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NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO UPDATED METHODOLOGY FOR STATISTICAL

COMBINATION OF UNCERTAINTIES

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE UNIT 2

DOCKET NO. 50-389

1.0 INTRODUCTION

By letter dated August 27, 1993, Florida Power and Light Company, licensee for St. Lucie Unit 2, requested approval of an improved methodology for the statistical combination of uncertainties associated with Departure from Nucleate Boiling (DNB) related analog protection and monitoring system setpoints. The methodology, "Extended Statistical Combination of Uncertainties," (ESCU) is described in Combustion Engineering report CEN-371(F)-P dated July 1989. This document updates the SCU methodology currently in use for combining uncertainties involved in calculation of the Thermal Margin/Low Pressure (TM/LP) Limiting Safety System Settings (LSSS) and DNB Limiting Conditions for Operation (LCO) and applies the updated methodology using plant-specific data for Unit 2. In addition, the licensee's August 27, 1993 letter requested approval of Technical Specification Bases revisions for Unit 2 that reflect use of the ESCU methodology in the calculation of a revised minimum DNBR. (The revised and current values are 1.20 and 1.28, respectively.)

The ESCU methodology and its application to analog protection and monitoring system setpoint calculations was previously described in CEN-348(B)-P for use with CE 14x14 fuel assemblies at Calvert Cliffs. This document was reviewed by NRC and approved by letter dated October 21, 1987. The report reviewed in the present Safety Evaluation, CEN-371(F)-P, differs from CEN-348(B)-P only in the plant-specific design differences existing between St. Lucie Unit 2 and Calvert Cliffs, such as the use of CE-16x16 fuel assemblies instead of the CE 14x14 assemblies.

The benefit to the licensee in adopting the updated methodology is the increased plant operational flexibility resulting from a reduction in the current overly conservative uncertainty penalties applied in the DNB LSSS and LCO setpoint calculations.

2.0 EVALUATION

In the current SCU methodology, uncertainties are divided into two groups. Group one consists of uncertainties in system physical parameters and uncertainties associated with the Critical Heat Flux (CHF) correlation. This group is statistically combined to generate a DNBR probability density function (pdf) using a thermal-hydraulic response surface which relates MDNBR to system and CHF parameters. The 95/95 probability/confidence limit value of this pdf is then used as the DNB Specified Acceptable Fuel Design Limit (SAFDL), i.e. MDNBR, in the setpoint analyses.

Group two consists of measurement uncertainties in state parameters which are monitored during operation as well as axial shape index (ASI) uncertainties and processing uncertainties. This group is statistically combined through stochastic simulation models of TM/LP LSSS and DNB LCO processes using the CETOP code to generate a DNB overpower pdf. The 95/95 probability/confidence limit value of this pdf is then employed as a percent overpower penalty in the setpoint analyses.

Even though the uncertainties within each of the above two groups were combined statistically, the resulting penalty derived from each group (i.e. the 95/95 limit values of the MDNBR and DNB overpower pdf) was applied separately in the TM/LP LSSS and DNB LCO setpoint calculations. As a result, the overall penalty allowance was overly conservative. In the updated ESCU methodology, the DNBR pdf which is generated from the statistical combination of the group one uncertainties is incorporated into the TM/LP LSSS and DNB LCO stochastic simulation models which previously included only the group two uncertainties. A single DNB overpower pdf is then generated which reflects the aggregate uncertainties of both groups. The 95/95 probability/confidence limit value of this pdf is then applied as the overall penalty in the TM/LP LSSS and DNB LCO setpoint analyses. This penalty is reduced in comparison to the overall penalty derived using the current SCU methodology.

The calculational and measurement uncertainties employed in the application of ESCU methodology to St. Lucie Unit 2 are identical to those in previously approved core reload analyses for Unit 2 using CE 16x16 fuel assemblies. The ESCU methodology, as noted above, has been previously approved for application to TM/LP LSSS and DNB LCO setpoint calculations at another CE plant. The revised MDNBR value of 1.20, used in conjunction with the ESCU methodology, will continue to provide at least a 95 percent probability at a 95 percent confidence level that the hottest fuel rod in the core will not experience DNB during normal operation or design basis Anticipated Operational Occurrences initiated within the LCO limits. The current DNB-related minimum acceptable margin of safety therefore remains unchanged.

3.0 CONCLUSION

On the basis of the above evaluation, we find that the ESCU methodology is acceptable for application to St. Lucie Unit 2 when CE 16x16 fuel assemblies are utilized.

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2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady-state peak linear heat rate below the level at which centerline fuel melting will occur. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the CE-1 correlation. The CE-1 DNB correlation has been developed to predict the DNB heat flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to the DNB-SAFDL of 1.20 in conjunction with the Extended Statistical Combination of Uncertainties (ESCU). This value is derived through a statistical combination of the system parameter probability distribution functions with the CE-1 DNB correlation uncertainty. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show conservative loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the DNB-SAFDL is not violated for the family of axial shapes and corresponding radial peaks shown in Figure B 2.1-1. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.1-1. The area of safe operation is below and to the left of these lines.

The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

The Thermal Margin/Low Pressure and Local Power Density Trip Systems, in conjunction with Limiting Conditions for Operation, the Variable Overpower Trip and the Power Dependent Insertion Limits, assure that the Specified Acceptable Fuel Design Limits on DNB and Fuel Centerline Melt are not exceeded during normal operation and design basis Anticipated Operational Occurrences.

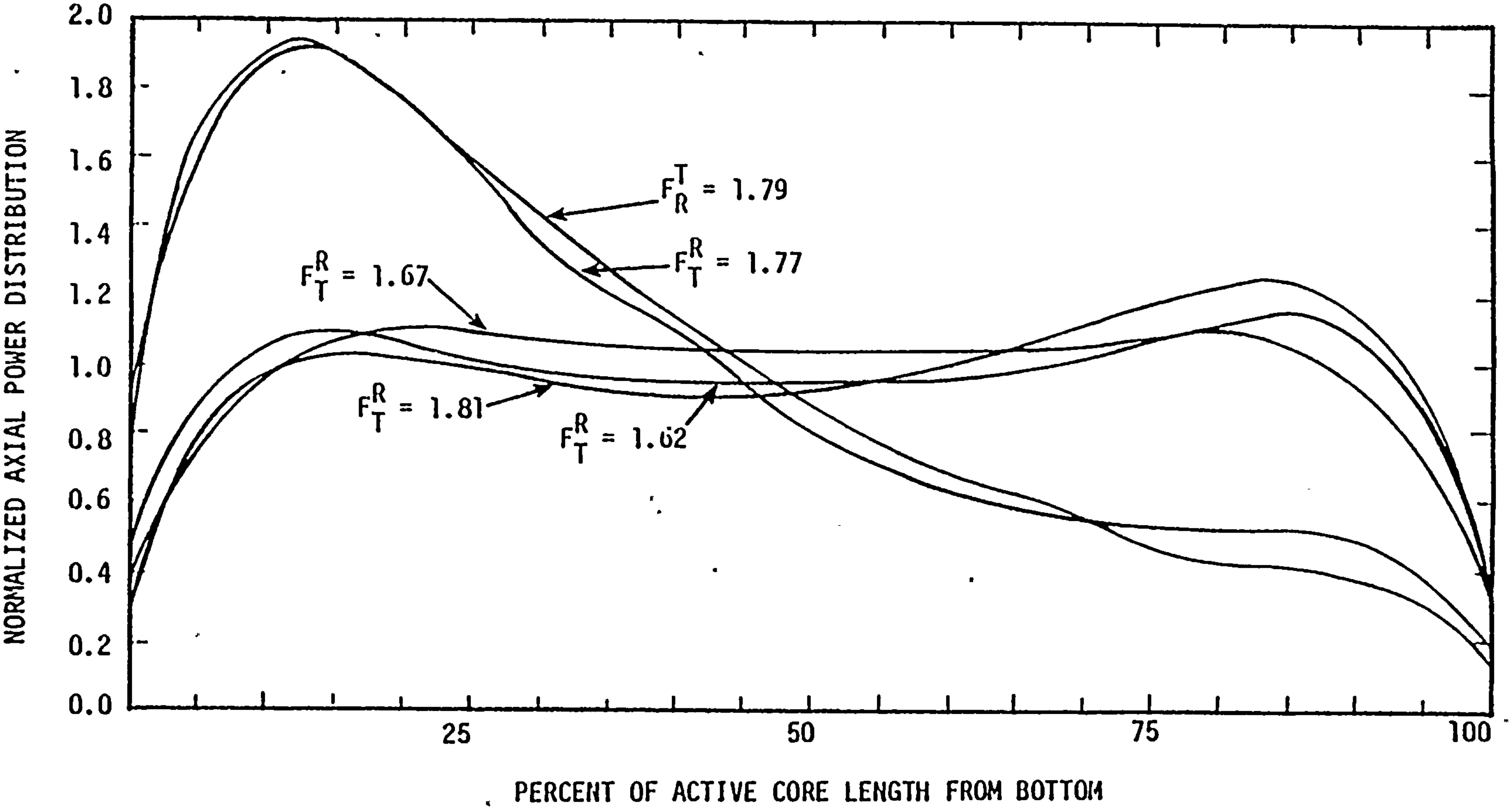

Figure B 2.1-1

Axial power distribution for thermal margin safety limits

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The Reactor Coolant System components are designed to Section III, 1971 Edition including Addenda to the Summer, 1973, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System was hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Variable Power Level - High

A Reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure Trip.

The Variable Power Level High trip setpoint is operator adjustable and can be set no higher than 9.61% above the indicated THERMAL POWER level.Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The trip setpoint has a maximum value of 107.0% of RATED THERMAL POWER and a minimum setpoint of 15.0% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is the value used in the safety analysis.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam line safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at less than or equal to 2375 psia which is below the nominal lift setting 2500 psia of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than the DNB-SAFDL of 1.20, in conjunction with ESCU methodology.

The trip is initiated whenever the Reactor Coolant System pressure signal drops below either 1900 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

The Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits through application of appropriate allowances for equipment response time measurement uncertainties and processing error. A safety margin is provided which includes: an allowance of 2.0% of RATED THERMAL POWER to compensate for potential power measurement error; an allowance of 3.0°F to compensate for potential temperature measurement uncertainty; and a further allowance of 125 psia to compensate for pressure measurement error and time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit. The 125 psia allowance is made up of a 55 psia pressure measurement allowance and a 70 psia time delay allowance.

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: (1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, (2) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied. and (3) the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for (1) a measurement-calculational uncertainty factor of 1.062, (2) an engineering uncertainty factor of 1.03, (3) an allowance of 1.01 for axial fuel densification and thermal expansion, and (4) a THERMAL POWER measurement uncertainty factor of 1.02.

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F^T_{xy} and F^T_r AND AZIMUTHAL POWER TILT - T_q

The limitations on F^T_{xy} and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F^T_r and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, the Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F^T_{xy}, F^T_r or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the

assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid.

An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The requirement that the measured value of T_q be multiplied by the calculated values of F_r and F_{xy} to determine F_r^T and F_{xy}^T is applicable only when F_r and F_{xy} are calculated with a non-full core power distribution analysis code. When monitoring a reactor core power distribution, F_r or F_{xy} with a full core power distribution analysis code the azimuthal tilt is explicitly accounted for as part of the radial power distribution used to calculate F_{xy} and F_r.

The Surveillance Requirements for verifying that F_{xy}^T, F_r^T and T_q are within their limits provide assurance that the actual values of F_{xy}^T, F_r^T and T_q do not exceed the assumed values. Verifying F_{xy}^T and F_r^T after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and safety analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of ≥ 1.20, in conjunction with ESCU methodology throughout each analyzed transient.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18-month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12-hour basis.