DUKE POWER COMPANY

OCONEE NUCLEAR STATION

RELOAD DESIGN METHODOLOGY

Technical Report

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ABSTRACT

This Technical Report describes Duke Power Company's Reload Design Methodology for the Oconee Nuclear Station. Included in this report are descriptions of Fuel Design, Fuel Cycle Design, Fuel Mechanical Performance, Maneuvering Analysis, Thermal Hydraulic Design, Technical Specifications Review and Development, Accident Analysis Review, and the Development of Core Physics Parameters.

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4. FUEL MECHANICAL AND THERMAL PERFORMANCE

4.1 Introduction

Each fuel cycle design requires that thorough fuel mechanical and thermal assessments be performed. A reload design utilizes fuel designs that are bound by previous fuel assembly design analyses. Occasionally, however, minor differences in the design will occur (such as a change from 94% TD fuel to 95% TD fuel). These changes must then be assessed in regard to the following:

- Cladding creep collapse,
- Cladding strain,
- Cladding stress,
- Fuel pin temperature, and
- Fuel pin pressure

Design analyses that envelope the operation of all current fuel designs have been completed by the fuel vendor, and reanalysis is normally not required for a new fuel cycle design. Rather, a specific fuel cycle design is compared against the enveloping design analyses. The assessment must compare cladding and pellet designs against the pellet and cladding geometries and densities, etc., that have been considered in the enveloping design analyses. Further, the individual radial power histories during the fuel cycle (current and previous batches) must be compared against the generic radial power envelopes that have been used in the design analyses. In most cases, the design analyses will envelope the fuel cycle design being considered and no reanalysis is required. However, in some cases, either the radial power history or fuel geometry may lie outside of the enveloping design analyses, thus requiring partial or full reanalysis. The following subsections describe the types of comparisons that must be made to justify a fuel cycle design without reanalysis and provides some detail concerning the types of analyses that must be performed if required by either the fuel cycle design or by changes in the fuel design itself.

Table 4-1 presents a summary of all types of fuel thermal and mechanical performance assessment criteria that are used to determine whether a fuel cycle design, the cladding, and the pellets are enveloped by existing analyses. As shown in Table 4-1, several of these analyses require either a comparison against a fuel pin power versus burnup envelope or a comparison against an assembly radial power versus burnup envelope. Examples of these power history envelopes are presented in Figures 4-1 and 4-2. These envelopes change, as reanalysis is occasionally required, resulting in an expanded power history envelope. Figure 4-3 presents a flow chart for the fuel pin pressure and linear heat rate to melt analyses. Figure 4-4 is a mechanical analysis flow diagram.

4.2 Cladding Collapse

Cladding creepdown under the influence of external (system) pressure is a phenomenon that must be evaluated during each reload fuel cycle design to ensure that the most limiting fuel rod does not exceed the cladding collapse exposure limit. Cladding creep is a function of neutron flux, cladding temperature, applied stress, cladding thickness, and initial ovality. Acceptability of a fuel cycle design is demonstrated by comparing the power histories of all the fuel assemblies against the generic assembly power history used in existing design analyses, similar to Figure 4-2. The generic power history must be completely enveloping to avoid reanalysis. Duke Power Company uses its own PDQ edit code to automatically perform this comparison for all fuel assemblies at each depletion step. Changes in pellet or cladding design are also assessed in a similar manner: direct comparison with the fuel rod geometries of Table 4-1 and reanalysis, if necessary. Four separate fuel designs have been analyzed to form the generic cladding creep collapse analysis.

The CROV[1] computer code calculates ovality changes in the fuel rod cladding due to thermal and irradiation creep and is used to perform the fuel rod creep collapse analysis when required. CROV predicts the conditions necessary for collapse and the resultant time to collapse. Conservative inputs to the CROV cladding collapse analysis include the use of minimum cladding wall thickness and maximum initial ovality (conservatively assumed to be uniformly oval), all as allowed by manufacturing specifications. Other conservatisms included are minimum prepressurization pressure and zero fission gas release. Internal pin pressure and cladding temperatures, input to CROV, are calculated by TACO2[6] using a radial power history similar to that of Figure 4-2, a generic pin to assembly local peak, and a standard axial flux shape.

The conservative fuel rod geometry and conservative power history are used to predict the number of EFPH required for complete cladding collapse. To demonstrate acceptability, the maximum expected residence time of the cycle is compared against the EFPH required for complete collapse.

4.3 Cladding Strain Analysis

The limit on cladding strain is that uniform strain of the cladding should not exceed 1.0%.

A generic strain analysis has been completed by the fuel vendor using TACO2 to ensure that the strain criterion above is not exceeded. To determine whether the fuel and fuel cycle designs are enveloped by existing analyses, the criteria of Table 4-1 are reviewed.

Should reanalysis be required, TACO2 will be used to determine the fuel rod dimensional changes that occur between the two power levels considered by the conservative design power ramp used in the strain analysis. Then, the maximum tensile (elastic and plastic) strain, which occurs at the cladding I.D., is determined from the following equation:

$$\text{Strain} = \frac{(\text{Pellet O.D.})\ \text{peak} - (\text{Pellet O.D.})_o \times 100}{(\text{Pellet O.D.})_o} \leq 1\%$$

where (Pellet O.D.) peak = the maximum pellet O.D. at the local power peak, and

$(\text{Pellet O.D.})_o$ = pellet O.D. prior to and after a local power ramp.

Pellet O.D. dimensions are used to calculate cladding strain because the strain itself is caused by pellet thermal expansion.

The strain analysis is completed in two parts:

- Part 1 employs TACO2 to determine when pellet contact occurs. A conservative fuel rod geometry is used in conjunction with a $\leq$ 1.5 axial flux shape, and the core average linear heat rate at 100% power to characterize gap closure. If contact occurs prior to 30,000 MWD/MTU, then Part 2 will use a ramp from 2 KW/FT to a final linear heat rate that is consistent with centerline fuel melt. Whereas, if contact occurs after 30,000 MWD/MTU, then the ramp's peak linear power is reduced to a lower value that is consistent with maximum local powers that could occur at burnups greater than 30,000 MWD/MTU.

- Part 2 of the strain analysis is the power ramp calculation, also performed on TACO2, which calculates the change in fuel pellet O.D. that occurs from the change in power level induced by the power ramp. The change in pellet O.D. is then used to perform the hand calculation of cladding strain using the equation above. The cladding and pellet are assumed to be in hard contact at the initiation of this ramp.

Thus, there are two major conditions in this scenario that make it conservative. The first is the extreme power change that is used to simulate the worst case peaking. The second is that the pellet is assumed to be in hard contact at inititation of the ramp. This is a conservative assumption since the power ramp is initiated from 2 KW/FT, and pellet/cladding contact is not expected to occur at this low linear heat rate.

4.4 Cladding Stress Analysis

The cladding stress analysis for a new fuel cycle design is similarly bounded by a conservative design analysis that uses Section III of the ASME Boiler and Pressure Vessel Code as a guide in classifying the stresses into various categories, assigning appropriate limits to these categories, and combining these stresses to determine stress intensity. Each new fuel cycle design is assessed against the criteria stated in Table 4-1 to determine if reanalysis is required. The stress analysis is very conservative, and reanalysis should not be required for standard Mark B reloads. However, an assessment is made for each reload design using the criteria of Table 4-1.

The fuel rod stress analysis considers those stresses that are not relaxed by small material deformation, and this anaysis complies with the following design critera:

- All fuel cladding stresses (primary and secondary) shall not exceed minimum unirradiated yield strength for condition I and II occurrences.

- The stress intensity value of the primary membrane stresses in the fuel rod cladding, which are not relieved by small material deformation of the cladding, shall not exceed 2/3 of the minimum unirradiated yield strength.

The above criteria keep the primary loads well below material allowable.

In performing the stress analysis, all the loads were selected to represent the worst case loads and were then combined. This represents a conservative approach since they cannot occur simultaneously. This insures that the worst case conditions for condition I and II events are satisfied. In addition, these input parameters were chosen so that they conservatively envelope all Mk-B design conditions.

The primary membrane stresses result from the compressive pressure loading. Stresses resulting from creep ovalization are addressed in the creep collapse analysis.

Since the internal fuel rod pressure cannot exceed system pressure for condition I and II occurrences (at coolant temperatures greater than 425°F), the need to address tensile stresses at hot zero power (HZP) conditions and higher is eliminated. The tensile stresses were addressed at cold conditions. The minimum internal fuel rod pressure at HZP conditions is combined with the maximum design system pressure during a transient to simulate the maximum pressure differential across the cladding. The tensile stress analyzed occurs at cold (room temperature) conditions at BOL. This is the worst case since the grid loads will be maximum at that point.

The worst case compressive pressure loads were combined with the other worst case loads. These are described below:

- The maximum grid loads will occur at BOL. During operation, the contact force will relax with time due to fuel rod creepdown and ovalization as well as grid spring relaxation.

- The maximum radial thermal stress will occur at the maximum rated power (power level corresponding to centerline fuel melt). This stress cannot physically occur at the same time the maximum pressure loading occurs, but is assumed to do so for conservatism. (Maximum cladding temperature gradient is combined with minimum pin pressure.)

- The ovality bending stresses are calculated at BOL conditions. A linear stress distribution is assumed. The creep collapse analysis calculates the stress increase with time and ovalization.

- Flow induced vibration and differential fuel rod growth stresses are also addressed.

4.5 Fuel Pin Pressure Analysis

The pin pressure analysis is assessed against the design basis analysis criteria and envelopes as indicated in Table 4-1. If any of the parameters of this table are violated, then a reanalysis is performed.

Pin pressure analysis is performed using TACO2. The rod is assumed to have a 1.5 symmetric axial flux shape, with a pin power history similar to that presented in Figure 4-1. Incore fuel densification is minimized in this analysis to yield a smaller plenum volume and a maximum pin pressure.

Figure 4-5 presents the result of an analysis of pin pressure versus burnup, performed by Duke Power Company, using TACO2. This analysis was performed for an extended burnup fuel cycle design, using the pin power history indicated in Figure 4-1, but with lower, more realistic axial flux shapes than the 1.5 cosine shape that is used for Reload Design purposes. (Refer to Table 4-2 for the axial flux shapes used in this extended burnup analysis.) To satisfy mechanical design criteria, pin pressure must be less than system pressure (2200 psia).

4.6 Linear Heat Rate Capability

Linear heat rate capability of all fuel rods in a reload batch is assessed by comparison against the criteria and envelopes of Table

4-1. Any rod whose geometry or power history falls outside of those criteria must be reanalyzed.

The Linear Heat Rate to Melt (LHRTM) analysis is performed using TACO2, assuming maximum incore pellet densification. This analysis assumes a conservative pin power history, similar to that of Figure 4-1, and a 1.5 cosine axial flux shape. In this analysis, very small axial segments of the fuel rod are spiked to high linear heat rates at each burnup step until centerline fuel melt occurs. The resulting heat rate required to reach centerline fuel melt at each burnup is then plotted versus burnup.

Figure 4-6 is a plot of fuel LHRTM versus burnup for an extended burnup fuel cycle design. This TACO2 analysis, performed by Duke Power Company, represents the pin power history of Figure 4-1, but with more realistic axial flux shapes than the 1.5 cosine that is used for reload fuel cycles. (Refer to Table 4-2 for the axial flux shapes used in this analysis.) The minimum LHRTM occurs early in life due to fuel densification, but quickly increases due to the offsetting effects of cladding creepdown, pellet swelling, and fuel relocation. (No credit is taken for fuel relocation in LHRTM analyses).

TABLE 4-1
FUEL MECHANICAL PERFORMANCE ASSESSMENT CRITERIA

		Analysis Category				
Item No.	Parameter Reviewed[1]	Cladding Collapse[2]	Cladding Strain	Cladding Stress	Pin Pressure	Linear Heat Rate Capability
1	Pin Power History vs Burnup	NA	NA	NA	Figure 4-1	Figure 4-1
2	Radial Assembly Power History vs Burnup	Figure 4-2	NA	NA	NA	NA
3	Clad O.D.	Yes	Yes	Yes	Yes	Yes
4	Clad I.D.	Yes	Yes	Yes	Yes	Yes
5	Clad Thickness	Yes	Yes	Yes	Yes	Yes
6	Clad Initial Ovality	Yes	NA	NA	NA	NA
7	Pellet Diameter	Yes	Yes	Yes	Yes	Yes
8	Pellet Density	Yes	Yes	Yes	Yes	Yes
9	Initial Prepressure	Yes	Yes	Yes	Yes	Yes

NOTES:
1. These criteria are the more significant items reviewed for a reload fuel cycle design, and do not include minor assumptions that are part of the bases.
2. The cladding collapse review actually is performed separately for each type of Mark B fuel design (four sets of parameters exist, corresponding to four separate fuel designs).

TABLE 4-2

Axial Flux Shapes Used for Thermal Analysis
(Reference, Figures 4-5, 4-6)

Burnup Range	Peak of Axial Cosine Shapes
0 - 15,100	1.28
15,100 - 35,000	1.22
> 35,000	1.16

NOTE: Standard reload design analyses always employ a 1.5 P/P axial flux shape for pin pressure and LHRTM analysis.

FIGURE 4-1 PIN POWER VERSUS BURNUP ENVELOPE FOR THERMAL ANALYSIS ASSESSMENTS (EXAMPLE ONLY)

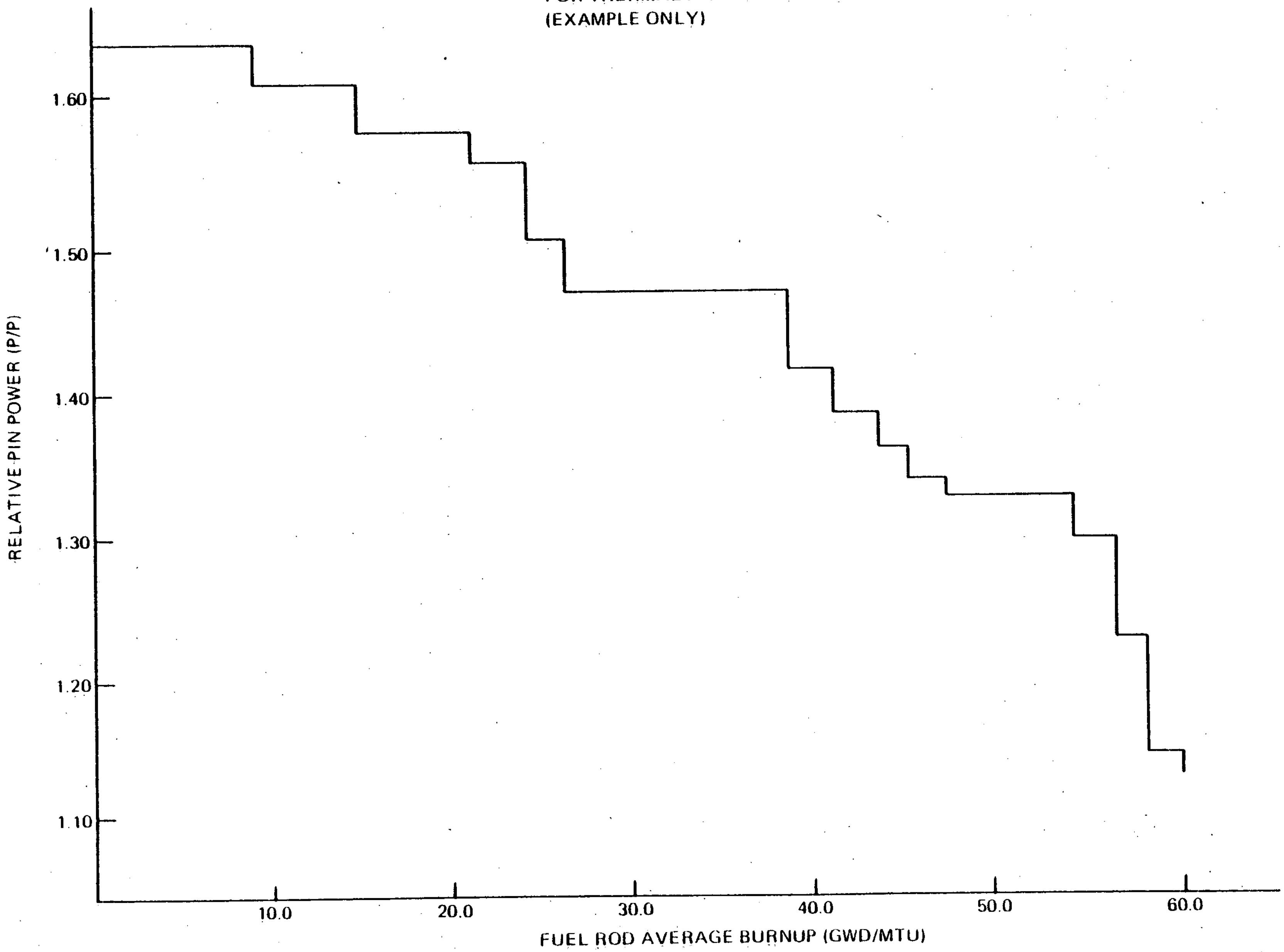


FIGURE 4-2 RADIAL ASSEMBLY POWER VERSUS BURNUP FOR CREEP COLLAPSE ANALYSIS ASSESSMENTS (EXAMPLE ONLY)

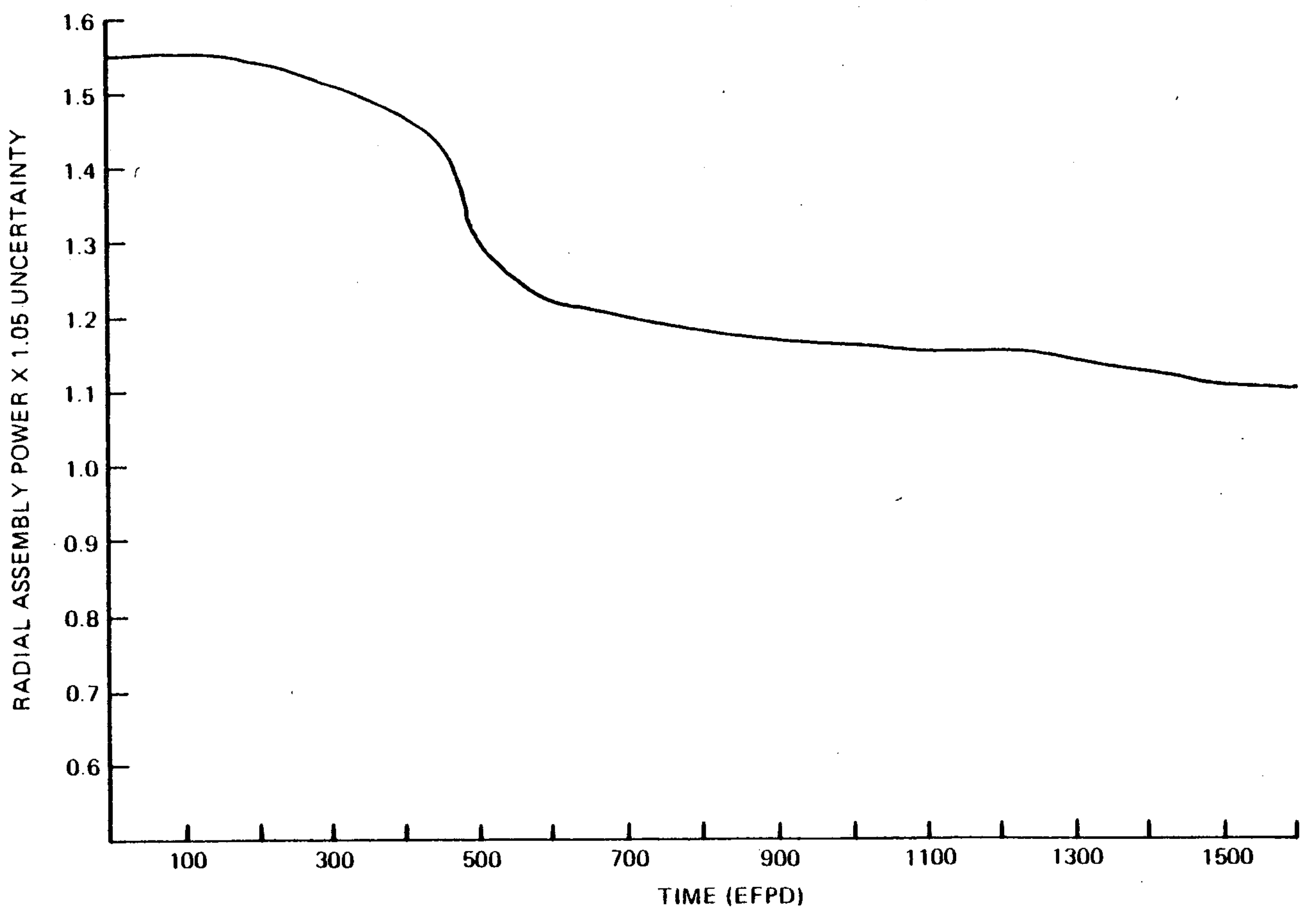


FIGURE 4-3 THERMAL ANALYSIS FLOW DIAGRAM

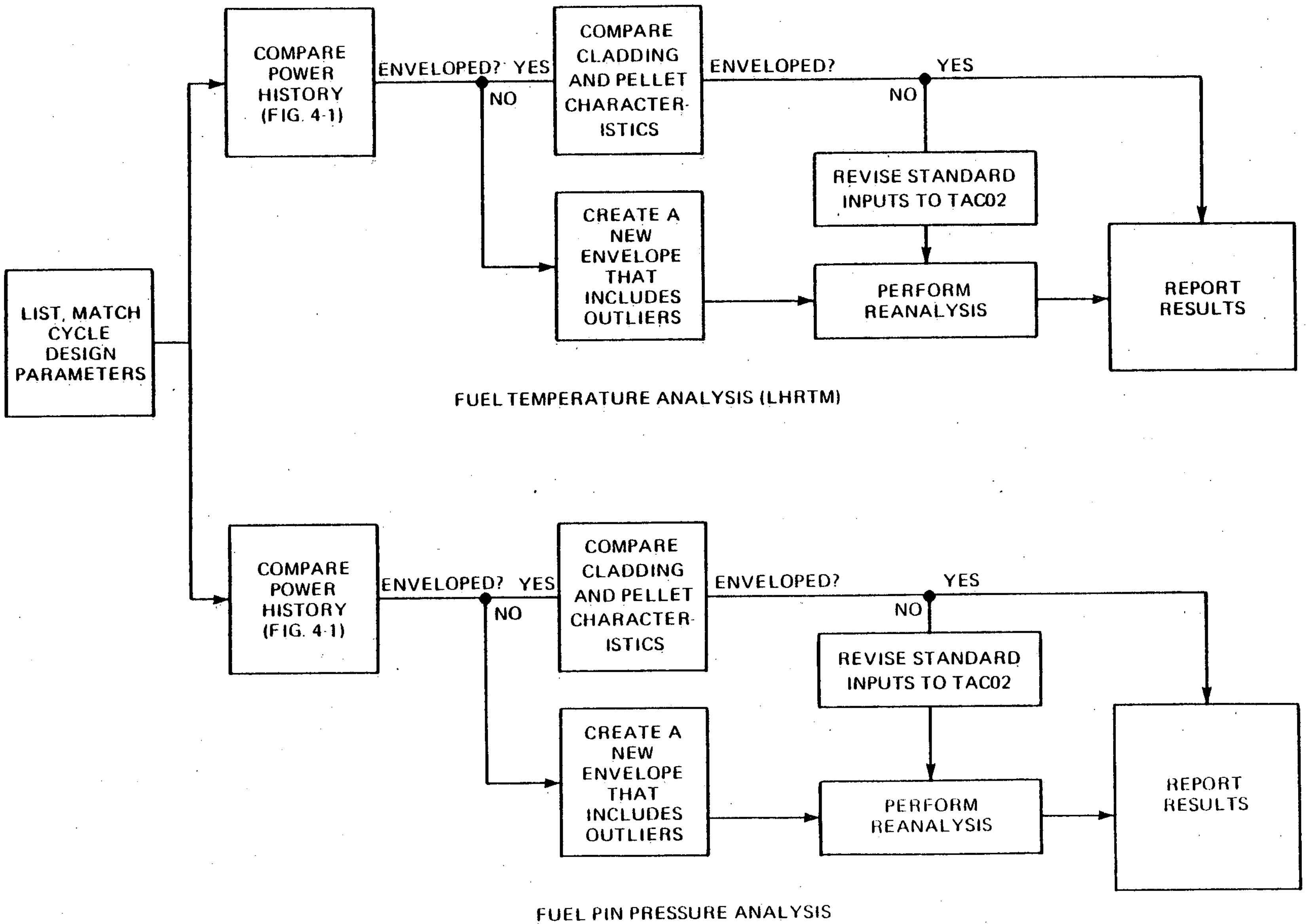
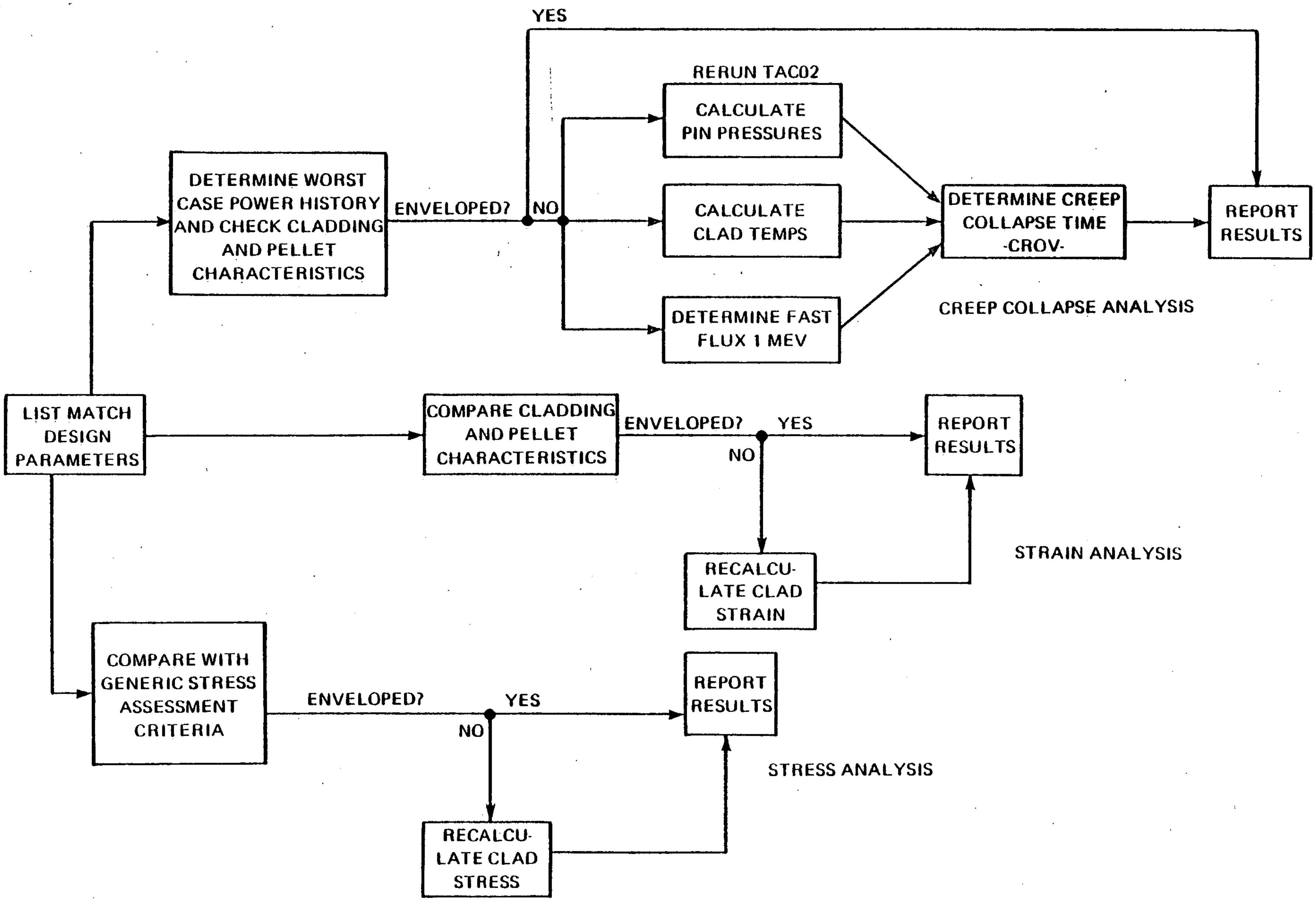


FIGURE 4-4 MECHANICAL ANALYSES FLOW DIAGRAM



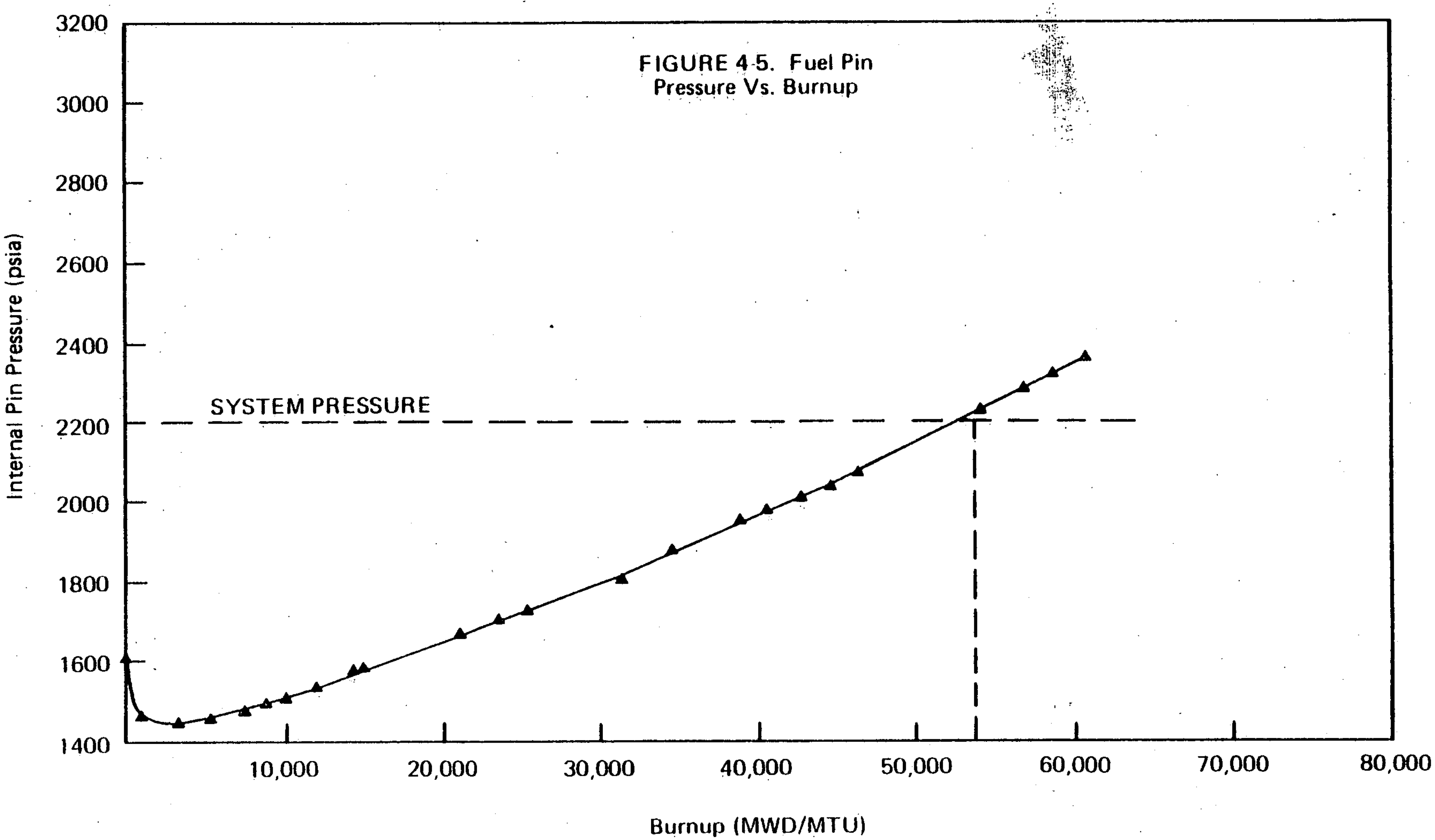
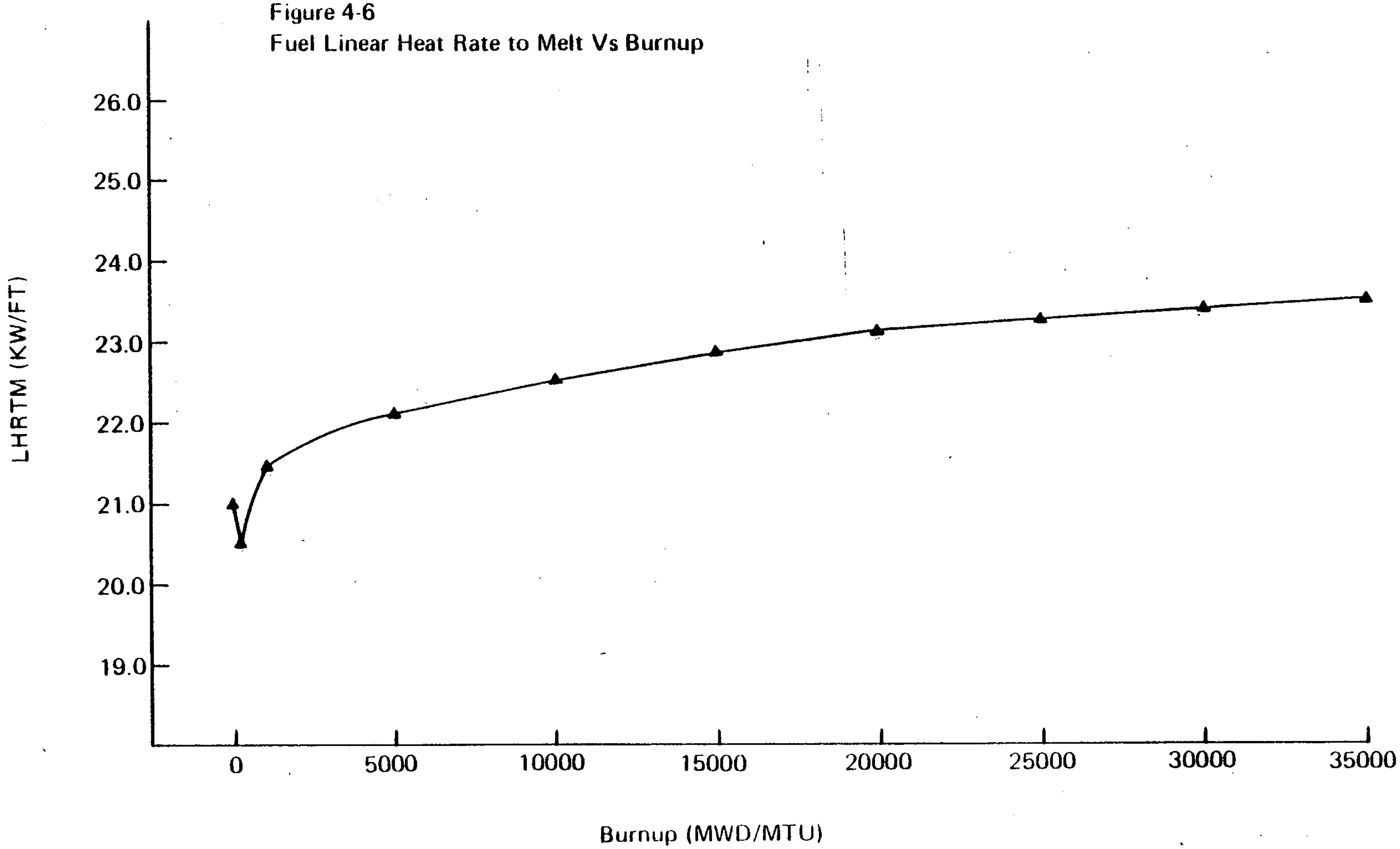


FIGURE 4-5. Fuel Pin Pressure Vs. Burnup

Figure 4-6
Fuel Linear Heat Rate to Melt Vs Burnup



pressure drop regardless of variations in local peaking and axial power shape. In other words, hot channel flow rate will be adjusted by the code to satisfy core-wide pressure drop as local conditions are varied. The axial power shapes input to these parametric hot channel runs are smooth cosine curves whose peak can be specified at various distances up the channel for each series of axial peaking factors. To obtain the maximum allowable peaking factor for each data point, power input to the channel is increased until the limiting DNBR of 1.4326 is reached. This process determines a maximum allowable total peak for a specified axial peak and its location.

After completion of these parametric analyses, two sets of generic DNBR curves or Maximum Allowable Peaking (MAP) curves are determined. One set is used for DNB operational offset limits, and the second set is used for RPS DNB offset limits. The generic DNBR curves used as operational limits are a conservative overlay of 1) the generic DNBR curves used for RPS offset limits, and 2) another set of MAP curves which have the reference design DNBR as their basis. Both sets of limits consider the extremities of the P-T core protection envelope (619°F and 1800 psig) as potential core operating conditions. Thus both the operational DNB offset limits and the RPS DNB offset limits have considered the worst case temperature and pressure envelope permitted by the RPS.
The last step in the thermal-hydraulic analysis is to take actual power shapes that gave the lowest DNBRs during the maneuvering analysis and input these irregularly shaped axial curves into the hot channel code to verify conservatism of the corresponding cosine curves used to develop the generic DNBR curves. A typical set of generic DNB curves is provided in Figure 6.3.

6.8.3 Hot Channel Factors

The following additional hot channel factors on local heat flux are utilized in the thermal-hydraulic analyses for developing the generic DNBR curves:

1.026 = penalty incurred to increase calculated axial powers since flux depressions at the spacer grids are ignored.

1.024 = the ratio of the total nuclear uncertainty of 1.075 to the radial nuclear uncertainty of 1.05.

Thus, in determining the generic DNB curves, the normal value of Fq" is increased from 1.014 to 1.065.

6.9 Transient Analysis - Determination of the Flux - Flow Ratio

During a loss of one or more reactor coolant pumps, the core is prevented from violating the 1.4326 minimum DNBR criterion by a reactor trip that is initiated by exceeding the allowable reactor power to reactor coolant flow ratio setpoint. Loss of one or more reactor coolant (RC) pumps is also detected by the RC pump monitors. That is, independently of the power to flow trip, loss of one RC pump will result in an automatic reactor runback. Similarly, loss of two or more RC pumps from above 55% full power will cause a reactor trip.

The thermal-hydraulic analysis that is used to set the power to flow trip setpoint for coastdown protection conservatively assumes the loss of two RC pumps. The transient is analyzed using the RADAR code to assure that the 1.4326 minimum DNBR criterion is not violated at anytime during the loss of one or more RC pumps.

The steady state thermal-hydraulic analysis provides the starting point for the transient analysis. The power to flow setpoint itself is derived from this analysis by varying the time of reactor trip following the loss of two RC pumps (that is by considering various trip setpoints) until the minimum ratio required to maintain the minimum DNBR of 1.4326 has been determined. Calculation of the actual (error corrected) power to flow setpoint used at the nuclear station is described in Section 7.3.2.

6.10 Application of the Rod Bow Penalty

In existing thermal-hydraulic analyses, a very conservative DNBR penalty is included to account for rod bowing effects. This penalty (11.2%), however, has been reduced by 1% because of the flow area (rod pitch) reduction factor already included in the thermal-hydraulic analysis.

2

For some reloads, additional credit can be applied based on the fact that primary coolant flow can be proven to be higher than the 106.5% design flow.

The resulting net penalty is applied directly to the final DNBR margins or by increasing the 1.3 DNBR criteria by the percent penalty, resulting in a DNBR criterion of 1.4326. 2

In future fuel cycle designs, this penalty will be revised to reflect the true effect of measured rod bowing on minimum DNBR (if any additional penalty is required). References 12 and 13 document the methods to be used for determining the true rod bow penalty. Then, a determination will be made to either maintain the current margin which exists or to eliminate part or all of this margin.

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