Journal of the Korean Nuclear Society
Volume 34, Number 6, pp.596~609, December, 2002

<Technical Note>

Improvement in the DNBR Modeling of RETRAN for Safety Analyses of Westinghouse Nuclear Power Plants

Ae-Ju Cheong and Yo-Han Kim
Korea Electric Power Research Institute,
103-16 Munjidong, Yuseung-gu, Daejeon 305-353, Korea
down@kepri.re.kr

(Receved August 20, 2002)

Abstract

Korea Electric Power Research Institute has developed the in-house safety analysis methodologies for non-LOCA(Loss Of Coolant Accident) events based on codes and methodologies of vendors and Electric Power Research Institute. According to the new methodologies, analyses of system responses and calculation of DNBR(Departure from Nucleate Boiling Ratio) during the transient have been carried out with RETRAN code and a sub-channel analysis code, respectively. However, it takes too much time to calculate DNBR for each case using the two codes to search for the limiting case from sensitivity study. To simplify the search for the limiting case, accordingly, RETRAN code has been modified to roughly calculate DNBR using hot channel modeling. The W-3 correlation is already included in RETRAN as one of the auxiliary DNBR models. However, WRB-1 and WRB-2 correlations required to analyze some Westinghouse type fuels are not considered in RETRAN DNBR models. In this paper, the RETRAN DNBR models using the correlations have been developed and the partial and complete loss of forced reactor coolant flow events have been analyzed for Yonggwang units 1 and 2 with the new methodologies to validate the models. The results of the analyses have been compared with those mentioned in the chapter 15 of the Final Safety Analysis Report.

Key Words : Non-LOCA, analysis methodologies, RETRAN, DNBR models, WRB-1, WRB-2

1. Introduction

Korea Electric Power Research Institute has launched the project, Development of operational transients analysis and new safety analysis technology for nuclear power plants, to develop the in-house safety analysis methodologies for non-LOCA(Loss Of Coolant Accident) by the fund of the Ministry of Science and Technology and Korea Electric Power Corporation. The methodologies have been developed based on the codes and the methodologies of vendors, such as Westinghouse, Combustion Engineering, Framatome, and Kraftwerk Union, and the

Table 1. Comparison among the Westinghouse Plants

	[1]KRN-1	KRN-2	KRN-3/4	[2]YGN-1	YGN-2
Rated Thermal Power (MWt)	1,724	1,876	2,775	2,775	2,775
No. of Loops	2	2	3	3	3
TH Design Method	ITDP	ITDP	ITDP	RTDP	ITDP
Fuel Type	OFA	Standard	V5H	V5H	V5H
DNB Correlation	WRB-1	W-3/R-Grid	WRB-2	WRB-2	WRB-2
$F\Delta H$	1.49	1.55	1.59	1.59	1.59
Fq	2.35	2.34	2.60	2.60	2.60

(Note) [1]KRN and [2]YGN mean Kori unit and Yonggwang unit, respectively.

Reactor Analysis Support Package of EPRI(Electric Power Research Institute). The RETRAN [4] had been developed under the sponsorship of EPRI for best-estimate thermal-hydraulic analysis of light water reactor systems. The code allows a variable nodes, neutron kinetic models, component models, etc. and has some special models, such as pressurizer, accumulator, separator, turbine, etc, which are useful to simulate the nuclear power plant systems. The code was licensed by USNRC(U.S. Nuclear Regulatory Commission) in 2001, has been used as the system analysis code in the methodologies. Non-LOCA events are usually analyzed to assure that the systems are designed to meet the limits of the system pressure, the minimum DNBR(Departure from Nucleate Boiling Ratio), the release of radioactive materials, and so on. In the methodologies, the system responses such as pressure and temperature and the DNBR as an index of the fuel integrity during transients are calculated with RETRAN code and a sub-channel analysis code, respectively. However, it takes too much time to calculate DNBR for each case using the two codes to search for the limiting case from sensitivity study.

To simplify the search for the limiting case, accordingly, RETRAN code has been modified to calculate roughly DNBR. The RETRAN code already includes the useful DNB correlation, W-3, for the standard fuel of Westinghouse as one of the auxiliary DNB models [5]. However, it does not contain the correlations, WRB-1 and WRB-2, required for the VANTAGE-5H(V-5H) and OFA fuels as mentioned Table 1 [7, 14]. So, through the review of the DNBR calculation schemes of the code and Westinghouse' s related documents, subroutines have been developed and added to the auxiliary DNBR models for the WRB-1 and the WRB-2 correlations. Additionally, the hot channel model has been developed to calculate the DNBR in hot channel.

To evaluate the feasibility of the DNBR models added, the events of partial and complete loss of forced reactor coolant flow, which are classified by USNRC [13] as the moderate frequent event and infrequent event, respectively and are treated as the typical DNBR-events, have been analyzed for Yonggwang units 1 and 2 using the new methodologies and the DNBRs during the events have been calculated using the DNBR models.

And then the DNBRs have been compared with those in the chapter 15 of FSAR(Final Safety Analysis Report). In Westinghouse' s methodology the minimum DNBR is estimated through statistical approach such as the ITDP(Improved Thermal Design Procedure), and the nominal values without any uncertainties are used to calculate the DNBR. So, the nominal values of such initial conditions and setpoints have been selected in this evaluation.

2. DNBR Modeling Using DNB Correlations

DNB(Departure from Nucleate Boiling) occurs when either the heat flux on the surface of the fuel rods becomes too great or the local fluid conditions deteriorate. From the situation the surface of the rod is blanketed with vapor and therefore insulated from the heat sink afforded by the liquid. This leads to a rapid increase in surface temperature since the generation of heat within the fuel rod is independent of the heat transfer rate. Operation beyond DNB entails high rod surface temperatures; thus, clad damage may result from mechanisms such as oxidation embrittlement, Zr-H_2O reaction, or in the extreme case, melting of the cladding material.

For ANS Condition I and II events, the design criterion is that the probability of the limiting fuel rod(s) not being in DNB must be greater than 95% at a 95% confidence level. This criterion is met in design by requiring the minimum DNBR to be greater than some value, where DNBR is defined as:

$$DNBR = \frac{q''_{DNB,\,predicted}}{q''_{actual}} \tag{1}$$

where $q''_{DNB,\,predicted}$ and q''_{actual} are the heat flux predicted by DNB correlation and the actual operating heat flux, respectively.

2.1. W-3 DNB Correlation

Early experimental studies of DNB were conducted in simple geometries with the fluid flowing inside a tube or an annulus with the walls heated. These tests were conducted at many laboratories around the world with a wide variety of diameters and heated lengths. The results of these experiments were analyzed by various investigators aiming to develop models for the DNB phenomenon. L. S. Tong compiled an extensive library of single tube DNB data from all the published sources available at the time. From this library, Tong developed the W-3 correlation [10] which is still in wide use in the design of PWRs(Pressurized Water Reactors); that is, W-3 correlation has been applied to rod bundle geometries even though it is based on tube data.

More specifically, the W-3 correlation is used in the design of some older plants and in the analysis of certain accident conditions at relatively low pressures. It is also used in special applications, such as analysis of competitor fuels or Westinghouse non-mixing vane grid designs. However, application of the W-3 correlation to reactor rod bundles requires the use of a cold wall factor to account for the presence of unheated surfaces within the rod bundle, and of a nonuniform factor to account for the nonuniform distribution of heat flux along the length of the heated rods. In the design of recent plants, the W-3 correlation is used with several additional compensation factor which were developed to bring the correlation into better agreement with experimental rod bundle data. The equation (1) is translated as follows according to the W-3 correlation.

$$DNBR = \frac{q''_{EU,W-3} \cdot CWF \cdot F_s'}{q''_{actual} \cdot F}, \tag{2}$$

where CWF, F_s', F, q''_{actual}, and $q''_{EU,W-3}$ mean cold wall factor, modified spacer factor, Tong's nonuniform F-factor, local heat flux, and equivalent uniform heat flux calculated with W-3 correlation, respectively. If the CWF is considered, D_h, equivalent heated hydraulic diameter, is used in estimating $q''_{EU,W-3}$. And if a particular factor is not applicable (such as CWF), the value of 1.0 is used. And the W-3 correlation is defined as :

$$\begin{aligned} q''_{EU,W-3} = 10^6 & \times[(2.022 - 0.0004302P) \\ & + (0.1722 - 0.0000984P) \\ & \times e^{(18.177 - 0.004129P)\chi}] \\ & \times[(0.1484 - 1.596\chi + 0.1729\chi|\chi|) \\ & \times G/10^6 + 1.037] \times [1.157 - 0.869\chi] \\ & \times[0.2664 + 0.8357e^{(-3.151D_e)}] \\ & \times[0.8258 + 0.000794(H_{sat} - H_{in})] \end{aligned} \tag{3}$$

Where P, x, G, D_e, H_{sat}, and H_{in} stand for pressure, quality, mass velocity, equivalent hydraulic diameter, saturation enthalpy, and inlet enthalpy respectively, and those are calculated in RETRAN control volume node.

The W-3 correlation is used for predicting DNB in channels which are entirely, or almost entirely, surrounded by heated walls. In many tests and reactor configurations this is not the case. The presence of thimble or instrument tubes, baffles, and test section walls introduces unheated surfaces. It is postulated that these unheated walls (cold walls) allow a cool liquid film to build up along them, thus depriving the remainder of the channel of some of its heat removal capacity. For equivalent cross-sectional average fluid conditions, the DNB heat flux may be less in a cold wall channel than in a channel with all heated walls. This effect was empirically determined from annulus data. The form of the CWF is given below.

$$
\begin{aligned}
CWF &= \frac{q''_{cold\ wall}}{q''_{W-3,D_h}} \\
&= 1.0 - R_u[13.76 - 1.372e^{1.78x} - 4.732\left(\frac{G}{10^6}\right)^{-0.0535} \\
&\quad - 0.0619\left(\frac{P}{10^3}\right)^{0.14} - 8.509D_h^{0.107}]
\end{aligned}
\tag{4}
$$

where R_u is defined as $(D_h - D_e)/D_h$, and P, D_h, and G are pressure, equivalent heated hydraulic diameter, and mass velocity calculated in RETRAN code, respectively. $q''_{cold\ wall}$ and q''_{W-3,D_h} are cold wall critical heat flux and equivalent uniform heat flux which are calculated with W-3 correlation.

The DNB phenomenon occurs at the interface between the fluid and the heated rod surface. The condition of the interface is influenced by what has occurred upstream of the position of interest. This is because heat is being carried downstream, along the heated rod surface, by bubbles and heated liquid. Under uniform heat flux condition, the interface effects build up continuously along the rod section length, and this phenomena should be accounted in the W-3 or W-3 cold wall correlations. In the case of a nonuniform heat flux, another term is necessary to reflect the effect of the changing heat flux upstream of the DNB location into the correlations. Tong and his colleagues introduced the concept of nonuniform F-factor and developed an analytical expression to predict the DNB under nonuniform heat flux based on that of uniform heat flux. The expression includes an empirical constant derived from several nonuniform heat flux shapes. The nonuniform F-factor is given below.

$$F = \frac{C}{[q''_{local\ at\ l_{DNB}}][1 - e^{-Cl_{DNB}}]} \int_0^{l_{DNB}} q''(z)e^{-C(l_{DNB}-z)}dz, \tag{5}$$

where l_{DNB} and z are the distance from inception of boiling and the distance from inception of boiling measured in the direction of flow, respectively. And C is defined as:

$$C = 0.15\frac{(1-\chi_{DNB})^{4.31}}{(G/10^6)^{0.478}}. \tag{6}$$

The core of the Westinghouse's PWR is not made up of closed heated channels, as was the geometry of all the data used to generate the previously discussed W-3 correlations. The reactor core is an open lattice of fuel rods held in place by grids. There is very little resistance to crossflow and mixing between the sub-channels. A large body of experimental data from electrically heated rod bundle test sections were analyzed to determine the flow conditions in each sub-channel from the DNB heat flux calculated with the single tube correlations. Inspection of rod bundle test data with mixing vane grids showed that the measured DNB heat flux was greater than that predicted by the W-3 correlation. The amount of this difference was varied with the grid types, number of grids, and the local fluid conditions.

2.2. WRB-1 DNB Correlation

The WRB-1 DNB correlation [11] was developed based on the extensive body of rod bundle data collected by Westinghouse before 1976. The WRB-1 correlation also includes the nonuniform F-factor (Eq. 5) like the W-3 correlation. However, the WRB-1 correlation does not utilize the cold wall factor, the spacer factor, or any of the other factors used in the modified W-3 correlations since the functions served by those factors have been built into the WRB-1 correlation already. According to the WRB-1 correlation, the DNBR (Eq. 1) is expressed as follow :

$$DNBR = \frac{q''_{WRB-1}}{q''_{actual} \cdot F}, \tag{7}$$

where F, q''_{actual}, and q''_{WRB-1} are Tong's nonuniform F-factor, local heat flux, and predicted equivalent uniform heat flux by WRB-1 correlation, respectively. Moreover, q''_{WRB-1} is represented as the function of pressure (P), local mass velocity (G_{loc}), local quality (x_{loc}), equivalent hydraulic diameter (D_e), equivalent heated hydraulic diameter (D_h), distance from beginning of heated length (L_H), performance factor (P_F), distance from last grid (d_g), and grid spacing (g_{sp}).

The DNBR correlation limit is that value of DNBR, based on a statistical analysis of the DNB test data, for which there is a 95% probability that DNB will not occur at a 95% confidence level. A correlation limit DNBR of 1.17 for the WRB-1 correlation has been approved by the NRC for the fuel types of 17×17 STD, 17×17 OFA, 15×15 OFA, 14×14 OFA, VANTAGE-5(V-5), and V-5H. Due to statistical variations in CHF test data, the NRC has imposed a correlation limit DNBR of 1.37 for 14×14 STD and 15×15 STD fuel.

2.3 WRB-2 DNB Correlation

DNB tests performed with the V-5 and APWR(Advanced Pressurized Water Reactor) geometries showed that the WRB-1 correlation under-predicted the DNB benefits of the reduced grid spacings. So the needs of modifying the WRB-1 correlation come to the fore to make efficient use of the benefits unveiled through the tests. The new modified correlation, WRB-2 [11], was developed using the data of 17×17 STD, 17×17 OFA, V-5 and APWR DNB tests. The correlation uses the same nonuniform F-factor as the WRB-1 correlation. The DNBR definition with WRB-2 correlation is as follow :

$$DNBR = \frac{q''_{WRB-2}}{q''_{actual} \cdot F}, \tag{8}$$

where F, q''_{actual}, and q''_{WRB-2} are Tong's nonuniform F-factor, local heat flux, and predicted equivalent uniform heat flux with WRB-2 correlation, respectively. The parameters in WRB-2 correlation are similar to those of WRB-1 correlation except the performance factor (PF). The factor is not considered in the correlation. The effect of the parameter is accounted by the others, however, the applicable fuel types are less than WRB-1 correlation.

It is recommended that the WRB-2 correlation be used only for 17×17 designs including V-5, V-5H, and APWR safety analyses. The NRC has approved a correlation limit DNBR of 1.17 for V-5 and V-5H. A report justifying the use of a 1.17 correlation limit for the 19×19 APWR fuel design has been approved by the NRC also.

2.4. Improvement in the DNBR Modeling

2.4.1. WRB-2 Correlation Coding

Prior to the reflection of the WRB-2 correlation as a option, which is represented as subroutines in the source code, of the RETRAN auxiliary DNBR models, the W-3 correlation model and some

Table 2. Comparison Between W-3, WRB-2 Correlation Related Parameters

W-3 Correlation Related		WRB-2 Correlation Related	
Pressure, P	psia	Pressure, P	psia
Local Quality, x	-	Local Quality, x	-
Local Mass Velocity, G	lbm/hr-ft^2	Local Mass Velocity, G	lbm/hr-ft^2
Equivalent Hydraulic Diameter, D_e	Btu/lbm	-	-
Inlet Enthalpy, H_{in}	in.	Equivalent Hydraulic Diameter, D_e	Btu/lbm
Equivalent Heated H. D., D_h	in.	Equivalent Heated H. D., D_h	in.
Total Heated Length, L	ft	Distance from Beginning of	in.
		Heated Length, L_H	ft
-	-	Distance from Last Grid, d_g	in.
-	-	Grid Spacing, g_{sp}	in.

Table 3. Comparison among W-3, WRB-2, and WRB-1 Correlation Related Parameters

W-3 and WRB-2 Correlation Related		WRB-1 Correlation Related	
Pressure, P	psia	Pressure, P	psia
Local Quality, x	-	Local Quality, x	-
Local Mass Velocity, G	lbm/hr-ft^2	Local Mass Velocity, G	lbm/hr-ft^2
Inlet Enthalpy, H_{in}	Btu/lbm	-	-
Equivalent Hydraulic Diameter, D_e	in.	Equivalent Hydraulic Diameter, D_e	in.
Equivalent Heated H. D., D_h	in.	Equivalent Heated H. D., D_h	in.
Total Heated Length, L[W-3]	ft		
Distance from Beginning of		Distance from Beginning of	
Heated Length, L_H [WRB-2]	ft	Heated Length, L_H	ft
Distance from Last Grid, d_g	in.	Distance from Last Grid, d_g	in.
Grid Spacing, g_{sp}	in.	Grid Spacing, g_{sp}	in.
-	-	Performance Factor, PF	-

other subroutines included in the code have been reviewed in detail. As mentioned in Table 2, some of the WRB-2 correlation related parameters, such as pressure and local quality, are used those corresponding to the W-3 parameters and others such as distance from last grid (d_g) and grid spacing (g_{sp}) are required to be defined additionally in the new subroutines. The distance from last grid and the grid spacing have been calculated from the V-5H fuel assembly data of Yonggwang unit 2.

2.4.2. WRB-1 Correlation Coding

The WRB-1 correlation is added as a RETRAN auxiliary DNBR model by the same manner mentioned in the section 2.4.1. Most of the parameters used in the WRB-1 are similar to those of W-3 or WRB-2 correlations except the performance factor (PF) (Table 3). The PF is the geometric parameter determined by the outside diameter of the fuel rods [11]. The data used in the model have been generated based on the fuel design data of Yonggwang unit 2.

2.4.3. Hot Channel Modeling

The DNBR is calculated with radial and axial distributions and peaking factors, which are generated by core physics/fuel management analysis. Westinghouse's plants employ a design radial power distribution which is characterized by the enthalpy rise hot channel factor, $F^N_{\Delta H}$. $F^N_{\Delta H}$ increases with decreasing power level as follow:

$$F^N_{\Delta H} = F^N_{\Delta H,RTP}\{1+PF_{\Delta H}(1-P)\} \quad (9)$$

where $F^N_{\Delta H,RTP}$ as the enthalpy rise hot channel factor at RTP varies between 1.446 and 1.59, $PF_{\Delta H}$ as part-power multiplier is 0.3, and P defined as the ratio of power to RTP is 1.0 for 100% RTP [7].

Use of the highest values of the enthalpy rise hot channel factor at full power, provides the maximum heat flux and the lowest value of DNBR. In conjunction with the maximum value of the enthalpy rise hot channel factor, the flattest fuel rod assembly power distribution is utilized to minimize inter channel mixing and produce the highest possible enthalpy in the hot channel. In general, the design axial power distribution is a 1.55 chopped cosine in Westinghouse's plant design.

3. Safety Analyses

3.1. Partial Loss of Forced Reactor Coolant Flow

A PLOF(Partial Loss Of forced reactor coolant Flow) can result from a mechanical or electrical failure in a RCP(Reactor Coolant Pump), or from a fault in the power supply to the pump or pumps supplied by a RCP bus. Normal power for the RCPs is supplied through busses from a transformer connected to the generator. When a generator trip occurs, the busses are automatically transferred to a transformer supplied from external power lines, and the pumps will continue to operate. Following any turbine trip without electrical faults which require tripping the generator from the network, the generator remains connected to the network. Hence, the RCPs will continue to operate after the reactor trip before any transfer is made.

If the PLOF is initiated at full power, a rapid increase in the coolant temperature may occur. This increase, which could result in DNB with subsequent fuel damage, will be terminated by tripping the reactor. The negative moderator temperature coefficient and insertion of control rods by reactor protection system cause reduction in neutron power. Because of the thermal capacity of the fuel, the heat flux is reduced more slowly than the neutron power. The incident can be generally characterized by a race between the decreasing flow and the decreasing heat flux. Since the flow is decreasing more rapidly than the heat flux, initially, the heat flux to flow ratio is increasing and hence, the margin to DNB is reduced. The closest approach to the DNB limit occurs in the neighborhood of the maximum heat flux to flow ratio. Heat flux to flow ratio is turned around and reduced by the insertion of control rods into the core. Reactor coolant system pressure tends to increase with the rapid rise in coolant temperature caused by the loss of forced primary coolant flow. The pressure increase is terminated by the eventual reduction of heat flux to flow ratio and the coolant temperatures.

The necessary protection against this event is provided by the low primary coolant flow reactor trip signal, which is actuated in any reactor coolant loop by two out of three low flow signals. Above Permissive 8 (30% to 50% Rated Thermal Power), low flow in any loop will actuate a reactor trip. Between approximately 10% RTP (Permissive 7) and the power level corresponding to Permissive 8, low flow in any two loops will actuate a reactor trip. Above Permissive 7, two or more RCP circuit breakers opening will actuate the corresponding undervoltage relays. This results in a reactor trip which serves as a backup to the low flow trip.

3.2. Complete Loss of Forced Reactor Coolant Flow

A CLOF(Complete Loss Of forced reactor coolant Flow) may result from a simultaneous loss of electrical supplies to all RCPs. This simultaneous loss of electrical supplies could occur as a result of a failure in the transformer supplying the two pump busses followed by a failure to transfer to external power lines or inadvertent open circuit of both pump busses. Another possibility might be a commom cause event which fails both busses. Generally, a CLOF event provides a greater challenge to the safety limits than a PLOF event.

The necessary protection against the CLOF incident is provided by reactor trip on RCP supply undervoltage or underfrequency. The reactor trip on reactor coolant bus undervoltage is provided to protect against conditions which can cause a loss of voltage to two or more RCPs. A voltage condition below the setpoint, as sensed by undervoltage relays either on RCP electrical bus or on the motor side of the RCP breakers will directly trip the reactor. This trip is bypassed below Permissive 7. The reactor trip on RCP underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid. If an underfrequency condition below the

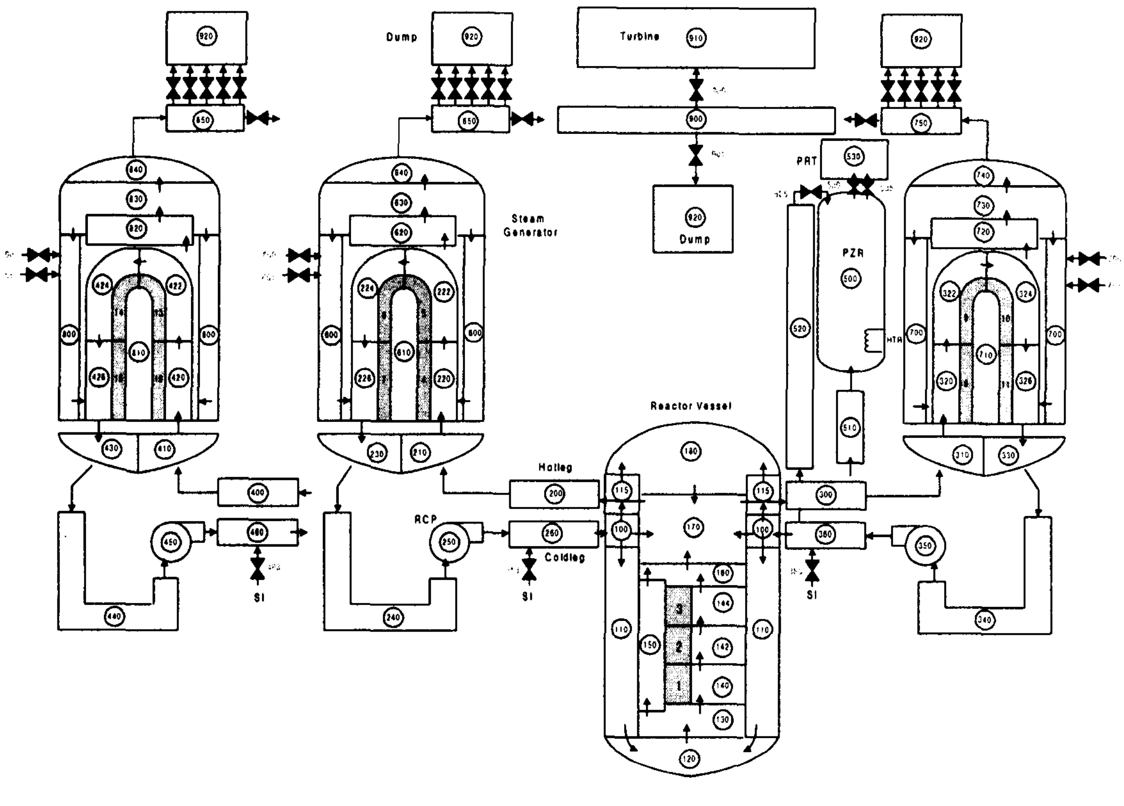

Fig. 1. Nodal Diagram for Yonggwang unit 2

underfrequency setpoint exists on the RCP buses, all RCP breakers and the reactor are tripped. RCP breakers are tripped for pump protection and because a rapid decrease in electrical frequency can decelerate the RCPs faster than a complete loss of power. Reactor trip on low primary coolant loop flow is provided to protect against loss of flow conditions which affect only one reactor coolant loop. These reactor trips ensure that DNBR during the transient is maintained within design limit [2].

3.3. Safety Analysis Methodologies

To analyze for PLOF and CLOF, the NSSS systems of the Yonggwang unit 2 are modeled with 66 volumes and 98 junctions (Fig. 1). Setpoints for reactor and turbine trips, valves of pressurizer and steam generators, etc., are controlled by 105 trip cards and 78 control block description cards. 3 RCPs are started and tripped by trip card and modeled with pump characteristic curve issued by the manufacturer. In steam generators, tubes and secondary sides are modeled with 4 vertical heat conductors and 5 volumes, respectively, based on the assumption that the thermal-hydraulic phenomena occurred in the curved section of the U-tubes is not dominant compared with those of straight regions of the tubes and could be represented as the straight tubes with the same thermal-hydraulic characteristics. Point kinetics model is assumed and then nuclear power is computed by the summation of reactivities. Fuels of the core are modeled with 3 stacked vertical heat conductors. To model PLOF or CLOF events, the steady-state operation for 10 seconds and the trip of one RCP or all RCPs has been assumed, respectively.

For the safety analyses of the events, the initial conditions are set as mentioned in Table 4 and key input parameters are as follow:

- the reactor trip setpoint and delay time;
- the control rod reactivity insertion as a function of time;
- the moderator temperature coefficient and Doppler coefficient;
- the initial reactor coolant system power;
- the initial core flow rate;
- the initial coolant temperature and pressure;
- the fuel properties.

For the events, the reactor trip setpoint for the low primary coolant flow is set by assuring that the DNBR design limit is not violated. It is normal practice to set the reactor trip setpoint as far as possible from normal operating conditions to avoid unwanted spurious reactor trips. This is accomplished by selecting an appropriate LSSS(Limiting Safety Systems Settings) setpoint value that provides ample margin between the LSSS setpoint and normal operation. This margin must accommodate delay times, measurement uncertainties, system perturbations, and spare capacity. A safety analysis setpoint is then selected by biasing the LSSS in the opposite direction to normal operation to account for delay times and measurement uncertainties. Delay times vary between 0.6 and 1.0 seconds delay from trip to initiation of rod insertion on low flow. These delay times account for delays in signal actuation, opening time of trip breakers, and the release of the rods by mechanism [2]. In this paper, if the primary flow rate decreases to 86.5% of nominal flow or under as described in the FSAR 15.3.1, then the reactor is tripped with 1.0 seconds delay time for PLOF event. In addition to, reactor trip is initiated by RCP undervoltage or under frequency signal which is modeled to be actuated manually and then the reactor is tripped with 1.5 seconds delay time for CLOF.

Control rod trip reactivity insertion as a function of time is obtained from combining the dropped control rod position as a function of time with the

Table 4. Initial Condition Utilized in Safety Analyses for PLOF and CLOF

Parameters	Nominal Values
Core Thermal Power (MWt)	2,775
Thermal Power Generated by RCP (MWt)	12
Vessel Average Temperature (°F)	588.5
Pressurizer Pressure (psia)	2,250
Reactor Coolant Flow per Loop (gpm)	97,900
Total Reactor Coolant Flow (10^6 lbm/hr)	109.3
Total Steam Flow (10^6 lbm/hr)	12.3
Steam Pressure at SG Outlet (psia)	964
Assumed Feedwater Temperature at SG Inlet (°F)	440
Average Core Heat Flux (Btu/hr-ft^2)	197,200

reactivity worth of the control rods as a function of rod position. Control rod insertion time is provided in Limiting Condition for Operation of the Technical Specifications. For various axial flux shapes, reactivity worth of the control rods is computed using core physics codes. The physics code results are benchmarked by plant tests. In safety analysis, only minimum worth is used and includes the effect of the most reactive control rod stuck in the fully withdrawn position. Control rod worth is further reduced by 10% to account for uncertainty in the calculation [1, 2].

In analyzing the events, the minimum reactivity feedback effect is selected to minimize reactivity insertion due to the increased coolant temperature. That is, the moderator temperature coefficient and the Doppler coefficient are chosen the least negative value and the maximum value at BOC (Beginning of Cycle), respectively [2, 3].

The ITDP conditions for core power, coolant temperature, and pressurizer pressure lead to the minimum DNBR for safety analysis of each event. In addition to, use of the minimum reactor coolant system pressure is conservative, since DNBR decreases with decreasing pressure. In order to minimize the benefit of the increasing pressure during the transient, the pressurizer spray may be assumed to operate if its setpoint is reached. However, the pressurizer spray is assumed to be inoperable. The initial levels of pressurizer and steam generator are set as the nominal full power programmed level, and the maximum steam generator tube plugging level is assumed.

The initial core flow rate is selected at its minimum to minimize the initial available margin for the DNBR limit. System coastdown characteristics are governed primarily by system loop resistance and inherent pump characteristics, especially flywheel inertia and pump torque. System loop resistances can be inferred from in-plant measurements while pump characteristics are supplied by the manufacturer [2]. In this paper, the events are analyzed with 97,900 gpm per loop as the initial core flow rate. Feedwater flow is set equal to the steam flow, consistent with the initial power level. 90% of the nominal RCP motor inertia is used to calculate the flow coastdown [1]. Generally, safety analysis relies only on safety grade instrumentation and systems. And then none of the control systems are assumed to mitigate PLOF or CLOF transient.

4. Results

4.1. DNBR Calculation

Fig. 2 shows the DNBRs at the PLOF, which are calculated with the RETRAN options of W-3 correlation (indicated as "W-3" in the figure and WRB-2 correlation ("WRB-2") and WRB-1 correlation ("WRB-1") discussed in the section 2.4. To evaluate the feasibility of the options added, the DNBRs are compared with the those of FSAR 15.3 computed with Westinghouse' s THINC code and WRB-2 correlation (" FSAR"). The DNBRs at the CLOF is presented in the Fig. 3 with the same indicators as Fig. 2. By the review of the Fig. 2 and Fig. 3, the results using the new RETRAN options show the similar trends although there are some differences for the minimum DNBR.

4.2 Safety Analyses for PLOF and CLOF

Compared with the FSAR 15.3, the results of the safety analyses are presented in Fig. 4 to Fig. 7 for PLOF and in and Fig. 8 to Fig. 11 for CLOF. For PLOF and CLOF, the results show that the general trends for nuclear and thermal power, flow in reactor vessel and faulted loop, and pressurizer pressure are similar to those of FSAR. The maximum pressurizer pressures at PLOF and CLOF computed with RETRAN are lower than those of FSAR as showed in Fig. 7 and Fig. 11 respectively, because the pressurizer model in

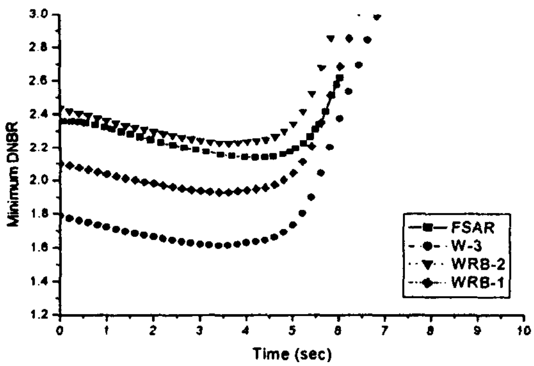

Fig. 2. DNBR Calculation with RETRAN for PLOF

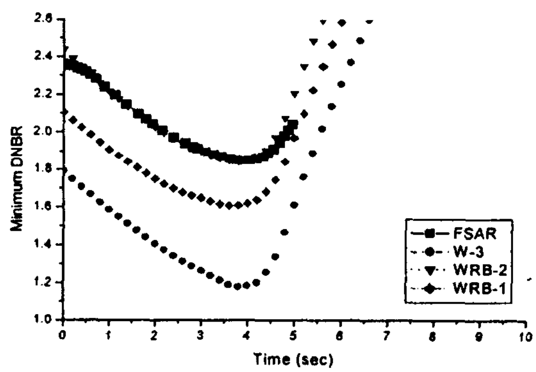

Fig. 3. DNBR Calculation with RETRAN for CLOF

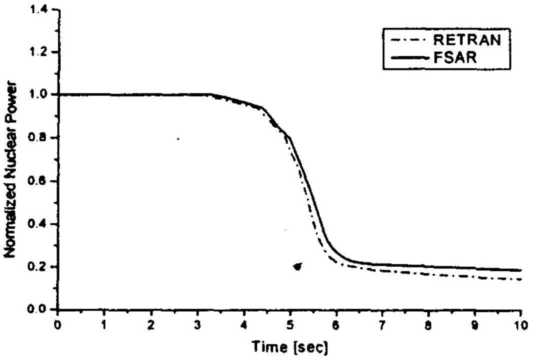

Fig. 4. Normalized Nuclear Power at PLOF

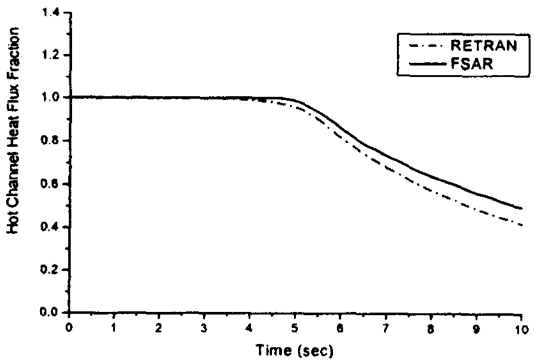

Fig. 5. Hot Channel Heat Flux Fraction at PLOF

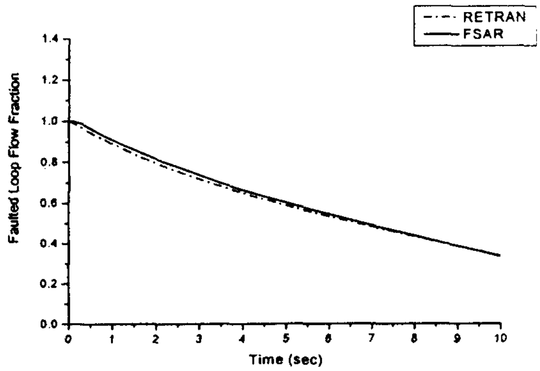

Fig. 6. Faulted Loop Flow Fraction at PLOF

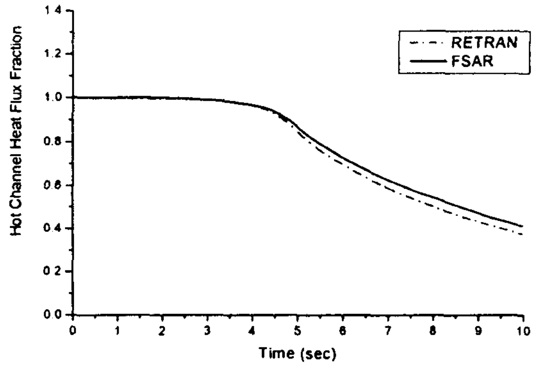

Fig. 7. Pressurizer Pressure at PLOF

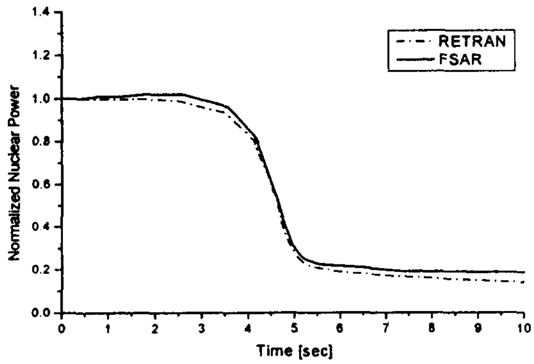

Fig. 8. Normalized Nuclear Power at CLOF

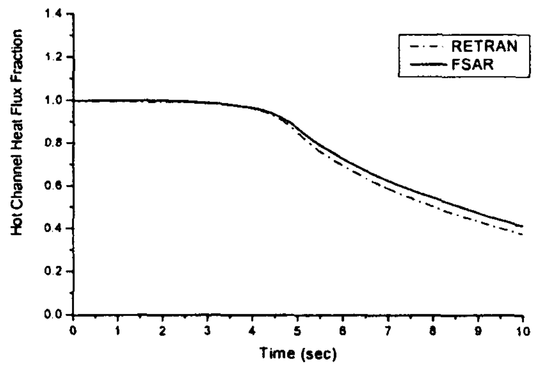

Fig. 9. Hot Channel Heat Flux Fraction at CLOF

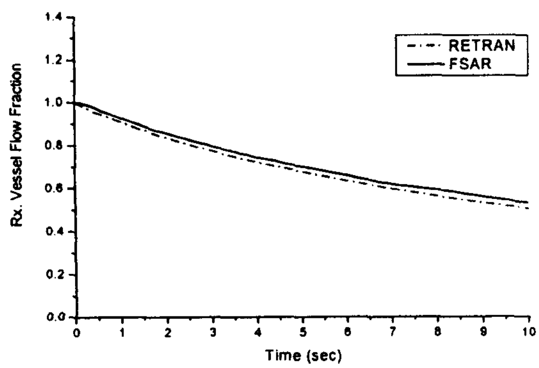

Fig. 10. Reactor Vessel Flow Fraction at CLOF

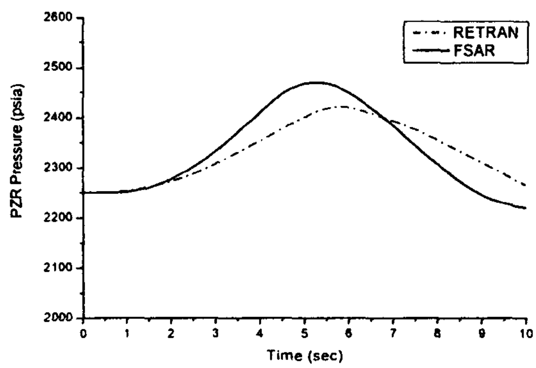

Fig. 11. Pressurizer Pressure at CLOF

RETRAN is more sophisticated [4, 6, 12] than that of the code used for the FSAR. Additionally, the lower maximum pressure in pressurizer results from earlier occurrence of the low flow signal in RETRAN calculation than in FSAR case for PLOF and the differences of nuclear power due to reactivity feedback effects during 4 seconds after CLOF initiation. In the RETRAN calculation, the earlier low flow signal occurrence causes earlier reactor trip and then less heat flux generated in core. The minimum DNBRs computed with RETRAN WRB-2 model, which vary higher than those of FSAR as showed in Fig. 2 and Fig. 3, result from the less conservative hot channel model than sub-channel analysis as well as the combination lower maximum pressurizer pressure and less heat flux generation.

5. Conclusions and Future Work

The safety analysis methodologies for non-LOCA has been developed. To search the limiting case at a standpoint of DNBR, two auxiliary DNBR models and hot channel model have been developed using WRB-1 and WRB-2 correlations for the Westinghouse' s fuel such as V-5H and OFA. To estimate the feasibility of the new models, PLOF and CLOF events have been analyzed with RETRAN. The results of the analyses have been compared with those mentioned in FSAR 15.3 and found the similarity among them except a little difference. For the next step of this work, DNBR will be calculated using sub-channel analysis code for the limiting case selected by the sensitivity study and also confirm the feasibility of the new models from the comparison with DNBR calculated with RETRAN. Moreover, safety analyses for other events will be performed with these models, and the models will be improved to be applicable to other Westinghouse' s plants.

Acknowledgement

This work has been accomplished under the fund of the Ministry of Science and Technology and Korea Electric Power Corporation.

References

1. *Safety Analysis Standard*, Rev. 6, Westinghouse Co. (1999).
2. Lance J. Agee, J.F. Harrison, D.M. Bucheit, W.C. Beck, H.D. Fulcher, J.O. Cermak, and J.M. Betancourt, The Reactor Analysis Support Package (RASP), NP-4498, Vol. 3, *Electric Power Research Institute* (1986).
3. Whee G. Choe, *TXU Reload Core Design and Safety Analysis Methodology*, Korea Electric Power Co. (2000).
4. J.G. Shatford, C.E. Peterson, J.H. McFadden, M.P. Paulsen, and G.C. Gose, *RETRAN-3D; A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems*, Vol. 1: Theory and Numerics, NP-7450, Vol. 1, Rev. 5, Electric Power Research Institute (2001).
5. J.G. Shatford, C.E. Peterson, J.H. McFadden, M.P. Paulsen, and G.C. Gose, *RETRAN-3D; A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems*, Vol. 3: User' s Manual, NP-7450, Vol. 3, Rev. 5, Electric Power Research Institute (2001).
6. G.H. Heberle, *LOFTRAN-Code Description and User's Manual*, WCAP-7878, Rev. 5, Westinghouse Co. (1989).
7. YGN 1&2 FSAR, Korea Electric Power Co. (1994).
8. YGN 1&2 PL&S, Rev. 3, Korea Electric Power Co. (1993).
9. *DYNODE-P RETRAN-3D Benchmark*, Kewaunee Nuclear Power Plant (2000).
10. Thermal-Hydraulic Design Procedure Manual,

Westinghouse Co. (1990).

11. *Thermal-Hydraulic Engineering Services Manual,* Westinghouse Co. (1990).
12. N.E. Todreas and M.S. Kazimi, *Nuclear Systems I - Thermal Hydraulic Fundamentals*, Hemisphere Publishing Co. (1990).
13. *Standard Format and Content of Safety Analysis Report for Nuclear Power Plant*, Regulatory Guide 1.70, Nuclear Regulatory Commission. (1978).
14. *Vantage 5H Fuel Assembly*, WCAP-10444-P-A, Westinghouse Co. (1988).