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Coupled 3 -D Reactor Kinetics and Thermal-Hydraulics Analysis for SLB Accident of an Operating Nuclear Power Plant by Using the RELAP5/PARCS Code[1]

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ABSTRACT

A study on the 3-dimensional core behaviors for accidents of nuclear power plants and their sensitivity in the reactor parameters is being undertaken. This paper provides a progress result showing the 3-dimensional core behaviors for steam line break transient of a currently operating commercial pressurized water reactor. The coupled RELAP5/PARCS code is used for the analysis. Cross section data for PARCS input are obtained from CASMO. The results are compared with those of RELAP5 for a typical steam line break.

1. INTRODUCTION

As an operating cycle is longer, the discharge burnup and uranium enrichment have gradually increased. The usage of the higher burnup fuel not only alleviated the rate of fuel depletion providing some relief with regard to the demand for spent fuel storage capacity at operating plants, but also improves the economics. However, several experiments on the behaviors of high-burnup fuel have shown that the current acceptance criteria for coolability during accidents may be called into question. Many safety issues have been provided by the fact that the integrity of cladding depends on its mechanical properties and the behavior of the cladding is strongly influenced on burnup (OECD/GD 197, 1995).

Conservative methods have been used for the accident analysis in the past, but with the more stringent fuel acceptance criteria for higher burnup and with more rigorous calculation methods available, it is expected that best-estimate core analysis methods will be used in the future (Diamond, 1998). Also, the concept designs of some small and medium reactors have already used the best-estimate methodologies for core analysis.

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The coupled RELAP5/PARCS code was released by U.S. Nuclear Regulatory Commission (Barber, 1998). This code utilizes an internal integration scheme in which the solution of the system and core thermal-hydraulics is obtained by RELAP5 and only the spatial kinetics solution is obtained by PARCS (Joo, 1998). In this scheme, PARCS utilizes the thermal-hydraulics solution data calculated by RELAP5 to incorporate appropriate feedback effects into the cross sections, and RELAP5 takes the space-dependent powers calculated in PARCS and solves for the heat conduction in the core heat structures.

When the license basis is moved away from the conservative method into best-estimate methods, the best-estimate methods require an uncertainty analysis. The identification of the uncertainty may be done with various sensitivities in the reactor parameters. Diamond (1999) showed the sensitivity studies for the rod ejection accident using PARCS code, with relative change in energy deposition per relative change in key reactor parameters such as the reactivity worth of the ejected control rod, the delayed neutron fraction, the Doppler coefficient, and the specific head of the pellet.

This study clarifies the trend of the 3-dimensional core behaviors for the accidents and their sensitivity on the reactor parameters by using the real scale data of a nuclear power plant (NPP). The objective of this study is to find out the licensing issues in the application of the coupled 3-D best estimate core kinetics and thermal-hydraulics methods to a NPP.

2. INPUT DESCRITION

It has been recognized that the applications of the coupled 3-D neutronics and thermal-hydraulics method for licensees may have many problems to be solved. This study is to justify the actual issues for them by carrying out the analysis of the multi-dimensional core performance for accidents of the operating commercial NPP. Hence, design data of Kori unit 1 NPP, which is currently operating, are chosen for the analysis. The Kori unit 1 has been operating for about 20 years, and it is currently focusing on the reanalysis for life extension. Core parameters are given in Table 1.

Table 1 Design Parameters of Kori 1 NPP

Parameter	Value
Thermal Power	1757 MWt
Core Mass Flow Rate	8365 kg/sec
Number of Fuel Assembly	121
Fuel Rods per Assembly	14×14
T_{in}	282 ^oC
T_{out}	320 ^oC

In PARCS, a set of base macroscopic cross sections generated at a reference thermal-hydraulics condition are assigned to each composition and four sets of derivative cross sections are also provided to describe the boron and thermal-hydraulics feedback effects (Joo, 1998). The core physics design for Cycle 19 EOC of Kori unit 1 is chosen for the analysis. The core is classified into 14 compositions including 3 types of reflector and 11 types of fuel.

These base cross sections and derivative cross sections were obtained from CASMO-3 (Edenius, 1991). CASMO-3 is a multigroup two-dimensional transport theory code for burnup calculations on fuel assemblies. CASMO output is transformed into PARCS cross sections by an interface program.

For PARCS input, the radial core is modeled on assembly-wised nodalization and the axial core is divided into 18. For RELAP5 input, the radial core is also modeled on assembly-wised nodalization, and the axial core is divided into 6. Because the RELAP5 radial core, with the same size as the PARCS, is very fine, it is expected that thermal-hydraulics feedback effects during the transients may be strictly. Nodalizations for Kori unit 1 modeling are given in Figures 1 and 2.

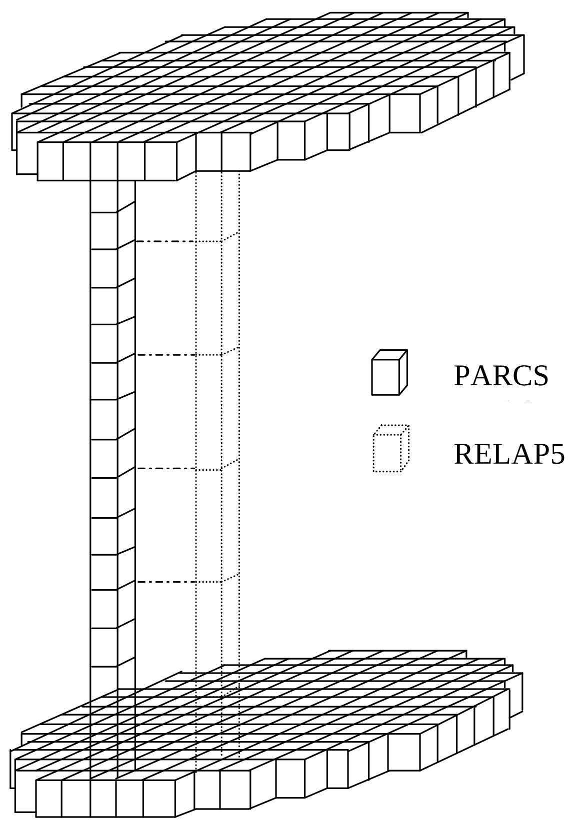

Figure 1 PARCS and RELAP5 Core Nodalization

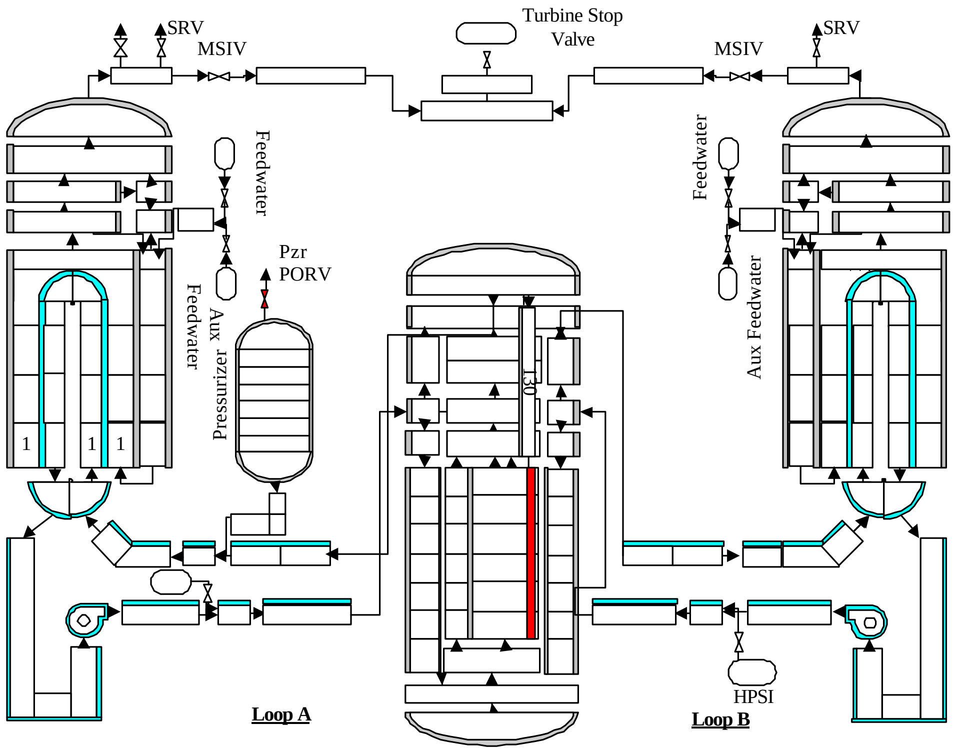

Figure 2 RELAP5 System Nodalization

3. RESULTS

This work presents steam line breaks that represents non-LOCAs. The steam release arising from a steam line break would result in an initial increase in steam flow, which decreases as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. This cooldown results in an insertion of positive reactivity due to a negative moderator feedback. The overpower trip and steam generator low level/pressure trip signals generally occurs in conjunction with receipt of the safety injection signal. It is assumed that the most reactive control rod assembly is stuck in its fully withdrawn position after the reactor trip. There is a possibility that the core will become critical and return to power. A return to power following a steam line break is a potential problem because of the high power peaking factors. The core is ultimately shut down by the boric acid provided by the safety injection.

A spectrum analysis for steam line break sizes and safety injection (SI) setpoint is performed. For small break sizes, the core reaches a new steady-state condition at a higher power equivalent to the increased steam release, without a reactor trip. Reactor trip depends on the SI setpoint of low steam pressure. For example, when this setpoint is set high, for large break a reactor trip is generated as a result of SI actuation. When this setpoint is set low, for large break a reactor trip is generated by high neutron flux trip signal before SI signal.

This work uses the lowest steam pressure SI setpoint for the break size of 2.0 ft^2, which is the smallest break size that results in a trip on high neutron flux trip before SI actuation, and that overpower trip occurs latest. It is assumed that offsite power is available. A realistic moderator temperature feedback curve based on EOC is used for RELAP5 calculations. Table 2 describes the sequence of events from RELAP5/PARCS and only RELAP5.

Table 2 Event Sequence of a Main Steam Line Break

Sequence	RELAP5 (sec)	RELAP5/PARCS (sec)
Break initiation	0.0	0.0
Reactor trip signal (118% FP)	112.9	37.8
SI signal	114.9	56.4
SI Injection	126.9	68.4
Return to power starts	134.0	~130
Maximum return-to-power	226.0	364

Figure 3 shows that the power following a steam line rupture increases due to negative moderator temperature feedback, but the reactor is shut down due to high neutron flux trip signal of 118 %FP.. Figures 4 and 5 are core inlet moderator temperature and steam pressure, respectively. Figure 6 and 7 are radial power and moderator temperature distributions at the 4th axial core, respectively. The results show that high neutron flux trip occurs earlier for RELAP5/PARCS than for RELAP5. This explains that the negative moderator temperature feedback for RELAP5 is stronger than for RELAP5/PARCS.

It is typical that the power increase in 3-D model would be slower than that in point-kinetics model for given temperature feedbacks. However, in this study, the moderator and fuel temperature feedback curves for RELAP5 stand alone calculations, which are from a nuclear design report (NDR), is independently obtained from cross sections of PARCS. Therefore, the power increase rates in RELAP5 and RELAP5/PARCS can not be compared each other. It is recommended that point-kinetics feedback curves from PARCS cross sections would be used for RELAP5 stand alone.

It is evaluated that the groupings of core fuel types, burnup history, cross section generations, and etc. for PARCS calculations take an influence on the results, and PARCS is not converged for some cases.

4. FURTHER STUDIES

This work is a progress result to show the 3-D core behaviors for a certain transient of a currently operating NPP. The final object is to identify the licensing issues in the application of the coupled 3-D best-estimate core kinetics and thermal-hydraulics methods for accidents of a NPP. For this, many kinds of sensitivity analysis will be carried out for key parameters such as burnup, core kinetics constants, feedback parameters, and etc.

Avvakumov (2000) concluded that the hottest fuel rod during the rod ejection accident did not necessarily belong to assembly with peak power, and hence an assembly-by-assembly model may result in underestimation of the local peak enthalpy obtained from a flux reconstruction. PARCS calculates power on the basis of an assembly-average approach, with using the nodal expansion method and the analytic nodal method. Necessity of pin-power calculation for each accident will be examined on following the above analysis.

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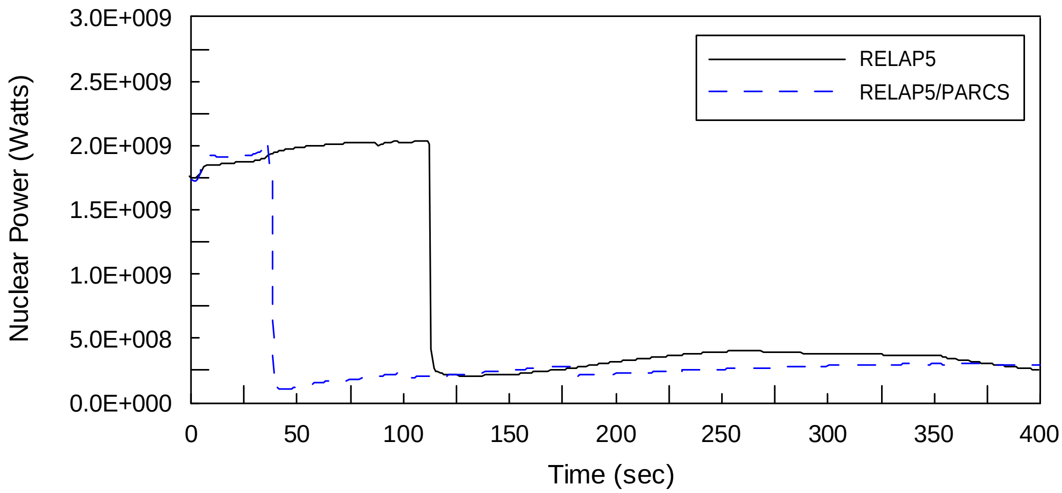

Figure 3 Nuclear Power

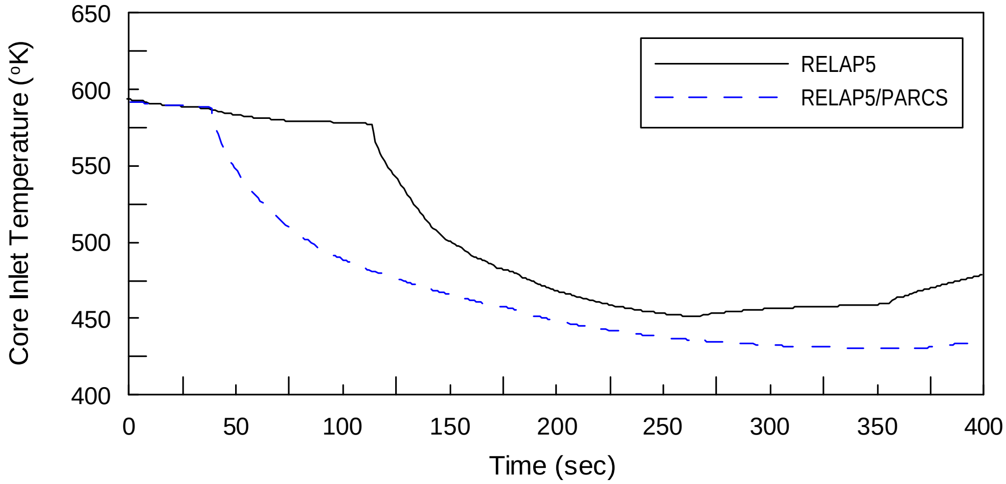

Figure 4 Core Inlet Tempeature in LOOP A

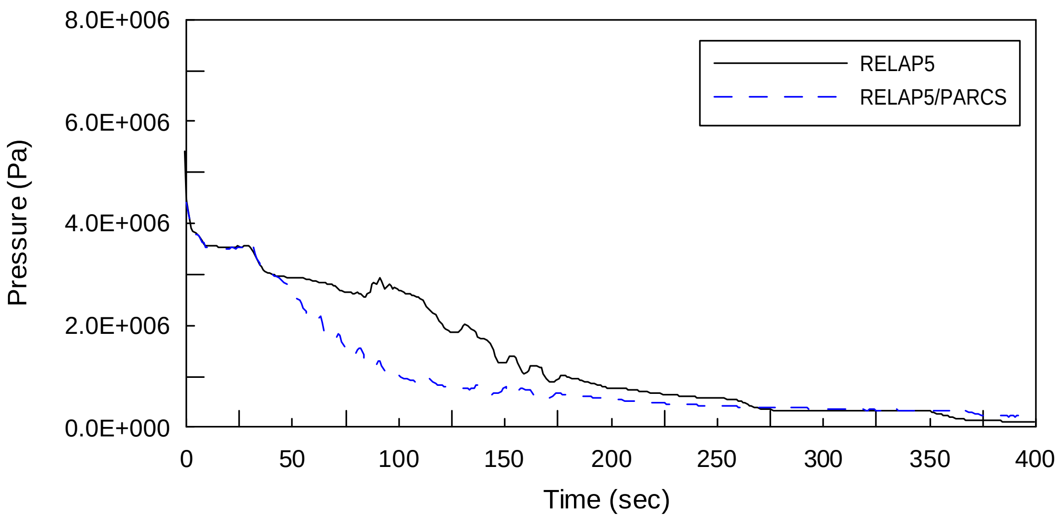

Figure 5 S/G A Pressure

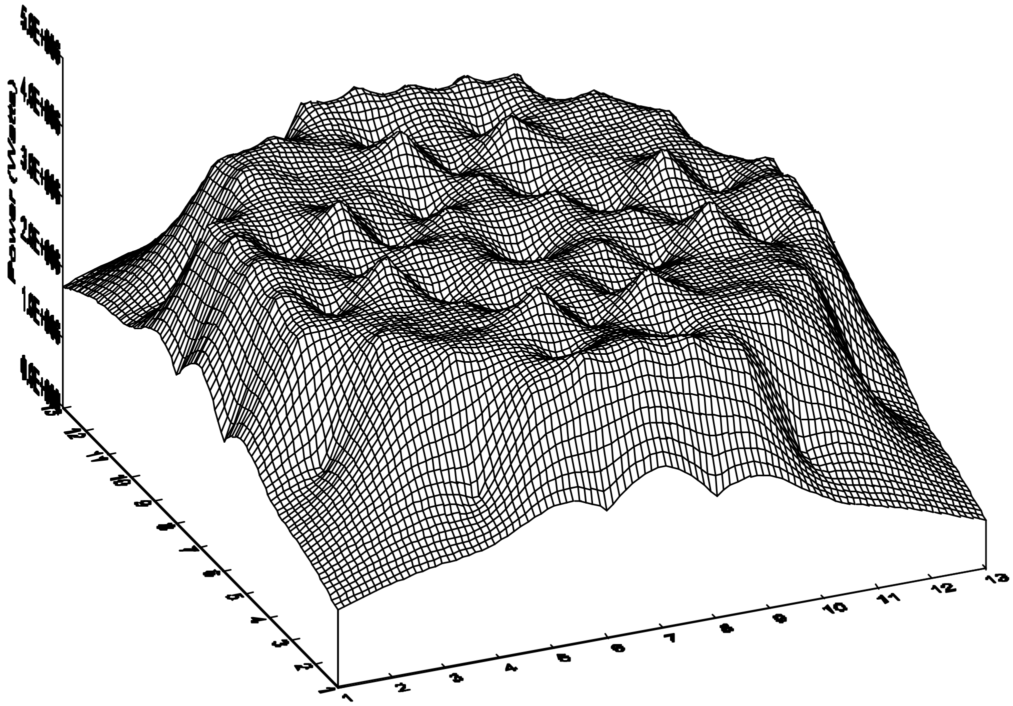

Figure 6 Nodal Power Distribution at 35 sec

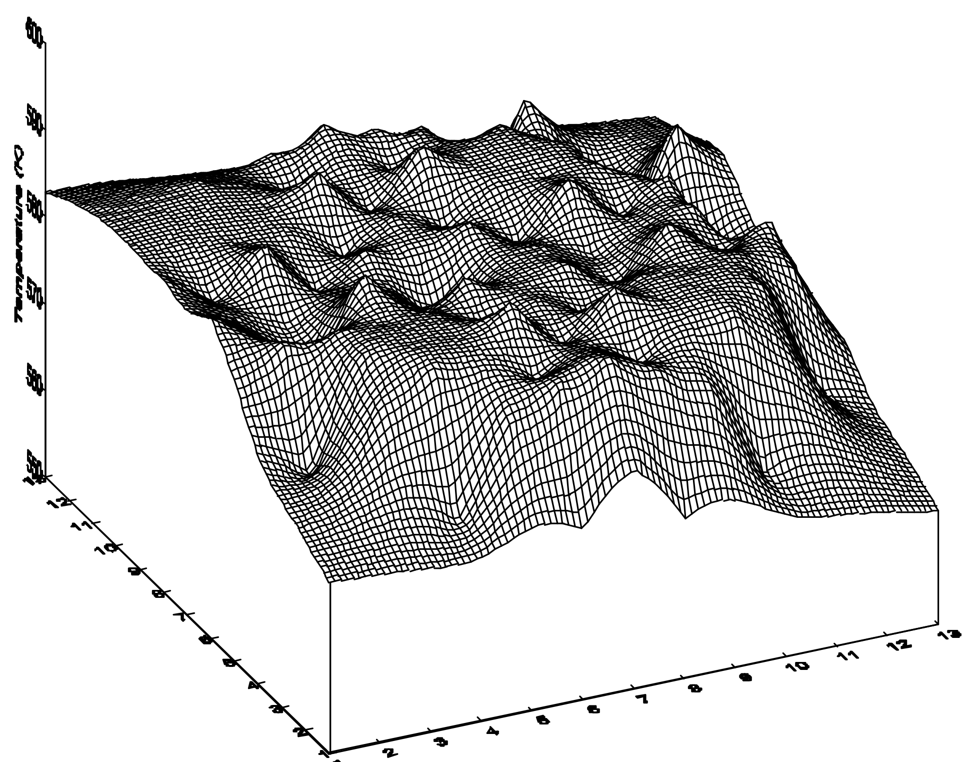

Figure 7 Nodal Temperature Distribution at 35 sec