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Responds to 781115 request for addl info re "Qual Reactor Physics Methods for Appl to Kewaunee". Addl info describing methodology & procedures to eval key physics parameters to determine if reanalysis of accidents is req set for 790115.

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P.O. Box 1200, Green Bay, Wisconsin 54305

December 5, 1978

Mr. A. Schwencer, Chief  
Operating Reactors Branch #1  
Division of Operating Reactors  
U. S. Nuclear Regulatory Commission  
Washington, D. C. 20555

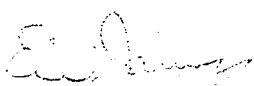
Gentlemen:

Kewaunee Nuclear Power Plant  
Docket No. 50-305  
Operating License DPR-43  
Request for Additional Information Concerning Your  
Review of "Qualification of Reactor Physics Methods  
for Application to Kewaunee"

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Enclosed please find five (5) copies of our response to your November 15, 1978, request for additional information concerning your review of the referenced topical report. Several of your requests require detailed information and descriptions which are not easily answered individually. In response to these questions and probable future questions, a detailed description of the methodology and procedures employed to evaluate key physics parameters to determine whether or not reanalysis of any accidents are required will be provided. Our current schedule will permit transmittal of this report to you by mid-January, 1979.

Very truly yours,

  
E. W. James  
Senior Vice President  
Power Supply & Engineering

snf

Encl.

*Acc  
1/11*

7812080108

ENCLOSURE

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

QUALIFICATION OF REACTOR PHYSICS METHODS

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1. Describe in more detail the specific ARMP calculational procedures used for each physics parameter calculated, i.e., an expanded version of Figure 2.1 and App. B of your qualification report.

Response: *Attachment A to this enclosure contains the WPS Core Analysis System Description. This description identifies the codes employed, the flow path and methodology employed to transfer and utilize data to and from the various codes, the resulting output, and the pertinent physics parameters of the various stages. This description is an expansion of the summary presented to members of the NRC staff during our November 1, 1978, meeting.*

2. Describe in more detail the thermal-hydraulic models and options selected including items such as the DNB correlation used in steady-state and transient analyses.

Response: *A more detailed description of the thermal-hydraulic models and options selected is also included in Attachment A.*

3. For each accident or transient, provide a list of the key physics parameters, an indication of their influence on the analysis results, and the procedures used to determine whether or not a reanalysis is required.

Response: *Due to the detailed in depth response needed to answer this question and several others, we intend to provide for your review a complete description of the methodology employed for determining whether the physics parameters calculated fall within existing accident analyses. This report will be accompanied by a preliminary Cycle 5 reload evaluation for direct comparison to the methods employed. Our anticipated transmittal date is mid-January, 1979.*

4. Provide background information on your experience in accident analysis to justify expertise required to make the determinations stated in item 3.

Response: *This information will also be provided as an attachment to the report responding to the information requested in item 3.*

5. Provide additional information on Exxon fuel design such as nuclear and thermal characteristics which may differ from the Westinghouse fuel and their associated effect on reload analysis, including LOCA analysis (e.g., gap conductance, power spikes, clad thickness, etc.).

Response: *This information is outside the scope of our qualification of Reactor Physics methods for the Kewaunee Plant. The analyses necessary to provide this information is being performed by Exxon Nuclear Company. Their current schedule calls for a report to be transmitted to us by mid-January, 1979. We expect to transmit this report to the NRC in early February following our receipt and review of it.*

6. If the present CAOC or PDC schemes are not used (a possibility mentioned at the 11/1/78 meeting) state what technology must be developed and approved if WPSC originates its own power distribution control technique.

Response: *At this time it is our intention to use one of the approved CAOC or PDC schemes. The evaluation presented in response to question 3 should verify the validity of one or either of these schemes for Cycle 5. If Cycle 5 or any future cycle evaluation requires the use of a different power distribution control scheme, relevant documentation will be provided in the proper time frame.*

7. Discuss planned startup testing and acceptance criteria to be provided as part of reload submittal.

Response: *At this time we intend to perform the same startup tests that have been employed in previous reload cores utilizing the same acceptance criteria. We intend to use the Rod Swap technique for verification of control rod worths as was done in Cycle 4 startup. Details of the startup tests and acceptance criteria for Cycle 5 startup will be provided along with the Reload Safety Evaluation for Cycle 5.*

8. Discuss any planned setpoint changes and additional information required.

Response: *System setpoint values are selected to meet conformance criteria for Accident Analyses. Analyses being performed by Exxon Nuclear Company will either verify existing setpoints or provide justification for any changes in setpoints.*

9. Provide information on the evaluation procedure used to assure that Tech. Spec. PDIL values are valid for all Cycle 5 operation.

Response: Power dependent Insertion Limits are dependent upon the shutdown margin requirements that result from the evaluation of the reload core. The documentation submitted in response to question 3 will describe the shutdown margins calculated for Cycle 5 and the evaluation procedures employed to assure that the PDIL values are valid.

WPS CORE ANALYSIS

SYSTEM DESCRIPTION

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## I. OVERVIEW

The description of the ARMP model and data flow are described in detail in the ARMP documentation sections:

Part I, Chapter 1 "Overview", and

Part I, Chapter 6 "Design Implementation"

This documentation describes the model calculational procedures which were followed during model construction at WPS, i.e., RCC and BPR parameterization, color set geometrics, etc.

The individual flow charting and descriptions of the ARMP computer codes are detailed in the ARMP documentation section:

Part II, Chapter 1 "Code Summary".

This information will not be repeated in any detail here, however, the flow chart diagrams presented in figures 1 and 2 illustrate the model data flow and the calculational flow respectively.

The implementation of the ARMP model and its interfaces with the WPS core analysis system will be described in the following sections.

## II. WPS-ARMP ANALYSIS SYSTEM SUMMARY

A simplified diagram of the WPS implementation of the ARMP model is presented in Figure 3.

The Spectral code, EPRI-CELL (ARMP, Part II, Chapter 5), produces initial nuclide concentrations, depletion and fission product chain data, and tables of microscopic and macroscopic cross-sections varying with burnup.

The NUPUNCHER code (ARMP, Part II, Chapter 8) serves as a data processor between EPRI-CELL and PDQ7/HARMONY.

CPM (ARMP, Part I, Chapter 6) provides lumped absorber data for burnable poisons and control rods to PDQ7/HARMONY.

PDQ7/HARMONY (XY-diffusion-depletion) is run in both the quarter core mode and in the unit assembly (color set) mode. The quarter core mode is used for nodal code normalization, local peaking factor generation, and for the establishment of assembly loading patterns for advanced cycles. In the color set mode, PDQ7/HARMONY supplies input data for the nodal code, EPRI-NODE-P (ARMP, Part II, Chapter 14).

The processing codes EPRI-FIT (ARMP, Part II, Chapter 10) and SUPERLINK-P (ARMP, Part II, Chapter 12) convert PDQ output data files into nodal code input.

The processing codes PCENSUS, NIFTIE, and QUEDEE convert PDQ output data files into INCORE analytic input.

Processing codes such as BETAV, NCOED and ROD\$, convert 3D nodal output into core physics parameters for use in evaluating core performance and safety as described in Section 5.

Monthly core performance reviews are performed by comparing the measurements to model results to provide continuing assurance of the model applicability.

### III. PRELIMINARY RELOAD CORE DESIGN

The following discussion defines the process and flow of information regarding the development of a Reload Core Design. The computer codes employed are displayed in Figure 4 along with the information flow.

OPTIMA is a two-dimensional nodal code (FLARE) capable of performing automated fuel loading and assembly loading patterns to minimize power peaking factors. It will also automate burnable poison loading and calculate fuel cycle costs. Input to OPTIMA is very similar to the 3D input. Output from OPTIMA includes a defined cycle lifetime (MWD/MTU), an assemblywise x-y power distribution, the assumed feed enrichment, the number of feed assemblies, burnable poison rods needed, and a core loading pattern. (Note 8th core symmetry is assumed.) An assembly loading pattern developed by OPTIMA is used as input to NSTASH.

NSTASH will expand the 8th core loading pattern to a full core loading pattern for Kewaunee, preserving core symmetry. Fuel assembly inventory, prior burnup history and symmetric assembly locations are all accounted for to eliminate core loading errors. The output pattern defines the shuffled assembly identification in its correct full core location. New fuel locations are provided with randomly generated new fuel identification to assure manufacturing tolerances on assembly loading will not affect core symmetry. The output also includes full core shuffled assembly rotations, reflections and translations for a quarter core symmetric shuffle of the PDQ concentration files.

End of cycle PDQ concentration files and output from NSTASH related to the quarter core shuffle are input to the SHUFFLE code.

Output from SHUFFLE includes (Beginning of Cycle) PDQ concentration files representing the shuffled reload core. A quarter core PDQ is depleted using as its initial step, the file generated by SHUFFLE. PDQ output files are maintained for each depletion step.

Input to PCENSUS includes the partition power files generated by PDQ. Output consists of assembly and rod power distributions. Each PDQ depletion step is processed through PCENSUS. The output of PCENSUS is verified to assure that peak rod power (FDHN) are well within technical specifications for Kewaunee for the entire cycle.

If the rod powers are unacceptable, the process starting with OPTIMA is repeated. If they are acceptable, and all other system constraints (i.e., fuel enrichment, cycle length, etc.) are acceptable, a Preliminary Reload Core Design has been defined. Core physics calculations and Reload Safety Evaluation commence.

#### IV. POWER DISTRIBUTION ANALYSIS

The flow diagram detailing the power distribution analysis is presented in Figure 5.

Power distribution analysis is based on nodal calculations combined with quarter core PDQ local pin to assembly power ratios. The remaining codes shown in Figure 5 perform the automated editing of peaking factors and application of appropriate reliability factors as defined in the Topical Report, "Qualification of Reactor Physics Methods for Application to Kewaunee".

The PDQ quarter core representing the preliminary core reload design (Ref. Section III) is used to create output data files representing partition power and pointwise fluxes, at core exposure steps expected through the cycle.

The PCENSUS code processes the PDQ-7 partition power files to obtain fuel rod average powers. The rod powers are sampled and sorted into relative power "bins". The highest powered fuel rods in each fuel assembly are identified by location and alphanumeric identifiers and stored on disk pack for input to the NIFTIE code.

The NIFTIE and QUEDEE computer codes edit and format the fuel rod power output from PCENSUS and PDQ-7 flux file data to obtain analytic INCORE input data and pin to assembly factors for use with the nodal power distribution.

The FQ computer code sequence automates the synthesis of the nodal power distributions and the quarter core PDQ pin to assembly power ratios and edits the resulting peaking factors. The model reliability factors are also applied in the POSTCORE portion of this sequence in a manner consistent with the definitions given in the Topical Report, "Qualification of Reactor Physics Methods for Application to Kewaunee".

#### V. CORE PARAMETER ANALYSIS

The FLOW diagram for core physics parameter analysis is presented in Figure 6.

The 3D nodal core model is used to compute reload core parameters used for reload safety evaluations, plant operations, and core performance evaluations. Aside from peaking factors, DNB calculations and delayed neutron parameters, the remaining of the core parameters used in the reload safety evaluation are inferred directly from the 3D nodal code without auxiliary code interface.

Rodworth values used for startup and operations are computed by the ROD\$ computer code using pairs of Core Keff and control rod position computed by the 3D nodal model. Output consists of plots of integral and differential rod worths as well as tables of rod position and reactivity.

Reactivity coefficients are edited by the NCOED computer code using values of soluble boron concentration, coolant temperature and core power supplied by the 3D nodal model. Pressure, power, temperature and boron coefficients of reactivity are output.

Calculations of the core delayed neutron characteristics are used for both the reload safety evaluations and interpretation of core behavior during start-up testing. The fission rates by isotope are obtained from PDQ-7 and input to the BETAV code. BETAV combines the isotopic fission rates with isotopic delayed neutron yields to obtain the 6-group delayed neutron parameters. The core average prompt neutron lifetime is also computed from the PDQ-7 group constants supplied to the BETAV code.

The delayed neutron parameters are input to the GAMMA code to obtain axial exposure effects and 3D nodal power weighting.

Delayed neutron parameters are processed by the NHOOR code to transform them into startup rate and reactor period as a function of core reactivity.

## VI. THERMAL HYDRAULIC ANALYSIS

The basic method employed for thermal hydraulic analysis is described in the ARMP documentation:

ARMP Part II, Chapter 16 "EPRI-THERM-P".

The method description will not be repeated here. A description of the WPS implementation of the method and its interface with the Core Analysis System will be discussed.

The EPRI-THERM-P computer code is a closed channel thermal hydraulic model based on the NAI HYDRO-P computer code employed in the WPS core analysis system. This code, used in the WPS core analysis system, differs from EPRI-THERM-P in the following respects:

1. The DNB correlation employed is the Westinghouse W-3 with the cold wall factor and the L grid spacer factors updated to those described in "Fuel Densification Kewaunee Nuclear Power Plant".
2. The subchannel model is adapted to perform T&H analysis of assemblies based on the fuel rod power distributions obtained from INCORE output. Entire assemblies may be analyzed for up to 61 axial nodes and 256 subchannels based on either measured or predicted power distributions.

## VII. SUMMARY

The overall WPS core analysis system is presented with details of data flow and computer code functions described. The implementation of the ARMP code package and its interface with the system are explained. The ARMP model setup procedures and data flow are adequately described in the referenced ARMP documentation and are not repeated here.

This report in conjunction with an understanding of the ARMP documentation provides a detailed description of the model used in preparation of the WPS topical "Qualification of Reactor Physics Methods for Application to Kewaunee". This core analysis system will be employed in the Reload Safety Evaluation methods.



FIGURE 1

OUTLINE OF PWR DATA FLOW

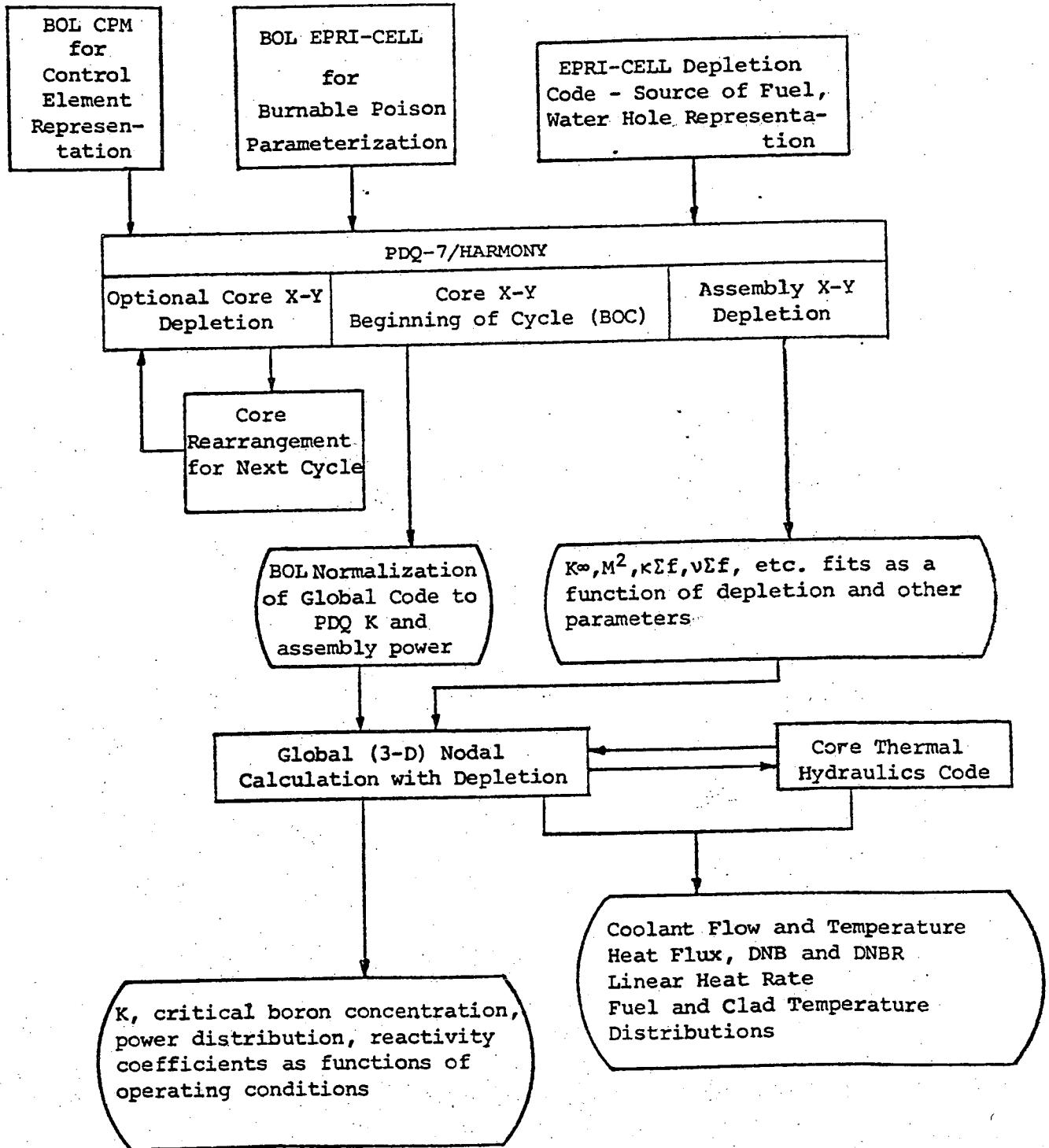


FIGURE 2  
PWR CALCULATIONAL FLOW

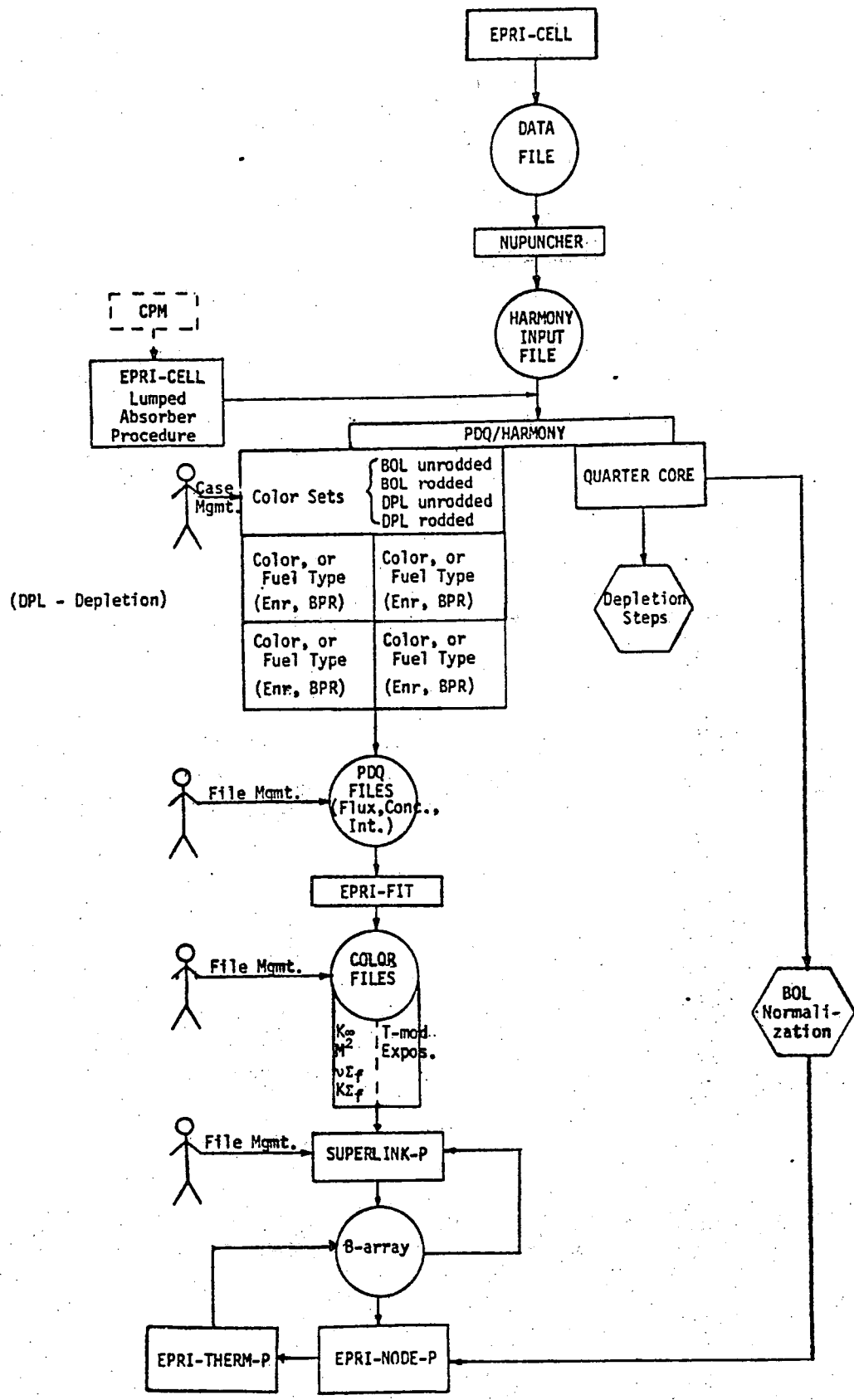


FIGURE 3

W.P.S. ANALYSIS SYSTEM

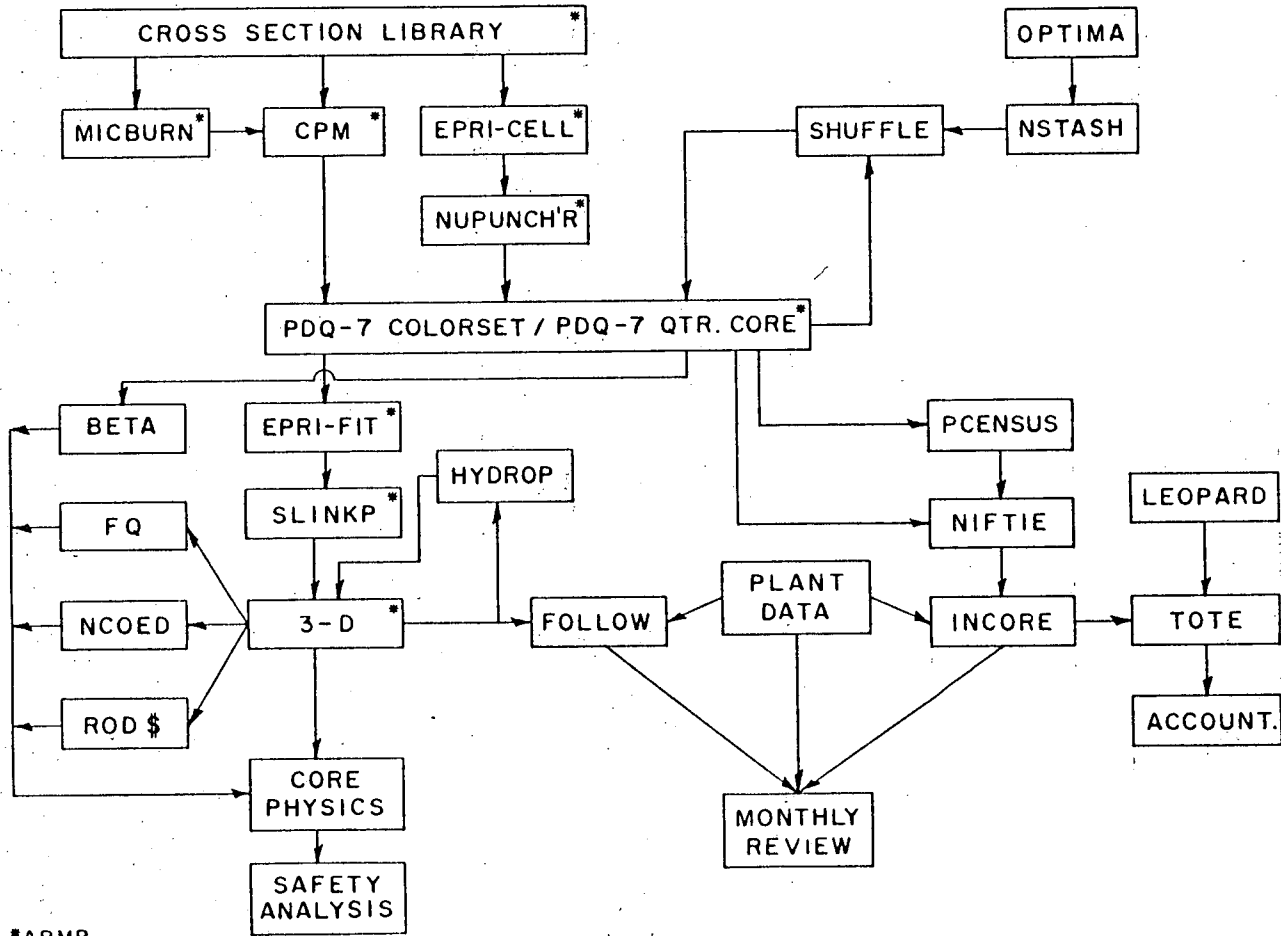


FIGURE 4  
RELOAD CORE DESIGN

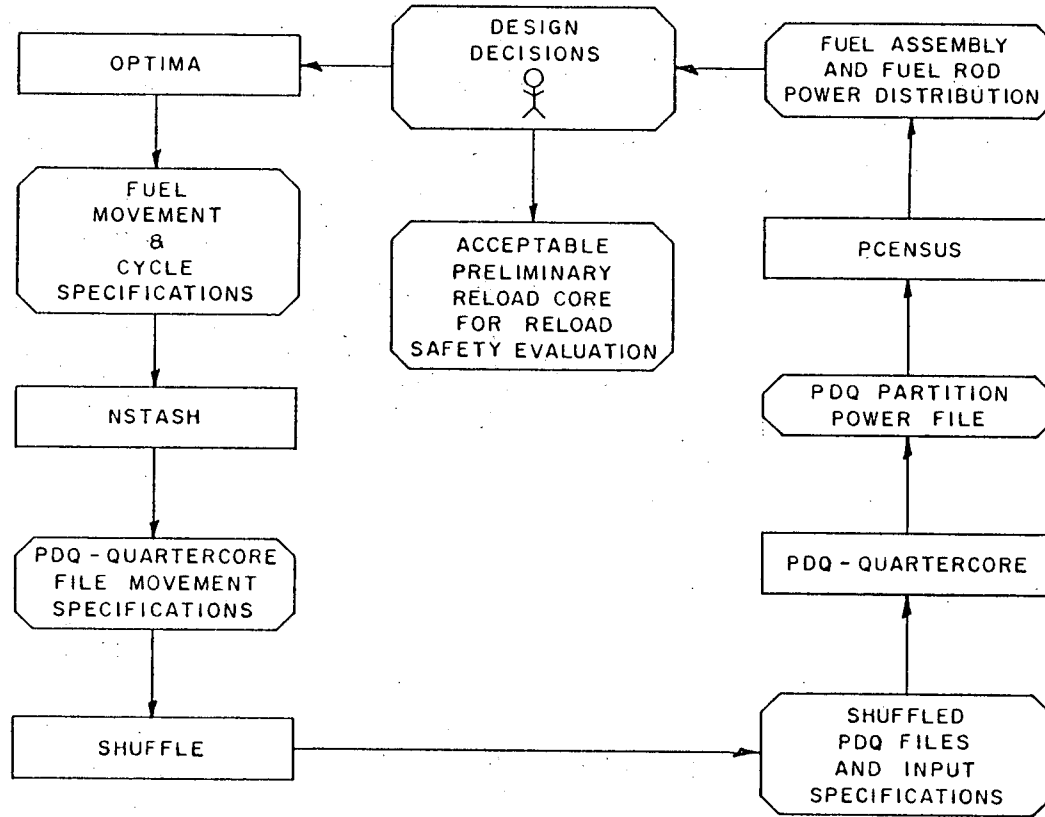


FIGURE 5

POWER DISTRIBUTION ANALYSIS

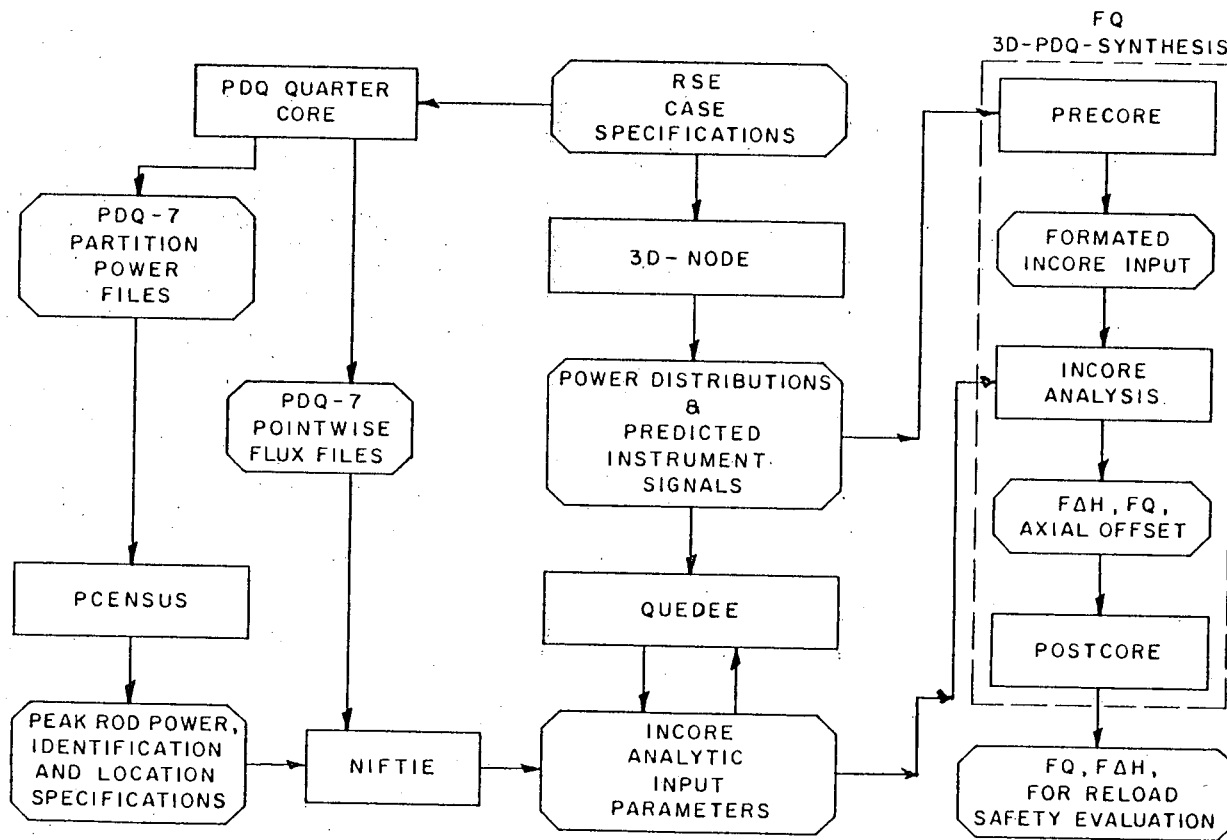


FIGURE 6  
PHYSICS PARAMETER ANALYSIS

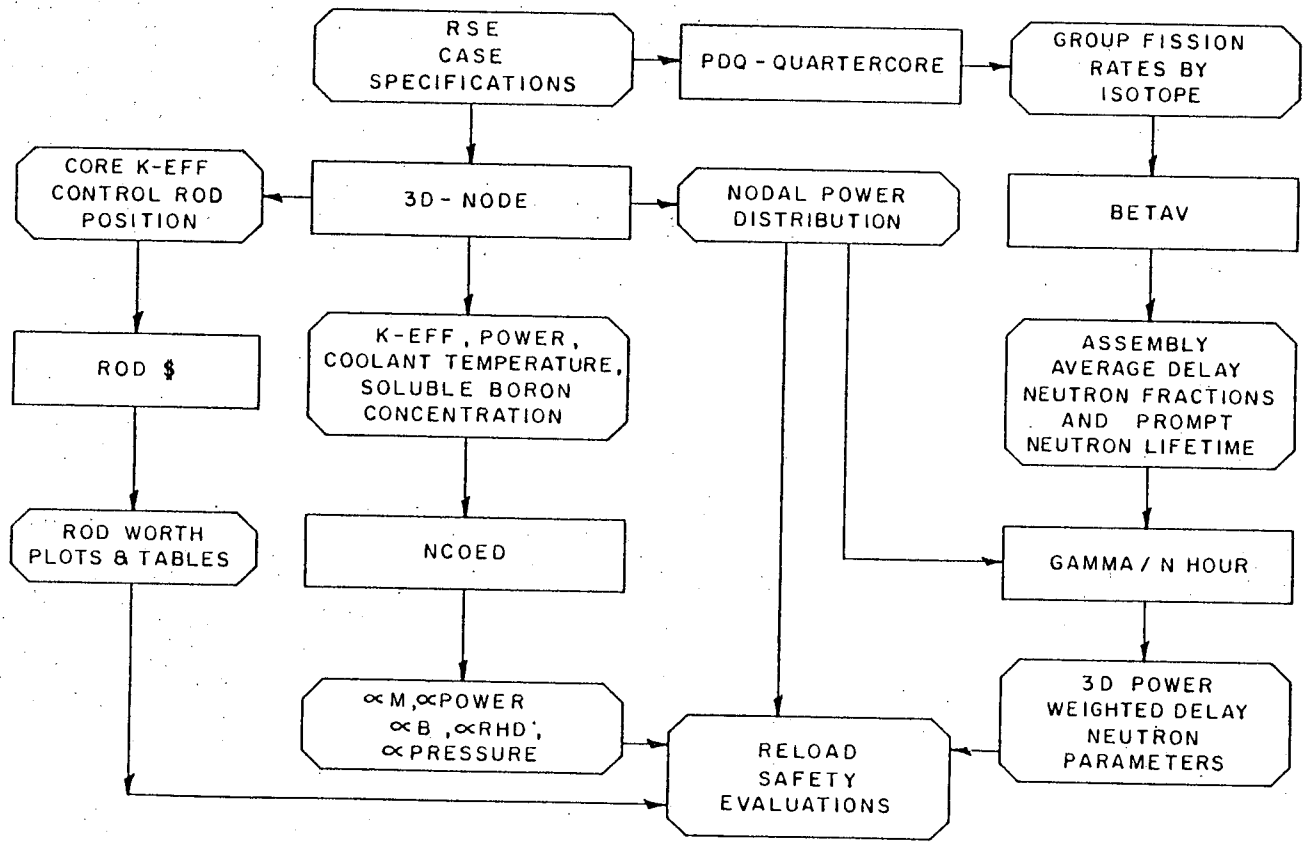


FIGURE 7  
THERMAL HYDRAULIC ANALYSIS

