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《Original》

A Point Kinetics Approach to the Analysis of Overpower Transients of the Ko-ri Unit 1 Reactor

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點 近似 動特性 모델을 이용한 고리 원자력 1호기의 過度出力 轉移 해석

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Abstract

The dynamic behavior of the Ko-ri Unit 1 nuclear reactor following some credible and postulated accidents has been analyzed to a certain extent by means of neutronics and temperature equations formulated in terms of point reactor model. In general, the result of numerical calculation is harnessed to be incorporated in more elaborate models so as to predict transient behavior in a reliable mode as a part of accident analysis. It is shown in the case of power response upon an uncontrolled withdrawal of rod cluster control assembly at hot full power that the point reactor kinetics model proves to be good enough to reproduce the generic features described in the final safety analysis report of the Ko-ri Unit 1.

요 약

고리 원자력 1호기에서 일어날 수 있는 가상사고에 의한 동특성 현상이 점근사 원자로 모델에 의한 중성자 및 온도 방정식을 사용하여 해석되었다. 일반적으로 수치해석 결과는 사고해석에 있어서 확실한 동특성 시간전이 현상을 예견하기 위해서는 보다 정밀한 계산모델을 사용해야 된다는 것을 지시한다. 전출력 상태에서 RCCA 인출에 따르는 출력반응의 경우는 점근사 원자로 모델이 고리 1호기의 최종 안정성 분석 보고서의 해석결과와 우수한 일치를 보여줬다.

1. Introduction

One of the most important criteria in an operating nuclear reactor is to ensure the prescribed safety features in a safe integrity so that reactor core may not be overridden by any severe transients even after postulated accidents. It is, therefore, essential to bring forth a reliable and timely prediction in regard to the transient power level and its distribution as a part of accident analysis work. Despite that every possible effort has been made available to solve the space-time neutron distribution problem with respect to different aspects which have become indispensable by the advent of large high-power reactors, the format of most reactor accident analysis results still remains unchanged, i.e. being in the form of point kinetics representation as easily seen in the safety analysis reports.

The fundamental assumption of point kinetics consists in the separability of neutron density into a product composed of constant shape function and time-dependent amplitude function. This assumption holds true so long as the time-dependent behavior of a nuclear reactor is asymptotic. In all other cases, however, neutron spatial distribution is practically distorted due to perturbation. As a result, the computed point kinetics parameters can no longer be exact, which is then susceptible to serious errors for problems with or without feedback.[1]

In this context, three types of accident, namely, uncontrolled withdrawal of rod cluster control assembly (RCCA) at hot zero power (HZP) as well as hot full power (HFP) condition and RCCA ejection accident, in the case of Ko-ri Unit 1[2], have gone through our analysis, bearing in mind the constraints of the conventional point kinetics approach. In connection with temperature effects, one-dimensional heat transfer equations are employed in the single channel with a cylindrical fuel pin surrounded by coolant, under the condition that time delay of trip signal is accounted for. In order to carry out this task, a computer program was developed so as to be applicable to pressurized water reactors in general, making use of the computer code, entitled AIREK-3[3], as a reference program. The numerical algorithm of the fifth order Runge-Kutta method is adopted herein in an attempt to simultaneously solve a set of differential equations.

2. Physical and Mathematical Model

2.1 Reactor Kinetics and Decay Heat

The conventional point kinetics equations are given as follows:

$$\frac{dN(t)}{dt}=\frac{\beta_{eff}}{\Lambda}\left\{[r(t)-1]N(t)+\sum_{j=1}^{J}f_jW_j(t)+S\right)\cdots \quad (1)$$

$$\frac{dW_j(t)}{dt}=\lambda_j\{N(t)-W_j(t)\},\ j=1,2,\cdots J \quad \cdots(2)$$

where

$N(t)$ =neutron density,

$r(t)=\frac{\rho}{\beta_{eff}}$,

$W_j(t)=\frac{\Lambda\lambda_jC_j(t)}{\beta_{eff}f_j}$,

$\beta_{eff}=I\cdot\beta$,

$S=\frac{\Lambda S_o}{\beta_{eff}}$,

$f_j=\frac{\beta_{effj}}{\beta_{eff}}$,

Λ=neutron generation time,

J=total number of delayed neutron precursor groups, and ρ, S_o, β_{eff}, λ_j, C_j and I are reactivity, external source, effective delayed neutron fraction, precursor decay constant and precursor concentration for delayed group j, and delayed neutron importance factor, respectively. The instantaneous power, Q(t), is calculated by the sum of contributions from instantaneous fission and from decay of eleven fission products. Taking the neutron density, N(t), as the fraction of initial total power, the instantaneous power, Q(t), is expressed as follow:

$$Q(t)=\{1-P(o)\}N(t)+\sum_{j=1}^{11}p_j(t) \quad \cdots\cdots(3)$$

where $P(o)=\sum_{j=1}^{11}G_j$,

and $\frac{dp_j(t)}{dt}=\lambda_j^*\{N(t)G_j-p_j(t)\},\ j=1,2,\cdots 11$ ⋯(4)

$P(o)$ is the fraction of decay power for total power at time zero, and $p_j(t)$ is decay power for decay heat group j and is expressed as fraction of total power. Yield fraction (G_j) and decay constant (λ_j^*) of the eleven group decay heat source are quoted from the reference[4]. Values used for the

Tablel 1. Delayed Neutron Data for the First Cycle Core

Delayed group	BOL	
	β_j	λ_j (sec^{-1})
1	0.000217	0.0125
2	0.001463	0.0308
3	0.001352	0.1155
4	0.002825	0.3116
5	0.000963	1.2479
6	0.000323	3.3531
Total delayed neutron fraction, β	0.007143	
Prompt neutron lifetime (μ sec), l	18.44	
Importance factor, I	0.970	

delayed neutron precursors are listed in Table 1, and the conservative value of the total yield fraction β is also employed in this study.

2.2 Feedback Equations

The transient response of a reactor system strongly depends on feedback effects resulted from changes in the moderator and fuel temperature. The heat conduction equation for a cylindrical fuel pin with a uniform volumetric heat source, $q'''(t)$, in the radial direction takes the form:

$$\text{Fuel: } \frac{\partial T_F}{\partial t}=\lambda_F\frac{\partial^2 T_F}{\partial r^2}+\frac{\lambda_F}{r}\frac{\partial T_F}{\partial r}+\frac{\lambda_F q'''(t)}{k_F} \quad \cdots(5)$$

$$\text{Cladding: } \frac{\partial T_C}{\partial t}=\lambda_C\frac{\partial^2 T_C}{\partial r^2}+\frac{\lambda_C}{r}\frac{\partial T_C}{\partial r} \quad (6)$$

where λ is thermal diffusibility, $\frac{k}{\rho c_p}$

In order to solve the eqs. (5) and (6), a fuel pin is divided into 8 rings of equal thickness, and another region is assumed for the cladding as shown in Fig. 1. Assuming that the temperature distribution in each ring is parabolic and remains so during transients and that there is no heat

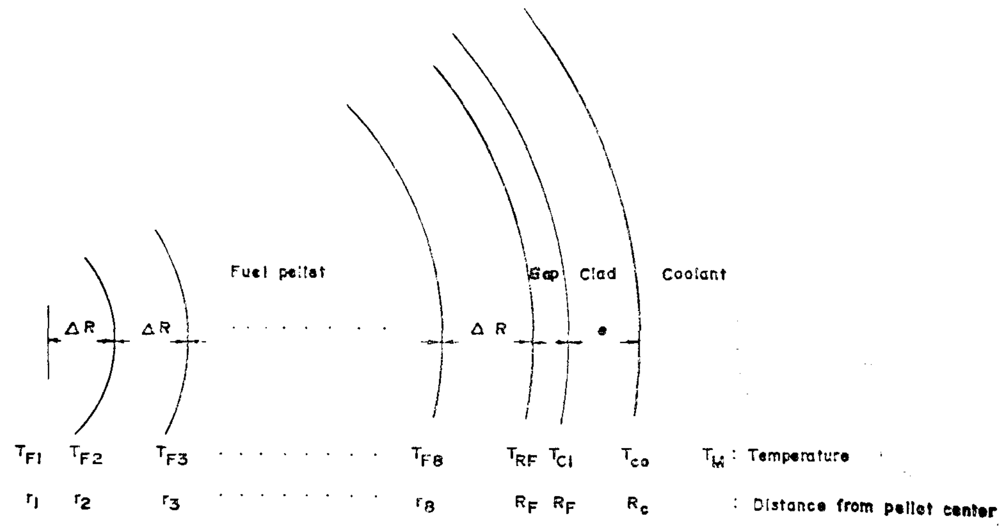

Fig. 1. Schematic of Subdivision of Fuel and Cladding for Finite Difference Equation

generation in the cladding, gas gap and coolant, and no heat is stored in the gas gap, then the heat transfer through the gas gap, fuel cladding and coolant film can be expressed as the finite difference equations for the spatial derivative[5]. Volumetric average fuel temperature is expressed as,

$$T_{Favg}=\frac{1}{192}\sum_{i=1}^{8}\{(3i-2)T_{Fi}+(3i-1)T_{Fi+1}\}\cdots \quad (7)$$

The expression for the average bulk coolant temperature ($\overline{T}_M$) with a uniform axial power distribution is given by the following model:

$$\rho_M C_{PM} A_M \frac{d\overline{T}_M}{dt}=2\pi R_C h(T_{co}-\overline{T}_M)+\frac{2\dot{m}C_{PM}}{L}(T_{in}-\overline{T}_M) \quad (8)$$

where

$$\overline{T}_M=(T_{out}-T_{in})/2,$$

ρ_M, C_{PM}, $\dot{m}$, h, A_M, L, T_{in} and T_{out} are coolant density, specific heat, mass flow rate, heat transfer coefficient, cross-sectional area of channel, channel length, inlet and outlet coolant temperature, respectively. The heat transfer coefficient, h, is obtained from the Dittus-Boelter correlation which reads:

$$h=\frac{ck_M}{D_e}R_e^{0.8}P_r^{0.4} \quad (9)$$

where c is the Colburn correlation factor for fluid flow parrallel to tube bundles, which yields 0.03134 for the Ko-ri reactor. The heat transfer coefficient for the gap is adjusted such that the steady state temperature agrees with that predicted by the computer program.

2.3 Trip Function

The abnormal behavior of temperature or power is sensed by some instrument, which has its own response to signal. This signal is finally relayed to a servomechanism which in turn initiates tripping.

Let $F_K(t)$ represent the response function of a certain instrument K. Then the desired equation of the instrument response to the power can be represented as follow:

$$\frac{dF_K(t)}{dt}=\frac{1}{\tau}\{N(t)-F_K(t)\}\cdots \quad (10)$$

where τ is response time of instrument K. The rapid shutdown is to be provided by the free-fall insertion of full length RCCA. The trip function used in this work is the power range high neutron flux trip[2], which is two independent trip settings: a high setting for the protection during the normal operation and a low setting for the protection during the start-up. In this analysis, the high setting is set at 118% of nominal power, and the time delay of the instrument response is 0.5 second. The low setting is set at 35% of nominal power, and the time delay is 0.5 second.

2.4 Reactivity

In control rod ejection and uncontrolled withdrawal of control rods, a ramp input equivalent to the malfunctioned rod worth is added to the reactor. When the reactor power reaches the trip set-point, the negative shutdown reactivity is added to the reactor by a tabulated function. The negative reactivity insertion characteristics is shown as a function of time in the FSAR. The values of the temperature coefficient of reactivity for fuel (α_F) and moderator (α_M) are obtained from the FSAR and the reference[6].

The reactivity prior to the reactor trip is represented by

$$r(t)=r_O(t)+\alpha_M\Delta T_M+\alpha_F\Delta T_F;\ t\leq(t_S+\tau) \quad (11)$$

and the reactivity following the trip is given by

$$r(t)=r_O(t)+\alpha_M\Delta T_M+\alpha_F\Delta T_F+r_S(t);\ t>(t_S+\tau) \quad (12)$$

where ΔT_M and ΔT_F are the temperature changes of the moderator and fuel from their steady state values, respectively, t_S is time at which the trip signal is sensed, and $r_O(t)$ and $r_S(t)$ are the perturbed and trip reactivity, respectively.

3. Accident Analysis and Discussions

3.1 Uncontrolled RCCA Withdrawal from a Subcritical Condition

The RCCA withdrawal accident is defined as an uncontrolled addition of reactivity to the core caused by withdrawal of RCCA resulting in a power excursion. Should a continuous RCCA withdrawal accident occur, the transient will be terminated by the reactor trip system. A conservative value of the Doppler coefficient (the lowest absolute magnitude: $\alpha_F=1.4\times10^{-5}\Delta k/^\circ F$)[2] is chosen in this analysis. The contribution of the moderator coefficient is negligible during the initial phase of the transient, since the time required for heat transfer from fuel to moderator is much longer than that of the neutron flux response. However, after the accident is terminated by the trip, a conservative value ($\alpha_M=-1.9\times10^{-5}\Delta k/^\circ F$)[3] is used in the analysis. The reactor is assumed to be at the hot zero power (10^{-13} of the nominal power) of the first cycle. The reactor trip is initiated by the power range high neutron flux (low setting), and the reactivity insertion rate describing the accident is assumed to be a ramp insertion of 7.5×10^{-4} Δk/sec. The initial effective multiplication factor is assumed to be 1.0.

The results are compared with the FSAR data in Figs. 2 and 3. The overall discrepancies observed in these figures are mainly due to the inherent limitations of the point kinetics model, especially at zero power[7-8], as well as the uncertainties of the parameters used in the calculation and of feedback effects from the single channel analysis. In Fig. 2 the sudden drop of the power after the peak is attributed to large reactivity feedback following the high temperature rise (See Fig. 3). The power behavior after that is characterized by the feedback and shutdown reactivity effects. When the values of neutron generation time and delayed neutron fraction are reduced, the present result following the perturbed reactivity insertion falls in with the range of the FSAR curve. This is partially by the delayed neutron holdback effect[8-9] in a large reactor. Analysis of this accident in case of the Ko-ri reactor is necessary to use more reliable model. The neutron flux overshoots the full power nominal value, but this ocurs only for a very short period of time. Hence, the subsequent fuel temperature increases are relatively small.

3.2 Uncontrolled RCCA Withdrawal at Power

The initial condition of maximum core power (102% of the nominal) at BOL is assumed, and a zero moderator coefficient and a conservatively small negative Doppler coefficient ($\alpha_F=-1.3\times10^{-5}$ $\Delta k/^\circ F$) are used. The effective total delayed neutron fraction adopted herein is 6.9287×10^{-3}. In Fig. 4 the response of neutron power to RCCA withdrawal incident with the maximum reactivity insertion rate (7.5×10^{-4} Δk/sec) is shown. The calculated time of reaching the high neutron flux trip set point is 1.25 seconds, whereas that described in the FSAR is 1.29 seconds. This discrepancy is derived from the differences in physical and mathematical modeling and in parameters used. The single channel analysis tends to oversimplify the heat transfer from the fuel to the coolant.

3.3 RCCA Ejection

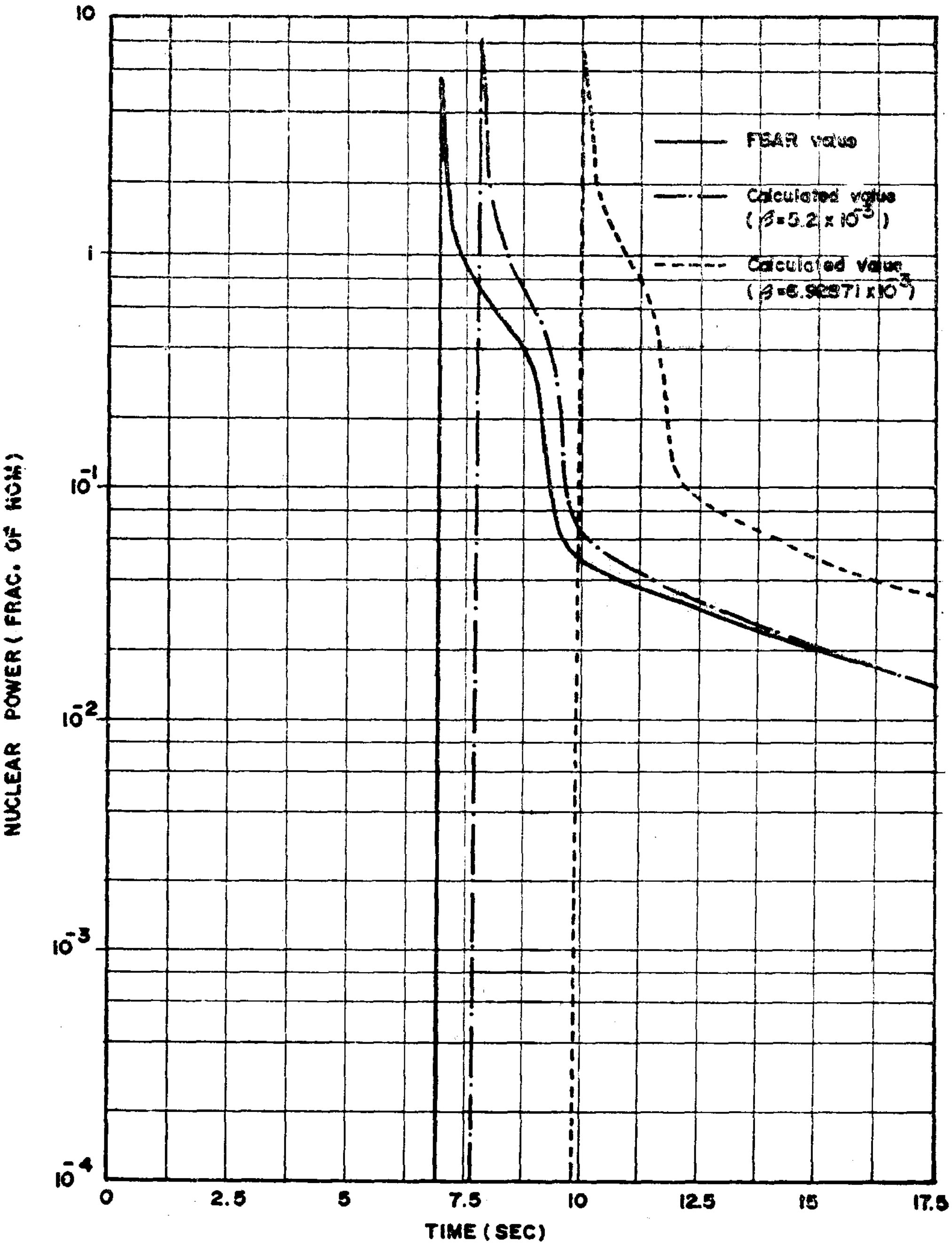

Fig. 2. RCCA Withdrawal from a Subcritical Condition Terminated by High Neutron Flux Trip (Low setting) (Reactivity insertion rate: $7.5\times10^{-4}\Delta$ k/sec)

The rod ejection accident occurs when the control rod drive mechanism housing fails and the reactor coolant pressure in the core drives RCCA to its fully withdrawn position, thus causing a large and rapid positive reactivity insertion into the core.

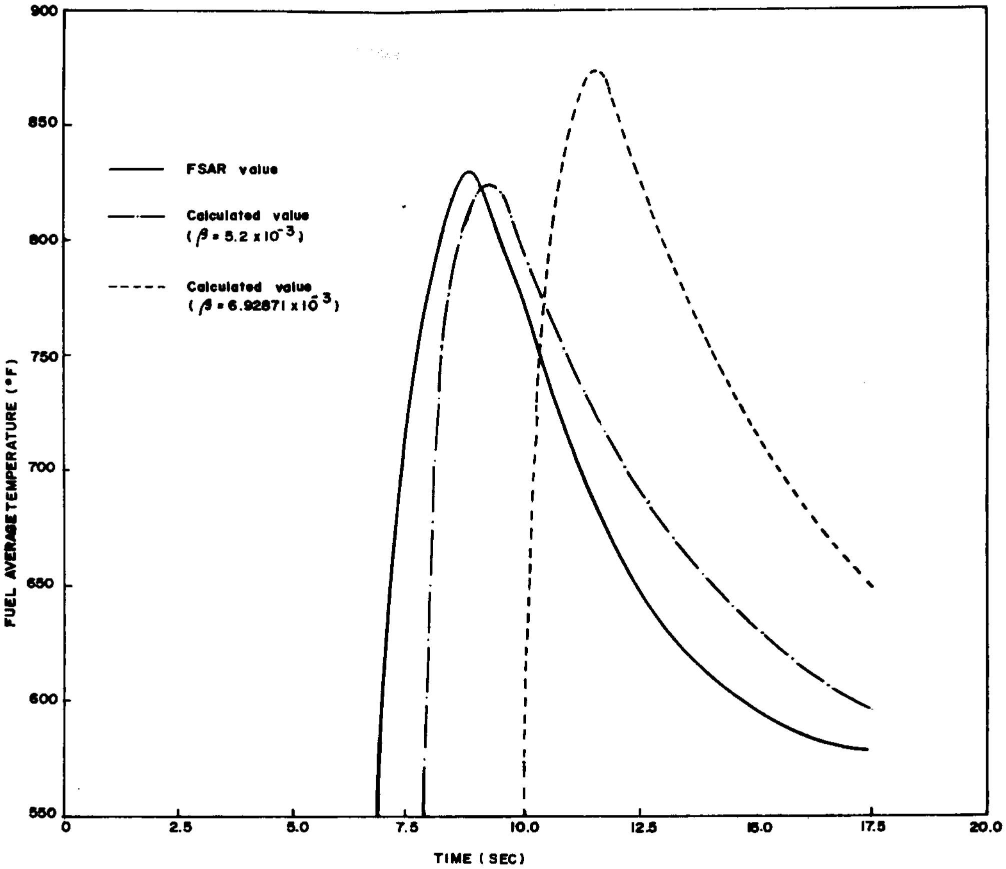

Fig. 3. RCCA Withdrawal from a Subcritical Condition Terminated by High Neutron Flux Trip (Low setting) (Reactivity insertion rate: $7.5\times10^{-4}\Delta$ k/sec)

The rod ejection accident can best be analyzed by a three-dimensional space-time kinetics model due to the highly localized reactivity effects.

Since it is assumed that the radial temperature profile over the core is flat in the single channel model, a weighting factor is applied to the fuel temperature feedback to account for the appropriate flux shape[10]. However, the axial weighting and the weighting for the moderator temperature feedback are not considered in this analysis. The conservative estimates of β_{eff} as presented in Table 2 are used in the analysis, and the ejected rod worths are assumed to be inserted linearly during 0.1 second.

Table 2. Parameters Used in the Analysis of the RCCA Ejection Accident

(1st cycle core)

Time in life	BOL	
Power level (Fraction of the nominal)	HFP (1.02)	HZP (10^{-13})
Ejected rod worth (Δk)	0.2×10^{-2}	0.83×10^{-2}
Delayed neutron fraction	5.2×10^{-3}	5.2×10^{-3}
Trip reactivity (Δk)	4.0×10^{-2}	3.0×10^{-2}
Feedback reactivity weighting	1.3	2.0

The numerical result for the accident at

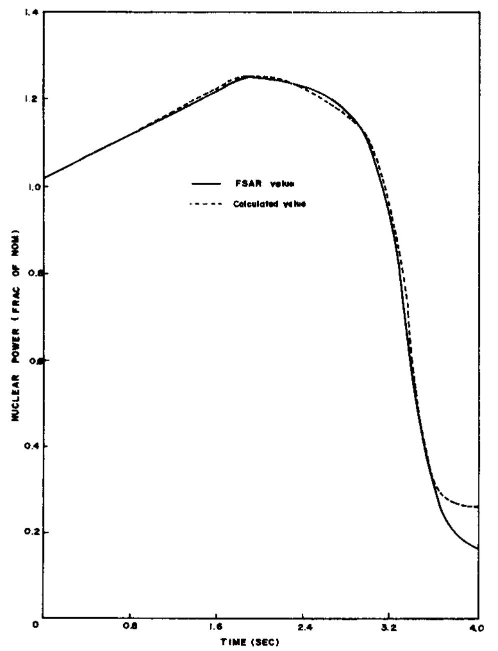

Fig. 4. RCCA Withdrawal from Full Power Terminated by High Neutron Flux Trip (Reactivity insertion rate: $7.5\times10^{-4}\Delta k$/sec)

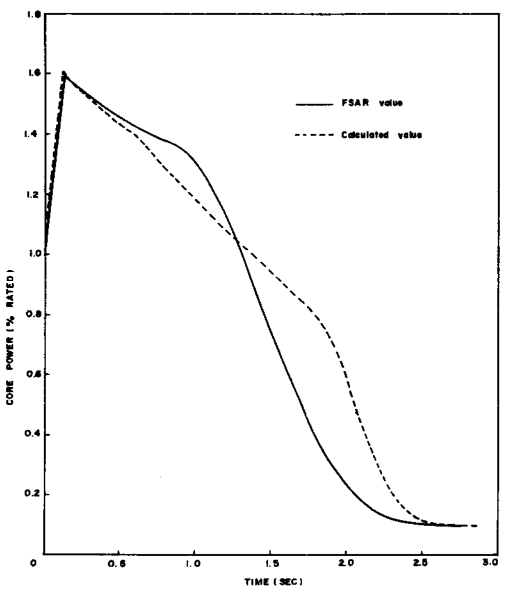

Fig. 5. RCCA Ejection Accident Terminated by High Neutron Flux Trip (BOL HFP)

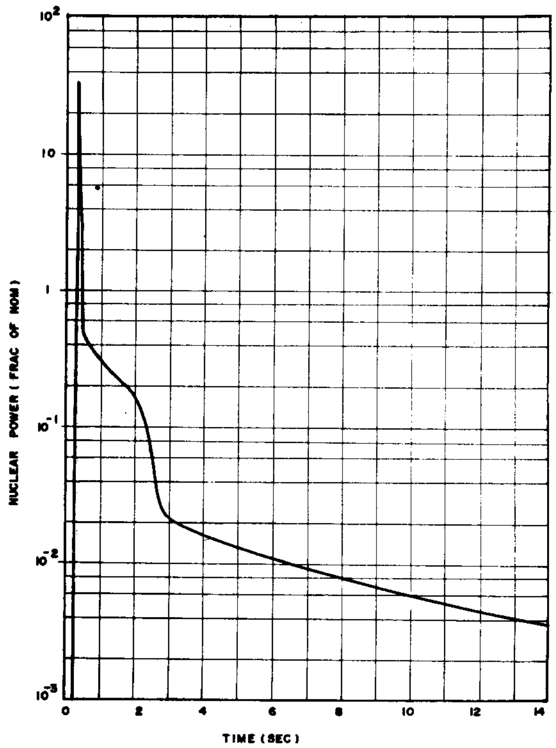

Fig. 6. RCCA Ejection Accident Terminated by High Neutron Flux Trip (BOL HZP)

the hot full power at BOL is presented in Fig. 5. The discrepancy with the FSAR result obtained from one-dimensional (axial) space-time analysis shows the inadequancy of the conventional point kinetics model[10-11] and of the simple feedback model. Because the effect of locally peaked shapes due to the rod ejection is not included in the Doppler reactivity calculation, the resultant Doppler reactivity feedback is considerably underestimated, while the magnitude of the power excursion is overestimated. The numerical result of the power behavior following the accident at the hot zero power is shown in Fig. 6.

4. Summary

The prediction of the single channel

analysis by the point reactor model on the power behavior of the Ko-ri Unit 1 following an uncontrolled RCCA withdrawal at the hot full power is in good agreement with the FSAR data. It is also indicated that the point kinetics approach with simple feedback model is inaccurate for the prediction of the transient behaviors following the RCCA withdrawal at zero power or the RCCA ejection. Generally. the conventional point kinetics is inadequate for handling the delayed neutron holdback effect in a large reactor, the rapid reactivity transient, and the problems associated with local perturbation which induces severe flux distortion. It is necessary to use more elaborate models to cover the full spectrum of transient analysis of the Ko-ri reactor. This work is primarily intended to clarify plausible generic features of physical effects inherent in the aftermath of transients rather than simply attempting to comprehend an accurate temporal behavior or even presupposing an approximate estimate of time-dependent behavior which are usually obtainable from a point kinetics approach when applied to a large high power reactor.

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