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May 2, 1983

Mr. Jerome Saltzman
Assistant Director of State
and Licensee Relations
Office of State Programs
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Toledo Edison Company
ANI/MAELU Policies NF-236, MF-92

Dear Jerry:

On behalf of the Toledo Edison Company, I have enclosed eight certified copies each of Endorsement 44 to NF-236 and Endorsement 34 to MF-92.

Please contact me if there are any questions.

Very truly yours,

Michael P. Golden
Nuclear Consultant

MPG:aj
Enc.

cc: R. Ertle

M001

Nuclear Energy Liability Insurance

NUCLEAR ENERGY LIABILITY INSURANCE ASSOCIATION

ADVANCE PREMIUM AND STANDARD PREMIUM ENDORSEMENT

CALENDAR YEAR 1982

It is agreed that Items 1a. and 1b. of Endorsement No. 42 are amended to read:

1a. ADVANCE PREMIUM: It is agreed that the Advance Premium due the companies for the period designated above is: $289,323.00.

1b. STANDARD PREMIUM AND RESERVE PREMIUM: In the absence of a change in the Advance Premium indicated above, it is agreed that, subject to the provisions of the Industry Credit Rating Plan, the Standard Premium is said Advance Premium and the Reserve Premium is: $217,560.48.

Return Premium: $ 2,893.24.

This is to certify that this is a true copy of the original Endorsement having the endorsement number and being made part of the Nuclear Energy Liability Policy (Facility Form) as designated hereon. No Insurance is afforded hereunder.

J. L. Quattrocchi

John L. Quattrocchi, Vice President-Liability Underwriting
American Nuclear Insurers

Effective Date of this Endorsement January 1, 1982
12:01 A.M. Standard Time

To form a part of Policy No. NF-236

Issued to The Toledo Edison Company

Date of Issue April 21, 1983

For the subscribing companies

By [signature]
General Manager

Endorsement No 44

Countersigned by

NE-36

NUCLEAR ENERGY LIABILITY INSURANCE

MUTUAL ATOMIC ENERGY LIABILITY UNDERWRITERS

1. AMENDMENT OF ADVANCE PREMIUM ENDORSEMENT

2. STANDARD PREMIUM AND RESERVE PREMIUM ENDORSEMENT

3. RETURN PREMIUM DUE

1. Advance Premium

It is agreed that the Amended Advance Premium due the companies for the calendar year 1982 is $83,997.00.

2. Standard Premium and Reserve Premium

Subject to the provisions of the Industry Credit Rating Plan, it is agreed that the Standard Premium and Reserve Premium for the calendar year designated above are:

Standard Premium $83,997.00

Reserve Premium $63,162.72

3. Return Premium $839.97.

Effective Date of this Endorsement January 1, 1982

To form a part of Policy No. MF-92

Issued to The Toledo Edison Company

Date of Issue April 21, 1983

For the Subscribing Companies

MUTUAL ATOMIC ENERGY LIABILITY UNDERWRITERS

By J. L. Quattrocchi

Endorsement No. 34

Countersigned by ______________

Authorized Representative

This is to certify that this is a true copy of the original Endorsement having the endorsement number and being made part of the Nuclear Energy Liability Policy (Facility Form) as designated hereon. No Insurance is afforded hereunder.

ME-36

J. L. Quattrocchi

John L. Quattrocchi, Vice President-Liability Underwriting
American Nuclear Insurers

Attachment B to AEP:NRC:0745C
Safety Evaluation of Reload

1.0 INTRODUCTION

D. C. Cook Unit 1 is operating with an all Exxon Nuclear Company (ENC) fueled core during Cycle 7. For subsequent cycles, it is planned to refuel Unit 1 with 15x15 optimized fuel assembly (OFA) regions supplied by the Westinghouse Electric Corporation (W). As a result, future core loadings would range from approximately a 40% OFA and 60% ENC fueled core to eventually an all OFA fueled core. The W 15x15 OFA fuel design is similar to the W 15x15 LOPAR (low parasitic) fuel which has had substantial operating performance in a number of nuclear plants. The major difference introduced by the W 15x15 OFA design is the use of five intermediate Zircaloy grids replacing five intermediate Inconel grids for the LOPAR fuel. The 15x15 Zircaloy grid design is similar to the W 17x17 OFA grid design. The W 17x17 OFA design has been generically approved by the NRC via their review of the W 17x17 OFA Reference Core Report.[1] Operating experience has been obtained for six demonstration 17x17 OFAs which contain Zircaloy intermediate grids.[2] Two assemblies have satisfactorily completed three cycles of irradiation to about 28,000 MWD/MTU burnup, two have completed two cycles to about 19,400 MWD/MTU, and two have completed one cycle in excess of 9,000 MWD/MTU. The demonstration OFAs have been examined and provide reason to expect good performance from the 15x15 OFA design.

This report summarizes the results of the W analyses which justify the transition from an all ENC core, through a mixed OFA/ENC fueled core to an all OFA core. Although it is planned to operate D. C. Cook Unit 1 Cycle 8 at the current licensed maximum power level of 3250 MWt, the core evaluations/ analyses summarized in this report have been performed at a reactor power level of 3411 MWt, with the exception of the large break LOCA which was analyzed at 3250 MWt. This conservative design basis provides early identification of those safety/accident analysis limits for a potential uprating.

All analyses were performed utilizing W standard methods, which are described in the W Reload Safety Evaluation Methodology Topical.[3] The approved Westinghouse Improved Thermal Design Procedure (ITDP) is used in the DNB analyses of both W and ENC fuel. The W WRB-1 correlation is used in the OFA DNB analyses. Both the ITDP and WRB-1 correlation were previously used to license D. C. Cook Unit 2 operation. The ENC fuel is analyzed using the W-3 DNB correlation. Other features being introduced with the Cycle 8 reload include the Westinghouse Wet Annular Burnable Absorber (WABA) rods and a revision to the Westinghouse fuel thermal safety model (PAD Code) used in the safety analyses. Westinghouse has submitted topical reports[4,5] on these subjects and is supporting the NRC's generic review, in order to obtain approval well before the planned Cycle 8 startup.

2.0 SUMMARY AND CONCLUSIONS

The Westinghouse Reload Safety Evaluation Methodology[3] was used to evaluate the transition from ENC fuel to W 15x15 OFA fuel for D. C. Cook Unit 1. Parameters were chosen to maximize the applicability of the transition evaluations for each reload cycle and to facilitate the safety evaluation of future reload cores. Transition core effects were considered in the mechanical, thermal and hydraulic, nuclear, and accident evaluations described in Chapter 18 of Reference 1. The summary of these evaluations for the D. C. Cook Unit 1 transition to an all W 15x15 OFA core is given in the following sections of this submittal.

The transition design and safety evaluations are based on the following maximum power conditions: 3411 MWt reactor power and 577.1°F vessel average temperature.

The results of evaluations/analyses and tests discussed in this report lead to the following conclusions:

1. The Westinghouse OFAs are mechanically and hydraulically compatible with the ENC fuel assemblies, control rods, and reactor internals interfaces.

2. Changes in the nuclear characteristics due to the transition from ENC to W 15x15 OFA fuel will be within the normal variations from cycle-to-cycle due to fuel management effects. W 15x15 OFA fuel up to and including a 4.00% nominal enrichment can be stored in the fresh and spent fuel areas.

3. Demonstration experience with W 17x17 OFAs containing Zircaloy grids provides reason to expect satisfactory operation from 15x15 OFA Zircaloy grids.

4. The WABA rod, as described in its generic topical[4], is compatible with the W 15x15 OFA and satisfies all performance requirements for its design life.

5. The proposed Technical Specification changes presented in Attachment A are applicable to cores containing any combination of W 15x15 OFA and ENC fuel.

6. All design criteria for the W 15x15 OFA fuel are satisfied.

7. A reference is established upon which to base future cycle safety evaluations for W OFA reload fuel.

3.0 MECHANICAL EVALUATION

The mechanical design requirements and criteria for the 17x17 OFA design are described in Reference 1, which was approved by the NRC. The 15x15 OFA design meets these same basic requirements and criteria.

ENC, in estabishing their assembly design, demonstrated their fuel's compatibility with the W LOPAR design which was the initial D. C. Cook Unit 1 fuel. W has demonstrated compatibility of its 15x15 OFA design with its LOPAR design. Compatibility of the OFA and ENC fuel is thereby demonstrated.

Figure 1 and Table 1 present a comparison of the W 15x15 OFA and ENC fuel assemblies. The W and ENC fuel rods have similar length and clad OD dimensions. The W 15x15 OFA rods have the same design as the LOPAR W 15x15 fuel rods which have exhibited good in-core performance in many operating reactors.

The top and bottom Inconel grids of the OFA are the same as the Inconel grids of a W LOPAR fuel assembly. The five intermediate OFA Zircaloy-4 grids have thicker and wider straps than the OFA Inconel grids (See Figure 1) in order to closely duplicate the Inconel grid strength. The ENC assembly grids are bimetallic, consisting of Zircaloy-4 straps with Inconel grid springs. Both the OFA Zircaloy and ENC bimetallic grids have grid heights of 2.25 inches. Elevation of the grids was established to ensure satisfactory axial alignment during operation.

Due to thicker Zircaloy grid straps and a resulting reduced cell size, the OFA guide thimble tube ID (above dashpot) has a 12 mil reduction compared to the ENC thimble tube ID of 0.511 inches. Below the dashpot, the OFA and ENC fuel thimble tubes have the same dimensions. The OFA guide tube thimble ID provides sufficient nominal diametral clearance for control rods as well as source rods, burnable absorber rods, and OFA thimble plugs. Due to reduced OFA diametral clearance, the control rod

scram time to the dashpot is increased from the current 1.8 seconds to 2.4 seconds. This increase in rod drop time was determined from conservative analytical calculations. The 2.4 second scram time is used in all the accident reanalyses.

The OFA design has minor differences in the overall height of the top and bottom nozzles, the adapter plate flow-slot configuration and hold-down leaf springs as compared to the ENC fuel assembly design. These minor differences have no adverse impact on the interaction of W 15x15 OFA and ENC assemblies during fuel handling operations or reactor operations. The W 15x15 OFA design uses a 3-leaf holddown spring design compared to the 2-leaf springs in the ENC assembly. The W OFA 3-leaf spring has been previously used in 15x15 LOPAR assemblies, as well as on the 17x17 OFA demonstration assemblies. The 3-leaf spring provides additional holddown force margin compared to the 2-leaf spring. The OFA bottom nozzle has similar design features and dimensions compared to the ENC nozzle. The OFA bottom nozzle design has a reconstitutable feature, as shown in Figure 2, which allows it to be easily removed. A locking cup is used to lock the thimble screw of a guide thimble tube in place, instead of the lockwire as used for the standard W LOPAR nozzle design. The reconstitutable nozzle design facilitates remote removal of the bottom nozzle and relocking of thimble screws as the bottom nozzle is reattached.

As stated in the 17x17 OFA Reference Core Report (1), for a given burnup, the magnitude of rod bow for the W OFA is conservatively assumed to be the same as that of a W LOPAR fuel assembly. The most probable causes of significant rod bow are rod-grid and pellet-clad interaction forces and wall thickness variation. Since the OFA fuel rods are the same as the W LOPAR fuel rods, there will be no difference in predicted bow due to rod considerations. The OFA design will have reduced grid forces due to the Zircaloy grid springs. Therefore, this component is predicted to decrease OFA rod bow compared to LOPAR fuel.

The wear of fuel rod cladding is dependent on both the support provided by the grids and the flow environment to which it is subjected. OFA and ENC assembly flow test results were evaluated. ENC hydraulic test results show the cross flow between ENC and W 15x15 LOPAR assemblies is very similar to that obtained during W flow tests on side-by-side W 15x15 OFA and W 15x15 LOPAR assemblies. These tests showed only a small cross flow between assemblies and no significant fuel rod wear due to rod vibration. Extrapolation of the results from flow tests involving OFA and LOPAR assemblies shows that fuel rod wear would be less than ten (10) percent of the cladding thickness for at least 48 months of reactor operation. This assures that clad wear will not impair fuel rod integrity.

The above conclusions on OFA rod wear and integrity have also been supported by analytical results. The analysis accounted for rod vibrations caused by both axial and cross flows, and the effect of potential fuel rod to grid gaps.

4.0 NUCLEAR EVALUATION

The nuclear design of cores with W OFA and ENC fuel is accomplished by using the standard calculational methods as described in the W Reload Safety Evaluation Methodology[3]. The dimensional and material differences between the W and ENC assemblies are small so that the W computer codes and methods are also valid for the ENC fuel. Dimensions and composition for each of the two fuel designs were used to establish the models. The burnup distribution of the ENC fuel assemblies remaining in Cycle 8 has been obtained by depleting the loading patterns from earlier cycles using two dimensional and three dimensional models of the applicable cores.

Changes in the nuclear characteristics during the transition cycles from an ENC fueled core to a W 15x15 OFA core will be primarily due to fuel management considerations (number of feed assemblies, feed enrichment, cycle burnup, etc.) and not due to the differences in fuel assembly design. Each reload core design will be evaluated to assure that design and safety limits for the OFA and ENC fuel are satisfied according to the W reload safety evaluation methodology. For the evaluation of the worst-case $F_Q(Z)$ envelope, axial power shapes are synthesized with the limiting Fxy values chosen over three overlapping burnup windows during the cycle. The design and safety limits will be documented in each cycle specific reload safety evaluation report which serves as the basis for any significant changes requiring NRC review.

In order to accommodate potential increases in future feed enrichments, a criticality analysis of the fuel storage areas was performed for nominal enrichments up to and including 4.00 Wt.% U235 in W 15x15 OFA fuel. These analyses confirm that all current safety criteria applicable to fuel storage are satisfied[6].

5.0 THERMAL AND HYDRAULIC EVALUATION

Results of hydraulic compatibility tests performed by the Exxon Nuclear Company for the ENC and W 15x15 LOPAR assemblies were compared to hydraulic test data for the W 15x15 LOPAR and OFA assemblies. The data show that the W 15x15 OFA fuel assemblies are hydraulically compatible with the ENC fuel assemblies. Pressure drop data were obtained over a range of fluid temperatures and flow rates. Pressure drops values were then extrapolated to core operating conditions. At typical reactor conditions, the ENC fuel assembly has a pressure drop within 0.7 percent of the W 15x15 OFA pressure drop.

The thermal hydraulic design of this core is conservatively analyzed at 3411 MWt core power with a 577.1°F vessel average temperature, even though the Cycle 8 core will continue to be limited to its current rated parameters of 3250 MWt core power and a 567.8°F vessel average temperature. The analyses employed the Improved Thermal Design Procedure(7) (ITDP) and the THINC IV(8,9) computer code. The WRB-1(10) DNB correlation was used in the W 15x15 OFA analyses, whereas the W-3 correlation was used to analyze the ENC fuel. The thermal hydraulic design criteria remain the same as those presented in the D. C. Cook Unit 1 Updated FSAR(11). All design criteria are satisifed.

The design method employed to meet the DNB design basis is the ITDP(7). Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically, such that there is at least a 95 percent probability that the minimum DNBR will be greater than or equal to the limit DNBR for the peak power rod. Plant parameter uncertainties are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the DNBR limit, establishes a design DNBR value which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses are performed using values of input parameters without uncertainties. In addition,

the limit DNBR values are increased to values designated as the safety analysis limit DNBR's. The plant allowance available between the safety analysis limit DNBR values and the design limit DNBR values is not required to meet the design basis.

In this application, the WRB-1 DNB correlation[10] is employed in the thermal hydraulic design of the W 15x15 OFA fuel. Due to an improvement in the accuracy of the critical heat flux prediction with the WRB-1 correlation compared to previous DNB correlations, a correlation limit DNBR of 1.17 is applicable. The W-3 DNBR correlation [12,13] was used in the design of the ENC fuel assembly. A W-3 correlation limit DNBR of 1.30 is applicable.

The table below indicates the relationships between the correlation limit DNBR, design limit DNBR, and the safety analysis limit DNBR values used for this design.

	W 15x15 OFA		ENC 15x15	
	Typical	Thimble	Typical	Thimble
Correlation Limit	1.17	1.17	1.30	1.30
Design Limit	1.32	1.31	1.58	1.50
Safety Analysis Limit	1.69	1.69	1.58	1.50

The margin to the safety analysis DNBR limit is more than sufficient to cover the maximum 12.5 percent rod bow penalty at full flow Conditions[14] and a 5 percent transition core penalty, both applied to the OFA only. An additional rod bow penalty of 2.4% DNBR at loss of flow conditions[14] is covered explicitly in the loss of flow analysis for the W 15x15 OFA. The 5 percent transition penalty was determined by analyzing W 15x15 OFA and ENC

assembly loading patterns at various core conditions in the same manner as the W 17x17 OFA/LOPAR fuel analysis which was reviewed and approved by the NRC[15]. The 5 percent transition penalty for OFA is due to the higher OFA mixing vane loss coefficient compared to that of the ENC fuel. This results in localized flow redistribution from the OFA to the ENC assembly near mixing vane grid positions. When the full transition is complete (all ENC assemblies removed from core), the transition core penalty will no longer apply to OFA assemblies.

The ENC fuel assembly would be expected to have less gap closure than the W 15x15 OFA, due to the ENC fuel's thicker cladding, as shown in Reference 16. Data obtained by other investigations[17,18] show that gap closures up to 55% have no measurable effect on DNB. Therefore, no resultant rod bow DNBR penalty is required for ENC 15x15 fuel.

6.0 ACCIDENT ANALYSES AND EVALUATION

6.1 NON-LOCA ACCIDENT ANALYSES AND EVALUATION

The effects of the transition from the resident ENC fuel to W OFA on the non-LOCA accident analyses have been addressed. The standard Westinghouse reload methodology described in Reference 3 was used. All of the non-LOCA accidents* in the D. C. Cook FSAR were reanalyzed to include three major design changes:

1. The analyses were performed at a conservative reactor power level of 3411 MWt. This affects all of the transients that are limiting at full power.

2. The ITDP was used with both the WRB-1 and WRB-3 DNB correlations. This impacts all of the DNB limited accidents. A conservative set of core thermal safety limits, overtemperature delta T and overpower delta T setpoints were generated that are applicable for both the transition and complete OFA cores. These limits are valid for reactor power levels up to and including 3411 MWt.

3. The control rod scram time to the dashpot is increased from 1.8 seconds to 2.4 seconds. This increased drop time primarily affects the fast reactivity transients but was used in all of the analyses requiring this parameter.

Also included in the analyses were fuel temperatures based on the revised PAD code. A +5 pcm/degree F moderator temperature coefficient (MTC) existing at full power was conservatively used for heatup events. This is conservative since the Technical Specifications require a non-positive MTC at or above seventy (70) percent power.

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*With exception of startup on an inactive loop. This transient cannot occur above 10% rated thermal power and thus was not reanalyzed.

The acceptance criterion used in the non-LOCA safety analyses is independent of fuel vendor. Thus, the results of the FSAR Chapter 14 accident reanalysis and evaluation, which are contained in Attachment C, show that the transition to OFAs can be accommodated with margin to the applicable FSAR safety limits for power levels up to and including 3411 MWt.

6.2 LARGE BREAK LOCA (@ 3250 MWt)

Description of Analysis Assumptions for W 15x15 OFA Fuel, Including Transition Impact

The large break loss-of-coolant accident (LOCA) analysis for D. C. Cook Unit 1, applicable to a full W 15 x 15 OFA core, was analyzed to develop W 15 x 15 OFA fuel specific peaking factor limits. This analysis is consistent with the methodology employed in Reference 1. The currently approved 1981 large break ECCS evaluation model[19] was utilized for a spectrum of cold leg breaks. The revised PAD fuel thermal safety model[5] generated the initial fuel rod conditions. The D. C. Cook Unit 1 analysis was performed for an assumed steam generator tube plugging level of five (5) percent, and was analyzed for both minimum and maximum safeguards (safety injection flows) assumptions, in accordance with Reference 20. A revised FSAR chapter 14.3.1.1, given in Attachment D, contains a full description of the analysis and assumptions utilized for the W OFA ECCS LOCA analysis. The ENC fuel ECCS analysis contained in FSAR section 14.3.1.2 remains unchanged.

When assessing the LOCA impact of transition cores, it must be determined whether the transition core can have a greater calculated peak clad temperature (PCT) than either a complete core of the reference fuel

design or a complete core of the new fuel design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch.

For the ENC and W 15x15 OFA designs, this difference in fuel assembly resistance (K/A^2), is less than one percent. The different flow resistances for the two assembly designs impact two portions of the LOCA analysis model. One is the reactor coolant system (RCS) blowdown portion of the transient, analyzed with the SATAN VI computer code, where the higher resistance W OFA assembly has less cooling flow than the ENC assembly. While the SATAN VI computer code models the cross flow between the average core flow channel (N-1 fuel assemblies) and a hot assembly flow channel (one fuel assembly), experience has shown that the SATAN VI results are not significantly affected by small differences in the hydraulic resistance ($\pm$ 10%) between these two channels. Since small resistance mismatches in the core are insignificant when compared to the total system resistance, and since the total core resistance is uniformly distributed in the SATAN VI code, the effect on the large break LOCA blowdown transient of modeling hydraulic resistance mismatch can be neglected. Therefore, it is not necessary or meaningful to perform a new SATAN VI analysis for this transition core configuration because the hydraulic resistance mismatch is much less than $\pm$ 10 percent.

The other portion of the LOCA evaluation model impacted by the hydraulic resistance difference is the core reflood transient. Since the hydraulic mismatch is so small, only crossflows due to the smaller rod size and different grid designs need to be evaluated. The maximum reflood axial flow reduction for the W 15x15 OFA fuel at any location in the core, resulting from crossflows to adjacent ENC assemblies, has been conservatively calculated to be three percent. Analyses have been per-

formed, which demonstrate that a reduction of five (5) percent in reflood axial flowrate results in a 19°F PCT increase. Therefore, the maximum PCT penalty possible for W 15x15 OFA fuel during the transition period is 12°F. After this transition, the W ECCS analysis will apply to a full core without the crossflow penalty.

The resident ENC fuel is shown to have axial flowrates always greater than the nominal design flowrate, for core axial elevations where PCT's can possibly occur. Therefore, the ENC ECCS analysis is not detrimentally affected by assembly crossflow and remains applicable to the ENC fuel for transition cycles.

The method of analysis, including assumptions and codes used, are described in detail in the revised FSAR Chapter 14.3.1.1 provided in Attachment D.

The results of this analysis, including tabular and plotted results of the break spectrum analyzed, are provided in Attachment D.

Conclusions

For breaks up to and including the double ended severance of a reactor coolant pipe, the emergency core cooling system will meet the acceptance criteria as presented in 10 CFR 50.46. That is:

1. The calculated peak fuel element clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed one (1) percent of the total amount of Zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling. The localized cladding oxidation limit of seventeen (17) percent is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after the break.

5. The core temperature is reduced and decay heat is removed for an extended period of time as required by the long-lived radioactivity remaining in the core.

The time sequence of events for all breaks analyzed is shown in Table 14.3.1-6 of the revised FSAR Chapter 14.3.1.1, presented in Attachment D.

The large break W 15x15 OFA LOCA analysis for D. C. Cook Unit 1 utilizing the currently approved 1981 evaluation models resulted in a PCT of 2170°F for the 0.4 C_D (discharge coefficient) LOCA Maximum Safeguards Injection (Max. SI) case at a total peaking factor of 2.00

The small impact of crossflow for transition core cycles is conservatively evaluated to be at most a 12°F effect on the W fuel, which is easily accommodated in the margin to 10 CFR 50.46 limits.

The ENC ECCS analysis is not detrimentally affected by assembly crossflow; consequently the ENC peaking factor limits remain valid for the ENC fuel during the transition period.

It can be seen from the results contained in Chapter 14.3.1.1 of the revised FSAR section that this ECCS analysis for D. C. Cook Unit 1 remains in compliance with 10 CFR 50.46 of Appendix K.

6.3 SMALL BREAK LOCA (@ 3411 MWt)

Description of Analysis Assumptions for 15x15 OFA Fuel Including Transition Impact

The small break loss-of-coolant accident (LOCA) analysis for D. C. Cook Unit 1, applicable to a full W 15x15 OFA core, was analyzed to develop W 15x15 OFA fuel specific peaking factor limits. This is consistent with the methodology employed in Reference 1. The currently approved October 1975 small break ECCS evaluation model(21), was utilized for a spectrum of cold leg breaks. The revised PAD fuel thermal safety model(5), generated the initial fuel rod conditions. Revised FSAR chapter 14.3.2, given in Attachment E, contains a full description of the analysis and assumptions utilized for the W OFA ECCS LOCA analysis.

When assessing the impact of a LOCA on transition cores it must be determined whether the transition core can have a greater calculated peak clad temperature (PCT) than either a complete core of the reference fuel design or a complete core of the improved fuel design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch.

The WFLASH computer code(21) is used to model the core hydraulics during a small break event. Only one core flow channel is modelled in WFLASH since the core flowrate during a small break is relatively low and this provides enough time to maintain flow equilibrium between fuel assemblies (i.e. cross flow). Therefore, hydraulic resistance mismatch is not a factor for small break. Thus it is not necessary to perform a small break evaluation for transition cores, and it is sufficient to reference the small break LOCA for the complete core of the W 15x15 OFA design.

The methods of analysis, including assumptions and codes used, are described in detail in the revised FSAR Chapter 14.3.2 in Attachment E.

The results of this analysis, including tabular and plotted results of the break spectrum analyzed, are provided in Attachment E.

Conclusions

The small break optimized fuel LOCA analysis for D. C. Cook Unit 1, utilizing the currently approved 1975 Small Break Evaluation model, resulted in a peak clad temperature of 1630°F for the 4 inch diameter cold leg break. The analysis assumed the worst small break power shape consistent with a LOCA F_Q envelope of 2.32 at core midplane elevation and 1.5 at the top of the core.

Analyses presented in the revised FSAR Chapter 14.3.2 show that the high head portion of the ECCS, together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperature well below the required limits of 10 CFR 50.46. Adequate protection is therefore afforded by the ECCS in the event of a small break LOCA.

7.0 TECHNICAL SPECIFICATION CHANGES

Based on the preceeding evaluations, a number of technical specification changes for D. C. Cook Unit 1 are required to support the transition to OFA. These changes are given in the proposed Technical Specification page changes (see Attachment A of this submittal).

8.0 REFERENCES

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5. Rahe, E. P., Westinghouse Letter to C. O. Thomas of NRC, Letter Number NS-EPR-2673, October 27, 1982, Subject: "Westinghouse Revised PAD Code Thermal Safety Model," WCAP-8720, Addendum 2 (Proprietary).

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

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11. D. C. Cook Unit 1 Updated FSAR, updated thru 1982.

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15. Letter from C. O. Thomas (NRC) to E. P. Rahe (Westinghouse), Subject: Supplemental Acceptance Number 2 for Referencing of Licensing Topical Report WCAP-9500, NRC SER Letter dated January 24, 1983.

16. Letter from G. F. Owsley (ENC) to T.A. Ippolito (NRC); Subject: XN-75-32, Supplement 1, "Computational Procedure for Evaluating Fuel Rod Bowing," July 17, 1979.

17. Matkowski, et. al., "Effect of Rod Bowing of CHF in PWR Fuel Assemblies," ASME paper 77-HT-91.

8.0 REFERENCES (Continued)

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TABLE 1

Comparison of OFA and ENC Assembly Design

Parameter	15x15 W Optimized Fuel Assembly Design	15x15 ENC Fuel Assembly Design
Fuel Ass'y. Length, in.	159.765	159.71
Fuel Rod Length, in.	151.85	152.07
Assembly Envelope, in.	8.426	8.426
Compatible with Core Internals	Yes	Yes
Fuel Rod Pitch, in.	0.563	0.563
Number of Fuel Rods/Ass'y.	204	204
Number of Guide Thimbles/Ass'y.	20	20
Number of Instrumentation Tube/Ass'y	1	1
Compatible w/Movable In-Core Detector System	Yes	Yes
Fuel Tube Material	Zircaloy-4	Zircaloy-4
Fuel Rod Clad OD, in.	0.422	0.424
Fuel Rod Clad Thickness, in.	0.0243	0.030
Fuel/Clad Gap, mil	7.5	7.5
Fuel Pellet dia., in.	0.3659	0.3565
Guide Thimble Material	Zircaloy-4	Zircaloy-4
Guide Thimble ID, in.*	0.499	0.511
Structural Mat'l-Five Inner Grids	Zircaloy-4	Zircaloy-4 Straps Inconel Springs

*Above dashpot