1 and 2



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 53 TO FACILITY OPERATING LICENSE NO. DPR-39

AND AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. DPR-48

COMMONWEALTH EDISON COMPANY

ZION STATION, UNITS 1 AND 2

DOCKET NOS. 50-295 AND 50-304

Introduction

By letters dated March 22, 1979, as supplemented May 3, 1979 and January 25, 1980. The Commonwealth Edison Company (the licensee) requested an amendment to Facility Operating License Nos. DPR-39 and DPR-48 for the Zion Station Unit Nos. 1 and 2. The application was in support of a request to modify the Technical Specifications, Appendix A to the licenses, to increase the allowable LOCA peaking factor from 1.86 to 1.93 based on an ECCS reanalysis. The letter contains a LOCA analysis and proposed Technical Specification changes in connection with the operation of Units 1 and 2 with 1 percent of steam generator tubes plugged and a peaking factor F_Q of 1.93.

The changes to the Technical Specifications requested by the licensee are the following:

(a) Change of F_Q to 1.93 for plant operation with 1 percent of steam generator tubes plugged.

(b) Change of the Hot Channel Factor Normalized Operating Envelope for Units 1 and 2 (Fig. 3.2-9).

Since the limiting value of F_Q is below the level at which the excore detectors could provide reliable readings and because the "18 case FAC analyses" performed for both units indicated that the maximum predicted F_Q exceeded the LOCA determined limits, the licensee is required either to operate the plant with the augmented power distribution surveillance or at the suitably reduced power levels.

Evaluation

The licensee has provided an evaluation of the performance of Emergency Core Cooling System (ECCS) for Units 1 and 2 corresponding to the hot channel peaking factor value of F_Q=1.93 and assuming a steam generator tube plugging level of 1 percent, a plant specific initial pellet temperature and a removal of 65°F fuel temperature conservatism in the PAD 3.3 fuel performance evaluation code.

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In performing analyses of the Loss of Coolant Accident, the Westinghouse method starts with a calculation of the volumetric average fuel temperature. For conservatism, an additional temperature increase is added to the calculated value. This increase consists of two components, one of which is a 65°F margin for modeling uncertainties. This is the claimed difference between the predicted value of the volumetric average temperature and the measured value. The licensee has requested removal of this value from the LOCA calculations on the grounds that adequate conservatism remains.

The licensee provided comparisons of the PAD 3.3 code with appropriate data on fuel pellet temperatures as measured with thermocouples located in the fuel. This data included fuel rods having dimensions, fuel densities, powers and burnups in the range of interest. These comparisons demonstrated that even without the 65°F modeling uncertainty, the PAD 3.3 code was sufficiently conservative.

In addition the staff performed an independent calculation which utilized studies on fuel temperature uncertainties by Battelle Pacific Northwest Laboratories (Reference 2), and EG&G Idaho Inc. (Reference 3).

Based on these estimates of fuel temperature uncertainty and our calculations, we conclude that the PAD 3.3 computer code meets the criterion of bounding a large portion of the expected volumetric average fuel temperatures when using nominal input conditions without the 65°F. We also find that there is sufficient remaining margin of conservatism to bound the expected uncertainty in other state of the art fuel performance computer codes.

It is therefore acceptable to delete the 65°F from the PAD 3.3 computer code. The detailed description of our evaluation is included in Annex 1.

In addition to the request to drop the 65°F model uncertainty, the licensee has requested the use of as-fabricated fuel parameters applicable to Zion fuel rather than more bounding values usually used by Westinghouse in LOCA analyses. The staff has reviewed the statistical methods and assumptions which the licensee will use for determining the dimensions to be used in the LOCA analyses and finds these acceptable. The licensee has proposed taking credit for a 20°F decrease in the volumetric average temperature, due to the difference in assumptions about the as fabricated fuel parameters. This is a conservative estimate of the expected change.

The LOCA analysis was performed using the February 1978 version of the Westinghouse Evaluation Model (Reference 2) which was reviewed and approved by us. It was performed for a spectrum of three double ended cold leg guillotine breaks (DECLG) with discharge coefficients of C_D=0.6, 0.8 and 1.0. The input parameters assumed in the analysis are listed below:

Core Power: 102 percent of 3250 MWt (rated power)
Peak Linear Power: 102 percent of 13.086 kw/ft
Peaking Factor: 1.93
Accumulator Water Volume: 818.65 cu ft/each

The results of the analysis indicate a peak cladding temperature of 2157°F, a maximum local Zr-water reaction of 6.71 percent and a total Zr-water reaction of less 0.3 percent, all these values occurring at the critical break size of C_D=0.8.

The licensee has performed the "18 case FAC analyses" for Cycle 4 in Units 1 and 2 (Reference 1) because the limiting peaking factor in the LOCA analysis was below the value for which the excore detectors could give reliable results. The results of these analyses have indicated that for both units, the predicted maximum peaking factor exceeds the limiting value of F_Q. The licensee is therefore required either to limit power to the rated power multiplied by the ratio of 1.93 divided by the predicted peaking factor or to operate the plant at higher power levels with augmented axial power distribution surveillance in order to ascertain that the peaking factor would not exceed the limiting value of 1.93. This requirements could be lifted any time during plant operation if the licensee demonstrates by the "18 case FAC analysis" that the maximum predicted F_Q is within the LOCA determined limit.

Conclusions

Based on the review of the submitted documents, we conclude that the results of the LOCA analysis performed with F_Q=1.93 are conservative relative to the 10 CFR 50.46 criteria. We consider the resultant changes to the Technical Specifications acceptable for operating Units 1 and 2 with a maximum 1 percent of steam generator tubes plugged.

References

1. Letter from Cordell Reed (Commonwealth Edison) to H. R. Denton (NRC), dated March 22, 1979.

2. M. E. Cunningham, "Stored Energy Calculation: The State of the Art," Battelle Pacific Northwest Laboratories, PNL-2581, May 1978.

3. D. R. Coleman, E. T. Laats and N. R. Scofield, "FRAP-S3: A Computer Code for the Steady-State Analysis of Fuel Rods, Volume 2, Model Verification Report, EG&G Idaho, Inc. Report TFBP-TR-228, August, 1977.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: March 13, 1980

SAFETY EVALUATION
OF THE REQUEST FOR
REMOVAL OF CONSERVATISM
IN THE WESTINGHOUSE PAD COMPUTER CODE

MARCH 1980

CORE PERFORMANCE BRANCH

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

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SAFETY EVALUATION OF THE REQUEST FOR REMOVAL OF CONSERVATISM IN THE WESTINGHOUSE PAD COMPUTER CODE

1. Introduction

The thermal conditions within the fuel of a light water reactor during its normal lifetime must be described in the safety analysis for each reactor. The fuel temperatures are used as initial conditions in describing the response of the reactor to a number of hypothetical transients and accidents, such as the loss-of-coolant accident (LOCA).

Commonwealth Edison Company (Com-Ed), the owner of Zion Station 1 and 2, has requested a license amendment for these two plants to increase the allowable LOCA peaking factor limit from 1.86 to 1.93. This change is based on reanalysis of the loss-of-coolant accident wherein reduced initial fuel temperatures are assumed. The reduced fuel temperature are a result of the removal of some of the margin of conservatism in the fuel performance code used in the analysis. Both Commonwealth Edison and the code developer, Westinghouse Electric Corporation, believe that the remaining conservative features of the code are adequate for the safety analysis.

A review of the proposed revisions to the fuel code, PAD-3.3, and our evaluation of these changes, are presented in the following sections. The discussion will consist of a technical review of the submittal, comparision of the Westinghouse code with a traditional staff audit code, and the development and application of a new criterion for margin of conservatism in codes of this type. All of these methods lead us to conclude that the remaining conservative features of the code are adequate for safety analysis.

2. Request for Reduction in Conservatism

On March 22, 1979, Commonwealth Edison Company requested (Ref. 1) a license amendment to increase the allowable LOCA peaking factor limit. This request was based on revised analysis of the emergency-core-cooling-system (ECCS) to meet 10 CFR 50.46 (Ref. 2) requirements. The LOCA peaking factor, also known as the limiting heat flux hot channel factor or the limiting F_Q, is defined as the maximum local heat flux on the surface of a fuel rod divided by the core average fuel rod heat flux. The maximum allowable local heat flux is calculated in the plant safety analysis, usually for the loss-of-coolant accident. For most reactors, the LOCA peaking factor limits the operational flexibility or power maneuvering capability, but not the total power generating capability, of the plant. In the case of Zion Station Unit 1, there is some evidence (Refs. 3-6) that the LOCA peaking factor may also limit the power production capability of the plant.

The potential peaking factor limitation caused Com-Ed to reanalyze the loss-of-coolant accident for the Zion facility. An increase in the allowable LOCA peaking factor from 1.86 to 1.93 was projected based on reduced fuel temperatures. The reduced temperatures were calculated with a modified version of Westinghouse PAD-3.3 code (Ref. 7).

We reviewed the Com-Ed submittal and requested (Refs. 8-9) additional information with regard to the proposed changes. Com-Ed responded to these requests with additional information (Refs. 10-11), which has also been reviewed. The details of the Com-Ed proposal are discussed in the next section.

3. Conservatism in the Westinghouse PAD Computer Code

The Westinghouse PAD computer code iteratively calculates the interrelated effects of temperature, pressure, cladding elastic and plastic deformation, fission gas release, and fuel densification and swelling as a function of time and power density. The most recent version of the code, PAD-3.3, was described by Westinghouse in a Licensing Topical Report (Ref. 7). This report was previously reviewed and approved by the NRC staff (Ref. 12).

As part of the emergency-core-cooling-system evaluation requirements (Ref. 13) for plant safety analysis, the PAD code utilizes a number of conservatisms in the prediction of fuel temperatures. These include conservative inputs to the code, conservatisms within the code itself, and conservative margin applied to the code output. Some of these conservatisms, such as the 102% of maximum allowable power that is input to the code, are specifically required by the regulations. Other conservatisms, such as the conservative margin applied to the code output, are not specifically required by law. These additional conservatisms were submitted by Westinghouse as part of earlier safety analysis reports or were required by the NRC staff during the review process. It is the second category, those conservatisms which are not specifically required by law, which is the subject of this report.

The derivation and application of conservative margin applied to the PAD code output have been described previously (Ref. 14). The margin is due to uncertainties in the fuel temperature predictions due to manufacturing variations. The parameters considered include:

a) cladding inside diameter
b) pellet outside diameter
c) pellet density
d) pellet sintering temperature

Variations in the first two parameters affect the calculated fuel-to-cladding gap size. Variations in the last two parameters affect the densification behavior of the fuel.

For safety analyses, nominal design values of the above parameters are used as input to the PAD code and allowance for manufacturing variations are then added. The allowance for each of these four parameters is determined by using a bounding value for each quantity. The allowance is simply the difference between the nominal and the bounded input code prediction. As an example, the PAD code will predict higher temperatures if an upper bound cladding inside diameter is used as input rather than the nominal design value of this parameter. The difference between the two predictions, in degrees Fahrenheit, is the allowance for manufacturing variations in cladding inside diameter. The bounding value for each input parameter is derived on a normally applied 95% probability basis at a 95% confidence level. Each allowance is calculated at the time in life when fuel temperatures are maximum and at a power level of 15 kW/ft, the approximate LOCA limit.

The allowance calculated for each of the four input parameters is statistically combined with the others to form the total fabrication uncertainty. To the total fabrication uncertainty, a second, so-called model uncertainty, of 65°F is added. It is the Westinghouse position that this additional 65°F margin was added to ensure that the best-estimate model predictions would bound most of the experimentally measured fuel temperature data. The best estimate model is the same PAD-3.3 code using nominal input values and no explicit internal or external code conservatisms. At the time the 65°F margin was accepted, it was the staff's opinion that this margin was used to account for uncertainties not explicitly considered elsewhere. Both Commonwealth Edison and Westinghouse believe that the 65°F margin is already considered elsewhere in the analysis and that the remaining conservative features of the code are adequate.

In addition to request to drop the 65°F model uncertainty, Commonwealth Edison has requested that the use of as-fabricated, rather than as-designed, fuel conditions be allowed in the safety analysis. Generally, this would permit the use of nominal values of cladding inside diameter, pellet outside diameter, pellet density, pellet sintering temperature and their respective uncertainty allowances based only on the fuel supplied to each Zion Unit, rather than the entire Westinghouse product line. In practice, the dimensional parameters (i.e., cladding I.D. and fuel O.D.) and their uncertainties do not change significantly. As a result, the use of as-fabricated fuel conditions affects only the pellet density, pellet sintering temperature and their respective uncertainties.

A complicating feature of the request is a change in the current product line analysis by Westinghouse. For the current Westinghouse fuel designs, the fuel is sintered in such a manner that the statistical lower bound of the actual sintering temperature is always above the sintering temperature used as input to the PAD code for safety analyses. This means the code input value is lower than virtually all of the sintering temperatures used in manufacturing the fuel. As a result, the code predicts more densification than is expected, but the allowance for uncertainty in sintering temperature becomes zero.

The Com-Ed request for the use of as-fabricated values would cause the analysis to revert back to its original form. Namely, the use of a nominal sintering temperature and a non-zero allowance for uncertainty in this temperature. Because these values would be based only on fuel in each Zion Unit, the result is a higher sintering temperature, an almost unchanged total fabrication uncertainty, and a reduction in average fuel temperature predictions of approximately 20°F.

3.1 Design vs. As-Fabricated Conditions

The request for the use of as-fabricated, rather than as-designed fuel parameters is fundamentally sound. The LOCA analysis should, in general, apply to a specific plant rather than a generic design. We therefore agree with the proposed use of plant-specific input conditions.

We have, however, examined the proposed change to determine whether the approach is indeed applicable to Zion and is statistically valid. Commonwealth Edison has stated that they will use generic fuel parameters (i.e., sintering temperature and pellet density) which are bounding to the actual values determined for each Zion reload core but not for the entire Westinghouse product line. These parameters will be selected on the basis of previous fuel reload data and anticipated future reload data. However, the fuel for each future reload will be measured to ensure that it meets the acceptance criteria for all fuel batches used in that reload. The statistics for the reload region are based on the complete set of data for all batches. At each Zion reload review, Commonwealth Edison will verify that the specific fuel parameters are bounded by the Zion generic fuel parameters.

Westinghouse, the fuel supplier for Zion Units 1 and 2, has also described (Ref. 11) the statistical methods on which the change will be based. For Zion Unit 1, Region 7, the data include over 28,000 density measurements and the sintering temperature for each sintering boat used in manufacturing the fuel for the region (3.3 million pellets). The large number of observations used in the process is well in excess of the levels required for proper statistical analysis. We conclude that the request to use as-fabricated, rather than as-designed, fuel parameters is acceptable.

3.2 Fuel Model Conservatism

As discussed previously, safety analyses with the PAD code currently applies a conservative margin to the fuel temperature predictions of the code. This margin is composed of a component due to fabrication uncertainties (cladding and pellet diameters, pellet density and sintering temperature) and the 65°F component termed the model uncertainty. Westinghouse has stated that the final component was added to ensure that the best estimate code predictions would bound most of the experimentally measured fuel temperature data. Because the evaluation model or conservative version of PAD, rather than the best estimate version, is used in safety analyses, it is not necessary (from the regulatory standpoint) that the best estimate version bound any data. It is our position, rather, that whichever version of PAD is used in plant safety analyses should conservatively predict fuel temperatures. This requires an explicit or implicit consideration of uncertainties, including uncertainties in the models used.

Both Commonwealth Edison and Westinghouse believe that the 65°F margin is already considered elsewhere in the analysis. They further state that the remaining conservative features of the code are adequate for ECCS analysis. The bases for this statement are:

1. The "best estimate" version of PAD bounds the majority of the experimental fuel temperature data considered.

2. The evaluation model or safety analysis version of PAD always predicts fuel temperatures greater than or equal to those predicted by the "best estimate" version of the code.

3. The limiting time in life for ECCS analyses is such that the conservative version of PAD always predicts fuel temperatures greater than those predicted by the best estimate version.

4. The overall conservatism in the calculation of fuel temperature, the conservative application of those temperatures in the LOCA analysis, and the conservatism associated with the overall LOCA/ECCS evaluation warrant the elimination of the 65°F fuel temperature model uncertainty.

We will discuss each of these bases individually.

1. Westinghouse has submitted the results of a number of comparisons between the best-estimate predictions of the PAD-3.3 code and experimentally determined fuel centerline temperature data. The data were taken from the Halden Heavy Boiling Water Reactor (AE-318, HPR-80 and IFA-226) and the Materials Testing Reactor (WAPD-228). Westinghouse selected these data because they represent helium-filled cylindrical fuel rods near beginning-of-life with densities and gap sizes typical for the standard product line. We agree that these data are representative of the current Westinghouse product. However, it is clear that the "best-estimate" version of PAD-3.3 is not a best-estimate code at all, but a conservative one. In other words, even with nominal input values and the removal of explicitly conservative models within the code, PAD-3.3 tends to overpredict the experimental data.

2. When the PAD code is used in safety analysis, certain evaluation model options are activated. These include the fuel densification, gap conductance (gap closure) and cladding creep models. Because all of these processes are time-dependent, the difference between the conservative and best-estimate options is zero at time zero. At all non-zero exposures, the evaluation model options do result in higher fuel temperature predictions as stated by Westinghouse.

3. Appendix K of 10 CFR 50 states that "the steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the

highest calculated stored energy.)" (Ref. 13) In the Westinghouse safety analysis, the limiting burnup occurs shortly after beginning-of-life, at the point of maximum fuel densification. If the best-estimate version of PAD is used, however, the burnup at which maximum burnup occurs is different than that calculated with the evaluation model options. For the best-estimate code, this burnup is at, rather than shortly after, beginning-of-life. We believe that the margin of conservatism between the best-estimate and evaluation model versions of PAD is misleading when measured at any specific burnup. We conclude that the difference between the maximum temperatures predicted by each version of the code is a more appropriate basis of comparison. The margin calculated in this fashion is significant, but less than that assumed by Westinghouse.

4. The overall conservatism in the calculation of fuel temperatures and the overall conservatism in the LOCA analysis have not been rigorously demonstrated by Commonwealth Edison or Westinghouse. Only individual details of the analysis, such as the impact of LOCA peaking factor uncertainties on fuel temperatures, have been described. The impact of the proposed modification on the overall analyses has not been addressed. As a result, we are unable to consider the overall conservatism in the LOCA analysis as a basis for the removal of the 65°F model uncertainty in PAD-3.3. This conclusion is discussed further in Section 4.1 of this evaluation.

We conclude that Westinghouse has shown the evaluation model or safety analysis version of PAD-3.3 to be conservative in calculating steady-state fuel temperatures for LOCA analysis. This alone is not sufficient to demonstrate that the degree of conservatism, when used in LOCA analysis, is sufficient to warrant elimination of the 65°F model uncertainty. We will discuss this further in Section 4.

3.3 Audit of the Revised Analysis

The historical method of regulatory review of fuel codes may be divided into three areas: (1) establishing the technical validity of the methods and supporting data described by the applicant, (2) verifying the existence of conservatism in the analyses, and (3) determining the degree of conservatism relative to traditionally accepted audit codes. The technical validity of the methods used in the Westinghouse PAD-3.3 code was established in Sections 3.1 and 3.2 of this report as well as earlier staff evaluations (e.g., Ref. 12). The fact that the code is indeed conservative without the 65°F model uncertainty was also established in Section 3.2 of this report. The remaining item, a comparison of the PAD-3.3 predictions with a traditionally accepted audit code, is presented in this section of the report.

The GAPCON-THERMAL-2 code (Refs. 15 and 16) is one of a series of fuel thermal performance codes developed by Battelle Pacific Northwest Laboratories for the Core Performance Branch of the Nuclear Regulatory Commission. Since 1975, it has been used by the staff to audit vendor fuel code submittals, including the Westinghouse PAD-3.3 code. GAPCON-2 predicts fuel temperatures, fuel densification and swelling, fission gap release and other fuel conditions as a function of time and power in a fashion much like that of PAD-3.3. GAPCON-2 also has a number of conservative model options similar to PAD.

In order to audit the proposed modifications to the PAD-3.3 code, a current version of the GAPCON-2 code was used to calculate (Ref. 17) volume average fuel temperatures as a function of burnup for the Westinghouse 15x15 fuel design used in Zion Unit 1. The results of these calculations are shown in Figure 1. The two lines represent the best-estimate and conservative predictions for GAPCON-2. Figure 2 of Ref. 11 is similar, showing (1) the best estimate PAD-3.3 prediction, (2) the conservative PAD-3.3 prediction, (3) the conservative

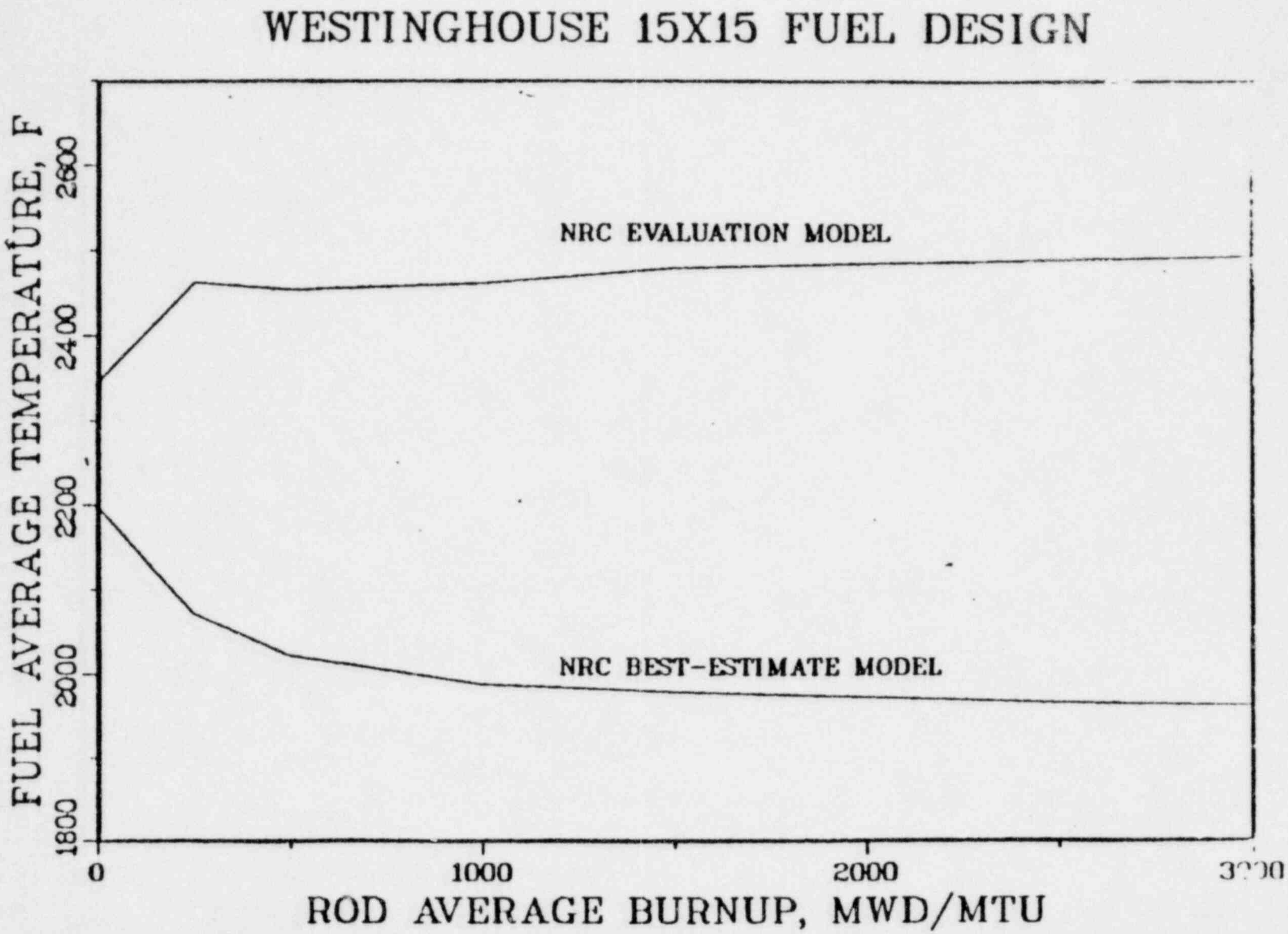
WESTINGHOUSE 15X15 FUEL DESIGN
NRC EVALUATION MODEL
NRC BEST-ESTIMATE MODEL
FUEL AVERAGE TEMPERATURE, F
1800
2000
2200
2400
2600
0
1000
2000
3000
ROD AVERAGE BURNUP, MWD/MTU

Figure 1

[illegible] prediction plus the margin for fabrication uncertainties, and (4) the conservative PAD-3.3 prediction plus the margin for fabrication uncertainties <u>and</u> the 65°F model uncertainty. The fourth Westinghouse curve (the highest) is representative of the current LOCA analysis. The third Westinghouse curve is the version requested by Commonwealth Edison. All calculations (both GAPCON and PAD) were made at a local linear power of 15 kW/ft, approximately the LOCA limit.

A number of conclusions can be drawn in examining these two figures. First, the current (highest) Westinghouse curve is higher than all of the other curves, including the conservative GAPCON-2 prediction traditionally accepted for audit of LOCA analyses. It is not until much higher burnups are reached that the Westinghouse prediction is exceeded by the conservative version of GAPCON-2. The change at higher exposures is due to the effect of cladding creepdown, which is considered in PAD-3.3, but not in GAPCON-2. The second observation to be made from these figures is that all of the PAD predictions are higher than the best-estimate version of GAPCON-2. This confirms our earlier conclusion that even the "best estimate" version of PAD-3.3 is not best-estimate at all, but conservative. There is additional evidence, not presented here, that even the best-estimate version of GAPCON-2 is conservative with respect to the data.

In order to obtain a more representative comparison between these two figures, a second set of results were generated in which cladding creepdown (which is already considered in PAD) was included in the GAPCON-2 predictions. The creepdown values used by GAPCON-2 were generated with the Zircaloy creep model from a second code called BUCKLE (Ref. 18). The results are shown in Figure 2. The revised results are very similar to those shown previously except both GAPCON calculations exhibit a significant decrease in fuel temperatures as a function of burnup. The conservative versions of both GAPCON-2 and PAD predict rising fuel temperatures from

WESTINGHOUSE 15X15 FUEL DESIGN

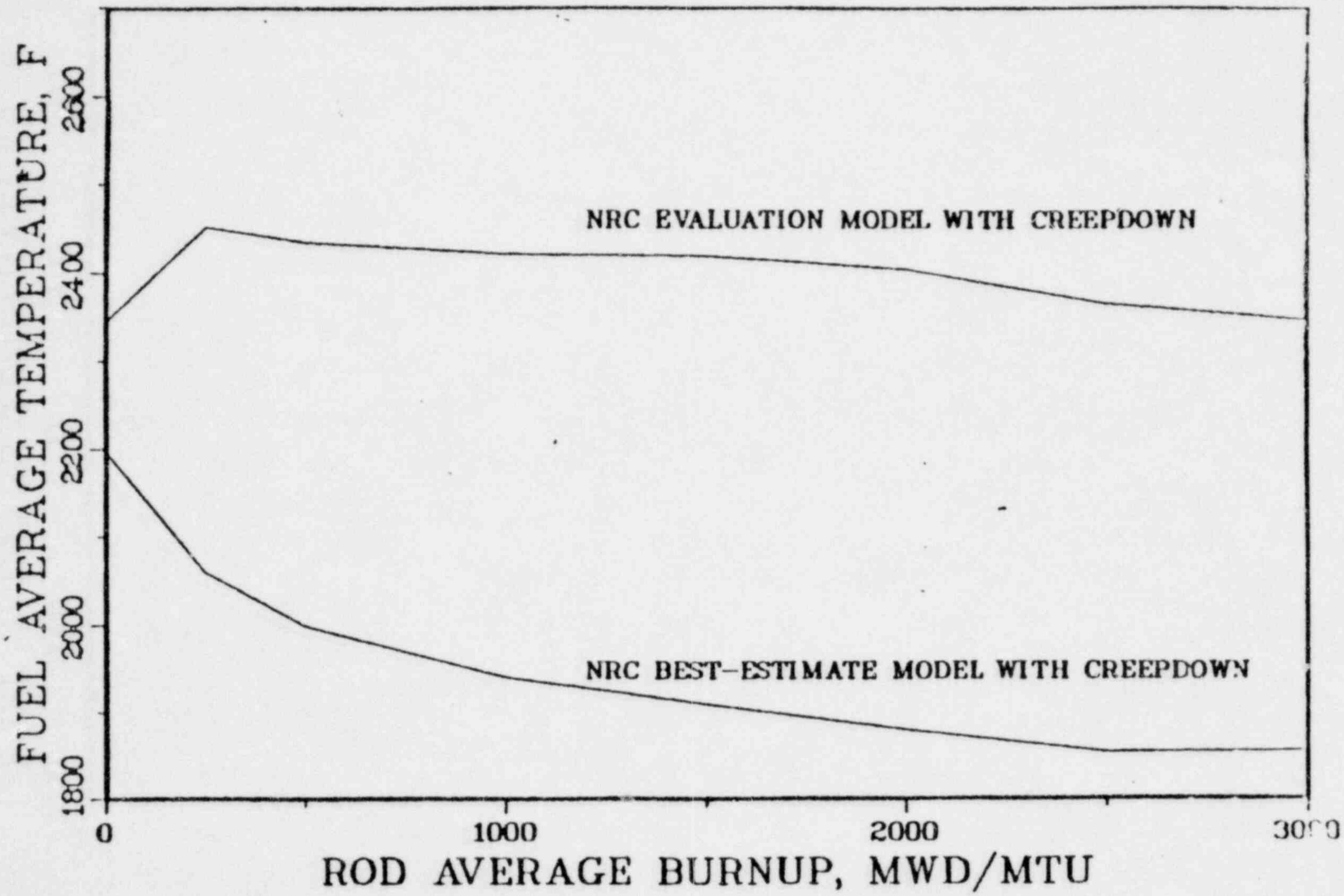

Figure 2

> pe[illegible]-of-life to a low burnup peak then falling temperatures. It should be noted that the conservative GAPCON-2 and PAD predicted maximums occur at different burnups, but the maximum value calculated by the <u>proposed</u> version of PAD is not unreasonable compared to the conservative version of GAPCON. It may also be noted the best-estimate versions of both GAPCON-2 and PAD-3.3 predict high temperatures at beginning-of-life and monotonically decreasing temperatures thereafter. For the burnup range considered, the best-estimate version of GAPCON was continually and substantially over-predicted by the best-estimate version of PAD.

From our audit calculations of the PAD-3.3 code, we observe a similar, but not identical, behavior between this code and GAPCON-2. We also note that the proposed modifications to PAD result in peak volume average fuel temperatures in reasonable agreement with that traditionally accepted for LOCA analyses with a conservative version of the NRC audit code, GAPCON-2.

4. Staff Criterion for Margin of Conservatism

The Appendix K requirements for fuel thermal performance codes state that:

> "The steady-state temperature distribution and stored energy in the fuel before the hypothetical accident shall be calculated for the burn-up that yields the highest calculated cladding temperature (or, optionally, the highest calculated stored energy.) To accomplish this, the thermal conductivity of the UO_2 shall be evaluated as a function of burn-up and temperature, taking into consideration differences in initial density, and the thermal conductance of the gap between the UO_2 and the cladding shall be evaluated as a function of the burn-up, taking into consideration fuel densification and expansion, the composition and pressure of the gases within the fuel rod, the initial cold gap dimension with its tolerances, and cladding creep." (Ref. 13)

There is no explicit requirement within this section of the Code for conservatism in the fuel performance codes.

4.1 Basis for Margin of Conservatism

Although there is no explicit requirement for conservatism in the calculation of the initial stored energy of the fuel, the Commission has expressed an opinion on the subject.

> "The assumption of 102% of maximum power, highest allowed peaking factor, and highest estimated thermal resistance between the UO_2 and the cladding provides a calculated stored heat that is possible but unlikely to occur at the time of a hypothetical accident. While not necessarily a margin over the extreme condition, it represents at least an assumption that an accident happens at a time which is not typical." (Ref.19)

This opinion establishes the requirement for conservatism in the calculation of initial fuel temperatures for Appendix K calculations. However, the degree of conservatism for this heat source was never established. It is possible that even a best-estimate fuel code would predict conservative fuel temperatures in the LOCA analyses because of conservatisms imbedded elsewhere in the calculation. For the number of vendor fuel codes (including the Westinghouse PAD-3.3 code) that have previously been approved by the staff, this is probably the case. They also exhibit various degrees of conservatism by themselves, depending on the vendor and the type of calculation performed. Similar behavior is exhibited by the fuel performance codes utilized by the NRC staff. These are discussed in Sections 3.3 and 5.2 of this report.

It may be noted, however, that a staff opinion has been developed for another heat source, the energy due to the decay of fission products.

> "A best judgment evaluation of these factors leads to the conclusion that a suitable probability level is 95%.... A change to 99% or 99.9% would increase these margins but not substantially (i.e., not produce a fundamental change in the nature of the margins). This level is viewed as the intent of the Appendix K rule development." (Ref. 20)

and further that

> "An additional factor to be considered is the interaction with criteria of other energy sources such as stored energy. Logically they too should be developed with the same uncertainty probability levels as used for decay heat." (Ref. 20)

There are other examples of the application of a 95% probability level in the calculation of heat sources and other portions of the LOCA analysis. The choice of probability level appears to be more traditional than analytical.

"There is no rigorous explicit basis for a 95% level, however it appears to be a conservative level and is compatible with other acceptance levels. The uncertainty and small variation analysis is inbedded in a large LOCA matrix where, for example, F_Q nominal seldom if ever occurs and other elements of required LOCA energy sources are undoubtedly conservative by some large but as yet undecided upon amount. Studies of power distributions for Westinghouse reactors as a function of reactor operation modes have indicated that F_Q limit levels are at least 15% (and usually over 25%) greater than steady state operation nominal peaking factors, and F_Q extremes of allowable load follow transients are reached, if at all, well less than 5% of the time during a cycle for any presently envisioned operation." (Ref. 21)

These two staff opinions suggest the acceptability, without a rigorously derived basis, of the probability and confidence levels proposed. Such levels are usually submitted in safety analysis reports and judged acceptable by the staff. We are aware of no submittals in which a basis for the 95% probability level has been established by the industry. It is recognized, however, that the establishment of such a basis should involve a statistical analysis of the entire LOCA problem. Such a study does not, as yet, exist.

We are also aware of a step in the direction of determining the overall Appendix K conservatism. Westinghouse has proposed (Ref. 22) a statistical combination of the uncertainties in the LOCA heat sources. The proposal was not accepted by the staff (Refs. 23 and 24). An appeal by Westinghouse (Ref. 25) resulted in a second rejection by the staff (Ref. 26) on the basis that modification to explicitly-required conservatisms (such as is the case for decay heat) in Appendix K analyses should be implemented by a change in the regulation itself. We conclude that a reduction in the margin of conservatism in Appendix K stored energy analysis cannot, at this time, be based on conservatisms in other portions of the LOCA analysis. Because there is no explicit requirement for margin of conservatism in the stored energy analysis itself, we also conclude that the conservative margin of this heat source may be established through the review, rather than rulemaking, process. Indeed, this has been the practice within the NRC in the past.

4.2 The Staff Criterion

In order to develop a more uniform review of stored energy codes, we intend to use the following criterion for these models:

> Assuming best-estimate input conditions, an acceptable fuel performance model shall yield a required output parameter such that the predicted value bounds a large proportion of the experimental values for this parameter.

This means that an acceptable fuel performance code, given best-estimate input values, will, at a high probability level, correctly predict the peak fuel temperatures, fuel stored energy, fuel-cladding gap conductance or other parameter required as input to subsequent LOCA analysis codes. We believe an appropriately high probability level is 0.95 or 0.95/0.95 where a confidence level is required.

5. Application of the Criterion

The principal required output parameter from the Westinghouse PAD-3.3 code is volume-average fuel temperature. Higher volume-average fuel temperatures are conservative for LOCA analysis. Therefore, to meet the proposed acceptance criterion, the PAD-3.3 code should show the ability to overpredict volume-average fuel temperatures 95% of the time at a 95% confidence level based on experimental data.

This criterion should be established with experimental data prototypic of the Westinghouse product line and, where possible, taken near LOCA conditions.

As discussed in Section 3.2 of this report, Westinghouse believes the experimental fuel temperature data shown in Ref. 11 to be representative of their standard product line. These data were taken at linear power levels of up to 15 kW/ft, which is approximately the LOCA limit. A limitation of these data, however, is that they are based on fuel centerline thermocouple measurements. Therefore, the data are an indication of fuel centerline temperature rather than volume average fuel temperature. We are not aware of any experimental data which directly measure in-reactor volume average fuel temperatures. It is possible, however, to relate fuel centerline and volume average temperatures analytically. This is shown in Figure 3, where best-estimate BOL fuel centerline and volume average fuel temperature predictions from GAPCON-2 are shown as a function of linear power. This figure indicates that the fuel centerline temperature rises much more rapidly than volume average temperature as a function of power. This is an expected result because fuel surface temperatures remain relatively close to the coolant temperature whereas the fuel centerline temperature rises. The volume average temperature may be approximated by the average of the fuel surface and centerline temperatures. Figure 4 shows the same prediction replotted with fuel volume average temperatures expressed as a function of fuel centerline temperature. We will use this figure to relate uncertainty in centerline temperature to uncertainty in volume average temperature.

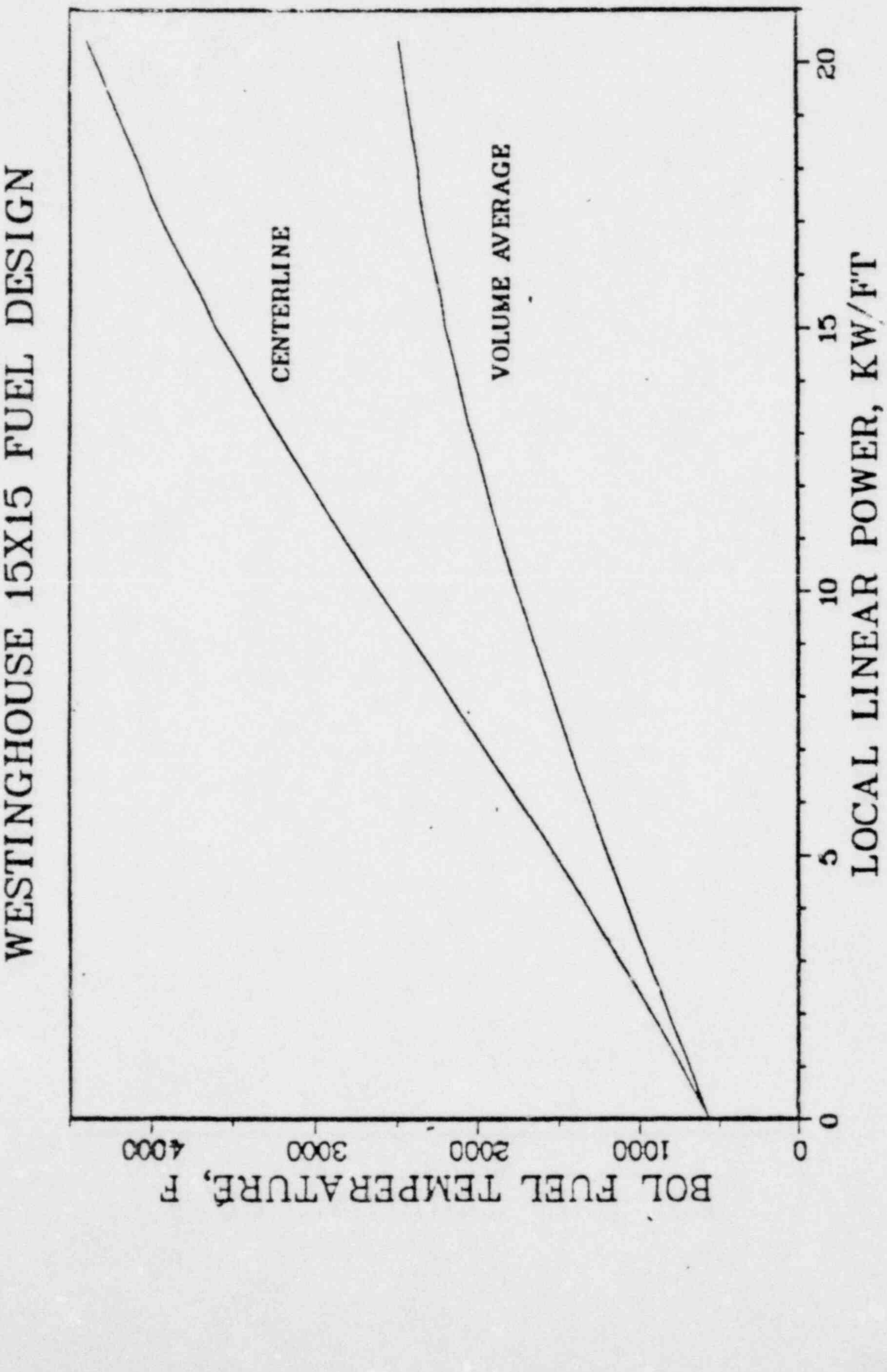
WESTINGHOUSE 15X15 FUEL DESIGN
CENTERLINE
VOLUME AVERAGE
BOL FUEL TEMPERATURE, F
LOCAL LINEAR POWER, KW/FT
0
1000
2000
3000
4000
0
5
10
15
20

Figure 3

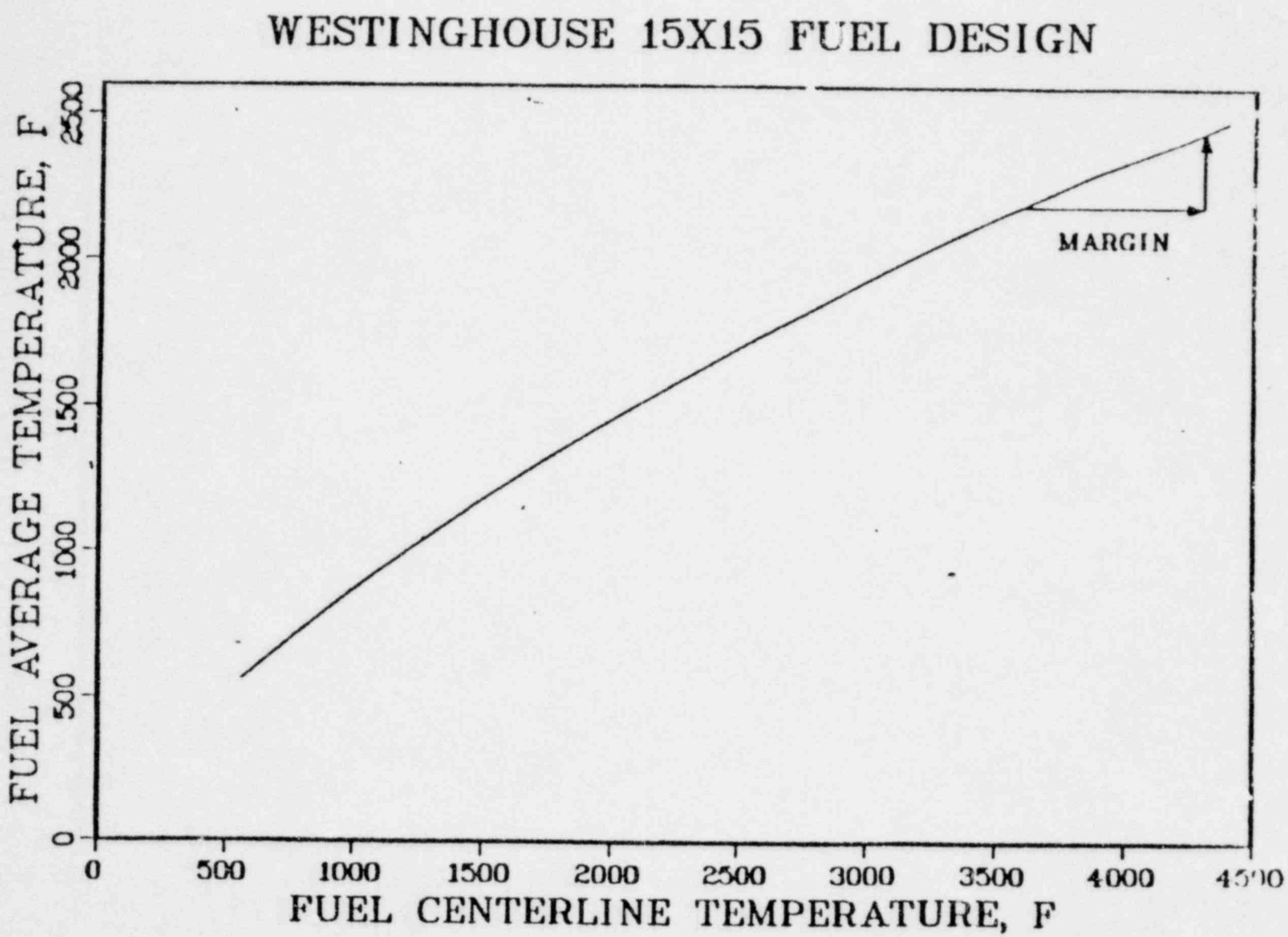
WESTINGHOUSE 15X15 FUEL DESIGN
FUEL AVERAGE TEMPERATURE, F
0
500
1000
1500
2000
2500
MARGIN
0
500
1000
1500
2000
2500
3000
3500
4000
4500
FUEL CENTERLINE TEMPERATURE, F

Figure 4

It should also be noted that the PAD-3.3 predictions of the experimental data utilize nominal input values. This is in keeping with the criterion. While we recognize (Ref. 27) the conservatism which may exist in the input values to PAD-3.3 when used in LOCA analyses, such uncertainties are difficult to quantify from experimental data. As discussed in Section 4 of this report, we also believe it to be inappropriate to base the reduction of conservatism in one segment of the LOCA analysis on possibly excessive conservatism in another. Such a change is more appropriate to the rulemaking process in which the conservative margin in the overall problem may be examined. For the purpose of this review, we shall assume that the PAD-3.3 code is provided nominal input conditions as part of the LOCA analysis.

5.1 The Westinghouse PAD Computer Code

A statistical analysis was performed on the measured and predicted fuel centerline temperature data shown in Figure 2 of Reference 11. We determined the mean and standard deviation of the difference (e.g. measured temperature minus predicted temperature) assuming the distribution to be normal. As the proposed criterion requires the PAD-3.3 code to conservatively predict volume average fuel temperature with a 95% probability at a 95% confidence level, the margin to be added to the best-estimate prediction should be 1.64 times the standard deviation plus the best-estimate code bias (data mean), if any.

To relate this uncertainty in fuel centerline temperature to volume average fuel temperature, we refer to Figure 3. At a LOCA limit of 15 kW/ft, we find best-estimate fuel volume average and fuel centerline temperature of 2200°F and 3600°F respectively. Adding the margin due to uncertainty in fuel centerline temperature to the best-estimate centerline temperature yields a conservative prediction of centerline temperature. From Figure 4, we find the corresponding best-estimate, conservative and equivalent margin values for volume average temperature. Using this process, we conclude that the PAD-3.3 code meets the proposed criterion based on the experimental data comparison supplied by Westinghouse.

5.2 The NRC Fuel Performance Codes

In order to check the validity of our conclusions regarding the overall uncertainty in the PAD-3.3 code, we reviewed a number of other fuel performance codes and their predictive uncertainties. We have attempted to show that the predictive uncertainty in the PAD-3.3 not only meets the proposed criterion, but is also representative of similar fuel performance codes.

5.2.1 FRAP-S3

The FRAP-S3 code (Ref. 28) was developed by Idaho National Engineering Laboratory for the thermal and mechanical analysis of light water reactor fuel rods. The code considers the effects of fuel and cladding deformation, temperature distribution, internal gas pressure, and material properties like PAD and GAPCON. FRAP-S3 was developed by Idaho National Engineering Laboratory for NRC's Office of Reactor Safety Research. It is a representative example of a state-of-the-art fuel performance code.

The FRAP-S3 verification report (Ref. 29), presents predicted versus measured fuel centerline temperatures based on thermocouple measurements from approximately 100 rods, representing over 800 data points. All fuel rods used by Westinghouse, except WAPD-228 rods 22-3 and 22-4 were included in this study. The standard error between measured and predicted fuel centerline temperature was stated to be 356°F and 457°F for unpressurized and pressurized rods respectively. Assuming the standard deviation is independent of fuel centerline temperature (as was assumed in the study), this would result in a maximum uncertainty in fuel centerline temperature of 1.64 x 457°F = 750°F. This is a .95/.95 statistical tolerance interval.

To relate this uncertainty in fuel centerline temperature to volume average fuel temperature, we again refer to Figure 3. Adding 750°F margin to the best estimate centerline temperature yields a conservative centerline temperature prediction of 4000°F. From Figure 4, we find that this 750°F margin on centerline temperature is approximately equal to a margin of 273°F on volume average fuel temperature. We conclude that FRAP-S3 volume average fuel temperature predictions have a maximum uncertainty of 273°F at a .95/.95 tolerance level. There is evidence (Ref. 30) that these values would drop considerably if the data base were restricted to helium pressurized rods near beginning-of-life with typical densities and gap sizes as proposed by Westinghouse. We have not considered that possibility in this evaluation.

5.2.2 FRAPCON-1

The FRAPCON-1 code (Ref. 31) is a more recent version of the FRAP-S3 code discussed previously. This computer program is the most recent of the fuel performance codes developed for the NRC. As is the case for its predecessor, FRAPCON-1 is intended to calculate the effects of power and burnup on fuel behavior under normal operating conditions.

The FRAPCON-1 verification report (Ref. 32) presents the results of predicted versus measured fuel centerline temperatures for approximately the same number (93 rods/740 data points) of fuel centerline thermocouple measurements as FRAP-S3. The standard deviation between measured and predicted values is 306°F for unpressurized rods and 529°F for the pressurized rod data. The latter value is larger than that calculated for FRAP-S3 but no explanation for the regression in predictive ability is presented in the report.

The FRAPCON-1 assessment report again assumes that the standard deviation is constant for the range of centerline temperatures considered. Using the same process described for FRAP-S3, we calculated maximum FRAPCON-1 predictive uncertainties of 868°F for fuel centerline temperature and 319°F for volume average fuel temperature at a 95/95 tolerance limit.

The GAPCON series of computer codes, which are also utilized by the NRC staff, have not been subjected to the same verification process used for FRAP-S3 and FRAPCON-1. All of the GAPCON series codes have been verified with experimental data but the measured and predicted values have not been statistically analyzed. However, the developers of GAPCON, Battelle Pacific Northwest Laboratories, have attempted to establish the predictive uncertainty in these codes from first principles (Refs. 33 and 34).

A recent investigation by Cunningham et al. (Ref 35) determined the effect of input and model uncertainties on fuel temperature and stored energy calculations. The study identified analytical models necessary for calculating stored energy and then utilized both the method of linear propagation and Monte Carlo technique to determine prediction uncertainties. Results were generated for a typical BWR fuel rod, but the study is also applicable to PWR fuel designs. The authors estimate the maximum uncertainties for fuel centerline temperature at a linear power of 500 W/cm (15.2 kW/ft) to be 15.5% for the Monte Carlo technique and 18.2% for the linear propagation method. These figures are given at a 3σ (99.9%) confidence level, but will be assumed .95/.95 tolerance intervals in this report.

Using these figures, we concluded that the expected uncertainty in predicting the fuel centerline temperature of a PWR rod operating at 15 kW/ft to be 558°F and 655°F by two first-principles methods. These values correspond to 206°F and 239°F uncertainties on the volume average fuel temperatures.

5.3 The State of the Art

We have examined a sample of state-of-the-art fuel performance codes for an indication of the expected uncertainty in predicting volume average fuel temperatures at approximately 15 kW/ft. This sample included two similar data-prediction studies (Refs. 29 and 32) and two similar first principles methods (Ref. 35). The maximum uncertainties assumed in this sample are summarized below:

Study	Maximum uncertainty in volume average fuel temperature at 15 kW/ft
Data-prediction (Ref. 29)	273°F
Data-prediction (Ref. 32)	319°F
Linear propagation (Ref. 35)	206°F
Monte Carlo (Ref. 35)	239°F
Average	259°F

To determine the state-of-the-art uncertainty in volume average fuel temperature, we have taken the average of these values. The value obtained is 259°F above the data mean. The Westinghouse margin for volume average fuel temperature is also the difference between the evaluation model prediction and the data mean. This value is the maximum volume average temperature predicted by the evaluation model version of the code minus the maximum volume average temperature predicted by the best estimate version plus the bias in the best estimate code, if any. We have determined this value and conclude that the PAD-3.3 meets the proposed criterion without the use of the 65°F model uncertainty and that the remaining margin of conservatism is similar to the expected uncertainty in other state-of-the-art fuel codes.

6. Conclusions

We have examined the proposed revisions to the Westinghouse fuel code, PAD-3.3, as described by the code developer and Commonwealth Edison Company. These changes consist of the use of as-fabricated, rather than as-designed, values of fuel density, fuel sintering temperature and their associated tolerances, and the deletion of the 65°F model uncertainty from the Westinghouse fuel thermal performance analysis. Based upon our technical review of the submittal, comparison of the Westinghouse code predictions with a traditional staff audit code, and the development and application of a new criterion for margin of conservatism in codes of this type, we conclude that these changes are acceptable. This acceptance is limited to the current [illegible]ion of the Westinghouse PAD-3.3 code as approved by the staff (Ref. 14) for application in LOCA analyses, but acceptance is not limited to Zion.

7. References

1. C. Reed (Com-Ed) letter to H. R. Denton (NRC) on "Zion Station Units 1 and 2 Proposed Change to Facility Operating License Nos. DPR-39 and DPR-48," dated March 22, 1979.

2. Title 10, Code of Federal Regulations, Part 50, Section 46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors," January 1979.

3. C. Reed (Com-Ed) letter to H. R. Denton (NRC) on "Zion Station Unit 1 Base Load Operation Under Revised ECCS Evaluation" dated January 24, 1979.

4. C. Reed (Com-Ed) letter to H. R. Denton (NRC) on "Zion Station Unit 1 Additional Information for Base Load Operation Under Revised ECCS Evaluation" dated Janaury 25, 1979.

5. C. Reed (Com-Ed) letter to H. R. Denton (NRC) on "Zion Station Units 1 and 2 Revised ECCS Evaluation" dated January 29, 1979.

6. C. Reed (Com-Ed) letter to D. G. Eisenhut (NRC) on "Zion Station Units 1 and 2 LOCA Reanalysis Review" dated July 16, 1979.

7. J. V. Miller, Ed., "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations," Westinghouse Electric Corporation Report WCAP-8720, October 1976 (proprietary) and WCAP-8785, October 1976 (non-proprietary).

8. P. S. Check (NRC) memorandum to A. Schwencer (NRC) on "Questions Concerning Removal of Conservatism from the Westinghouse PAD Code" dated April 18, 1979.

9. P. S. Check (NRC) memorandum to A. Schwencer (NRC) on "Zion Request for Technical Specification Change to F_Q Based on Conservatism in the PAD Computer Code" dated August 30, 1979. Enclosure reprinted as Appendix A to Ref 11.

10. W. F. Naughton (Com-Ed) letter to H. R. Denton (NRC) on "Zion Station Units 1 and 2 Additional Information for Increase in F_Q Peaking Factor" dated May 3, 1979.

11. W. F. Naughton (Com-Ed) letter to H. R. Denton (NRC) on "Zion Station Units 1 and 2 Additional Information for Increase in F_Q Peaking Factor, NRC Docket Nos. 50-295 and 50-304" dated January 25, 1980 (proprietary).

12. J. F. Stolz (NRC) letter to T. M. Anderson (W) on "Safety Evaluation of WCAP-8720" dated February 9, 1979.

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

13. Title 10, Code of Federal Regulations, Part 50, Appendix K, "ECCS Evaluation Models" January 1979.

14. R. Salvatori (W) letter NS-RS-133 to D. B. Vassallo (AEC) with Supplemental Information on Fuel Densification dated February 5, 1974. Also Appendix D.1 of "Fuel Densification Experimental Results and Model for Reactor Application," J. M. Hellman, Ed., Westinghouse Electric Corporation Report WCAP-8218-P-A, March 1975 (proprietary) and WCAP-8219-A, March 1975 (non-proprietary).

15. C. E. Beyer, C. R. Hann, D. D. Lanning, F. E. Panisko, and L. J. Parchen, "GAPCON-THERMAL-2: A Computer Program for Calculating the Thermal Behavior of an Oxide Fuel Rod," Battelle Pacific Northwest Laboratory Report BNWL-1898, November 1975.

16. C. E. Beyer, C. R. Hann, D. D. Lanning, F. E. Panisko, and L. J. Parchen, "User's Guide for GAPCON-THERMAL-2: A Computer Program for Calculating the Thermal Behavior of an Oxide Fuel Rod," Battelle Pacific Northwest Laboratories Report BNWL-1897, November 1975.

17. K. Kniel (NRC) memorandum to S. Fabic (NRC) on "GAPCON Input Parameters for the WRAP Program" dated September 27, 1979.

18. P. J. Pankaskie, "An Analytical Computer Code for Calculating Creep Buckling of an Initially Oval Tube," Battelle Pacific Northwest Laboratories Report BNWL-1784, May 1974.

19. Opinion of the Commission, "In the Matter of Rulemaking Hearing, Acceptance Criteria for Emergency Core Cooling Systems for Light Water-Cooled-Nuclear Power Reactors," USAEC Docket No. RM-50-1, December 28, 1973.

20. H. J. Richings (NRC) memorandum to P. S. Check (NRC) on "Potential Changes in CPB Position on Decay Heat for LOCA" dated July 14, 1976.

21. H. Richings (NRC) memorandum to P. S. Check (NRC) on "Policy Questions Arising from W Maxi Convolution" dated April 6, 1978 (Proprietary).

22. C. C. Little, S. D. Kopelic and H. Chelemer, "Consideration of Uncertainties in the Specification of Core Hot Channel Factor Limits," Westinghouse Electric Corporation Report WCAP-9180, September 1977.

23. D. F. Ross, Jr. (NRC) memorandum to R. J. Mattson (NRC) on "A Proposed Change to LOCA Peaking Factor Uncertainty Requirements" dated July 24, 1978.

24. R. J. Mattson (NRC) letter to T. Anderson(W) dated August 9, 1978.

25. T. M. Anderson (W) letter NS-TMA-1929 to R. J. Mattson (NRC) dated September 8, 1978.

26. R. J. Mattson (NRC) letter to T. Anderson (W) dated October 26, 1978.

27. H. Richings (NRC) memorandum to P. S. Check (NRC) on "Some Notes on PWR (W) Power Distribution Probabilities for LOCA Probabilistic Analyses" dated July 5, 1977.

28. J. A. Dearien, G. A. Berna, M. P. Bohn, J. D. Kerrigan and D. R. Coleman, "FRAP-S3: A Computer Code for the Steady-State Analysis of Oxide Fuel Rods, Volume 1, FRAP-S3 Analytical Models and Input Manual," EG&G Idaho, Inc. Report TFBP-TR-164, March 1978.

29. D. R. Coleman, E. T. Laats and N. R. Scofield, "FRAP-S3: A Computer Code for the Steady-State Analysis of Fuel Rods, Volume 2, Model Verification Report, EG&G Idaho, Inc. Report TFBP-TR-228, August, 1977.

30. J. D. Kerrigan and D. R. Coleman, "Standard Design Analysis - A Statistical Determination of Corewide Initial Accident Conditions - Fuel Stored Energy Results," Table X, EG&G Idaho, Inc. Report CAAP-TR-034, December 1978.

31. G. A. Berna, M. P. Bohn, D. R. Coleman and D. D. Lanning, "FRAPCON-1: A Computer Code for the Steady-State Analysis of Oxide Fuel Rods," EG&G Idaho, Inc. Report CDAP-TR-78-032, August 1978.

32. E. T. Laats, G. B. Peeler and N. S. Scofield, "Independent Assessment of the Steady State Fuel Rod Analysis Code FRAPCON-1," EG&G Idaho, Inc. Report CAAP-TR-050, May 1979.

33. D. D. Lanning, C. R. Hann and E. S. Gilbert, "Statistical Analysis of Gap Conductance Data for Reactor Fuel Rods Containing UO_2 Pellets," Battelle Pacific Northwest Laboratories Report BNWL-1832, August 1974.

34. C. R. Hann, D. D. Lanning, R. K. Marshall, A. R. Olsen and R. E. Williford, "A Method for Determining the Uncertainty of Gap Conductance Deduced from Measured Fuel Centerline Temperatures," Battelle Pacific Northwest Laboratories Report BNWL-2091, February 1977.

35. M. F. Cunningham, D. D. Lanning, A. R. Olsen, R. E. Williford and C. R. Hann, "Stored Energy Calculation: The State of the Art," Battelle Pacific Northwest Laboratories Report PNL-2581, May 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-295 AND 50-304

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 53 to Facility Operating License No. DPR-39, and Amendment No. 50 to Facility Operating License No. DPR-48 issued to the Commonwealth Edison Company (the licensee), which revised Technical Specifications for operation of Zion Station, Units 1 and 2 (the facilities) located in Zion, Illinois. The amendments are effective as of the date of issuance.

The amendments modify the Technical Specifications, Appendix A to the licenses, to increase the allowable LOCA peaking factor from 1.86 to 1.93 based on an ECCS reanalysis.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on May 7, 1979 (44 FR 26816). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated March 22, 1979, as supplemented on May 3, 1979 and January 25, 1980, (2) Amendment Nos. 53 and 50 to License Nos. DPR-38 and DPR-49, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Zion-Benton Public Library District, 2600 Emmaus Avenue, Zion, Illinois 60099. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 13th day of March, 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors