CONF-900917--16

DYNAMIC ANALYSIS OF THE EBR-II PLANT WITH THE DSNP-ND SIMULATION PACKAGE

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Submitted for Presentation

at the

American Nuclear Society
The Safety, Status and Future of Non-commercial Reactors and Irradiation Facilities
September 30 - October 4, 1990
Boise Idaho

*Work supported by the U.S. Department of Energy, Reactor Systems, Development, and Technology, under Contract No. W-31-109-Eng-38

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ABSTRACT

Development and initial testing have been completed for a special version of the DSNP simulation language software package, DSNP-ND, which simulates EBR-II based on the models of the primary system components used in the NATDEMO program. Preliminary results are included for a comparison with an original NATDEMO prediction for the SHRT-45 transient.

I. INTRODUCTION

The purpose of the paper is to present the development and initial testing of a special version of the Dynamic Simulator for Nuclear Powerplants[1] (DSNP) simulation language software package, DSNP-ND, for simulating EBR-II. This version has new library modules based on the models of the primary system components used in the NATDEMO program.[2] The paper first describes EBR-II and then gives a brief background on DSNP. The new modules are described and the results of testing are presented. A brief summary completes the paper.

II. DESCRIPTION OF EBR-II REACTOR

The Experimental Breeder Reactor II (EBR-II) plant is a sodium- cooled, pool-type fast breeder reactor plant with a 62.5 MW thermal power rating. Figure 1 shows the components of EBR-II's primary system, which are completely submerged in the primary tank. The components are described below in detail sufficient to support the explanation of the new models in the DSNP-ND software package. More details of the design of the EBR-II primary system are found in References 3 and 4.

There are two inlet plena for the reactor, a high pressure one that feeds the inner seven rows and a low pressure one that supplies coolant to Rows 8 through 16. The plena serve to mix the flows from the two main pumps

and distribute them to the reactor assemblies.

The total reactor loading consists of 637 assemblies in three zones - core, reflector and blanket zones. There are five types of assemblies, but all of them have hexagonally-shaped ducts through which the coolant passes carrying the heat from the reactor. Each type has a different design inside the duct to perform its function. The core zone contains three types: driver, control and experimental assemblies. Driver assemblies have a fissile region, an above-core fission gas plenum, and upper and lower reflectors. There are 91 wire-wrapped elements in a driver assembly which forms the fissile and plenum regions. In the fissile region the elements contain metal alloy fuel pins, a sodium bond, and stainless steel cladding. A gas volume occupies the space in the pins above fuel. The axial reflector regions are made of stainless steel blocks with coolant passages. The reflector assemblies have four axial regions, plus a short upper adapter region, all composed of stainless steel and sodium. Blanket assemblies consist of 19 elements of depleted uranium pellets and sodium-bond in steel cladding, surrounded by sodium coolant. These assemblies are divided into five axial regions, the uppermost being the fission gas plenum. The sodium coolant enters all assemblies through the nozzles attached to the bottom of their hexagonal cans, passes upward and exits through the upper adapters into the outlet plenum.

The outlet plenum or (upper plenum) is bounded by the reactor vessel wall, assembly exits, and the reactor vessel cover which is removable for fuel handling. The Control Rod Drive Lines (CRDLs) pass through the upper plenum, penetrate the cover and then pass through the tank sodium to the tank cover. The coolant exits the vessel through a single horizontal nozzle. Leakage flow through the cover passages exist along the CRDLs and between the cover and wall, and heat is lost from the coolant through the wall and cover into the tank sodium.

The hot-leg pipe, called the Z-pipe because of its shape, transports the sodium from the upper plenum to the Intermediate Heat Exchanger (IHX). The hot sodium in the pipe is insulated from the colder sodium in the tank by a steel liner which maintains an annulus of stagnant sodium between it and the pipe. In the last segment of the Z-pipe the sodium passes through an auxiliary E-M pump before entering the IHX.

The IHX is a tube-and-shell, countercurrent heat exchanger. The primary system sodium enters a small baffled region at the top of the IHX and flows down the shell side of the IHX. The secondary sodium first passes through a center pipe and then distributes itself at the bottom of the IHX to the 3026 tubes and flows upward on the tube side. During normal operation, the axial temperature profiles are essentially linear on both primary and secondary sides. An insignificant amount of heat is transferred between the IHX and the sodium tank.

Upon leaving the IHX the primary sodium enters the sodium tank and flows toward the inlets of the two main pumps. On its way to the pump inlets the primary sodium mixes with some of the tank sodium and enters the pumps at the temperature of the resulting mixture. Since the pump inlets are located at different distances from the IHX outlet, the degree of mixing with

the tank sodium may be different for the two flows. The motor-driven, centrifugal pumps, which are rated at 270 kg/s, provide the hydraulic driving head for the primary coolant circuit. The sodium flows from each pump into a short discharge pipe and then separates into high- and low- pressure streams. The low-pressure flows are adjusted by manually-set throttle valves. The high- and low- pressure pipes from each pump are connected to the high- and low- pressure inlet plena, respectively.

III. DEVELOPMENT OF DSNP-ND

DSNP-ND (Dynamic Simulator for Nuclear Power-plants, Version NATDEMO) has component models initially developed for the NATCON program[5] or its successor, the NATDEMO program.[2] A little background on DSNP is helpful as a preface to the next section. DSNP[1] is a simulation language which generates modular FORTRAN programs that solve the steady-state and dynamic equations describing reactor components, systems or entire plants. The simulation software consists of a precompiler and up to five libraries. The precompiler generates the FORTRAN programs by translating the DSNP statements in the simulation problem specified by the user. The libraries contain component modules (e.g., a pump), neutronic process modules (e.g., point kinetics model), plant control functions (e.g., time delays), mathematical functions (e.g., integration routines), physical properties (e.g., sodium heat capacity), thermal hydraulics correlations (e.g., Dittus-Boelter equation) and MACRO translation rules, all written in the DSNP language. The development of DSNP-ND consisted of converting the NATDEMO models for EBR-II components into the DSNP language and adding them to the libraries.

IV. CONVERSION OF PRIMARY SYSTEM MODELS

A total of fourteen new modules were developed and added to the libraries of the DSNP-ND version. The models for these modules are discussed in the following sub-sections.

A. Reactor Channel Module, FUELR2

The driver, blanket and reflector assemblies are each represented by a single channel module which calculates the radial and axial temperature distributions from input values of the flow and power distributions. A channel consists of four concentric rings - a center pin, a cladding/moderator annulus (including a gap between it and the center pin), a coolant passage, and an outer annular ring with an adiabatic boundary. In the axial direction, the number of regions, the dimension and the number of nodes in each region, and the number of the 'core' region are set by the user. Radially the number of nodes in the pin and the dimensions and the materials in the four (plus a fuel-cladding gap) radial regions are specified. The user may specify different materials in the fueled and non-fueled regions for the pin and gap. Values may also be given for flow and heat transfer maldistribution, fraction of fuel theoretical density and pin pitch/diameter ratio. The channel model does not calculate axial heat transfer nor phase changes in any of the materials.

The temperatures or enthalpies are obtained from the steady-state and dynamic energy balances in the radial direction for each axial node. The dynamic equations for the coolant and an interior fuel node are given in equations 1 and 2:

$$M_c(I)\frac{dH_c(I)}{dt} = Q_c(I) + UA\left[T_1(I) - \overline{T_c}(I)\right] - UA_d\left[\overline{T_c}(I) - T_d(I)\right] - W_c\left[H(I+1) - H(I)\right] \quad (1)$$

at I^{th} axial node

$$M_f(I,J)C_p\frac{dT_f(I,J)}{dt} = UA_f(I,J-1)\left[T_f(I,J-1) - T_f(I,J)\right] - UA_f(I,J)\left[T_f(I,J) - T_f(I,J+1)\right] + Q_f(I,J) \quad (2)$$

at J^{th} radial node.

B. Channel Pin Heat Transfer Coefficient Module, UCORL2

This module calculates the coefficient for heat transfer from the outer node of the pin, across the gap, and to the node in the cladding/moderator. The thermal resistances for each of the three regions are calculated from the regional dimensions and thermal properties provided by the core channel module. The UCORL2 module tests for a zero gap thickness, and if present, sets the gap resistance to zero. The basic equation in the module is:

$$U = 1.0 \Big/ \left[\frac{r_f}{8k_f} + \frac{1.0}{h_g} + \frac{r_g}{k_l}\ln\left(\frac{r_g + r_l}{2r_g}\right)\right] . \quad (3)$$

C. Reactivity Feedback Module, RFDBK2

The reactivity feedback module calculates the individual reactivity feedback effects and their algebraic sum for the steady-state and dynamic conditions in the reactor. The feedback reactivity is evaluated for seven effects: fuel expansion, Doppler, sodium density, steel density, assembly bowing, grid plate expansion and CRDL expansion. The fuel expansion and Doppler effects are calculated only for the fissile region of the driver channel. The sodium density effect is obtained by summing the contribution from all axial nodes of the driver and reflector channels. The stainless steel density effect is calculated for the full length of the reflector channel only. For each of these four effects a weighting factor is used in summing the contributions from the axial nodes. The general form of the equations for all these effects is equation 4.

$$\Delta K = \sum_{i=1}^{nodes} f_i R_i (T_i - T_o) \quad (4)$$

The calculation of the assembly bowing reactivity effect is based on the effective rise in the coolant temperature above its inlet value. The bowing reactivity effect is obtained from a user-supplied lookup table, where Y (equation 5) is the independent variable. The relative contribution (F_z) by different types of core channels is specified by the user.

$$Y = \left[\frac{P_o/P_{fp}}{W_o/W_{fp}}\right]\sum_{z}^{zones}\frac{F_z}{L_z}\left\{\sum_{i}^{nodes}\left[(T_i - T_n)/(T_i - T_n)_o\right]x_i\right\} \quad (5)$$

The grid plate reactivity expansion effect is calculated from the thermal expansion of the inner eleven rows in the grid plate (Row 1 is not included). The equation for grid plate expansion includes a weighting factor by row. The effect is obtained by equation 6:

$$\Delta K = R_{grid}\,\alpha \sum_{y=1}^{10} f_y r_y \left(T_{ny} - T_o\right) \quad (6)$$

The reactivity (ΔK) is delayed using a second order lag function to account for heat transfer between flowing sodium and the grid plate.

The control rods are suspended from the tank cover by CRDLs which includes the control-rod subassembly duct and the core is supported from the bottom of the tank. Thermal expansion of the CRDLs and the core support structures drives the rods farther into the core and produces a negative reactivity effect. The thermal expansion of the tank wall tends to drive the rods out of the core. The reactivity effect due to the expansion of these components is calculated from equation 7:

$$\Delta K = R\left\{\begin{matrix}[\alpha x(T_c - T_o)]_{CRDL} + [\alpha x_q(T_q - T_o)]_{Core\ Support} \\ - [\alpha x_t(T_t - T_o)]_{Tank\ Wall}\end{matrix}\right\}. \qquad (7)$$

D. Fission Product Decay Rate Modules, DFFUN1 and DSFUN1

These two modules calculate the relative decay heating rates of fission product decay power in fuel and decay power in stainless steel at any time after transient initiation as a function of power history and the current power level. The DFFUN1 module contains a correlation for the decay power deposition in fuel, and the DSFUN1 module has the correlation for deposition in stainless steel. The decay heating rates are given as a fraction of the total power, and the basic equation for both types of decay power deposition has the following form:

$$P = P_o[At^a - Bt^b]. \qquad (8)$$

The values of the four time-dependent coefficients are obtained by interpolation of user-supplied data.

E. Decay Power Module, DHEAT1

The decay heat module calculates the total decay power deposited in the fuel and steel for use in the thermal power module. The module calculates various initial power terms and also uses the decay rate correlations in DFFUN1 and DSFUN1 to calculate the current decay power from each of the past operating cycles and from the power operation since the beginning of the simulation. The user supplies up to 50 power-duration cycles over the past operating history. In the steady-state portion of the modules the decay powers in the fuel and steel are calculated for each cycle and then summed for all cycles to give the values at the beginning of the simulation. In the dynamic portion, the decay power for each cycle is up-dated with the current values of total reactor power and time since the start of the simulation. This type of calculation is done in the driver, reflector and blanket zones.

F. Point Kinetics Module, NEUTN1

The point kinetics module calculates the relative fission power with the model selected from three choices. Two options use a space-independent kinetics model having up to six delayed-neutron groups. One solves the neutron population and precursor concentration equations exactly and the other uses the prompt-jump approximation. The third one produces the relative fission power by interpolation of a power history table supplied by the user. The equations solved in the option are the standard point kinetics equations, as shown below in equations 9 and 10:

$$\frac{dN(t)}{dt} = \frac{(\Delta K/K - \beta)}{l} N(t) + \sum_{i=1}^{6} C_i \lambda_i(t) \qquad (9)$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{l} N(t) - \lambda_i C_i(t) \qquad (10)$$

G. Power Distribution Modules, PDIST2 and TPOWR2

These modules sum the fission and the two decay powers from NEUTN1 and DHEAT1, respectively, assign fractions of these powers to the three core channels and the material regions within a channel, and then distribute those fractions to the axial nodes in the channels. The PDIST2 module writes the FORTRAN statements which assign the fraction of the total power to the pin, cladding/moderator and coolant regions and then distributes the power axially to the nodes in the regions of each channel. The identity of the three types of powers (i.e., fission, decay in fuel and decay in steel) is maintained, so they may be assigned different axial distributions. In addition to the calculations directed by the PDIST2 module, the TPOWR2 module allows options for determining the decay power fraction (of the total power) for each channel, sums the three types of powers in the channels and calculates the total power generated in the reactor.

H. Upper Plenum Module, UPPLE1

The UPPLE1 module represents the upper plenum as two perfect mixing volumes in series, one volume bounded by the subassembly exits and the cover, and a second volume bounded by the cover and the vessel wall plus plenum outlet. The coolant leakage between the plenum and the tank is modeled by a branch in the flow from the first to the second volume. The flow and mixed-mean temperature entering the first volume are calculated at the connection between the subassembly exits and the upper plenum inlet. The fluid temperature in the first volume is obtained from an energy balance containing terms for the coolant inflow and outflow and the heat loss to the cover. The temperatures of the cover nodes are calculated from energy balances at each of the three nodes, using the coolant and tank sodium temperatures as boundary conditions. The perfect mixing assumption means that the exit temperature is the same as the mixed-mean temperature in the volume. The inlet temperature and flow for the second volume are calculated from mass and energy balances at the connection of the two volumes. The temperatures in the second volume and the wall are calculated in the same manner as those in the first volume. The basic equations (11 and 12) for obtaining the temperatures in the upper plenum second volume are given below:

$$V\rho C_p \frac{dT_c}{dt} = WC_p[T_c - T_n] - UA[T_c - T_w(I)] \qquad (11)$$

$$MC_p \frac{dT_w(J)}{dt} = UA(J)[T_w(J-1) - T_w(J)] - UA(J+1)[T_w(J) - T_w(J+1)] \qquad (12)$$

for J=1, $T_w(J-1) = T_c$

J = 3, $Tw(J+1) = Tt$.

I. IHX Baffle Region Module, DPLEN1

The baffle/plenum region at the entrance to the shell-side of the IHX is modeled as two perfect-mixing plena in series. The two plena are assumed to be adiabatic and equal in volume. The basic equation used to calculate the coolant temperatures in each of the plena is:

$$\rho V\frac{dT_c}{dt} = 2W[T_n - T_c] \qquad (13)$$

J. Intermediate Heat Exchanger (IHX) Module, GIHXR1

This module simulates the heat transfer process in a counterflow, tube-and-sheet heat exchanger. GIHXR1 models the heat transfer in a single channel of the heat exchanger which consists of a tube with fluids on both sides. The user specifies the physical dimensions of the heat exchanger and materials of the wall and fluids. The module selects the appropriate heat transfer correlations for the assumed condition of single phase fluids in the exchanger. The primary and secondary time-dependent inlet temperatures and flow rates are the driving functions for the module. The primary and secondary outlet temperatures, in addition to the internal nodal temperatures, are the response variables calculated in the module. The module has several special features. One is a flow dependent weighting factor in the differential equations describing the heat transfer process to accelerate the solution. A second feature is an additional node at each end of the IHX which aids in calculating the relatively large temperature changes at the ends of the heat exchanger that occur under off-normal flow conditions. The temperatures of the primary sodium, secondary sodium and tube wall are obtained from energy balances at the nodes. The equations for the three temperatures are given below in equations 14, 15, and 16.

$$V\rho C_p\frac{d\overline{T_p}(I)}{dt} = W_pC_p[T_p(I) - T_p(I-1)] - UA[\overline{T_p}(I) - \overline{T_w}(I)] \qquad (14)$$

$$MC_p\frac{d\overline{T_w}(I)}{dt} = UA[\overline{T_p}(I) - \overline{T_w}(I)] - UA[\overline{T_w}(I) - \overline{T_s}(I)] \qquad (15)$$

$$V\rho C_p\frac{d\overline{T_s}(I)}{dt} = W_sC_p[T_s(I+1) - T_s(I)] + UA[\overline{T_w}(I) - \overline{T_s}(I)] \qquad (16)$$

K. Thermal Mixer Module, MIXER1

The MIXER1 models the energy and mass exchanges between the tank sodium and the primary sodium leaving the IHX, as it flows from the IHX to the suctions of the two main primary pumps. The module assumes that the exchanges take place in a perfect-mixing plena with two entering and two exiting streams and that the ratio of the flows of each mixer stream to the total flow is a user-specified constant. In the case of EBR-II, the temperature of the IHX primary outlet (T_p) and tank sodium (T_t) usually differ by only a few degrees (in steady state), so the heat capacities of the streams are taken to be equal. The temperatures of the mixer exit streams (T_x), which are equal because of the perfect-mixing assumption, are obtained from an energy balances in the mixers. The temperature of the fluids at the exit of the mixer is given by:

$$\frac{dT_x}{dt} = \frac{W_p}{G}\left\{\left[T_p + \frac{W_t}{W_p}T_t\right] - \left[1 + \frac{W_t}{W_p}\right]T_x\right\}. \qquad (17)$$

Two mixer modules are used in EBR-II simulations, in between the IHX exit and each of the pump suctions. The fraction of the total sodium entering a mixer and the ratio of the primary-to-tank sodium flows may be different for each mixer, because the physical flow paths to the two pump suctions are different.

L. Reactor Tank Module, LXPLE1

The LXPLE1 module, which is used to simulate the bulk sodium region of the EBR-II reactor tank as a perfect-mixing plenum, allows any number of inlet streams, outlet streams, heat sink and heat sources to be specified. The module calculates the temperature of the tank sodium from an energy balance that takes into account the specified streams, sinks and sources. The module also models a submerged stainless steel mass and a stagnant sodium volume with which the tank sodium can exchange energy with the active sodium. The basic equation used to determine the temperature of the tank sodium is:

$$V\rho C_p \frac{dT_t}{dt} = \sum^{In} W_n C_p T_n - \sum^{Out} W_x C_p T_x - UA(T_t - T_m) - UA(T_t - T_q) + \Sigma Q_{in} - \Sigma Q_{out} \quad . \quad (18)$$

For EBR-II simulations the tank volume has three inlet streams (upper plenum leakage, exit streams from the two mixers), two outlet streams (entering the two pumps), four sources (heat gains from the upper plenum, Z-pipe, pumps and tank heaters) and five sinks (heat losses to the IHX, shutdown coolers, purification system, shield and instrumentation cooling systems, and tank wall).

V. COMPARISON WITH NATDEMO CALCULATIONS

Each new module was first tested by running it individually (i.e. without other component modules connected). Input data typical of EBR-II were provided as boundary conditions, and an arbitrary forcing function was specified. Atypical results from these calculations led to the correction of most programming and modeling errors. Once the module produced results in the range of the expected values, it was tested further by comparing the results to those obtained from a prior NATDEMO calculation with identical boundary and initial conditions. The comparison identified any remaining differences between the NATDEMO and DSNP-ND codes in solving the basic models and led to a correction of a few additional errors in the modules. An example comparison is shown in Fig. 2, which presents the steady-state axial temperature distributions in the driver fuel obtained from the FUELR2 module and the NATDEMO program. The plots show that the temperature profiles are in good agreement.

Next the fourteen modules were combined to simulate two portions of the EBR-II primary system. The first ten modules listed above simulated the reactor and upper plenum, and the last four modules describe the components between the exit of Z-pipe and the inlet to the pumps. Any interfacing problems among the new modules were resolved during these test runs. The results from these test runs were also compared to those from a NATDEMO calculation for the same transient conditions.

The modules were then tested in a simulation of the full EBR-II primary system. The DSNP problem was set up to simulate the Shutdown Heat Removal Test[6] 45 (SHRT 45), a loss-of-flow test without scram conducted at EBR-II in 1986. A transient similar to SHRT 45 but with the auxiliary pumps off was simulated by DSNP-ND and also by the NATDEMO.[7] The values for the normalized power and power-to-flow were selected from the DSNP-ND and NATDEMO calculations and plotted in

figures 3 and 4. Good agreement in the calculation of reactor power between the DSNP-ND and NATDEMO is shown in Fig. 3. The shapes of the two power/flow curves (Fig. 4) are about the same from to 60 seconds, but they diverge after 60 seconds. A number of factors may contribute to the discrepancy in the curves. Different material property routines were used in DSNP and NATDEMO calculations, and this difference was identified in earlier comparisons as being significant. Another contributor is that the hydraulic modeling used in each code is different, and differences can be expected in the calculation of flow at the lower flow rates.

VI. SUMMARY

The models from the NATDEMO program which represent the thermal hydraulic response of EBR-II's primary system components were translated into the DSNP language and placed in the library of the DSNP-ND version of the software package. The new DSNP modules were tested at the individual and system level by comparing their results with those from a NATDEMO calculation with the same boundary conditions. The good comparison between the results shows that the models were correctly translated from the NATDEMO program to the DSNP-ND library.

VII. NOMENCLATURE

- A - Time-dependent coefficient in the decay correlation.
- a - Time-dependent exponent in the decay correlation.
- B - Time-dependent coefficient in the decay correlation.
- b - Time-dependent exponent in the decay correlation.
- C - Delayed neutron precursor concentration, atoms/cm^3.
- Cp - Specific heat, in J/kg/°C.
- F - Fractional core zone contributions.
- f - Weighting factor.
- G - Mass of mixer, in kg.
- H - Enthalpy, in J/kg.
- h - Heat transfer coefficient, watts/m^2/°C.
- k - Thermal conductivity, in watts/m/C.
- L - Total length, in m.
- l - Prompt neutron lifetime, second.
- M - Mass, in kg.
- N - Neutron density, in n/cm^3.
- P - Thermal power, in watts.
- Q - Power source, in watts.
- R - Reactivity coefficient, in ΔK/K/°C or ΔK/K/m.
- r - Radius, in m.
- T - Temperature, in °C.
- t - Time, in seconds.
- U - Overall heat transfer coefficient, in watts/m^2/°C.
- UA - Heat transfer rate, in watts/°C.
- V - Volume, in m^3.
- W - Mass flow rate, in kg/s.
- x - Length, in m.
- Y - Factor for bowing reactivity effect.
- ΔK - Reactivity effect.
- α - Thermal expansion coefficient, 1/°C.
- β - Delayed neutron fraction.
- ρ - Density, in kg/m^3.
- λ - Decay constant, 1/s.
- τ - Time delay.

Sub-scripts

- c - Coolant

d - Duct
f - Fuel
fp - Defines 100% power or flow condition.
g - Gap between fuel and cladding.
I - Axial index number
i - Axial node number
J - Radial index number
j - Radial node number
l - Cladding
m - Metal
n - Inlet
o - Reference or initial condition
p - Primary
q - Stagnant pool
s - Secondary
t - Tank
w - Wall
x - Exit
y - Row number
z - Zone number

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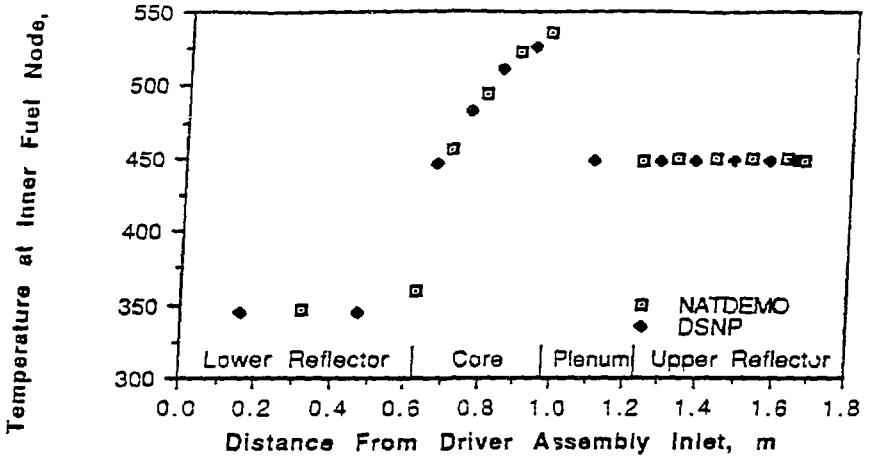

Fig. 2. DSNP-ND and NATDEMO Fuel Temperature Profiles at Inner Node under Steady-State Conditions

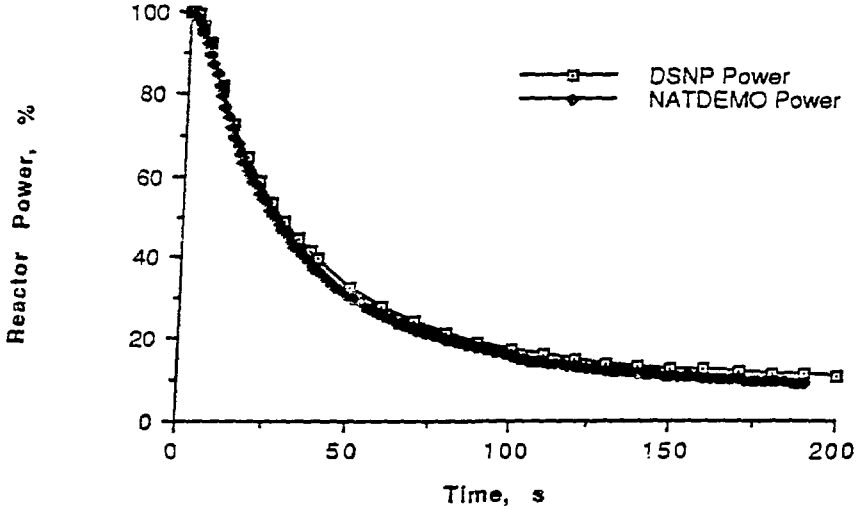

Fig. 3. Comparison of DSNP-ND and NATDEMO Simulated Reactor Power Following Loss of Pumping Power Without Reactor Scram

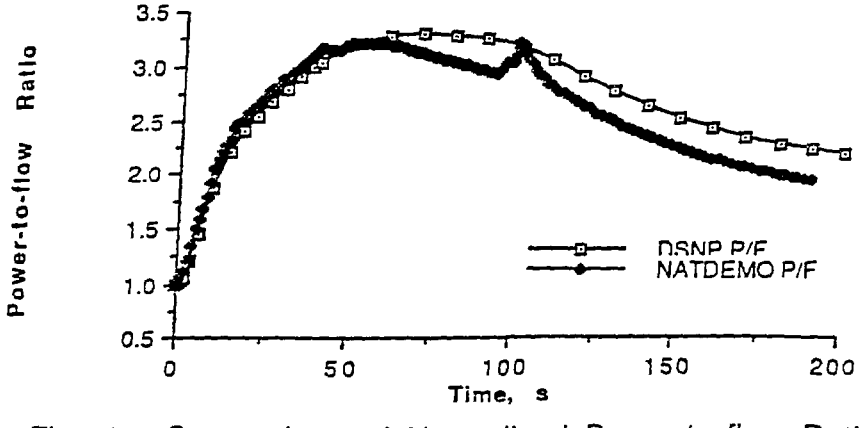

Fig. 4. Comparison of Normalized Power-to-flow Ratios From DSNP-ND and NATDEMO Simulation of SHRT-45