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Southern Nuclear Operating Company
the southern electric system

June 17, 1994

Docket No. 50-348

U.S. Nuclear Regulatory Commission
ATTN.: Document Control Desk
Washington, D.C. 20555

Joseph M. Farley Nuclear Plant - Unit 1
Technical Specification Amendment for
Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

Gentlemen:

In order to provide greater flexibility in the Farley Unit 1 core design, Southern Nuclear is requesting an increase in the nuclear enthalpy rise hot channel factor ($F^N_{\Delta H}$) limit contained in the technical specifications. All analyses required to raise the limit from 1.65 to 1.70 for Vantage 5 fuel have been completed. Furthermore, Southern Nuclear proposes to revise the associated action statement to more closely reflect the guidance contained in the improved standard technical specifications, NUREG-1431.

Enclosure 1 provides a description of the proposed changes and the basis for the change. Enclosure 2 provides the basis for a determination that the proposed changes do not involve a significant hazard. Enclosure 3 contains the proposed changed technical specification pages in support of the amendment.

In accordance with 10CFR50.46, the large break LOCA peak clad temperature reported in Enclosure 2 is the result of a reanalysis and is considered a new, baseline peak clad temperature value.

Southern Nuclear has determined that the proposed license amendment will not significantly effect the quality of the environment.

In accordance with 10CFR50.91, the designated state official will be sent a copy of this letter and all enclosures.

A001

If there are any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

J. D. Woodard

REM/clt:fdhts.doc

Attachment

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 17th DAY OF June 1994

Notary Public

My Commission Expires: 9-14-94

cc: Mr. S. D. Ebneter
Mr. B. L. Siegel
Mr. T. M. Ross
Dr. D. E. Williamson

Enclosure 1

Safety Evaluation for Technical Specification Amendment for

Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

Safety Evaluation for Technical Specification Amendment for Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

Proposed Change

The proposed change to the J.M. Farley Unit 1 Technical Specification 3/4.2.3 involves a change in the Vantage 5 $F^N_{\Delta H}$ from "≤ 1.65 [1 + .3 (1 - P)]" to "≤ 1.70 [1 + .3 (1 - P)]" and bases to Technical Specification 2.1.1 ($F^N_{\Delta H}$ changed from 1.65 to 1.70). Furthermore, Southern Nuclear proposes to revise the associated action statement to more closely reflect the guidance contained in the improved standard technical specifications, NUREG-1431.

Background

Currently Farley Nuclear Plant Unit 1 contains mixed cores of LOPAR (low parasitic) and Vantage 5 fuel types. Each fuel type has a specific technical specification for $F^N_{\Delta H}$. The LOPAR technical specification for $F^N_{\Delta H}$ is not affected by this proposed change. Since increased margin is available in the Vantage 5 fuel assembly, it is beneficial to use this available margin, since no reduction in safety exists because of the proposed change. In addition, operational and core design flexibility will be available because of the proposed change.

Analysis

$F^N_{\Delta H}$ is an important accident analysis parameter. All Vantage 5 related accident analyses were evaluated for the proposed change and only the large break LOCA and fuel handling accidents required reanalysis. The current core safety limits have been reviewed and bound the effects of $F^N_{\Delta H}$ = 1.70 for Vantage 5 fuel.

Large Break LOCA

The most limiting large break LOCA (LBLOCA) (Cd=.4) was reanalyzed using the increased $F^N_{\Delta H}$ of 1.70 at full power. The Westinghouse 1981 Evaluation Model with BART (reference 1) and BASH (reference 2) was used for this analysis. These are the current models of record for Farley Vantage 5 analyses. Only the most limiting LBLOCA needed to be reanalyzed since the proposed change will have approximately the same effect on all break sizes, and the limiting break size is not expected to change. It should be noted that there is approximately 400°F difference in PCT between the most limiting break size (Cd=.4) and the next most limiting.

The revised LBLOCA PCT is 1957°F which is less that the 10CFR50.46 limit of 2200°F. In addition, zirc-H_20 reaction remains below the limit prescribed by 10CFR50.46. All other aspects of 10CFR50.46 continue to be met.

Radiological Consequences

The fuel handling accident is the only radiological event requiring reanalysis due to the increase in $F^N_{\Delta H}$. All other previously calculated doses remain bounding and all acceptance criteria continue to be met.

The fuel handling accident (in containment and in the fuel handling building) has been reanalyzed using an $F^N_{\Delta H}$ of 1.70 (previously 1.65). All acceptance criteria (including

DNB design criteria) continue to be met. All 10CFR100 limits are met, and the conclusions of the Final Safety Analysis Report remain valid.

Other Analyses

All current analyses of record for non-LOCA (including mass and energy release for inside and outside containment analyses) and small break LOCA include the effects of a 1.70 $F^N_{\Delta H}$ and therefore the proposed change has no effect on any of these postulated events. The increase in $F^N_{\Delta H}$ has no effect on the steam generator tube rupture event. All acceptance criteria (including the DNB design criterion) continue to be met. The conclusions of the Final Safety Analysis Report remain valid.

As a result of the increase in the $F^N_{\Delta H}$ limit, there is no significant increase in the dose rate above the refueling canal and spent fuel pool. It has also been determined that there will be no significant increase in the dose rate at ground level outside the containment due to passage of fuel through the fuel transfer canal. No change to existing radiation controls is required.

Thermal calculations were performed for the discharge to the spent fuel pool of fuel assemblies with $F^N_{\Delta H}$ of up to 1.70. The clad temperature will increase by at most 2°F due to the increased heat flux. This small increase will not challenge fuel integrity.

Action Requirement Revisions

If $F^N_{\Delta H}$ exceeds its limit, under this amendment the unit will be allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value, e.g., static control rod misalignment, are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs while $F^N_{\Delta H}$ is above its limit. The increased allowed action time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.

References

1. Chiou, J.S., et al., "Models for PWR Reflood Calculations Using the BART Code," WCAP-10062 (Proprietary), March 1982.

2. Besspiata, J.J., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code, Power Shape Sensitivity Studies," WCAP-10266-P-A Revision 2 Addendum 1 (Proprietary), December 15, 1987.

Enclosure 2

Significant Hazards Evaluation for

Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

Significant Hazards Evaluation for Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)

As required by 10CFR50.91(a)(1), an analysis is provided to demonstrate that the proposed license amendment to increase $F^N_{\Delta H}$ for Vantage 5 fuel for 1.65 to 1.70 at full power involves no significant hazards consideration.

Proposed Change

The proposed change to the J.M. Farley Unit 1 Technical Specification 3/4.2.3 involves a change in the Vantage 5 $F^N_{\Delta H}$ from "$\leq$ 1.65 [1 + .3 (1 - P)]" to "$\leq$ 1.70 [1 + .3 (1 - P)]" and bases to Technical Specification 2.1.1 ($F^N_{\Delta H}$ changed from 1.65 to 1.70). Furthermore, Southern Nuclear proposes to revise the associated action statement to more closely reflect the guidance contained in the improved standard technical specifications, NUREG-1431.

Background

Currently Farley Nuclear Plant Unit 1 contains mixed cores of LOPAR (low parasitic) and Vantage 5 fuel types. Each fuel type has a specific technical specification for $F^N_{\Delta H}$. The LOPAR technical specification for $F^N_{\Delta H}$ is not affected by this proposed change. Since increased margin is available in the Vantage 5 fuel assembly, it is beneficial to use this available margin, since no reduction in safety exists because of the proposed change. In addition, operational and core design flexibility will be available because of the proposed change.

Analysis

$F^N_{\Delta H}$ is an important accident analysis parameter. All Vantage 5 related accident analyses were evaluated for the proposed change and only the large break LOCA and fuel handling accidents required reanalysis. The current core safety limits have been reviewed and bound the effects of $F^N_{\Delta H}$ = 1.70 for Vantage 5 fuel.

Large Break LOCA

The most limiting large break LOCA (LBLOCA) (Cd=.4) was reanalyzed using the increased $F^N_{\Delta H}$ of 1.70 at full power. The Westinghouse 1981 Evaluation Model with BART (reference 1) and BASH (reference 2) was used for this analysis. These are the current models of record for Farley Vantage 5 analyses. Only the most limiting LBLOCA needed to be reanalyzed since the proposed change will have approximately the same effect on all break sizes, and the limiting break size is not expected to change. It should be noted that there is approximately 400°F difference in PCT between the most limiting break size (Cd=.4) and the next most limiting.

The revised LBLOCA PCT is 1957°F which is less that the 10CFR50.46 limit of 2200°F. In addition, zirc-H_2O reaction remains below the limit prescribed by 10CFR50.46. All other aspects of 10CFR50.46 continue to be met.

Radiological Consequences

The fuel handling accident is the only radiological event requiring reanalysis due to the increase in $F^N_{\Delta H}$. All other previously calculated doses remain bounding and all acceptance criteria continue to be met.

The fuel handling accident (in containment and in the fuel handling building) has been reanalyzed using an $F^N_{\Delta H}$ of 1.70 (previously 1.65). All acceptance criteria (including DNB design criteria) continue to be met. All 10CFR100 limits are met, and the conclusions of the Final Safety Analysis Report remain valid.

Other Analyses

All current analyses of record for non-LOCA (including mass and energy release for inside and outside containment analyses) and small break LOCA include the effects of a 1.70 $F^N_{\Delta H}$ and therefore the proposed change has no effect on any of these postulated events. The increase in $F^N_{\Delta H}$ has no effect on the steam generator tube rupture event. All acceptance criteria (including the DNB design criterion) continue to be met. The conclusions of the Final Safety Analysis Report remain valid.

As a result of the increase in the $F^N_{\Delta H}$ limit, there is no significant increase in the dose rate above the refueling canal and spent fuel pool. It has also been determined that there will be no significant increase in the dose rate at ground level outside the containment due to passage of fuel through the fuel transfer canal. No change to existing radiation controls is required.

Thermal calculations were performed for the discharge to the spent fuel pool of fuel assemblies with $F^N_{\Delta H}$ of up to 1.70. The clad temperature will increase by at most 2°F due to the increased heat flux. This small increase will not challenge fuel integrity.

Action Requirement Revisions

If $F^N_{\Delta H}$ exceeds its limit, under this amendment the unit will be allowed 4 hours to restore $F^N_{\Delta H}$ to within its limits. This restoration may, for example, involve realigning any misaligned rods or reducing power enough to bring $F^N_{\Delta H}$ within its power dependent limit. When the $F^N_{\Delta H}$ limit is exceeded, the DNBR limit is not likely violated in steady state operation, because events that could significantly perturb the $F^N_{\Delta H}$ value, e.g., static control rod misalignment, are considered in the safety analyses. However, the DNBR limit may be violated if a DNB limiting event occurs while $F^N_{\Delta H}$ is above its limit. The increased allowed action time of 4 hours provides an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time.

Evaluation

On the basis of the information presented above, the following conclusions can be reached with respect to 10CFR50.92:

1. Will the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

The proposed increase in $F^N_{\Delta H}$ for Vantage 5 fuel does not involve a significant increase in the probability or consequences of an accident previously evaluated in the Farley Final Safety Analysis Report. No changes in the mechanical design of the fuel are necessary for this $F^N_{\Delta H}$ increase. No new performance requirements are being imposed on the fuel or any system or component because of this change. Vantage 5 fuel contains several features that provide increased margin to core limits. The proposed increase in $F^N_{\Delta H}$ is utilization of this margin with no violation of any acceptance criteria. Subsequently, overall plant integrity is not reduced. $F^N_{\Delta H}$ is not an accident initiator. Therefore, the probability of an accident has not significantly increased.

The radiological consequences of all accidents, including the fuel handling accident, remain within the previous appropriate acceptance limits as well as those included in 10CFR100. Therefore, the radiological consequences to the public resulting from any accident previously evaluated in the Final Safety Analysis Report has not significantly increased.

2. Will the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed specification change to the $F^N_{\Delta H}$ limit does not create the possibility of a new or different kind of accident from any previously evaluated in the Final Safety Analysis Report. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the increase in $F^N_{\Delta H}$. Any accident using the revised analytical assumption has been evaluated or reanalyzed and it has been determined that there is no adverse effect on, or do not challenge the performance of, any safety-related system. Therefore, the possibility of a new or different kind of accident is not created.

3. Will the proposed amendment involve a significant reduction in a margin of safety?

The Vantage 5 technical specification change for increasing $F^N_{\Delta H}$ does not involve a significant reduction in the margin of safety. The margin of safety for the Vantage 5 fuel parameters are defined by the accident analyses that are performed to conservatively bound the operating conditions defined by the technical specifications and to demonstrate that the regulatory acceptance limits are met. Performance of analyses (including the LBLOCA) and evaluations for the proposed inclusion of an increased $F^N_{\Delta H}$ for Vantage 5 fuel type confirmed that the operating envelope defined by the technical specifications continues to be bounded by the revised analytical basis, which in no case exceeds the acceptance limits. Therefore, the margin of safety provided by the analyses in

accordance with these acceptance limits is maintained and is not significantly reduced.

On the basis of the preceding information, it has been determined that the proposed change to the technical specifications of increased $F^N_{\Delta H}$ for Vantage 5 fuel does not involve a significant hazards consideration as defined in 10CFR50.92(c).

References

1. Chiou, J.S., et al., "Models for PWR Reflood Calculations Using the BART Code," WCAP-10062 (Proprietary), March 1982.

2. Besspiata, J.J., et al., "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code, Power Shape Sensitivity Studies," WCAP-10266-P-A Revision 2 Addendum 1 (Proprietary), December 15, 1987.