THERMAL-HYDRAULIC SAFETY ANALYSIS EVALUATION FOR ANGRA1 AFTER REPLACEMENT STEAM GENERATOR, 16NGF INSERTION AND POWER UPRATE

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ABSTRACT

After Replacement of Steam Generator (RSG) in ANGRA1 nuclear power plant, it is intended to increase the power and to insert the 16NGF FA (Next Generation Fuel Assembly) in the subsequent cycles. This fact required that the whole safety analysis be re-done for ANGRA1 for this purpose. Because that a series of analyses were re-done in several areas.

In the TH area, the analyses have been done for several postulated accidents starting from the following assumptions:

- New operational parameters according to plant reevaluation;
- 16NGF features were considered in the analyses (geometries, components);
- Revised Thermal Design Procedure (RTDP) has been considered for the DNBR calculations (statistical convolution of the uncertainties) and
- WRB-2 correlation has been used for 16NGF Fuel Assembly, after the applicability has been proved in the CHF tests performed at the Columbia University in 2003.

The performed analyses have demonstrated full TH compatibility of the 16NGF working at the considered conditions. Following results arose from the analyses:

- The minimum DNBR criteria have not been violated for any one of the postulated Condition II accidents.
- Sufficient DNBR margin has been preserved to account for possible penalties for transition core and rod bow.

1. INTRODUCTION

In order to update the design of the ANGRA-1 fuel assembly of the 70’s, the 16NGF fuel assembly was developed by a joint venture consisting of the companies INB, KNFC (South Korea) and Westinghouse (USA). The project activities have been performed in the years 2002-2003. With the decision of inserting the 16NGF fuel assembly just after replacing the ANGRA-1 steam generators and up-rate the power in 6.3%, it was necessary to repeat the
performed analyses for the 16NGF fuel assembly during the design phase under the new operational conditions given by the Westinghouse’s Performance Capability Working Group (mentioned thereafter as PCWG) as well as to perform specific safety evaluations.

This paper intends to give an overview of the main analyses done for the thermal-hydraulics area in the scope of the ANGRA-1 Safety Analysis project, to give an overview of the main achieved results and also to present the codes, methodologies and correlations that have been utilized.

2. CODES

VIPRE-W was the main code used for the analysis, especially to determine the Departure from Nuclear Boiling Ratio (DNBR). It is based on several versions of COBRA codes for thermal-hydraulics applications, developed by the Battele Pacific Northwest Laboratories under sponsorship of EPRI, including the determination of the minimum DNBR, the critical heat flux (CHF), the fuel and clad temperatures and also coolant state in normal operation and assumed accident conditions determinations.

For VIPRE-W, like the majority of core thermal-hydraulic codes, the modeling structure is based on sub-channel analysis. A large part of development work on VIPRE-W consisted of tailoring the code, the utilities’ analytical requirements, up-rating the code’s capabilities and improving the flexibility of its use.

VIPRE-W was submitted to USA National Regulatory Commission (NRC) for use licensing under WCAP-14565.

Two basic code input were built, one for 16NGF and another one for 16STD, considering the geometries and operational parameters for each one. These decks were the basis for the performed analyses.

THINC-IV is a three-dimensional thermal-hydraulic computer program which calculates the distribution of coolant density, mass velocity, enthalpy, temperature, vapor void, thermodynamic quality, static pressure, and the resulting DNBR distribution in a pressurized water reactor core. The THINC-IV computational method is unique since a perturbation method is used to simplify the governing equations which describe the thermal-hydraulic behavior of the core.

BANDIT code has the basic assumption that a single channel analysis can be made to match (with reasonable accuracy) the DNBR results of the VIPRE/THINC codes by matching local coolant conditions through the use of bias curves. Bias curves account for the effects of flow redistribution and thermal mixing on hot channel mass velocity and enthalpy rise, which cannot be directly accounted for in a single channel analysis.

STEBL code calculates the exit boiling limits for both Standard and Improved Methods. The vessel flow rate and power is used as input to the calculations. The core inlet temperature is determined such that the average enthalpy at the vessel exit will be equal to saturation ($T_{HOT}$...
The pressurizer pressure should be used to calculate the exit boiling limits even if an adjusted core pressure is used for DNBR and quality limits.

3. METHODOLOGIES

Revised Thermal Design Methodology (RTDP) was employed for the majority of the analyses. In this methodology the parameters uncertainties are statistically combined with the DNB correlation uncertainties to obtain the overall DNBR uncertainty factor used to define the Design Limit DNBR (DL DNBR). The uncertainty factor obtained is used to define the design limit DNBR which satisfies the DNB design criterion. The DNB design criterion is that the probability that DNB will not occur on the most limiting fuel rod is at least 95 percent at a 95 percent confidence level during normal operation and operational transients (Condition I events) and during transient conditions arising from faults of moderate frequency (Condition II events). It is a methodology widely employed in USA and is to be licensed in Brazil.

The statistical procedure defines a so-called DL DNBR (RTDP starting from the correlations limits WRB-1 / WRB-2) and applying the uncertainties of the operational parameters, engineering factors, hot channel factors and codes. The use of DL DNBR allows that the operational parameters be input in their nominal values and not in the most unfavorable value (including uncertainties) as in the Standard Thermal Design Procedure (STDP).

For the ANGRA-1 it was retained a margin of 15% over the DL DNBR, that is the Safety Analysis Limit DNBR (SAL DNBR). This margin accounts for possible DNBR penalties as for transition core and rod bow. The analyses of the postulated accidents require that all Condition II accidents do not violate the minimum DNBR criteria for the postulated accidents should not be lower than the SAL DNBR.

The Standard Thermal Design Procedure (STDP) was used for some specific analyses in conjunction with W-3 correlation in the case of the operational parameters are outside the range of WRB-1 / WRB-2 correlations. For these specific cases the W-3 correlation limit should not be violated.

4. CORRELATIONS

The WRB-2 correlation was employed to analyze the nuclear power core with the 16NGF FA and the WRB-1 correlation was employed to analyze the nuclear power core with the 16STD FA, respectively.

WRB-1 DNB Correlation was developed based exclusively on the extensive body of rod bundle data that Westinghouse had collected prior to 1976. WRB-1 correlation is already being employed for the 16STD Fuel Assembly.
WRB-2 DNB Correlation was developed to account the benefits of the reduced grid spacing and to take advantage of the actual benefits of the Intermediate Flow Mixer (IFM) Grids and Low Pressure Drop (LPD) Grids.

The Critical Heat Flux (CHF) tests were performed at the Columbia University in New York USA, during 16NGF design phase, to demonstrate the applicability of the WRB-2 correlation to the 16NGF Fuel Assembly. Licensing of WRB-2 correlation is being carried out through the licensing authorities in Brazil.

Some specific analyses have required the application of W-3 correlation in conjunction with STDP methodology, because for these analyses the operational parameters were outside the WRB-2 / WRB-1 correlations ranges.

5. PERFORMED ANALYSES

The nominal plant conditions (e.g., core thermal power, design flow rate, coolant inlet temperature, system pressure and bypass flow rate) are obtained from design documents such as PCWG Design Power Capability Parameters and Design initialization letters. For accident state-point analyses, the system conditions are provided by the transient analysis (TA). The 16NGF and 16STD analyses have been redone with the new operational parameters (PCWG) updated for Replacement Steam Generator (RSG) applying the limits of the RTDP methodology. Besides that, specific analyses were performed for postulated accidents after RSG and new PCWG.

The VIPRE code was used in the following 16NGF/16STD thermal-hydraulic calculations with updated operational parameters:

- Transition Core
- Flow Stability
- RTDP Sensitivity
- Void Content
- Crossflow
- Thimble Bypass Flow
- Lift Forces

The results are very close to the previous values predicted by 16NGF design phase in the majority of the analyses, since the changes in operational parameters are not significant.

The following specific analyses were done for the safety analysis area (TA):

- Core Thermal Limits
- Axial Offset Limits
- Loss of Flow
- Locked Rotor
- Steam Line Break
• Rod Misalignment
• Dropped Rod Limit Lines
• Rod Withdrawal at Power
• Rod Withdrawal from Sub critical

In accord with the analyses performed, with WRB-2 / WRB-1 correlations and RTDP methodology, in all cases the minimum DNBR calculated was over the SAL DNBR (with exception of Locked Rotor accident), that is considered a Condition IV event. For the analyses performed with W-3 correlation and STDP methodology, the minimum DNBR was over the correlation limit.

3. CONCLUSIONS

The performed analyses in the scope of the ANGRA-1 Safety Analysis project have shown full thermal-hydraulic compatibility for the use of 16NGF and 16STD Fuel Assemblies in ANGRA-1 after RSG and power up-rate.

No violation of minimum DNBR criteria was detected for the selected SAL DNBR’s for any one of the postulated accidents and enough margin in DNBR has been preserved to take in account possible penalties as for transition core and rod bow.

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REFERENCES


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