NEACRP-L-133 ~~SPAIN~~

CYCLE DESIGN AND FUEL MANAGEMENT FOR A PWR

A reference reactor of 510 Mwt was chosen to do a general study involving Fuel Management Evaluations of several cycles from a real reactor status, and Design Calculations of cycles already performed, as well as a new one.

These tasks have been performed according to a calculation scheme set up in the Reactor Technology Division of the J.E.N., using some computer codes acquired to foreing sources (NUS Corporation, NEA-CPL at Ispra) and other ones developed in the J.E.N.

- 1. A previous recopilation of technical documentation, criteria, and basis, related to the fuel management, was done.

For fuel management evaluations two cycles lengths (12 and 18 months) were fixed up prior the study.

A set of reactivity (K_∞ vs. burnup) and isotopic tables for different feed enrichments were needed to carry out these studies.

They were obtained by means of LEOPARD, a lattice code that uses a fuel rod effective temperature, a Boron programme, and other items, to simulate the operating conditions.

The fuel temperatures were determined with the TEMP code for each type of mechanical design and power density considered.

The Boron programme was deduced from the reactor "follow" data, expressed at rated conditions. Two different programmes were used, according to the two cycle lengths.

Reloads and feed enrichments were the main unknown magnitudes, both for the transition and equilibrium cycles.

The neutronic problem of the transient period was treated with the CICLON code, based on the Approximate Balance of Reac-

tivity (A.B.R.) developed and tested in the J.E.N. for PWR.

Alternative reload strategies were evaluated, and results were transmitted to the FUELCOST program, to compute the economic performances.

Equilibrium cycles were dealt in a similar way. Nevertheless, both the neutronic and the economic problem were solved by the same code, SOTHIS, that also uses the A.B.R. to determine the feed enrichment for a reload given. Fuel costs are computed with a simple model to treat an equilibrium batch, whose major characteristics are determined by the neutronic part of the code. Several economic situations can be entered to the program, in order to be applied to the same neutronic results, that is, to perform sensitivity calculations.

2. A wide recopilation of technical documentation about nuclear design of PWR was also done, mainly focused to the information of the reference reactor.

Experimental data from two cycles were compiled, and processed by NUDO, a code to deal in-core measurements, developed in the J.E.N. for this activity. Evolutions of some main megnitudes (power distribution, boron concentration) along one of these cycles were prepared with NUDO in a way suitable for checking the calculation models used throughout this work.

To perform the design of a new cycle two different systems were used, once that fittings to the experimental data obtained by simulation of that cycle were quite acceptable.

The first one is basically a nodal method constitued by the ROLLO, MELON and NUTRIX codes.

NUTRIX is a three-dimensional reactor representation, coupling the nodes by means of several magnitudes depending on neutronic properties and geometry. It needs a good evaluation of the nodal Migration Area and K-infinite, and it's obtained

accounting for all the major effects using correlation formula.

The correlation coefficients are supplied by the MELON code. It has been developed mainly in the J.E.N., and in the last version all the significant feedbacks are considered.

On the other hand, NUTRIX also requires a suitable normalization, and the accuracy of its results depends strongly on the precision of values asigned to the free parameters of the code.

This problem was deeply studied, and a program, ROLLO, was written incorporating the JEN experiences on this subject. ROLLO uses a set of data processed by NUDO from the reactor performances, and obtains the adequate values to normalize NUTRIX.

The other system is based on CITATION, a powerful code to solve the neutron diffusion equation and to calculate power distribution, poison searches, etc., and PENELOPE, a programme to generate the cross sections and nuclide concentrations to be used in the former one, accounting for all the reactivity effects to take place in PWR,s. PENELOPE also performs the burnup calculation (region-and zonewise) and a simple thermohydraulic analysis to obtain the water density.

CITATION and PENELOPE work coupled, using several tape units to transmit the information required.

A third code, LIBRA, prepares the general nuclear data from lattice cell calculations. Both this one and PENELOPE have been developed in the J.E.N.

The reactor was treated with this system by different models. The usual one was the "fine-mesh discret" calculation, but also "slot" and "coarse - mesh" representations were done.

Both systems were used in the cycle design.

After normalizing the NUTRIX code for the reference reactor, power and burnup distributions were computed, simulating the cycle at rated conditions, and searching the critical Boron concentration.

The PENELOPE-CITATION system was more widely used. R-Z, R, X-Y, and Z representations were involved in the calculations, choosing them according to each specific problem.

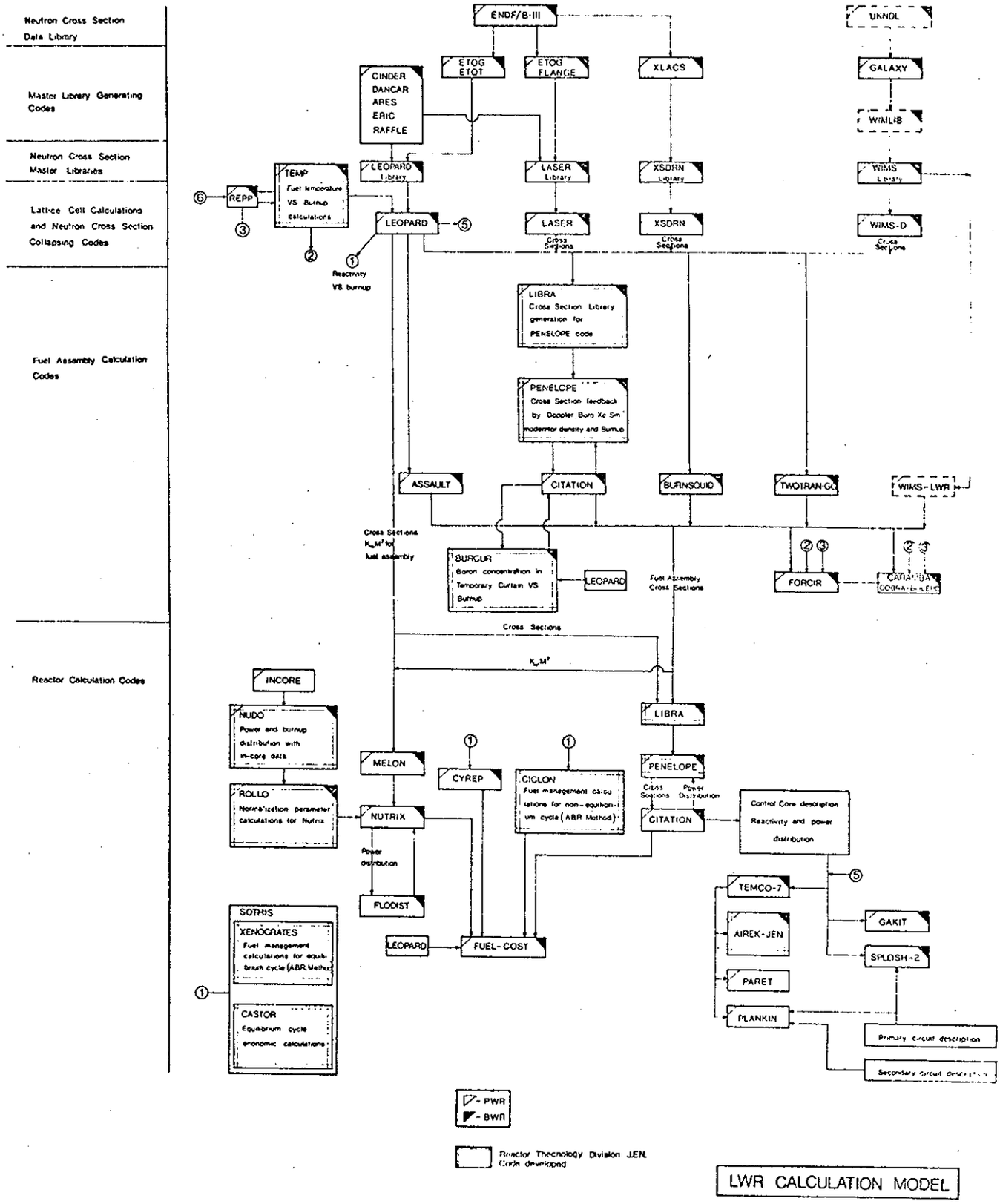
Reactor performances were done to simulate all the cycle. Coupled R-Z and R calculations determined a first guess of the Boron programme, as well as the axial Buckling at several burnup steps.

A X-Y and Z synthesis was carried out to represent the reactor as a set of fuel elements. All the magnitudes of interest were computed in this way, including burnups and isotopic concentrations at different heights of each fuel element.

Control rod studies were also accomplished, by means of reactor and assembly calculations. Last ones were used to determine the equivalent cross sections of assemblies with "rods-in", and core performances were carried out to show the power distribution shift, the control rod worth, and so on.

Reactivity coefficients were also evaluated with a similar synthesis calculations. Several perturbations on water density (enthalpy) and fuel temperatures were simulated at beginning and end of cycle.

Soluble Boron worth was determined at the same moments with the same procedure, and Xenon transients were studied, too.


Neutron Cross Section Data Library
Master Library Generating Codes
Neutron Cross Section Master Libraries
Lattice Cell Calculations and Neutron Cross Section Collapsing Codes
Fuel Assembly Calculation Codes
Reactor Calculation Codes
ENDF/B-III
UKNDL
ETOG ETOT
ETOG FLANGE
XLACS
GALAXY
CINDER DANCAR ARES ERIC RAFFLE
WIMLIB
LEOPARD Library
LASER Library
XSDRN Library
WIMS Library
TEMP Fuel temperature VS Burnup calculations
REPP
LEOPARD
LASER Cross Sections
XSDRN Cross Sections
WIMS-D Cross Sections
Reactivity VS burnup
LIBRA Cross Section Library generation for PENELOPE code
PENELOPE Cross Section feedback by Doppler, Burn Xe Sm moderator density and Burnup
ASSAULT
CITATION
BURNSQUID
TWOTRAN-GC
WIMS-LWR
Cross Sections K∞M² for fuel assembly
BURCUR Boron concentration in Temporary Curtain VS Burnup
LEOPARD
FORCIR
Fuel Assembly Cross Sections
Cross Sections
K∞M²
INCORE
NUDO Power and burnup distribution with in-core data
ROLLO Normalization parameter calculations for Nutrix
LIBRA
MELON
CYREP
CIGLON Fuel management calculations for non-equilibrium cycle (ABR Method)
PENELOPE
Cross Sections
Power Distribution
NUTRIX
CITATION
Control Core description Reactivity and power distribution
Power distribution
FLOODIST
SOTHIS
XENOCRATES Fuel management calculations for equilibrium cycle (ABR Method)
CASTOR Equilibrium cycle economic calculations
LEOPARD
FUEL-COST
TEMCO-7
AIREK-JEN
PARET
PLANKIN
GAKIT
SPLOSH-2
Primary circuit description
Secondary circuit description
- PWR
- BWR
Reactor Thecnology Division J.E.N. Code developed
LWR CALCULATION MODEL