

May 14, 1985

Docket Nos. 50-250
and 50-251

Mr. J. W. Williams, Jr., Group Vice President
Nuclear Energy Department
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408

<u>Distribution</u>	
<u>Docket file</u>	NRC PDR
ORB#1 RDG	L PDR
Gray file (4)	HThompson
CParrish	DMcDonald
OELD	LHarmon
EJordan	BGrimes
JPartlow	TBarnhart (8)
WJones	EButcher
ACRS (10)	OPA, CMiles
RDiggs	

Dear Mr. Williams:

SUBJECT: SUPPLEMENT TO THE SAFETY EVALUATION IN SUPPORT OF AMENDMENT
NOS. 99 AND 93 FOR THE TURKEY POINT PLANT

On December 23, 1983, the Commission issued Amendment No. 99 to Facility Operating License No. DPR-31 and Amendment No. 93 to Facility Operating License No. DPR-41 for the Turkey Point Plant Units 3 and 4, respectively. The amendments changed the Technical Specifications to support the integrated program for vessel flux reduction and to take credit for operation with the new steam generators. The amendments are the subject of a current proceeding, ASLB No. 84-504-07 LA.

The Board was notified by letter dated March 18, 1985 from your attorney, Mr. Michael Bauser, that it was necessary to revise the data transfer procedure between the WREFLOOD and BART codes which are part of the Westinghouse Emergency Core Cooling System (ECCS) evaluation model used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR 50. It was further stated that the new procedure resulted in an increase in calculated Peak Clad Temperature (PCT) but would be well below the 2200°F limit established in 10 CFR 50.46.

By letter dated March 22, 1985 from E. P. Rahe (Westinghouse) to D. G. Eisenhut (NRC), the NRC staff was informed of the problem with the data transfer procedure between the WREFLOOD and BART codes. Details which led to the identification of the problem, corrective actions taken and a reanalysis for the Turkey Point Plant were provided.

You confirmed the results of the reanalysis by letter dated April 16, 1985 from J. W. Williams (FPL) to S. A. Varga (NRC). The new results show an increase in the calculated PCT of 79°F to 2051°F from the original calculated PCT of 1972°F. The new value is well below the 2200°F limit established in 10 CFR 50.46. In our initial Safety Evaluation (SE) we indicated that the results of an analysis using the previously approved ECCS evaluation model using the FLECHT correlation in lieu of the BART code also resulted in a PCT less than the 2200°F limit.

We have concluded in the enclosed SSE that the ECCS evaluation model using the BART code meets the requirements of 10 CFR 50, Appendix K, and the results of the reanalysis are acceptable. Our initial conclusions have remained valid based on the results of the analyses using the FLECHT correlation and all other conclusions regarding BART and the results of the large break LOCA analysis (Section 4.2 of our SE) remain valid.

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Mr. Williams

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May 14, 1985

As the result of our SSE, we have revised the PCT value of 1972°F to 2051°F in Sections 4.2 and 7 and the value of the reduction in PCT has been revised from 160°F to 81°F in Section 7 of the SE, dated December 23, 1983, issued in support of the subject amendments. The revised SE dated May 14, 1985 is included as Enclosure 2 and replaces our initial SE.

The acceptability of the new methodology for data transfer procedure between the WREFLOOD and BART codes is reflected in an SSE on the BART computer code methodology. Reference 14 of Enclosure 2, which references the initial SE for the BART code has been revised to also include a reference to the SSE for the BART code.

Sincerely,

/s/SAVarga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Enclosures:

1. SSE Related to Amendment 99 to
DPR-31 and Amendment No. 93 to
DPR-41 for Turkey Point Units 3 & 4
2. Safety Evaluation, Rev. 1
dated May 14, 1985

cc w/enclosures:
See next page

*SEE PREVIOUS WHITE FOR CONCURRENCE

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05/01/85

BC-ORB#1:DL*
SVarga
05/01/85

OELD*
MYoung
05/13/85

AD:OR:DL*
GLainas
05/14/85

conclusions regarding BART and the results of the large break LOCA analysis (Section 4.2 of our SE) remain valid.

As the result of our SSE, we have revised the PCT value of 1972°F to 2051°F in Sections 4.2 and 7 and the value of the reduction in PCT has been revised from 160°F to 81°F in Section 7 of the SE, dated December 23, 1983, issued in support of the subject amendments. The revised SE dated 1985 is included as Enclosure 2 and replaces our initial SE.

The acceptability of the new methodology for data transfer procedure between the WREFLOOD and BART codes is reflected in an SSE on the BART computer code methodology. Reference 14 of Enclosure 2, which references the initial SE for the BART code has been revised to also include a reference to the SSE for the BART code. Therefore, the ECCS evaluation model using the BART code meets the requirements of 10 CFR 50, Appendix K, and is acceptable as a reference for the Turkey Point Plant.

Sincerely,

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

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Mr. Williams

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The acceptability of the new methodology for data transfer procedure between the WREFLOOD and BART code is being reflected in an SSE on the BART computer code methodology.

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

May 14, 1985

Docket Nos. 50-250
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Nuclear Energy Department
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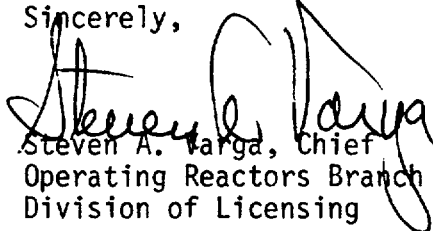
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J. W. Williams, Jr.
Florida Power and Light Company

Turkey Point Plant

cc: Harold F. Reis, Esquire
Newman and Holtzinger, P.C.
1615 L Street, N.W.
Washington, DC 20036

Mr. Jack Shreve
Office of the Public Counsel
Room 4, Holland Building
Tallahassee, Florida 32304

Norman A. Coll, Esquire
Steel, Hector and Davis
4000 Southeast Financial
Center
Miami, Florida 33131-2398

Mr. C. M. Wethy, Vice President
Turkey Point Nuclear Plant
Florida Power and Light Company
P.O. Box 029100
Miami, Florida 33102

Mr. M. R. Stierheim
County Manager of Metropolitan
Dade County
Miami, Florida 33130

Resident Inspector
Turkey Point Nuclear Generating Station
U.S. Nuclear Regulatory Commission
Post Office Box 57-1185
Miami, Florida 33257-1185

Mr. Allan Schubert, Manager
Public Health Physicist
Department of Health and
Rehabilitative Services
1323 Winewood Blvd.
Tallahassee, Florida 32301

Intergovernmental Coordination
and Review
Office of Planning & Budget
Executive Office of the Governor
The Capitol Building
Tallahassee, Florida 32301

Administrator
Department of Environmental
Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

Dr. J. Nelson Grace
Regional Administrator, Region II
U.S Nuclear Regulatory Commission
Suite 2900
101 Marietta Street
Atlanta, GA 30303

Martin H. Hodder, Esquire
1131 N.E. 86th Street
Miami, Florida 33138

Joette Lorion
7269 SW 54 Avenue
Miami, Florida 33143

Mr. Chris J. Baker, Plant Manager
Turkey Point Nuclear Plant
Florida Power and Light Company
P.O. Box 029100
Miami, Florida 33102

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

SSE RELATED TO AMENDMENT 99 TO DPR-31 AND AMENDMENT NO. 93
TO DPR-41 FOR TURKEY POINT UNITS 3 AND 4

The purpose of this SSE is to address the change in input methodology for the BART computer code as reported to the Board on March 18, 1985. In consideration of that issue some changes are made in the subject SER (reference 1).

In reference 2, the NRC staff was informed by Westinghouse Electric Corporation of an input methodology problem for the BART computer program. BART is one of the programs in the Westinghouse Emergency Core Cooling System (ECCS) evaluation model used to demonstrate compliance with 10 CFR 50.46 and Appendix K to 10 CFR 50.

Core inlet flooding rate (V_{in}) calculated as a function of time in the WREFLOOD computer code is used as input to the BART code. However, only a limited number of V_{in} points are made available from WREFLOOD. During the first few seconds of the core reflooding transient, the change in V_{in} as a function of time is oscillatory. Therefore using a limited number of points from WREFLOOD did not allow an accurate representation of V_{in} or the integral of V_{in} used in BART. In particular the integral of V_{in} and consequently the water level in the core was too high as used in BART.

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Westinghouse modified the data transfer procedure so that good agreement now exists between WREFLOOD and BART. The analysis procedure also instructs the analyst to assure that for all times during reflood the integrated value of V_{in} used in BART is equal to or less than that calculated by WREFLOOD. A reanalysis of the Turkey Point Units 3 and 4 was performed using the new methodology. The new results (References 2 and 3) showed a 79°F increase which results in a peak cladding temperature of 2051°F. This is well below the 2200°F limit specified in 10 CFR 50.46. Therefore, the number 1972 should be changed to 2051 on the 8th line of the first full paragraph on page 6 of reference 1. And, also on the 2nd line of the first paragraph and 6th line of the second paragraph of page 12. The number 160 should be changed to 81 on the 5th line of the second paragraph of page 12.

We have reviewed the information submitted by Westinghouse and find the new methodology satisfactory and meets the requirements of Appendix K to 10 CFR 50. This information is reflected in a SSE on the BART computer code methodology. (Reference 4)

We have concluded that the results of the reanalysis are acceptable, our initial conclusions have remained valid based on the results of the analysis using the FLECHT correlation and all other conclusions regarding BART and the results of the large break LOCA analysis for Turkey Point Plant Units 3 and 4 remain valid.

Dated: May 14, 1985

Principal Contributor:

G. N. Lauben

REFERENCES

1. Letter to Robert E. Uhing (FP&L) from Dan McDonald (NRC) on "Technical Specification Amendments to Support the Integrated Program for Vessel Flux Reduction and Operation with New Steam Generators" (Enclosure 3) dated December 23, 1983:
2. Letter from E. P. Rahe, (Westinghouse) to D. G. Eisenhut (NRC) on "BART-WREFLOOD Input Revision" dated March 22, 1985.
3. Letter from J. W. Williams (FPL) to S. A. Varga (NRC) "BART-WREFLOOD Input Revision" dated April 16, 1985.
4. Letter to EP Rahe (Westinghouse) from C. O. Thomas (NRC), Acceptance for Referencing of Special Report NS-NRC-85-3025 (NP), "BART-WREFLOOD Input Revision"



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 99 TO FACILITY OPERATING LICENSE NO. DPR-31
AND AMENDMENT NO. 93 TO FACILITY OPERATING LICENSE NO. DPR-41
FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT PLANT UNIT NOS. 3 AND 4
DOCKET NOS. 50-250 AND 50-251

1. Introduction

Florida Power and Light Company submitted a request for Amendments to the Technical Specifications contained in Appendix A of Facility Operating Licenses DPR-31 and 41 by letter dated August 19, 1983 (Ref. 1) and September 9, 1983 (Ref. 2). The request was supplemented, to provide additional information, on September 20, 1983 (Ref. 15), October 4, 1983 (Ref. 6) and December 17, 1983 (Ref. 16).

These amendments propose changes to the Technical Specifications to support the integrated program for pressure vessel flux reduction and to take credit for operation with the new steam generators in an unplugged (maximum of five (5) percent tube plugging) configuration. Changes are requested to: (1) increase the hot channel $F_{\Delta H}$ limit from 1.55 to 1.62; (2) increase the total peaking factor F_0 limit from 2.30 to 2.32; (3) change the Overpower ΔT setpoints and thermal-hydraulic limit curves; and (4) delete restrictions and limits previously placed on operation with the old steam generators having tubes plugged in excess of five percent.

In connection with the review of these proposed changes, we have received Comments and a Request for Hearing and Petition for Leave to Intervene in this matter from the Center for Nuclear Responsibility and Joette Lorion (Ref. 3). We have addressed the concerns contained in the comments and the petition in the text of this Safety Evaluation Report where relevant.

In addition, we have addressed concerns not relevant to the present amendments, but related to pressure vessel embrittlement in Appendix A to this evaluation. Other concerns of the commenters related to reload core designs are addressed in our Safety Evaluation, Section 6, dated December 9, 1983, supporting Amendment Numbers 98 and 92.

2. Nuclear Design Evaluation

The existing nuclear design bases for the Turkey Point reactors as stated in the FSAR and applied to subsequent Reload Safety Evaluations are not altered in any way by this amendment. These bases address design criteria for items such as allowable fuel burnup, shutdown margin requirements, negative reactivity coefficients, and xenon stability. The standard calculational methods described in the "Westinghouse Reload Safety Evaluation Methodology" (Ref.5) continue to apply. As is current practice, each reload core design will be evaluated to assure that design and safety limits are satisfied according to this reload methodology.

As discussed in Section 4.2, a Loss of Coolant Accident (LOCA) analysis has been performed using a total heat flux peaking factor, F_0 , of 2.32. Basically, this reflects elimination of the need to operate the Turkey Point reactors at a reduced F_0 because of a substantial percentage of plugged steam generator tubes which existed prior to the recent steam generator replacement. There is an approved methodology for justification that the F_0 assumed as an initial condition in the LOCA analysis will not be exceeded in normal operation of the power plant. This methodology is described in Ref. 5. As a result of our questions, the licensee provided (Ref. 6) the specific results of application of this technology to Turkey Point Unit 3, Cycle 9. These results employ radial peaking factors in conformance with the $F_{\Delta H}$ change proposed in this amendment. We have reviewed the results for Unit 3, Cycle 9 and find them acceptable: That is, based upon these results, and the axial power distribution monitoring Technical Specifications in place for the reactor, we are confident that the F_0 limit of 2.32 will not be exceeded during normal operation of the power plant. Continued application of this methodology for future cycles of both units will permit the same conclusion to be drawn.

Although the previous cycle of operation had an F_0 limit of 2.30, the proposed change in F_0 to 2.32 and the reduction in the number of plugged tubes in the steam generator will not result in an increase in the coolant temperature and therefore will not result in any increase in the potential to produce a pressurized thermal shock to the reactor vessel. This is because the reduction in the number of plugged tubes allows more coolant flow and the reactor coolant inlet, average, and outlet temperature do not change with a change in peaking factor, as they would with a change in power level for which no change has been requested.

At full power the average core linear heat generation rate is 5.58 kW/ft. The product of the average heat generation rate and the peaking factor (F_0) yield a peak linear heat generation rate of 12.9 kW/ft. This peak linear heat rate is being increased from 12.8 kW/ft in the previous cycle. The accident analyses, particularly the LOCA, show acceptable results with this slightly increased peak linear heat rate.

3. Thermal-Hydraulic Design Evaluation

Since the proposed Technical Specification amendment will increase the hot channel factor, $F_{\Delta H}$, from 1.55 to 1.62 and increase the total peaking factor, F_0 , from 2.30 to 2.32, and since the future cycles will be reloaded with the 15X15 optimized fuel assemblies (OFA), the impact of operating at these higher peaking factors on thermal margin is evaluated.

One of the fuel design acceptance criteria is the minimum departure from nucleate boiling ratio (DNBR) which ensures with a 95% probability at 95% confidence level that the hot rod in the core does not experience a departure from nucleate boiling during normal operation or anticipated operational occurrences. The DNBR is defined as the critical heat flux, which is the maximum heat flux occurring just before a change of boiling heat transfer mode resulting in a fuel cladding temperature excursion, divided by the local heat flux. Since the critical heat flux (CHF) is dependent upon the fuel and flow conditions, the increase in $F_{\Delta H}$ will result in lower CHF as well as DNBR.

The licensee has determined that the increase of the $F_{\Delta H}$ from 1.55 to 1.62, an increase of 4.5%, will result in a DNBR penalty of 9%, that is, the minimum DNB ratio will be reduced by 9%. This is derived from using a conservatively estimated sensitivity factor of -2.0 for the rate of change of DNBR with respect to the $F_{\Delta H}$. This estimated sensitivity factor is a conservative value since a study performed by the Battelle Pacific Northwest Laboratories (Ref. 7) has shown a sensitivity factor of about -1.0 which would result in the DNBR penalty of less than 9%. This reduction in DNBR is offset by a number of calculational improvements with respect to other aspects of the overall thermal modeling.

In the previous Technical Specification change (Amendments 98 and 92) the fuel rod bow effect on DNBR was calculated using an older approved interim method (Refs. 8, 9) which resulted in a maximum rod bow penalty of 14.9%. This interim method for rod bow penalty calculation was developed by Westinghouse and approved by the NRC staff as a conservative calculational method. The licensee has recalculated the rod bow penalty using a more recently approved method, Westinghouse topical report number WCAP-8691, Revision 1 (Ref. 10). This method applies statistical convolution of the CHF test data and interfuel rod gap closure data to derive the rod bow penalty on DNBR. Since rod bow and gap closure increase with fuel burnup, the rod bow penalty on DNBR increases with burnup. However, for the purpose of calculating rod bow penalty, the maximum burnup used for the calculation is 33000 MWD/MTU. The 33000 MWD/MTU in the rod bow penalty calculation is used because the physical burnup effects at higher burnup are greater than the rod bowing effects. By the time the fuel exceeds a burnup of 33000 MWD/MTU it is not capable of achieving limiting peaking factors due to the decrease in fissionable isotopes and the buildup of fission product inventory. The rod bow penalties at 33000 MWD/MTU are 4.7% and 5.5%, respectively, for the 15X15 LOPAR fuel using the Westinghouse designated W-3 L-Grid CHF correlation and the 15X15 OFA fuel using the Westinghouse correlation designated WRB-1. The difference in rod bow penalties using the old interim method and the new approved method are 10.2% and 9.4%, respectively, for the low parasitic (LOPAR) and OFA fuels. These differences represent gains in DNBR margins which can be used to compensate for the estimated DNBR penalty of 9% resulting from the increase of $F_{\Delta H}$ from 1.55 to 1.62.

The licensee has performed the thermal-hydraulic analysis with the proposed $F_{\Delta H}$ of 1.62 using the same methods described in the Final Safety Analysis Report (FSAR). The licensee used a more representative densification power spike factor which is used as input to the fuel densification calculation performed with the approved fuel densification model (Ref. 11).

Since the analysis is performed with the assumption of homogeneous full core of either LOPAR or OFA fuel, a transitional mixed core penalty of 3% DNBR was imposed on the 15x15 OFA fuel to account for the mismatch in the hydraulic resistances between the LOPAR and OFA fuel. In addition, since the WRB-1 CHF correlation is used for the DNBR calculation there may be a small (less than 2%) uncertainty due to the lack of CHF data on the 15x15 OFA fuel. This additional uncertainty was added even though the WRB-1 correlation has been approved for the 17x17 OFA fuel and that additional CHF data was submitted by Westinghouse for the 14x14 OFA fuel to support the application of the WRB-1 correlation to the 15x15 OFA fuel. The mixed core penalty and the WRB-1 application to the 15x15 OFA fuel has been identified in the NRC staff evaluation of the previous Technical Specification change (Amendments 98 and 92). These penalties and uncertainty along with the DNBR penalty due to rod bow are accounted for in the safety analysis.

For the 15x15 LOPAR fuel, the safety analysis uses a minimum DNBR limit of 1.30. The licensee has identified a total thermal margin of 11.1% in the use of such a value. This margin consists of three elements. A 4.8% margin from the use of 1.30 design DNBR limit instead of 1.24 which is the value of minimum DNBR derived from the 15x15 L-Grid CHF test data. A 3% margin from the use of a conservative thermal diffusion coefficient and a 3.3% margin from pitch reduction. These thermal margin components have been identified in Ref. 8 and have also been approved for other plants such as Zion Units 1 and 2. Therefore, a total of 11.1% DNBR margin is available to compensate for the remaining rod bow penalty of 4.7% for the LOPAR fuel. For the OFA fuel, the safety analysis minimum DNBR limit is 1.34 using the WRB-1 CHF correlations. This DNBR limit is 12.7% higher than the allowable DNBR limit of 1.17 derived from WRB-1. This margin is sufficient to compensate for the 5.5% remaining rod bow penalty as well as the transitional mixed core penalty of 3% DNBR imposed on the 15x15 OFA fuel and the small (<2%) uncertainty associated with the application of the WRB-1 correlation to the 15x15 OFA fuel.

Therefore, plant operation with the proposed $F_{\Delta H}$ limit of 1.62 will still result in adequate DNBR margin for all analyzed ΔH transients to assure that the minimum DNBR derived from the W-3 and WRB-1 correlations will be exceeded for all normal operational and anticipated operational occurrences.

4. Accident Evaluation

The licensee also provided an evaluation on the effects of the increased $F_{\Delta H}$ and F_Q limits on non-LOCA and LOCA accidents.

4.1 Non-LOCA Evaluation

The Reactor Core Thermal and Hydraulic Safety limits are recalculated using the new $F_{\Delta H}$ limit of 1.62. Based on these new protection limits, the licensee has performed calculations for the Overtemperature ΔT (OT ΔT) and Overpower ΔT (OP ΔT) setpoint equation constants using the standard Westinghouse method (Ref. 12). The results indicate that the Overtemperature ΔT setpoint equation in the current Technical Specification is conservative. Therefore no change in the OT ΔT equation and no reanalysis for the OT ΔT trip events are required. A change in the OP ΔT

setpoint is required. It was calculated with the methods described in Ref. 12. These methods have been used to calculate the safety limit curves and $OT\Delta T$ and $OP\Delta T$ setpoints for all Westinghouse initial and reload cores approved to date. We reviewed these methods as applied to the safety limit curves and $OP\Delta T$ setpoints changes submitted for this application and find the requested changes acceptable.

4.2 Large Break LOCA Evaluation

The large break LOCA analysis is performed with 102% of the rated thermal power of 2200 Mwt, a hot channel factor, $F_{\Delta H}$, of 1.62, a total peaking factor, F_0 , of 2.32 and an assumed steam generator tube plugging level of 5%. A sensitivity study is performed with break sizes ranging from 1 ft² area to a full double ended break of the cold leg, and various Moody discharge coefficients. The results show that the double ended cold leg guillotine break with a discharge coefficient of 0.4 is the worst large break LOCA case. It has the highest peak cladding temperature.

The analysis is performed with a modified version of the 1981 Westinghouse Emergency Core Cooling System (ECCS) evaluation model (Ref. 13). This modification to the 1981 evaluation model uses the revised PAD Fuel Thermal Safety Model for the calculation of the initial fuel conditions; the SATAN-VI code for the transient thermal hydraulic calculation during blowdown period; the WREFLOOD code for the calculation of the refill and reflood transient period; the LOCTA-IV code for the calculation of peak cladding temperature; and the COCO code for the calculation of the dry containment pressure history. The modified version of ECCS evaluation model uses the BART computer code (Ref. 14) to calculate the reflood heat transfer coefficient normally performed by the WREFLOOD code. The BART code provides a time and location dependent clad surface heat transfer coefficient for the reflood rates ranging from 0.6 to 1.5 inch/sec during the reflood stage of LOCA. The BART computer code and its application are described in the Westinghouse topical report WCAP-9561 (Ref. 14). The BART computer code without grid spacer model and its application in the Westinghouse evaluation model have been reviewed and approved by the staff in a Safety Evaluation Report regarding WCAP-9561. Since the spacer grid model to be used in BART is still under staff review, the licensee in its letter of September 20, 1983, (Ref. 15) submitted additional analysis of the large break LOCA using the ungridded BART model. The staff has reviewed this analysis. We find that approved methods and computer codes are used and the results show that the peak cladding temperature, metal-water reaction and cladding oxidation are within the acceptance criteria specified in 10 CFR 50.46 for LOCA analysis.

In addition, our review indicated that the reduction in peak cladding temperature (PCT) in the LOCA analysis resulting from the use of the BART code were not necessary to demonstrate that Turkey Point Units 3 and 4 meet the acceptance criteria specified in 10 CFR 50.46 and that use of the previously approved ECCS evaluation model using the FLECHT correlation in lieu of the BART code would still

result in a PCT less than 2200°F. We requested the licensee to provide verification of this indication. The licensee submitted the results of a large break LOCA analysis by letter dated December 17, 1983, (Ref. 16) which used the previously approved ECCS evaluation model using the FLECHT correlation. This analysis indicated a PCT of 2130° for the worst case break.

This analysis is applicable to both a full core 15X15 LOPAR and a full core 15X15 OFA fuel. For its application to the transition mixed core, the licensee has performed an evaluation to determine the effect of the flow distribution due to hydraulic resistance mismatch in the mixed core configuration. Since the 15X15 OFA increases the flow resistance by about 4.5%, the reflood flow rate for the 15X15 OFA fuel during the transitional mixed core period will be reduced by approximately 2.2%. This will result in approximately 10°F increase in the peak cladding temperature of 2051°F for the transition core which is still within the acceptance criteria imposed in 10 CFR 50.46. Since 5% tube plugging was assumed in the analysis, plant operation will be restricted to no more than 5% steam generator tube plugging.

4.3 Small Break LOCA Evaluation

The small break LOCA analysis is performed with the approved computer codes, i.e., (1) the revised PAD Fuel Thermal Safety Model for the calculation of the fuel initial conditions; (2) the WFLASH code for the calculation of the transient-depressurization of the reactor coolant system, fuel power, mixture height and steam flow past the uncovered part of the core; and (3) the LOCTA-IV code for the peak cladding temperature analysis. The evaluation is done at 102% of the rated thermal power with the hot channel factor of 1.62 and the total peaking factor of 2.32 at the core midplane. Various break sizes are performed and the results show that the worst break size to be a 3 inch diameter break which results in the highest peak cladding temperature of 1605°F, well below the acceptance criterion of 2200°F. This analysis is applicable to both 15X15 LOPAR and 15X15 OFA fuels. For a transition mixed LOPAR-OFA core, the flow redistribution due to mismatch in the fuel assembly hydraulic resistance may have an effect on the peak cladding temperature (PCT). However, since the PCT margin is so large, this effect will not cause the PCT to approach the acceptance criterion.

4.4 New or Different Accidents

Neither the licensee nor the NRC staff could identify any aspects of the requested changes in these amendments which would create the probability of a new or different kind of accident from any accident previously evaluated.

5. Technical Specifications

The specific Technical Specification changes and the reasons for their acceptability are:

Page vi

Figure 3.1-1 has been added to the List of Figures. This change is editorial and has no safety significance.

Figure 2.2-1

This figure has been modified to remove the "note", which is no longer applicable with the new steam generators.* The limits were recalculated to reflect the increase in the allowable $F_{\Delta H}$ limit, and is acceptable as discussed in Section 4.1.

Figure 2.1-1a & 2.1-1b

These figures are no longer required with the new steam generators.*

Page 2.3-2

The note is deleted as it is no longer applicable with the new steam generators.*

Page 2.3-3

The multiplier is modified in Overpower ΔT equation. The notes are deleted as they are no longer needed with the new steam generators.*

The modified multiplier in the Overpower ΔT is acceptable as discussed in Section 4.1.

Page 3.1-7

The notes are deleted as they are no longer applicable with the new steam generators.*

*The analyses discussed in this amendment, particularly the LOCA analysis were conducted with steam generator plugging levels up to 5%. This reflects the installation of new steam generators. The noted changes reflect the change to plugging levels up to 5%, and remove references to greater plugging levels previously allowed.

Page 3.2-3

The $F_{\Delta H}$ limit and part power multiplier are increased. Notes and references to plugging levels are deleted as they are no longer applicable with the new steam generators.*

F_Q is increased to 2.32 on the basis of LOCA analysis.

The $F_{\Delta H}$ limit change is acceptable for the reasons discussed in Section 4. The part power multiplier on the limit is changed from .2 to .3. This change allows an increase in $F_{\Delta H}$ linearly increasing from zero at full power to 30% at zero power. The .2 multiplier allowed an increase in $F_{\Delta H}$ of 20% at zero power. The purpose of this multiplier is to allow an increase in $F_{\Delta H}$ with decreasing power level to account for the effect of insertion of control rods and reduction in negative feedback with decreasing power level. It has been found generically that the .2 multiplier was too restrictive, and caused violations of the $F_{\Delta H}$ limits at very low power levels, when there is clearly no safety problem. Accordingly, licensees with Westinghouse designed reactors have been requesting, and we have been accepting, the change to the .3 multiplier. We have reviewed the licensee's evaluation of the effect of this change (Ref. 1, Section 3.1) and agree with his conclusion that its effect is negligible.

The F_Q increase is acceptable as discussed in Sections 2 and 4.2.

Figure 3.2-3 & 3.2-3a

These figures have been combined into Figure 3.2-3 and revised to present new limits from the LOCA analyses. We calculated this figure independently and agree that it is correct, and therefore this change is acceptable.

Page B2.1-2

The $F_{\Delta H}$ and part power multiplier have been increased.

Page B3.2-4

The increased $F_{\Delta H}$ limit is noted.

The last two changes are consequences of the changes on page 3.2-3 and are acceptable as discussed above.

*The analyses discussed in this amendment, particularly the LOCA analysis were conducted with steam generator plugging levels up to 5%. This reflects the installation of new steam generators. The noted changes reflect the change to plugging levels up to 5%, and remove references to greater plugging levels previously allowed.

6.0 Significant Hazards Consideration Comments

These proposed amendments were initially noticed (FR 48 45862 dated October 7, 1983) and significant hazards comments and a Request for Hearing and Petition for Leave to Intervene were received from the Center for Nuclear Responsibility and Ms. Joette Lorion (Petitioners). We have addressed the relevant comments and concerns related to these amendments in the text of this Safety Evaluation. In addition, we have addressed concerns not relevant to these specific amendments, however related to the reload core design, in Appendix A to this evaluation and in our Safety Evaluation, Section 6, dated December 9, 1983 supporting Amendment Numbers 98 and 92.

- A. The petitioners expressed concern relating to operating the units at higher fuel temperatures.

This concern is addressed in Sections 2, 3 and 4 of this Safety Evaluation. As noted in Section 2, the total heat flux peaking factor increase reflects the elimination of the need to operate the Turkey Point reactors at a reduced peaking factor due to the old steam generators (which had substantial number of plugged tubes) that have been replaced with new steam generators which allow for an increase in flow. The inlet, average and outlet temperatures do not change with the change in peaking factor. It is also noted that the units have previously operated with higher peaking factors than requested in this amendment.

The effects of increasing the hot channel factor is discussed in detail in Section 3. The increase in the operational limits will not result in the DNBR violating the specified acceptable fuel design limit as shown by analyses using calculational and analytical methods approved by the NRC staff.

As stated in Section 4, the thermal and hydraulic safety limits have been recalculated using the new protection limits requiring no change in the Overtemperature ΔT trip and a change in the Overpower ΔT trip which has been found acceptable. The use of approved methods and computer codes result in the peak cladding temperature, metal-water reaction and cladding oxidation being within the acceptance criteria imposed in 10 CFR 50.46.

The Turkey Point reactors have operated at a lower power density and licensed power level than other similar reactors using the same type of fuel. In fact, for Cycles 1 through 3, inclusive, the F_0 limit for Turkey Point Units 3 and 4 was 2.32. The peak allowable design operating limit on linear heat rate at full power in the initial cycles was 18 kW/ft. The authorized core power level is 2200 Mwt. This is the lowest for a Westinghouse 3 loop reactor. Others operate at power levels up to 2785 Mwt. The Turkey Point average full power linear heat generation rate of 5.58 kW/ft is the lowest of the Westinghouse designed reactors using 15X15 fuel. Of the others, six run at 5.7 kW/ft., five run at 6.2 kW/ft and three at 6.7 kW/ft. These comparisons illustrate that the Turkey Point reactors are operated at heat related conditions below other reactors in the design group.

- B. The petitioners have expressed concern that the staff has not published a proposed safety evaluation report that the Commission could review to determine whether the new Westinghouse fuel design or the accompanying Emergency Core Cooling System (ECCS) computer model comply with the Commission standards and criteria including especially the standards for ECCS.

The Westinghouse fuel design has been addressed in a published Safety Evaluation, Section 6, dated December 9, 1983 supporting Amendment Numbers 98 and 92. The computer model is discussed in detail in Section 4 of this evaluation and has been addressed in our Safety Evaluation of the Bart Code dated December 21, 1983. In addition, analyses have been performed (Ref. 16) using the FLECHT correlation in lieu of the BART code confirming that the acceptance criteria specified in 10 CFR 50.46 are met.

- C. Petitioners contend that the entirely new computer model used by the utility, for calculating reflood portions of an accident does not meet the Commission's ECCS Acceptance Criteria; specifically, whether a 2.2% reduction in reflood rate is misleading because for a small decrease in reflood rate, there results a large increase in fuel temperature. Reflood rates are critical if below 1 or 2 inches per minute.

As stated above, details are provided in Section 4 and our Safety Evaluation of the Bart Code dated December 21, 1983. It is noted in Section 4 that the analysis for the large break LOCA was performed using the ungridded BART model. (The results of analysis are within the acceptance criteria imposed in 10 CFR 50.46. This model and the confirmatory analysis (Ref. 16) included data bases for reflood rates less than 1.5 inches per second.)

- D. The petitioners contend that the amendments requested involve a significant hazards consideration because they might result in an increase in the authorized maximum core power level.

As stated in Section 2, there is no change to the authorized power level of the facility. The reactor coolant inlet, average and outlet temperatures do not change.

- E. Petitioners contend that the proposed departure from the nucleate boiling ratio (DNBR) would significantly and adversely affect the margin of safety for operation of the reactors. As the amount of heat decreases, the difference in temperature increases, driving heat flux higher. Then nucleate boiling may occur at the top of the active fuel rods at a time

there exists the need to drive the same amount of heat throughout the system. The heat flux the Commission and utility company would take to DNB is 1.3 or 30% below the heat flux that would cause an increase in fuel temperature.

The details relating to the DNBR for the LOPAR and OFA fuels is discussed in Section 2 and 3. The effects and impact of operating at the higher peaking factors on thermal margin is evaluated for both (gains and penalties) in DNBR margin. Operation with the higher F_{AH} limit will still result in adequate DNBR margin for all analyzed transients to assure that the minimum DNBR will be exceeded for all normal operational and anticipated operational occurrences.

- F. Petitioners contend that the increased fuel core temperatures generally would exceed safety margins and specifically would result in unacceptable swelling or bowing of fuel rods. During an accident, fuel rod swelling due to higher temperatures displaces cooling water and impedes insertion of control rods by that physical phenomenon of increased size. This could result in a significant increase in the possibility and/or consequences of an accident.

This concern is addressed in detail in Sections 3 and 4. Rod bow effects have previously been calculated using an NRC approved interim method. The use of NRC approved methods in WCAP-8691 result in a gain in margin for both LOPAR and OFA fuel. The fuel rod swelling was calculated using NRC approved methods in WCAP-9220, Rev. 1. As stated in response to the DNBR concern, both the gains and penalties associated with the change in operational limits are considered. We have concluded in Section 4 that the results of the analysis are within the acceptance criteria imposed in 10 CFR 50.46.

7. Final No Significant Hazards Consideration Determination

The Commission has provided guidance concerning the application of the standards for determining whether license amendments involve no significant hazards considerations by providing certain examples (48 FR 14870). The increase in the hot channel F_{AH} limit and the total peaking factor F_Q limit is similar to example (vi) of changes which are not likely to involve significant hazards considerations: A change which either may result in some increase in the probability or consequences of a previously analyzed accident or reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan: for example, a change resulting from the application of a small refinement of a previously used calculational model or design method.

The amendment follows this example. First, the calculated peak cladding temperatures (PCT) of 1605°F and 2051°F for small and large break loss of coolant accidents respectively. These are within the maximum limit of 2200°F specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems (ECCS) for Light Water Nuclear Power Reactors". Second, additional departure from nucleate boiling ratio margin is identified for Overtemperature ΔT and loss of flow conditions to accommodate the slight reduction in margin resulting from increasing the $F_{\Delta H}$ limit and the Overpower ΔT change is in the conservative direction. This is well within the Final Safety Analysis Report (FSAR) design basis. Third, the overpower ΔT setpoints are more restrictive to provide protection using the recalculated core limits and error allowances provided in the safety evaluation which indicate the safety margin is clearly within all acceptance criteria specified in the Standard Review Plan.

The Thermal Hydraulic Design Evaluation (Section 3) and Accident Evaluation (Section 4) of this Safety Evaluation include a number of improvements and refinements. The principal improvements were the use of the BART code and the methodology for calculating the rod bow penalty. The use of the BART code resulted in a relatively small reduction in PCT of 81°F, resulting in a maximum PCT of 2051°F. Confirmatory analysis using the FLECHT correlation in lieu of the BART code resulted in a maximum PCT of 2130°F. The results of both methods are below the previously maximum PCT of 2195°F. Thus, the use of BART simply improves the calculated margins. The use of the methodology for calculating rod bow is a significant improvement in the technique used for determining the effect of rod bow on DNBR while maintaining conservatism for performing licensing analysis.

The changes in $F_{\Delta H}$ and F_0 limits and Overpower ΔT setpoints do not affect the operating characteristics of any safety equipment nor otherwise affect the likelihood that such equipment may fail to function properly; accordingly, these changes do not affect the probability of any accidents as discussed above and in Sections 2, 3, and 4 of this Safety Evaluation. Similarly, these changes do not affect core inventory, or power level, or maximum temperature or pressure, or in any other way offset the consequences of accidents.

As indicated in Section 4.4 of this Safety Evaluation, these changes do not create the probability of a new or different accident from any accidents previously evaluated.

The changes in calculated DNBR all result from improvements in modeling techniques and in an improved data base. These changes result in models which more accurately reflect the thermal and hydraulic phenomena involved. Thus, margin changes are offset by improved modeling accuracy. Accordingly, the overall margin of safety taking into account modeling and data uncertainty has not been reduced as discussed above and in Sections 2, 3, and 4 of this Safety Evaluation. With respect to the change in calculated PCT, use of the previously approved ECCS model using the FLECHT correlation in lieu of BART code results in temperatures

satisfying the requirements specified in 10 CFR 50.46. The change in PCT resulting from the use of BART merely serves to improve the calculated margin.

The deletion of the technical specifications relating to the old steam generators is similar to example (v) of changes not likely to involve significant hazards considerations. This example deals with the situation when a license condition is imposed initially because some aspect of construction remains to be satisfactorily completed. Upon satisfactory completion of this item, the removal of the license condition is described by the Commission to be an example of an amendment involving no significant hazards consideration.

In this case, the deletions remove restrictions placed on the use of the old steam generators on the Turkey Point Plant, Units 3 and 4, for which license conditions require a new ECCS analysis be performed if credit is to be taken for the unplugged configuration (maximum of five (5) percent tube plugging). These analyses have been completed and found acceptable by the NRC staff as part of this current amendment. These results demonstrate that the restrictions placed on the old steam generators are no longer applicable and the new steam generators will function satisfactorily.

Therefore, the deletions which remove the restrictions placed on the use of the old steam generators which had tubes plugged in excess of five percent do not: 1) involve a significant increase in the probability or consequences of an accident previously evaluated based on the results of the ECCS analysis and the tube plugging limit of 5 percent as discussed above and in Section 4 of this evaluation, 2) create the probability of a new or different kind of accident from any accident previously evaluated as indicated in Section 4.4 of this evaluation, or 3) involve a significant reduction in a margin of safety in that the ECCS can perform its function and is within the acceptance criteria specified in the Standard Review Plan and 10 CFR 50.46, and the new steam generators are limited to five percent tube plugging, as discussed above and in Section 4 of this report.

Based on our review of the licensee's submittal, as described in our above evaluation, we have made a final determination that the amendment requests do not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the probability of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety; and therefore, do not involve a significant hazards consideration.

8.0 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.

9.0 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 the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 14, 1985

Principal Contributors:

M. Dunenfeld

Y. Hsii

S. Sun

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