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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 74 TO FACILITY OPERATING LICENSE NO. DPR-58

INDIANA AND MICHIGAN ELECTRIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NO. 1

DOCKET NO. 50-315

A. INTRODUCTION

By letter dated May 11, 1983, the Indiana and Michigan Electric Company (the licensee) submitted an application for the Donald C. Cook Nuclear Plant, Unit No. 1 reload for Cycle 8. The reload will include the first fuel batch fabricated by Westinghouse (W) of the 15X15 optimized fuel assembly design and the first use in the Donald C. Cook Nuclear Plant of the W Wet Annular Burnable Absorber (WABA) burnable poison rods. The reload fuel will have an enrichment up to 4.0 weight percent U 235 and may achieve extended burnup in future cycles to 39,000 MWD/MTU (average region discharge). The application has also defined a new term "design basis power level" of 3411 MWt at which a number of accidents and transients have been analyzed. However, no request has been made to increase the approved power level for operation and some of the more significant evaluations, i.e., large break loss of coolant accident (LOCA), have not been submitted at this higher power level.
The approved maximum power level for the Donald C. Cook Nuclear Plant, Unit No. 1 remains at 3250 Mwt.

On June 22, 1983, the NRC issued a "Monthly Notice: Amendments to Operating Licenses Involving No Significant Hazards Considerations; Duquesne Light Company et al." with the Office of the Federal Register for publication. That notice recognized the proposed core reload for Cycle 8 and the related changes to the Technical Specifications. In

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a related licensing action, on May 4, 1983, the NRC issued Amendments 73 and 55 to Facility Operating License Nos. DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, respectively. Those amendments revised the Technical Specification to permit storage of the W fuel with a uranium enrichment of less than or equal to 4.00 weight percent U-235.

Subsequent to the May 11, 1983 letter by the licensee, a number of supplements to the original proposal have been received and were used in the evaluation of the W fuel for Cycle 8 operation. The evaluation section includes a list of references to these supplements as well as other information used in the evaluation.

B. EVALUATION

1. Introduction:

By letter dated May 11, 1983, the Indiana and Michigan Electric Company (the licensee) made application to amend the Technical Specifications of the Donald C. Cook Nuclear Plant, Unit No. 1, in order to reload and operate the plant for Cycle 8. In support of the application, attachments A through G were appended to the letter. The Core Performance Branch has reviewed the application and prepared the following evaluation.

For Cycle 8 the licensee is switching fuel vendors from EXXON (ENC) to Westinghouse who performed the analyses for this reload. In addition, in anticipation of an application for a power increase from the currently licensed 3250 MWt to 3411 MWt, all analyses were performed at the higher power with the exception of the LOCA analysis.

2. Fuel Mechanical Design

The D. C. Cook, Unit 1, Cycle 8 reload core will consist of 80 Westinghouse 15x15 optimized fuel assemblies (OFAs) and 113 Exxon Nuclear (ENC) 15x15 fuel assemblies. Although the Westinghouse 15x15 OFA fuel is a new design, it is very similar to the Westinghouse 15x15 standard low parasitic (LOPAR) fuel design, which previously operated in Cook Unit 1 and has substantial commercial operating experience. The major change introduced by the 15x15 OFA design is the use of five intermediate Zircaloy grids replacing five intermediate Inconel grids in the LOPAR fuel. The Zircaloy grids have thicker and wider straps than the Inconel grids in order to closely match the Inconel grid strength. Furthermore, the 15x15 OFA Zircaloy grid design is similar to the Westinghouse 17x17 OFA grid design, which was decribed in WCAP-9500-A (Ref. 1), which has been reviewed and approved by the NRC.

In performing our review of the 15x15 OFA fuel for Cook, Unit 1, we asked the licensee to verify that the design criteria and evaluation methods used for 17x17 OFA in WCAP-9500-A were also used for Cook's 15x15 OFA. The licensee verified that both criteria and methods were exactly the same (Ref. 2). The balance of our review thus focused on those plant-specific issues identified in the SER for WCAP-9500-A insofar as they are applicable to Cook, Unit 1, Cycle 8. Our evaluation of those issues follows.

2.1 Cladding Collapse

The licensee uses an approved method described in WCAP-8377 (Ref. 3) to analyze cladding collapse. The result for Cook, Unit 1 shows that no cladding collapse is expected up to 40,000 EFPH (in excess of 50,000

MWd/MTU peak-rod average burnup) for the new Westinghouse fuel design. The ENC fuel remains bounded by the previously accepted analysis. We conclude, therefore, that no cladding collapse is expected during Cycle 8 operation.

2.2 Rod Bowing

The rod bow magnitude for the Westinghouse OFA fuel was calculated with an approved method described in WCAP-8691, Revision 1 (Ref. 4). The rod bow magnitude for the ENC fuel was calculated in an earlier Cook, Unit 1 reload safety analysis and found to be acceptable by the NRC staff. Penalties associated with these adequately calculated bow magnitudes are discussed in Section 4.0 of this evaluation.

2.3 Fuel Thermal Conditions

The D. C. Cook, Unit 1, Cycle 8 reload submittal (Ref. 5) is based, in part, upon fuel thermal analyses generated with a revised (Ref. 6) version of a previously approved Westinghouse code called PAD (Ref. 7). The single revision to the PAD code is currently under staff review. A request for additional information was issued (Ref. 8) and responses (Ref. 9) have been obtained from the fuel vendor (Westinghouse).

Due to unexpected computational difficulties, the responses obtained from Westinghouse have not shown that certain analytical assumptions (e.g., worst time in life) continue to be met with the revised version of PAD. Pending resolution of this problem, and to avoid impacting the Cycle 8 reload schedule, the licensee submitted an addendum (Ref. 10) to the Cycle 8 reload report which (partially) reverts back to the previously approved version of PAD. The reanalysis results in a slightly lower LOCA Fq limit of 1.97, compared to an Fq of 2.00 using the revised thermal safety model (Ref. 6). The lower Fq limit and its associated K(Z) envelope have been incorporated into the revised Technical Specifications for D. C. Cook, Unit 1.

The revised Fq limit is based on an updated large break LOCA analysis described in Attachment C to Reference 10. The worst break was reanalyzed (at 3250 MWt) using previously approved methods, including the approved version of PAD. Results show that the D. C. Cook, Unit 1 emergency core cooling system will meet the acceptance criteria in 10 CFR 50.46 for Cycle 8 conditions. We find this result, and the manner in which it was obtained, acceptable. The manner in which the revised Fq limit and associated K(Z) envelope have been incorporated into the plant Technical Specifications has also been examined (see Section 3.0 of this SER) and found acceptable.

Other non-LOCA analyses in the Cycle 8 submittal continue to rely on the unapproved version of PAD. However, Westinghouse has performed (Ref. 10) an evaluation to determine if the use of the revised PAD model impacts other core operating limits. The initial fuel conditions used in non-LOCA transients were re-examined and it was found that the revised PAD code has only a slight impact on the safety analysis. In all cases, the appropriate design bases are still met. The small break LOCA ECCS analysis was not reanalyzed because the event is not limiting. In addition, cladding heatup occurs after core uncovery for this event and is not sensitive to changes in initial stored energy.

We conclude that the methods used to determine fuel thermal conditions, including limited use of the unapproved, revised version of PAD, are acceptable in support of the D. C. Cook, Unit 1, Cycle 8 reload safety analysis and the resulting modifications to the plant Technical Specifications.

2.4 Cladding Swelling and Rupture

For large break loss-of-coolant accident analysis, the licensee used the approved 1981 large break ECCS evaluation model (Ref. 11), which includes an approved cladding swelling and rupture model. The use of this ECCS model obviates the need for supplemental ECCS calculations mentioned in the SER for WCAP-9500-A. We thus find that cladding swelling and rupture have been adequately treated in the Cycle 8 reload analysis.

2.5 Seismic and LOCA Loads

Three major fuel types have been recently analyzed for seismic-and-LOCA loads in Cook Unit 1. These fuel types are: (1) LOPAR (standard Westinghouse Inconel-grid 15x15 fuel, now completely discharged from Cook Unit 1), (2) ENC (Exxon Nuclear Zircaloy-grid 15x15 fuel that constitutes the entire Cycle-7 core), and (3) OFA (new Westinghouse 15x15 Optimized Fuel Assemblies to be loaded in one region of the core for Cycle 8). Exxon Nuclear previously performed a seismic (only) loads analysis for a mixed-core configuration of LOPAR and ENC fuel; that analysis demonstrated that fuel rod and guide tube integrity and core coolable gemoetry would be maintained (Ref. 14). As part of the present reload safety analysis, Westinghouse performed a seismic-and-LOCA loads analysis for a mixed-core configuration of ENC and OFA fuel; that analysis demonstrated that fuel rods and guide tubes (thimbles) have ample margin (almost a factor of 2) even when seismic-and-LOCA loads were combined (Ref. 2). In the Westinghouse analysis, spacer grids had adequate margin to withstand seismic-and-LOCA loads separately, but grid deformation in core-peripheral fuel assemblies would be expected if seismic-and-LOCA loads were combined.

Several circumstances are noteworthy. First, Cook Unit 1 is one of the plants covered by a Westinghouse Owners' Group analysis that shows that pipe cracks will leak before they break so that the large LOCA load will not be present (Ref. 15). In light of that analysis, Cook Unit 1 does not presently have an obligation to address LOCA loads in the conservative manner analyzed by Westinghouse. Second, Westinghouse has shown in other cases (Ref. 16) that grid deformation has small consequences even when it is assumed to occur (less than 20°F increase in LOCA peak cladding temperature). Third, both the Exxon Nuclear and Westinghouse analyses mentioned above involved assumptions about the competitor's fuel design since neither Westinghouse nor Exxon Nuclear possesses complete details of each other's fuel design.

In light of the above circumstances and results -- particularly the large margin on the important guide tubes (thimbles) -- we conclude that all combinations of LOPAR, ENC, and OFA in Cook Unit 1 meet the appropriate mechanical loads requirements.

2.6 Wet Annular Burnable Absorbers

Cycle 8 will utilize a new burnable poison design, the Wet Annular Burnable Absorber (WABA), in 68 of the OFA's. The WABA rod design consists of annular pellets of aluminum oxide and boron carbide (Al_2O_3-B_4C) burnable absorber material encapsulated within two concentric Zircaloy tubings. The reactor coolant flows inside the inner tubing and outside the outer tubing of the annular rod. The topical report describing the WABA design (Ref. 12) has been recently reviewed and approved (Ref. 13), and the utilization of WABA rods in D. C. Cook 1 would thus be automatically approved subject to certain conditions described in the NRC approval of the generic topical report (those conditions concern surveillance and the analysis of core bypass flow). The WABA surveillance is discussed in Section 2.7 and the analysis of core bypass flow is discussed in Section 4.0 of this evaluation.

2.7 Post-irradiation Surveillance

As indicated in SRP* Section 4.2.II.D.3, a post-irradiation fuel surveillance program should be established to detect anomalies or confirm expected fuel performance.

The licensee states that a routine fuel inspection program will be implemented on the irradiated and discharged OFAs from the initial reload region (Ref. 2). The program involves visual examination on a representative sample of assemblies from the initial fuel region during each refueling until this fuel is discharged. Visual examination includes, but is not limited to, crud buildup, rod bowing, grid strap conditions, and missing parts. Additional fuel inspections would be performed if coolant activity or visual inspections indicate a need. We conclude that this satifies the fuel surveillance guidelines in the SRP 4.2.

As for the WABAs, the licensee agrees to have a supplementary surveillance program as described in Reference 13 if D. C. Cook Unit 1 is the first or second lead plant to discharge the WABAs. We find this acceptable.

2.8 Conclusion

We have reviewed the fuel assembly mechanical design for Cook, Unit 1, Cycle 8. We conclude that the Cycle-8 fuel mechanical design, which includes the Westinghouse 15x15 Optimized Fuel Assemblies (OFAs) and the Wet Annular Burnable Absorbers (WABAs), is acceptable.

* SRP - Standard Review Plan

3. Nuclear Design

For this cycle, 80 of the ENC assemblies will be replaced by 80 Westinghouse 15x15 Optimized Fuel Assemblies (OFA). These assemblies are identical to the Westinghouse 15x15 LOPAR (low parasitic) assemblies except that five of the interior Inconel grids have been replaced by Zircaloy grids. The LOPAR assemblies have substantial operating experience in a number of plants. The Westinghouse OFA assemblies are nearly identical from a neutronics point of view to the ENC assemblies which they replace.

The nuclear design and analysis of the D. C. Cook core was performed with the Westinghouse Reload Safety Evaluation Methodology. This methodology has been previously employed for reload design in several reactors and we find its use acceptable for the present reload. The analyses were performed for a series of cycles which proceed from Cycle 8 to a core completely loaded with the Westinghouse OFA fuel. The neutronics parameters used as input to the safety analyses were then chosen to bound the values obtained from this series. In addition the analyses were done at a power level of 3411 MWt except for the LOCA analysis as noted above.

The licensee has included a listing of the neutronics parameters used in the safety analysis to provide bounding values against which cycle dependent parameters may be compared. We conclude that the nuclear design analysis is acceptable.

4. Thermal-Hydraulic Evaluation

The D. C. Cook Unit 1 Cycle 8 core consists of 80 Westinghouse 15x15 optimized fuel assemblies (OFA) and the 113 remaining Exxon 15x15 standard fuel assemblies. Sixty-eight (68) of the 80 OFA's employ the wet annular burnable absorber (WABA) poison rods. The OFA and standard fuel assemblies have been tested and the results show that they are hydraulically compatible with the pressure drops within 0.7 percent of each other.

The thermal-hydraulic analysis of this mixed core was performed using the improved thermal design procedures (ITDP) and the THINC IV code. The WRB-1 and W-3 CHF*correlations were used for the Westinghouse OFA and the ENC fuel assemblies, respectively. The ITDP, THINC IV code, and both CHF correlations have previously been approved by the staff. However, there are areas requiring additional evaluation regarding this transitional mixed core configuration. These areas are addressed as follows:

(a) The WRB-1 correlation was approved for the 17x17 OFA, and 17x17 and 15x15 standard LOPAR fuel assemblies with DNBR limit of 1.17 for R-grid. No CHF test data is available for the 15x15 OFA and, therefore, the application of the WRB-1 correlation to the 15x15 OFA is of concern. In response to staff questions, the licensee provided W 14X14 OFA CHF test data and additional proprietary information regarding the design of the 15x15 OFA. The 15x15 OFA design is virtually identical to the 15x15 R-grid design. A scaling technique was used in the 15x15 OFA grid design to ensure that the DNB performance is not affected by the OFA grid. This scaling technique has also been used for the design of the 17x17 and 14x14 OFA grids. In order to evaluate the effect of the geometry change on the accuracy of the WRB-1 correlation, Westinghouse also performed a statistical analysis using the T-tests and F-tests for the 17x17 standard/OFA data and the 14x14 standard/OFA data. The results show that the null hypothesis that the WRB-1 correlation predicts the DNB behavior of the OFA geometry with the same accuracy as the standard R-grid geometry can not be rejected at a 5% significance level. For the case where the F-test rejects the null hypothesis, the OFA data have an appreciably lower variance which is indicative of better correlation accuracy. Therefore, even though no 15x15 OFA CHF data is available, the statistical analysis performed by Westinghouse has provided the basis for the applicability of the WRB-1 correlation on the 15x15 OFA.

(b) The use of ITDP for the analysis of a transitional mixed core has been previously reviewed by the staff and approved with a condition requiring a

* CHF - Critical Heat Flux

penalty on DNBR to account for the uncertainty associated with the inter-bundle cross-flow in the mixed core.

The licensee has performed an analysis to determine the required penalty factor in the same manner approved for the 17x17 OFA/LOPAR mixed core analysis. The result shows that a 5% penalty is required on the OFA for the Cycle 8 transitional core.

(c) The Westinghouse WABA poison rod design is described in WCAP-10021, Revision 1 which has been approved by the NRC. In order to ensure no violation of the total core bypass flow limit, the total number of WABA rods in the core should be less than the upper limit established in Table 7.2 of WCAP-10021, Revision 1. Since only 68 OFA assemblies employ WABA with a total of 864 WABA rod for Cycle 8 core, the limit is not exceeded and is therfore of no concern.

(d) The Cycle 8 projected maximum assembly burnup is 36,800 MWD/MTU for the ENC fuel. The staff audit calculation has determined that the maximum gap closure will be 40.4% for the ENC fuel by the end of Cycle 8. Therefore, no rod bow penalty is required for the ENC fuel because investigations have shown that gap closure of less than 50% has no measurable effect on DNB.

(e) The core thermal-hydraulic analysis was performed by conservatively using 3411 MWt core power and 577.1°F average coolant temperature compared to the rated values of 3250 MWt and 567.8°F, respectively for the typical and thimble cells using the ITDP. The safety analysis DNBR limit is 1.69 for both typical and thimble cells. This safety limit is 28% higher than the design limit and the margin is more than enough to account for the rod bow penalty, the transitional mixed core penalty and any uncertainty associated with the application of WRB-1 on 15x15 OFA with DNBR limit of 1.17. For the ENC fuel, the W-3 correlation with DNBR limit of 1.30 was used, and the design safety limits are 1.58 and 1.50 for the typical cell and thimble cell, respectively. We conclude that the thermal-hydraulic analysis is acceptable.

5. Transient and Accident Analyses

All of the non-LOCA transients and accidents except startup of an inactive loop were reanalyzed to include three major design changes:

1. An increased power level of 3411 MWT
2. Use of the Improved Thermal Design Procedure with both the WRB-1 and W-3 DNB correlations
3. Increase of control rod scram time from 1.8 to 2.4 seconds. This change is necessitated by the reduction in ID of the thimbles in the OFA guide assemblies.

In addition, fuel temperatures were based on the revised PAD code and a 5 pcm/degree F MTC* at full power was used for heatup events. Standard Westinghouse codes and procedures were used for these analyses.

All the transients and accidents and the LOCA were done using approved methods and acceptable initial conditions. The results presented were acceptable since they did not violate the DNBR limit nor did they exceed the maximum pressure and temperature limits.

However, it is important to clarify that this SER approves the transient and accident analysis for operation of Cycle 8 only and in no way does it approve the plant to operate at the higher power level of 3411 MWt. If Cook 1 is planning to operate at the higher power level of 3411 an independent review of the LOCA and following transient accidents, is necessary.

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* MTC - Moderator Temperature Coefficient

1. malfunction of the CVCS
2. loss of reactor coolant flow
3. locked rotor event
4. loss of external load
5. loss of normal feedwater
6. excessive heat removal due to feedwater system malfunction
7. excessive load increase incident
8. loss of all AC power to station auxiliaries
9. rupture of a steam pipe

The following transients and accidents have been reviewed at the higher power level and a detailed discussion is presented. These are:

1. bank withdrawal at low power
2. bank withdrawal at power
3. rod cluster control assembly misalignments
4. rod ejection accident

5.1 Bank Withdrawal at Low Power (Startup Accident)

The consequences of the insertion of reactivity at a rate of 75 pcm/second were calculated assuming a moderator temperature coefficient of 5 pcm/°F. This insertion rate is greater than that due to the withdrawal of the two sequential banks having the greatest combined worth at maximum speed (45 inches/minute). The peak heat flux during the transient is less than 50 percent of that at full power. We conclude that fuel thermal limits are not violated and that the analysis is acceptable.

5.2 Bank Withdrawal at Power

This event is analyzed at 100 percent, 60 percent, and 10 percent of full power. Minimum and maximum reactivity feedback effects are included as well as reactivity insertion rates up to values greater than that for the simultaneous withdrawal of the combination of the two control banks having the maximum combined worth at maximum speed. Trip occurs on high neutron flux for the high withdrawal rates and on the overtemperatureΔT trip for the low withdrawal rates. The minimum DNBR is 1.8 at full power, 1.85 at 60 percent power and 3.96 at 10 percent power. This meets the safety analysis limit of 1.69 for OFA and 1.58 for ENC fuel.

Based on the fact that approved analysis procedures and methods are used and that the resulting minimum DNBR values meet the relevant safety limits, we conclude that the analysis of the rod withdrawal event at power is acceptable.

5.3 Rod Cluster Control Assembly Misalignments

This category includes statically misaligned rods, dropped rods and dropped rod banks. The methodology used is described in document NS-EPR-2595, "Dropped Rod Methodology for Negative Flux Rate Trip Plants" which has been reviewed and approved by the staff.

Two static misalignment cases are analyzed - Bank D inserted with one rod fully withdrawn and one rod fully inserted with Bank D withdrawn. In the first case the calculation determines the amount by which Bank D may be inserted before fuel thermal limits are violated. The result is used in establishing the Technical Specification limits on Bank D insertion (other considerations usually determine these limits). The consequences of the single rod completely inserted while the rest of Bank D is withdrawn is analyzed by computing the resulting DNBR including the effect of the increased peaking factor. Fuel thermal limits are met for this case. Inspection of peaking factors obtained when a rod from another bank is on the bottom shows that the analyzed case is limiting.

Most dropped rods or dropped banks will result in a negative flux rate trip at about 2.5 seconds. Since power is decreasing at this point no thermal limits are approached and the operator follows procedures for a reactor scram. For rods with insufficient worth to cause the trip two cases are analyzed - reactor in manual control and reactor in automatic control. In the first case the reactor reaches a new steady-state configuration at a power not higher than the initial power. This case is bounded by the case of a static rod completely inserted with the D bank withdrawn.

In the second case the automatic controller will respond to the initial reduction in power by withdrawing rods which, in the limiting case, results in a power overshoot. In a typical case a 10 percent power overshoot occurs. The range of potential dropped rod cases has been investigated and in all cases thermal limits were not violated.

On the basis that approved methods were used and the results do not show a violation of fuel thermal limits, we conclude that the analysis of the rod misoperation events is acceptable.

5.4 Rod Ejection Accident

This accident postulates the rupture of a control rod drive mechanism housing and the consequent rapid ejection of the control rod from the core. This event has been analyzed by standard Westinghouse methods which have been shown to be conservative with respect to the three-dimensional calculations.

Four cases were analyzed-full power at beginning-and end-of-life and zero power at beginning-and end-of-life. Conservative values of ejected rod worth were used along with conservatively low values of delayed neutron fractions. The calculated maximum fuel enthalpy values ranged from 147 to 186 calories per gram. These values meet the acceptance criterion for this quantity of 280 calories per gram as given in Regulatory Guide 1.77. Less than 10 percent of the hot pellet melts in the two full power cases.

Less than 10 percent of the rods in the core experience departure from nucleate boiling during the event. No significant pressure surge occurs and the maximum pressure does not exceed that for emergency conditions as required by Regulatory Guide 1.77. We conclude that the analysis of the rod ejection event is acceptable.

6. Technical Specification

Changes have been proposed to the Cook Unit 1 Technical Specifications in order to account for the use of the Improved Thermal Design Procedure (ITDP), the analysis of non-LOCA events at 3411 MWt, and the introduction of Westinghouse OFA Fuel into the core. Each proposed change from Ref. 5 and 17 is discussed below.

Definition 1.27

A new power term, DESIGN THERMAL POWER (3411 MWt) is introduced in order to take advantage of the fact that safety analyses were done at 3411 MWt. In particular, the Overtemperature ΔT and Overpower ΔT trips have been recalculated for the increased power. The RATED THERMAL POWER, appearing in most specifications, is still 3250 MWt. We find this definition acceptable.

Figure 2.1-1

This figure provides the low points of the thermal power, RCS pressure and average temperature as reactor core safety limit for 4-loop operation to avoid violation of the design DNBR limit using the improved thermal design procedure. This figure is identical to Figure 3 of the Attachment C to AEP:NRC: 07450 in which the "fraction of design thermal power" is used in the abscissa and a conversion factor of (design thermal power/rated thermal power) is needed to convert the abscissa to "fraction of rated thermal power".

Table 2.2-1 (Items 7 and 8)

The algorithms for the Overtemperature ΔT and Overpower ΔT trips have been altered to reflect the use of the ITDP, the use of two different DNB correlations (WRB-1 for the Westinghouse fuel and W-3 for the ENC fuel) and the analyses at 3411 MWt. On the basis that these algorithms have been constructed by the methods which have been successfully employed on other Westinghouse reactors, we find them to be acceptable.

Bases for Specification 2.1.1 and 2.2.1

These bases have been changed to reflect the fact that two different fuel types having different DNBR limits and values of $F^N_{\Delta H}$ are present in the core and that the ITDP is used. In addition, values of the design and safety analysis values of DNBR for the two correlations are given. These changes are acceptable.

Specification 3/4.1.1.1

This specification has been modified to change the required shutdown margin from 1.75% to 1.60% reactivity change. The new value is consistent with the new steamline break analysis and is acceptable.

Specification 3/4.1.3.3

The rod drop time in this specification has been increased to $<$2.4 seconds. The change is necessary to account for the smaller diameter of the guide tubes in the optimized fuel assemblies. Since the safety analyses performed for D. C. Cook Unit 1 used the new value we find the proposed Technical Specification change acceptable.

Figures 3.1.-1 and 3.1-2

These figures show the rod group insertion limits for three-loop and four-loop operation respectively. Since these were obtained by using standard Westinghouse methodology we conclude that they are acceptable.

Specification 3/4.2

This specification has been expanded to include both Westinghouse OFA and Exxon (ENC) fuel. The format of this - the $F_q(z)$ specification - has been retained from the current specification and the OFA fuel specification has been cast in the same format with appropriate curves for the various parameters. The peaking factor of 1.97 for the Westinghouse fuel is consistent with the Cycle 8 LOCA analysis and is acceptable.

Specification 4/3.2.3

The specification is revised to include the $F^N_{\Delta H}$ value for the Westinghouse Fuel. The limiting values reflect the use of the ITDP. This is acceptable.

Specification 4/3.2.4

The editorial changes made here for clarity are acceptable.

Specification 4/3.2.6

The changes in this specification consisted in adding the Westinghouse OFA specifications and inserting a reference to the peaking factor limit report which contains the V(z) function. These changes are acceptable under the condition that the peaking factor limit report is transmitted to NRC for review 60 days prior to the scheduled startup date for the new cycle.

Table 3.3-1

A footnote has been added to certain of the FUNCTIONAL UNITS in this table to indicate that the provisions of Specification 3.0.4, dealing with entry into another operational mode is not applicable. This is consistent with Westinghouse Standard Technical Specifications and is acceptable. An addition to Action Statement 1 permits the bypassing of one channel for up to 3 hours to permit surveillance. This time is required because of the increased complexity of the surveillance procedures and is acceptable. Other changes in the table are editorial in nature and are acceptable.

Specification 4.10.1.2

This specification has been altered to make it consistent with Specification 3/4.1.3.3 (see above) and is acceptable.

7. Radiological Consequences

The licensee does not propose to increase the operating power level of the Unit 1 and does not propose to increase burnup for Cycle-8 beyond the 37,000 MWD/MTU batch average at discharge which we have previously considered and found acceptable generically. Therefore, the conclusions stemming from accident radiological analyses of record at 3250 MWt for fuel at 37,000 MWD/MTU (or the existing average burnup in Cook Unit 1, whichever is higher) are still valid. A complete radiological consequence analysis will be required for any proposed increase in the operating power level.

8. Spent Fuel Pool Cooling

The proposed reload involves fuel enriched to 4.00 weight percent U 235 This will result in increased burnup and thus decay heat production in the spent fuel pool when the fuel is eventually removed from the core, i.e., at the end of Cycle 10. We have reviewed the licensee submittal from the standpoint of decay heat load and spent fuel pool cooling capability and conclude that the increased enrichment of the fuel produces a negligible addition to the total decay heat production profile. Thus we conclude that the existing spent fuel pool cooling system is capable of handling the increased heat load.

9. Summary

We have reviewed the information submitted on Cycle 8 reload for D. C. Cook Unit 1. We find the Cycle 8 operation acceptable for the fuel system mechanical design, nuclear design, thermal hydraulic, transients and accidents, the Technical Specification proposed, and radiological consequences. In addition, we find the enriched fuel to have insignificant effect on the spent fuel pool cooling capability when the fuel is eventually discharged.

However, as stated in Section 5, the transient and accident and LOCA design are acceptable for the Cycle 8 only and operation at the higher power level of 3411 MWt will require that additional review be performed independent of this evaluation.

10. References

1. R. L. Tedesco (NRC) letter to T. M. Anderson (Westinghouse), "Reference Core Report 17x17 Optimized Fuel Assembly", May 22, 1981.

2. R. F. Hering (AEP) letter to H. R. Denton (NRC), August 31, 1983.

3. V. Stello (NRC) memorandum to R. DeYoung (NRC), "Evaluation of Westinghouse Report WCAP-8377, Revised Clad Flattening Model", January 14, 1975.

4. L. S. Rubenstein (NRC) memorandum to T. M. Novak (NRC), "SERs for Westinghouse, Combustion Engineering, Babcock & Wilcox, and Exxon Fuel Rod Bowing Topical Reports", October 25, 1982.

5. R. S. Hunter (I&MEC) letter AEP:NRC:0745C to H. R. Denton (NRC) on "Application for Reload License Amendment Using Westinghouse Optimized Fuel Assemblies", dated May 11, 1983.

6. W. J. Leech, D. D. Davis and M. S. Benzvi, "Revised PAD Code Thermal Safety Model", Westinghouse Electric Corporation Report WCAP-8720, Addendum 2 (Proprietary), October 1982. Submitted by E. P. Rahe, Jr. (Westinghouse) letter NS-EPR-2673 to C. O. Thomas (NRC) dated October 27, 1982.

7. J. V. Miller et al., "Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations", Westinghouse Electric Corporation Reports WCAP-8720 (Proprietary), October 1976, and WCAP-8720, Addendum 1 (Proprietary), September 1979.

8. C. O. Thomas (NRC) letter to E. P. Rahe, Jr. (Westinghouse) on "Request Number 1 for Additional Information on WCAP-8720, Addendum 2", dated May 3, 1983.

9. E. P. Rahe, Jr. (Westinghouse) letter NS-EPR-2773 to C. O. Thomas (NRC) on "Response to Request for Additional Information on WCAP-8720, Addendum 2", dated June 2, 1983.

10. M. P. Alexich (I&MEC) letter AEP:NRC:0745F to H. R. Denton (NRC) on "Application for Unit 1 Cycle 8 Reload License Amendment: Addendums and Answers to NRC Questions", dated July 29, 1983.

11. E. P. Rahe, "Westinghouse ECCS Evaluation Model, 1981 Version", WCAP-9220-P-A (Proprietary), Revision 1, 1981.

12. E. P. Rahe, Jr. (W), letter to C. O. Thomas (NRC), "W WABA Evaluation Report", ECAP-10021, Revision 1, (Proprietary), October 18, 1982.

13. L. S. Rubenstein (NRC), memorandum for F. J. Miraglia, "SER of Westinghouse WABA Design", June 1, 1983.

14. "Lateral Core Seismic Analysis for ENC's 15x15 Reload Fuel Westinghouse Plants", XN-NF-52, September 1975.

15. "Mechanical Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack", WCAP-9558, Revision 2, May 1982; "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation", WCAP-9787, May 1981.

16. L. S. Rubenstein (NRC) memorandum, "SER Input for Millstone, Unit 2 Cycle 5 Reload", to T. M. Novak (NRC), February 18, 1982.

17. M. P. Alexich (IMEC) letter AEP:NRC:0745G to H. R. Denton (NRC) on "Additional Technical Specification Changes for Unit 1 Cycle 8," dated July 25, 1983.

18. R. F. Hering (IMEC) letter AEP:NRC:0745H to H. R. Denton (NRC) on "Further Answers to NRC Questions Concerning the Unit 1, Cycle 8 Reload" dated August 31, 1983.

C. 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.

D. CONCLUSION

We have concluded, based on the consideration 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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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