ATTACHMENT 2

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

Technical Report

NFS-1001

"Reload Design Methodology"

Revision 4

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

RELOAD DESIGN METHODOLOGY

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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 TACO[16] (or TACO2[6] when approved) | Rev. 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 TACO (or TACO2 when approved) to ensure that the strain | Rev. 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, TACO (or TACO2 when approved) will | Rev. 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 TACO (or TACO2 when approved) 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. Rev.

- Part 2 of the strain analysis is the power ramp calculation, also performed on TACO (or TACO2 when approved), 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. Rev.

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

- 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 TACO (or TACO2 when approved). 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. Rev. 4

Figure 4-5 presents the result of an analysis of pin pressure versus burnup, performed by Duke Power Company, using TACO (or TACO2 when approved). 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). Rev. 4

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 TACO (or TACO2 when approved), 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. Rev. 4

Figure 4-6 is a plot of fuel LHRTM versus burnup for an extended burnup fuel cycle design. This TACO (or TACO2 when approved) 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). Rev. 4

FIGURE 4-3 THERMAL ANALYSIS FLOW DIAGRAM

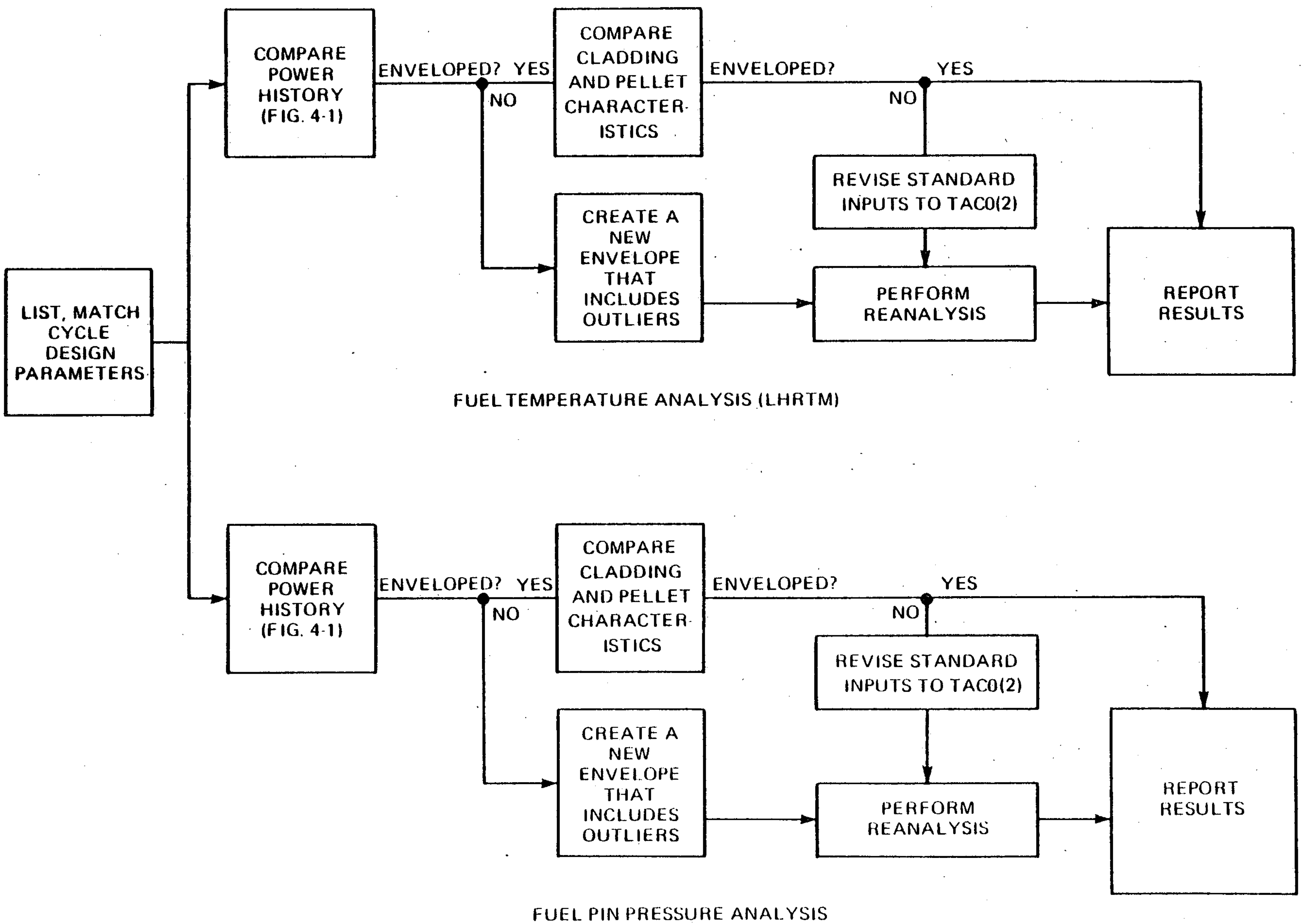
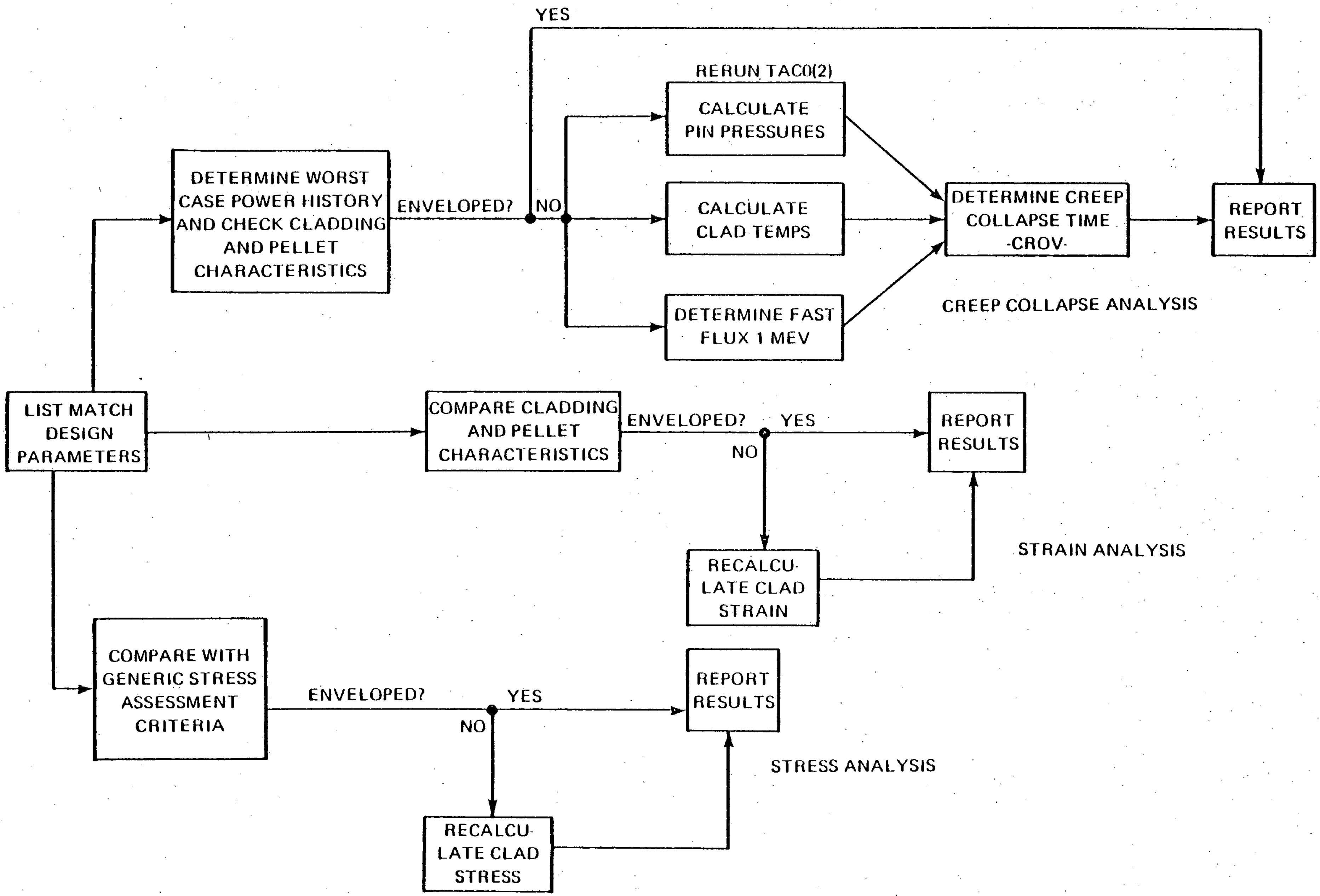


FIGURE 4-4 MECHANICAL ANALYSES FLOW DIAGRAM



fied whenever the P-T core safety limits are changed, P-T error adjustment factors are changed, or the high RC outlet temperature trip setpoints are changed, or the low RCS pressure trip setpoint is changed.

In order to determine the P-T trip setpoints, first the locus of pressure-temperature points constrained by the high RCS pressure trip setpoint (2300 psig), 4
the high RCS temperature trip setpoint (619°F), and the low RCS pressure trip setpoint (1800 psig) are identified on the Core Safety P-T Limit curve, as shown in Figure 7-4. Referring to Figure 7-4, the straight lines AB, BC, and DE respectively represent the locus of P-T points constrained by the high RCS pressure trip, the high RCS temperature trip, and the low RCS pressure trip setpoints. Next, the pressure-temperature points C and D are adjusted for the difference between the core pressure and the RCS pressure at the measurement location and for the errors in the temperature and pressure measurements by the RPS. Referring to Figure 7-4, C' and D' are the error adjusted points, and the straight line C'D' joining these points defines the locus of RPS P-T trip setpoints.

7.3.2 Determination of RPS Power-Flow-Imbalance Trip Setpoints

The power-flow-imbalance trip setpoints define the values of reactor power at which RPS trip should occur whenever the combinations of power, flow, and their uncertainties produce limiting values of power and flow which result in the design minimum DNBR during a flow transient and whenever the combination of power, imbalance, and their uncertainties correspond to the core safety limits on power-imbalance. This trip function is established by considering maximum allowable power-to-flow ratio and by considering the maximum allowable values of power as a function of imbalance. The maximum allowable power-to-flow ratio is constrained by the requirement that the minimum DNBR, in the event of a limiting flow transient, is equal to or greater than the design limit of 1.3. Thus the power-flow-imbalance trip setpoints ensure core protection during transients involving a flow reduction (by the power-to-flow trip portion of the trip function) and during conditions involving adverse power distributions (by the power-imbalance trip portions of the trip function).

In order to determine the power-flow-imbalance trip setpoints, first the maximum allowable power-to-flow ratio is to be obtained. The maximum allowable power-to-flow ratio (also called the flux/flow trip setpoint) is obtained by reducing the calculated flux/flow ratio (Section 6.9) by an error adjustment 4
factor, which takes into account the noise in the RPS flow signal and other electronic errors in the RPS flow instrumentation. Next, the core safety power-imbalance limits are error-adjusted both on the power level limit and the imbalance limit. The error adjustment factor for power level is 6.5% FP, which includes 4% FP allowance for the neutron flux error (uncertainty in correlating the RPS measured neutron flux to reactor power), 2% FP allowance for the calorimetric error, and ½% FP allowance for any setpoint error. The error adjustment factor for imbalance accounts for the uncertainty in the measurement of axial imbalance by the out-of-core detector system, and it is a function of the imbalance limit and the power level. To establish the RPS power-flow-imbalance trip setpoints, the error adjusted power and imbalance are plotted on a figure with imbalance as the horizontal axis and power as the vertical axis. The envelope obtained by the straight lines passing through pairs of these points and the horizontal straight line drawn passing through the point representing 112% power for the 4-pump case or the maximum power allowed by the flux/flow 1
trip setpoint, as illustrated in Figure 7-5.

7.4 Development of Limiting Conditions for Operation

The limiting conditions of operation generally requiring modification in conjunction with a reload cycle are the LOCA-limited power distribution limits, shutdown margin-limited control rod insertion limits, and the ejected rod worth-limited control rod insertion limits.

The LOCA-limited power distribution limits are limits on pertinent core parameters (such as control rod positions, axial imbalance, quadrant power tilt, and xenon conditions which influence the power distribution in the core) such that the power distributions in the core during normal operation are within the values assumed in the safety analysis of the loss of coolant accident.

The shutdown margin-limited control rod insertion limits are limits on the maximum allowable control rod insertions satisfying the shutdown margin

Table 7-1

Reactor Protection System Trip Functions

Reactor Trip	Monitored Parameter	Trip Setpoint During 4-Pump Operation	Purpose of Trip	
1. Overpower trip	Neutron flux	105.5% FP	To provide core protection during transients involving uncontrolled power increase.	
2. Power-flow-imbalance trip	Neutron flux, RC flow and power imbalance	Flux/Flow = 1.08	To provide core protection during transients involving a flow reduction and during core conditions involving excessive power peaking	1
3. RCS pressure-temperature trip	RCS pressure and RC outlet temperature	Function of RC outlet temperature	To provide core protection during transients involving a reduction in pressure or a reduction in core heat removal and to ensure reactor shut down during a LOCA.	
4. Low RCS pressure trip	RCS pressure	1800 psig	To provide core protection during transients involving a pressure reduction	
5. RC Pump Monitor trip	Neutron flux and pump contact monitor voltage	Loss of two pumps above 55% FP	To provide core protection during loss of RC pumps	
6. High RCS pressure trip	RCS pressure	2300 psig	To provide protection of RCS pressure boundary from excessive pressures	4
7. High RCS temperature trip	RC outlet temp.	619°F	To prevent excessive temperature in the RCS	
8. High RC pressure trip	RB pressure	4 psig	To ensure reactor shutdown during a LOCA and SLB inside containment.	

Figure 7-3 Core Protection Safety Limits

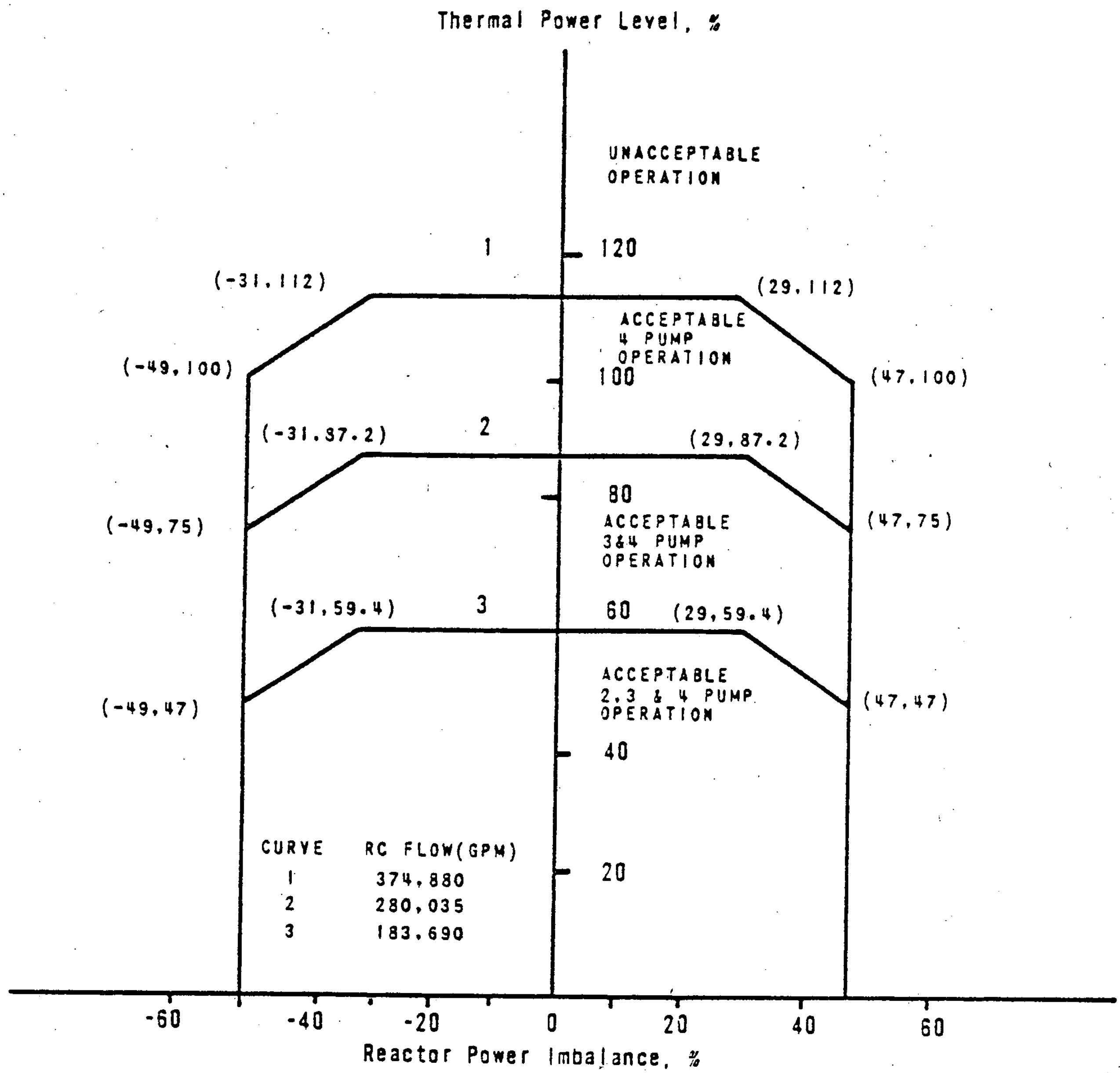
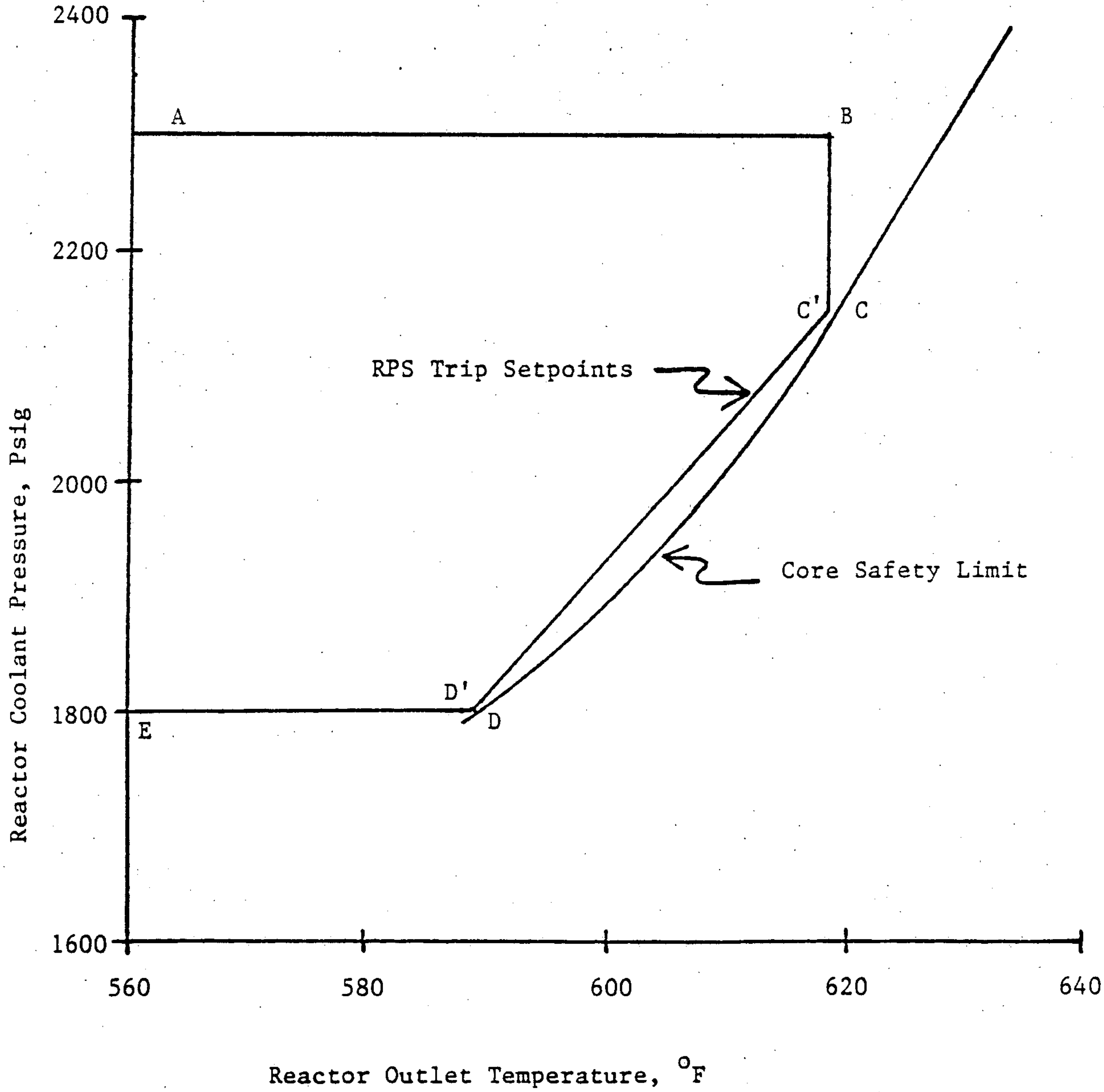


Figure 7-4

Determination of RPS P-T Trip Setpoints



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15. J. R. Gloudemans and H. C. Cheatwood, RADAR - Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown, BAW-10069A, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, October 1974. 3

16. TACO - Fuel Performance Analysis, BAW - 10087A, Rev. 1, Babcock & Wilcox, Lynchburg, Virginia, August 1977. 4

EPRI-SHUFFLE

The EPRI-SHUFFLE program will read a PDQØ7 concentration file, make certain modifications to this file, and write a new updated concentration file. This procedure is accomplished by defining "assembly regions" in the program input. Assembly regions are square arrays of mesh points containing depletable nuclide concentrations and superimposed on the original PDQØ7 geometry. These assembly regions are then used to describe the movement of existing nuclide concentrations by translation, reflection and/or rotation. In addition, new fuel concentrations can replace spent fuel concentrations in selected assembly regions described in the program's input.

EPRI-SUPERLINK

SUPERLINK accesses data on the files produced by EPRI-FIT and together with relevant input information for file management and for data processing control produces polynomial coefficients for use in EPRI-NODE.

PDQØ7

See EPRI-PDQØ7 Modifications.

NODE UTILITY CODE (NUC)

The NUC program is a package of subroutines that performs any necessary utility function to EPRI-NODE files. The major subroutines are:

I. FILE - this mode lists, merges, purges, adds, rearranges, edits, etc. the NODE cases on one or more history files.

II. FLEX - this mode takes an existing file, expands or collapses it to a new problem size, and then stores it on a new disk.

III. COPY - this mode copies a given history file from disk storage (working file) to magnetic tape storage (permanent backup file) and vice versa.

IV. MARGINS - this mode performs those operations which are necessary to calculate CFM, DNB, and LOCA margins from an input history file(s). It also plots the results in the form of a "fly speck" graph.

TACO2

TACO2 conservatively predicts fuel pin temperature and fuel pin pressure. It includes models for fuel densification, fuel swelling, fuel restructuring, gas release, cladding creep, and gap closure. 3

TEMP

TEMP is a steady state open channel thermal hydraulic code that considers energy mixing between channels and is used to calculate flow distribution among individual channels in an assembly or a cluster of fuel pins. It calculates flow, pressure drop, coolant parameters up the channel, and DNBR.

RADAR

RADAR performs a thermal analysis of a slow reactor transient such as the loss of a primary pump, computing as a function of time fuel pin and clad surface temperatures, DNBR, and coolant thermodynamic conditions when given pin power and either channel flow or pressure drop as a function of time.

TACO

TACO conservatively predicts fuel pin temperature and fuel pin pressure. It includes models for fuel densification, fuel swelling, fuel restructuring, gas release, cladding creep, and gap closure. 4

Appendix B

List of Revisions

Revision	Date Issued
1	May 1980
2	January 1981
3	April 1981
4	June 1981