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Conservatism of the Scram Curve Corresponding to ±0.3 Axial Shape Index in Safety Analysis

Seung Su Baek, Gyu Cheon Lee, Cheol Woo Kim, Chul Jin Choi and Jong Tae Seo

Korea Power Engineering Company, Inc.
150 DukJin-Dong, Yusong-gu
Taejon, Korea

Abstract

The conventional KSNP non-LOCA safety analysis has used the unrealistically over-conservative scram curve corresponding to a ±0.6 pre-trip ASI even though the power level is higher than 20% of design power. This report proposes the use of scram curve based on a ±0.3 ASI which is resonable and still conservative if core power is higher than 20% of design power. During normal operations, Technical Specification requires that ASI be maintained between -0.27 and +0.27 when COLSS is in-service. The axial shape variation from the initial condition to the reactor trip time is negligible for the initiating events which are not caused by CEA motion. Therefore, the use of the scram curve corresponding to a ±0.3 ASI for all non-LOCA events except those caused by a CEA malfunction or misoperation is verified as a valid method.

I. Introduction

The reactor physics design provided ±0.3 ASI for the power level higher than 20% and ±0.6 ASI for the power less than 20%. However, the traditional safety analysis methodology for the thermal-hydraulic analysis with CESEC-III uses an unrealistic and conservative scram rod worth based on the ±0.6 pre-trip ASI for all the non-LOCA event analysis, whereas the Departure from Nucleate Boiling Ratio (DNBR) calculation with the CETOP code uses ±0.3 ASI. This inconsistency provides overly conservative results in the CETOP DNBR calculation due to less shutdown worth after reactor trip for +0.6 ASI as shown in Figure 1.

The Axial Shape Index (ASI) is assumed to be constant during an event if the CESEC-III and CETOP codes are used. However, the value is dependent on the change in the core power, RCS temperature, RCS pressure and core flow rate. The purpose of this analysis is to investigate whether the initial ASI is within the acceptable range during transients.

Steam Line Break (SLB), Increase in Feedwater Flow (IFWF), Inadvertent Opening of SG Atmospheric Dump Valve (IOSGADV), Feedwater Line Break (FWLB) and Steam Generator Tube Rupture (SGTR) events were selected to evaluate the change of ASI.

When those events occur, the conditions of core inlet temperature, RCS pressure and core flow rate will be changed during the event. Therefore, it is necessary to verify the assumption of the constant ASI. And the use of reasonable scram rod worth, which is based on the ±0.3 ASI for the full power case, should be justified.

The thermal-hydraulic response of the system was analyzed using the CESEC-III computer code (Reference 1) and the DNBR was calculated using the CETOP computer code (Reference 2). To compare the results of CESEC-III, the HERMITE-1D computer code (Reference 3) was run with the same initial conditions as was used in the CESEC-III calculation.

II. Justification of the Use of ±0.3 ASI

The definition of ASI is the fraction of the difference in power distribution between the lower half of the core and the upper half to the total core power. In the current safety analysis, the scram rod worth calculated by the HERMITE code has been conservatively used. When the scram curve is generated, the limiting axial shapes for a ASI at each location are chosen. The conservative scram curve is generated by synthesizing the limiting shape. The CESEC-III computer code uses this rod worth according to the rods travel.

The Core Protection Calculator (CPC) has been modeled with the ASI range of ±0.6 and the setpoint of CPC range trip for ASI is ±0.5 as described in the Technical Specification (Reference 4). This wide range of ASI has been determined to cover the whole power range from zero to full power. In the safety analysis, the most conservative values of ±0.6 ASI has been very conservatively used as the initial ASI independent of the power level. However, in real plants, the ASI is calculated by COLSS and maintained by one of the following two methods during power operation.

a. Maintaining the COLSS-calculated ASI such that $-0.27 \le ASI \le +0.27$, or
b. With COLSS out of service, maintaining the CPC-calculated ASI such that $-0.20 \le ASI \le +0.20$, using any operable CPC channel.

This Limiting Conditions for Operation (LCO) is valid when the reactor power is above 20% of rated thermal power. If ASI is outside the LCO, the Technical Specification requires that the core average ASI should be within its limit within 2 hours or the thermal power should be less than 20% of rated thermal power within next 4 hours. Therefore, when the initial core power is above 20% of rated thermal power, the use of $-0.30 \le ASI \le +0.30$ as an initial condition for the safety analysis has been justified without any change in the design or Technical Specification requirements.

When an event occurs, the major operating parameters such as core power, core inlet and RCS temperatures, the RCS pressure and the core flow rate may be changed during the event. It is very difficult to evaluate the changes in ASI due to the change of the operating parameters before reactor trip. After reactor trip, the change of ASI due to the insertion of the shutdown and regulating rods has been already considered in the rod worth calculation.

III. Evaluation Methodology

The variation of the major operating parameters such as core inlet temperature, RCS pressure and core flow rate until the reactor trip are calculated from the CESEC-III computer code. Then, these data is used as input to the one dimensional HERMITE code to calculate the heat flux and the ASI vs. time. The DNBR in the event analyzed is calculated using the CETOP computer code.

The followings are the model description of CESEC-III and HERMITE-1D. The CESEC-III code, which numerically integrates the one-dimensional conservation equations, assumes a node/flow-path network to model the RCS. The energy source in the CESEC-III code is generated from the fission in the fuel. This fission energy consists of two parts, the instantaneous fission power and the decay power released by the fission products. The instantaneous power is determined by solving the standard point kinetics equations with six delayed neutron groups while the decay power is calculated from an 11 fission product group decay heat model. The present applications of the CESEC-III code utilize only one axial node in the core for the Doppler and moderator feedback calculations. The Doppler reactivity is determined from the reactivity versus fuel temperature data. The moderator reactivity is determined from either the reactivity versus moderator temperature data or moderator density.

The HERMITE-1D code performs the neutronic and thermal hydraulic calculations for the steady state or the transient using space-time coarse mesh method. Neutronics calculations are performed in one, two or three dimensions with thermal-hydraulic feedback, xenon feedback, and fuel depletion considered. HERMITE-1D uses a model based on two or four group diffusion theory. Thermal hydraulic calculations can include open or closed channel flow, fuel heat transfer to the coolant and void formation including subcooled voids. Space-time effects are important during a number of reactor transients including control rod ejections, asymmetric steam generator events and loss of flow events which require prompt scrams. In addition, certain hypothetical transients can lead to far off-nominal coolant flow conditions characterized by highly non-uniform temperature distributions, crossflow and void formation. In order not to penalize reactor performance unduly with overly conservative design methods, it is desirable to have the capability of performing detailed space-time neutronics and/or thermal hydraulic calculations for both anticipated transients and accidents.

To evaluate the variation of ASI before reactor trip, the one dimensional HERMITE-1D code was used since the CESEC-III code has no model for ASI. The DNBR of the CESEC-III code has more conservative than that of the HERMITE-1D code because of the simple modelling (point kinetics). The comparison of the predicted ASI and DNBR changes by HERMITE-1D and CESEC-III (constant ASI) are performed in the next section for various events.

The control rod movements are not credited in the analysis since it may results in non-conservative predictions of the fuel performance. An initial axial shape is arbitrarily chosen from the shapes in 0.30±0.02 ASI. And the effect of power differences between the CESEC-III and the HERMITE-1D codes on ASI is assumed to be negligible, since minor power differences between two codes will not make significant change in axial power distribution.

IV. Analysis Results and Discussions

IV.1 Increase in Feedwater Flow (IFWF) Event

Increase in Feedwater Flow (IFWF) event results in the increase in the secondary heat removal due to the excessive feedwater flow, then the core power and steam generator level increase. These may cause the reactor to trip depending on the extent of the excess feedwater flow. The behavior of the operating parameters is dependent on the value of Moderator Temperature Coefficient (MTC) and increased FW flow rate. In this study, the most negative MTC of $3.5x10^{-4}/°F$ and 140 % of design flow were chosen to investigate the changes in ASI. In this case, the reactor trip occurs at around 20 seconds into the transient. Using the CESEC-III results, the HERMITE-1D code was used to calculate the ASI and the average heat flux. Table 1 gives the initial conditions of CESEC-III and HERMITE-1D codes used in the IFWF event analysis. The initial conditon of CESEC-III are selected to initiate the event from POL. The CESEC-III code predictions are used as input to the HERMITE-1D code.

The RCS temperature and pressure will decrease due to the overcooling caused by the increased main feedwater flow. But, the core flow rate will increase slightly by the cooldown due to the increased feedwater flow. Since the RCS temperature decreases until the reactor trips, the core average heat flux increases by the most negative MTC and the least negative FTC as shown in Figure 2. Due to the changes in the T-H parameters, the ASI and DNBR are changed as shown in Figures 3 and 4. The ASI does not change in the CESEC-III code, whereas it changes in the HERMITE-1D simulation.

IV.2 Inadvertent Opening of SG Atmospheric Dump Valves (IOSGADVs) Event

In the IOSGADV event, one ADV is inadvertently opened at time zero. Due to the negative MTC, the core power increases from the initial value of 102 % of rated core power, reaching a new stabilized value of 113 %. Table 1 gives the initial conditions for the CESEC-III and HERMITE-1D codes used in the IOSGADV analysis. The initial conditons of CESEC-III are selected to initiate the event from POL. The CESEC-III code predictions are used as input to the HERMITE-1D code.

The RCS temperature and pressure will decrease due to the increased main steam flow. But, the core flow rate will increase slightly by the increased steam flow. Since the RCS temperature decreases until the reactor trips, the core power increases by the most negative MTC and the least negative FTC as shown in Figure 5. Due to the changes in the T-H parameters, the ASI and DNBR are changed as shown in Figures 6 and 7. The ASI does not change in the CESEC-III code, whereas it changes in the HERMITE-1D code simulation. And, the DNBR due to the changed ASI is less than one due to the constant ASI.

IV.3 Steam Line Break (SLB) Event

Steam Line Break event is characterized as cooldown events due to the increased steam flow rate, which causes excessive energy removal from the steam generators and the reactor coolant system. This results in a decrease in RCS pressure and temperatures, and steam generator pressures. The cooldown causes an increase in core reactivity due to the negative moderator and Doppler reactivity coefficients. Table 2 gives the initial conditions for the CESEC-III and HERMITE-1D codes used in the SLB event analysis. The initial conditons of CESEC-III are selected to initiate the event from POL. The CESEC-III code predictions are used as input to the HERMITE-1D code.

In this event, RCS temperature and pressure will decrease due to the increase in the secondary heat removal. But, the core flow rate will increase due to the cooldown. Due to the decrease in the temperature before the reactor trip, the core power increases by the most negative MTC and the least negative FTC as shown in Figure 8. Due to the changes of the T-H parameters, the ASI and DNBR are changed as shown in Figures 9 and 10. In this case, the differences between the ASI and DNBR predicted by the CESEC-III and HERMITE-1D code are very small.

IV.4 Feedwater Line Break (FWLB) Event

In the Feedwater Line Break (FWLB) event, SG water level will decrease and SG pressure and temperature will increase. The RCS pressure and temperature also increase until the reactor trip occurs due to the low steam generator water level or the high pressurizer pressure. Table 2 gives the initial conditions for the CESEC-III and HERMITE-1D codes used in the FWLB event analysis. The initial conditons for the CESEC-III are selected to initiate the event from POL. The CESEC-III code predictions are used as input to the HERMITE-1D code.

RCS pressure will increase due to the decrease in the secondary heat removal. But, the core flow rate will decrease due to the inventory decrease by opening pressurizer safety valves. Due to the temperature increase before the reactor trip, the core power decreases by the most positive MTC as shown in Figure 11. Due to the changes in the T-H parameters, the ASI and DNBR are changed as shown in Figures 12 and 13. The ASI does not change in the CESEC-III code simulation, whereas it changes in the HERMITE-1D code simulation. And, the DNBR due to the changed ASI is less than one due to the constant ASI.

IV.5 Steam Generator Tube Rupture (SGTR) Event

In the SGTR event, the reactor will be tripped by one of the following reasons; high steam-generator level, the CPC hot leg saturation or the CPC low DNBR. Before the reactor trip occurs, the RCS pressure and temperature will decrease. Table 3 gives the initial conditions for the CESEC-III and HERMITE-1D codes used in the SGTR event analysis. The initial conditons of the CESEC-III code are selected to initiate the event from POL. The CESEC-III code predictions are used as input to the HERMITE-1D code.

In the SGTR event, RCS pressure will decrease due to the decrease in RCS inventory. But, the core flow rate will decrease slightly due to the tube leakage. Before the reactor trips, the core heat flux decreases by the most positive MTC due to the temperature increase (Figure 14). Due to the changes in the T-H parameters, the ASI and the DNBR are changed as shown in Figures 15 and 16. As in the SLB analysis, the differences between the predicted ASIs and DNBRs by the two codes are very small.

V. Conclusion

YGN 5&6 Technical Specification 16.3.2.7 requires that ASI shall be maintained within the range of $-0.27 \leq ASI \leq +0.27$ in Mode 1 with thermal power above 20% rated thermal power. This requirement ensures that actual ASI values are within the range of $-0.30 \leq ASI \leq +0.30$ used in the safety analyses. Therefore, the use of $-0.30 \leq ASI \leq +0.30$ as an initial condition in the safety analysis is justified without any design change.

Based on the given initial ASI, five events of IFWF, IOSGADV, SLB, FWLB and SGTR were analyzed using the CESEC-III and HERMITE computer code. The ASI and DNBR of HERMITE-1D in each events were changed due to the core average heat flux, and other parameters. The results of CESEC-III (assumed constant at the initial value) are well close to those of HERMITE-1D. The difference between the ASIs of the CESEC-III and HERMITE-1D is very small. And, the minimum DNBR of CESEC-III is more conservative than that of HERMITE-1D.

According to the results of analysis, the amount of change in ASIs are well within $\pm$ 0.02 which can be covered by the conservative assumptions and methodology.

Finally, the use of the scram curve corresponding to a ± 0.3 ASI for all non-LOCA events except those caused by a CEA malfunction or misoperation is verified as a valid method.

VI. References

1. Enclosure 1-P to LD-82-001, CESEC digital simulation of a CE NSSS, January 6, 1982.
2. CEN-214(A)-P,CETOP-D code structure and modelling methods for ANO unit 2, July 1982.
3. CE-CES-91, REV 3-P, HERMITE 1.6 Mod 0, December 1997.
4. Technical Specification, SAR Chapter 16.

Table 1. Initial Conditions and Parameter Changes for IFWF and IOSGADV Events

PARAMETER \ EVENT	IFWF		IOSGADV	
	Initial Conditions	Changes	Initial Conditions	Changes
Core Power, %	102	N/A	102	N/A
Core Inlet Temperature, °F	570	-5	570	-5
RCS Pressure, psia	2336	-50	2336	-36
Core Mass Flow Rate, % of design flow	116	+1.11	116	+0.80
MTC	Most Negative	N/A	Most Negative	N/A
Doppler Coefficient	Least Negative	N/A	Least Negative	N/A
Fr	2.0197	N/A	2.0197	N/A
ASI	+0.3	N/A	+0.3	N/A

Table 2. Initial Conditions and Parameter Changes for SLB and FWLB Events

PARAMETER \ EVENT	SLB		FWLB	
	Initial Conditions	Changes	Initial Conditions	Changes
Core Power, %	102	N/A	102	N/A
Core Inlet Temperature, °F	570	-10	570	+5
RCS Pressure, psia	2336	-176	2336	+138, -233
Core Mass Flow Rate, % of design flow	116	+3.13	116	-1.24
MTC	Most Negative	N/A	Least Negative	N/A
Doppler Coefficient	Least Negative	N/A	Least Negative	N/A
Fr	2.0197	N/A	2.0197	N/A
ASI	+0.3	N/A	+0.3	N/A

Table 3. Initial Conditions and Parameter Changes for SGTR Event

PARAMETER \ EVENT	SGTR	
	Initial Conditions	Changes
Core Power, %	102	N/A
Core Inlet Temperature, °F	570	+1
RCS Pressure, psia	2336	-86.0
Core Mass Flow Rate, % of design flow	95	-0.07
MTC	Least Negative	N/A
Doppler Coefficient	Least Negative	N/A
Fr	1.7543	N/A
ASI	+0.3	N/A

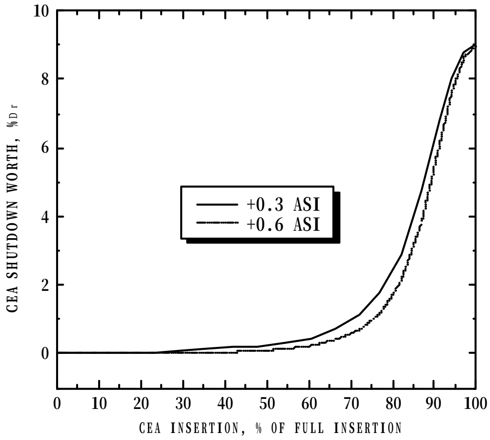

FIGURE 1. CEA SHUTDOWN WORTH

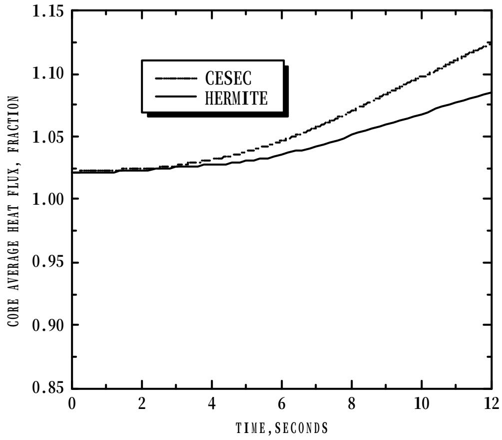

FIGURE 2. CORE AVERAGE HEAT FLUX (IFWF)

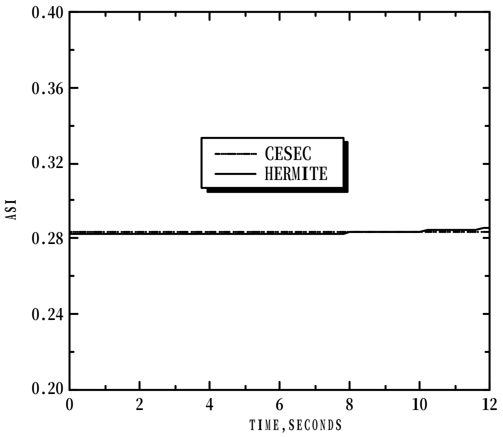

FIGURE 3. AXIAL SHAPE INDEX (IFWF)

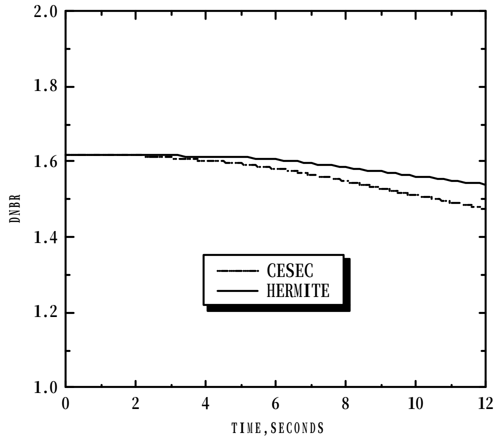

FIGURE 4. DNBR (IFWF)

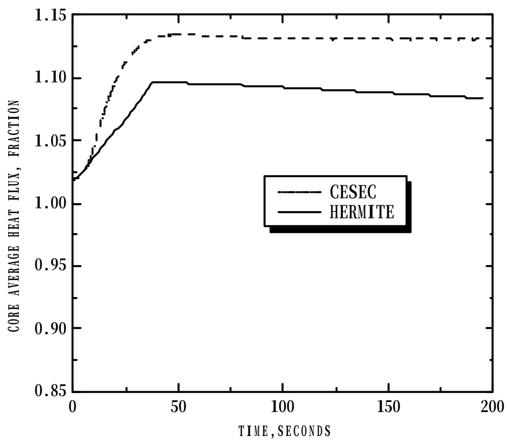

FIGURE 5. CORE AVERAGE HEAT FLUX (IOSGADV)

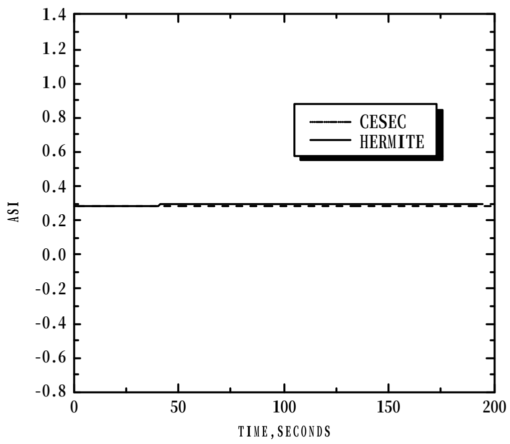

FIGURE 6. AXIAL SHAPE INDEX (IOSGADV)

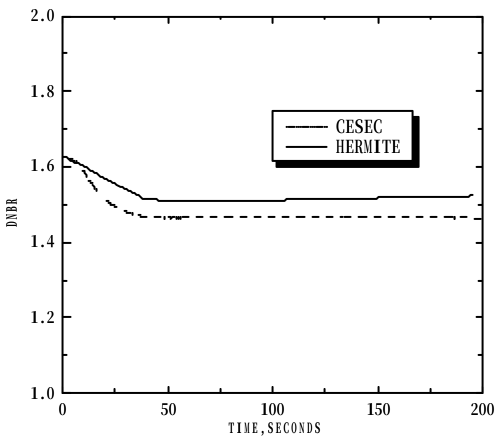

FIGURE 7. DNBR (IOSGADV)

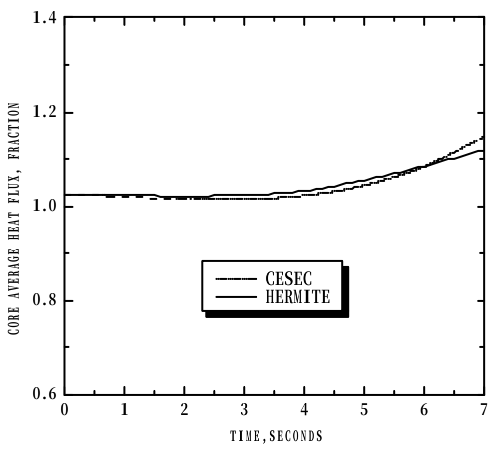

FIGURE 8. CORE AVERAGE HEAT FLUX (SLB)

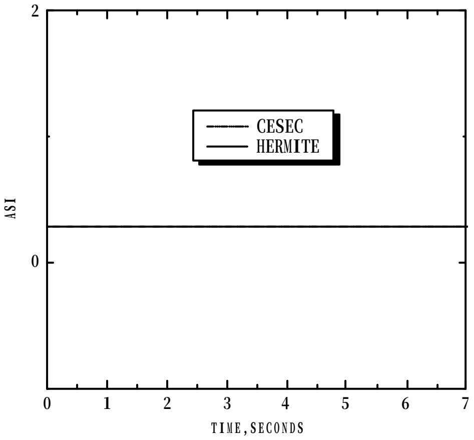

FIGURE 9. AXIAL SHAPE INDEX (SLB)

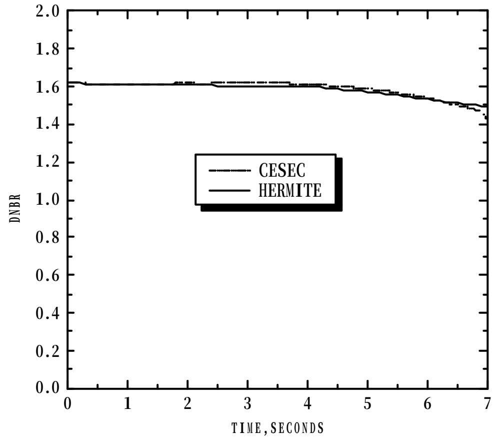

FIGURE 10. DNBR (SLB)

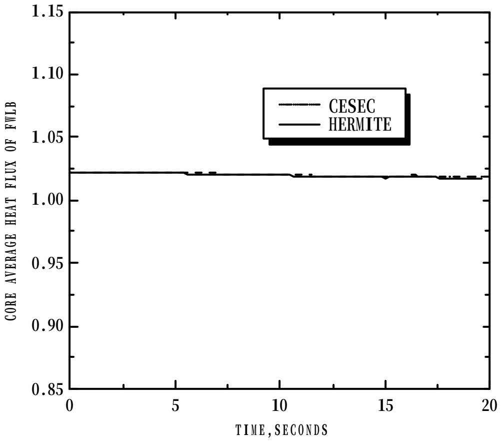

FIGURE 11. CORE AVERAGE HEAT FLUX (FWLB)

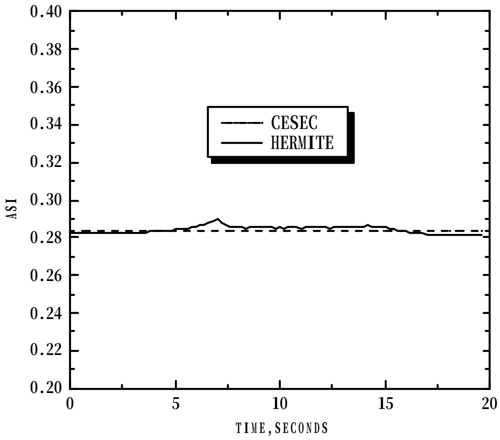

FIGURE 12. AXIAL SHAPE INDEX (FWLB)

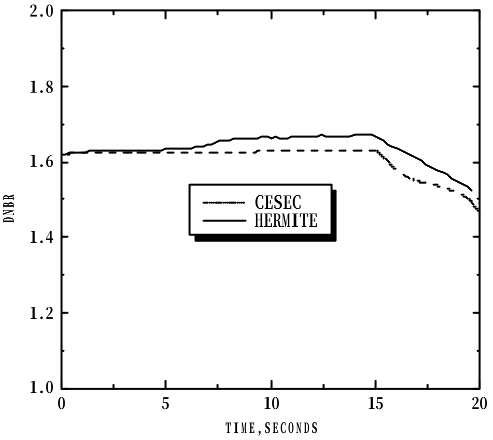

FIGURE 13. DNBR (FWLB)

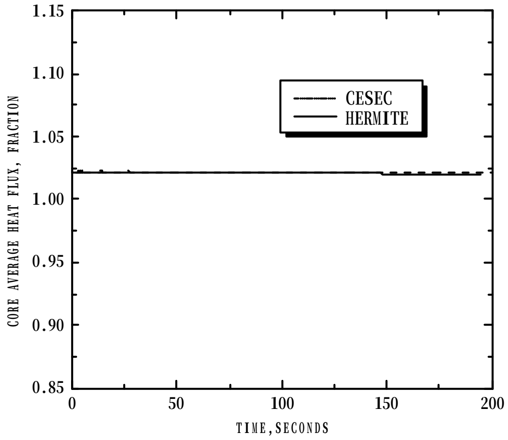

FIGURE 14. CORE AVERAGE HEAT FLUX (SGTR)

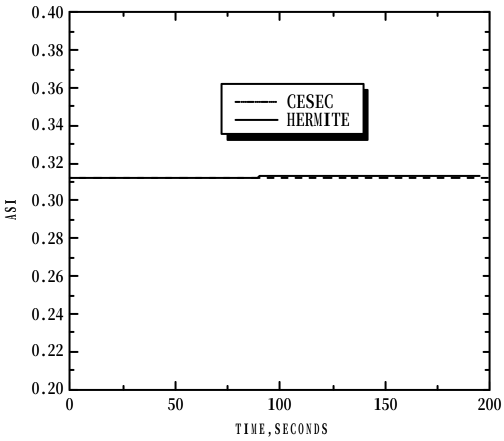

FIGURE 15. AXIAL SHAPE INDEX (SGTR)

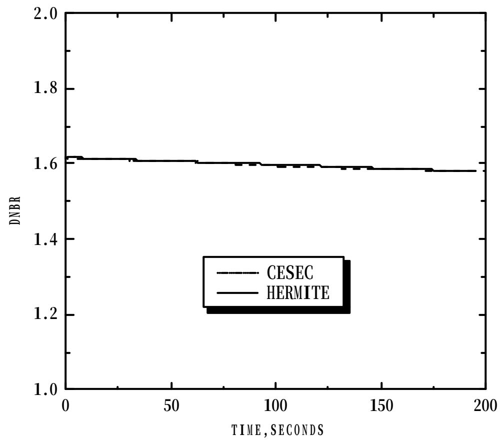

FIGURE 16. DNBR (SGTR)