Calculation and Analysis of Critical Heat Flux at LWR Passive Safety Design

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Abstract. Critical heat flux (MDNBR) is one parameter for safe nuclear reactor operation. The magnitude of MDNBR is determined by local heat flux. When it is sufficiently high, it will cause a change in heat equilibrium and deteriorate nuclear fuel structure. To obtain assurance for nuclear reactor safety in safe criteria, Calculation and analysis of Critical heat flux of 630 MWe passive LWR was carried out at steady state and transient condition. The study used COBRA IV-I sub channel program coupled with critical heat flux correlation of EPRI-Columbia and W-3. The program models the core into 1/8 sections, each of which is divided into 83 sub-channels in radial direction and 40 nodes in axial direction, including 8 sub-channels of the hottest assembly. Normalized coast down flow and reactor power decay at initial phase is 10 second after loss of flow accident (LOFA) caused by anticipated primary pump failure. The effects of all input parameters, such as radial nodes, axial nodes, mix coefficient, velocity of coolant mass entering the core and power variation, have been identified. Evaluation of the results of safety calculation indicates that MDNBR, cladding temperature, and hot channel temperature are 1.83, 349°C and 2033°C, respectively. As the licensing limits for those parameters are 1.3, 1850°C, and 2840°C, it is concluded that the design of 630 MWe passive LWR has met the licensing criteria of safety. Modelling for safety analysis of passive LWR employs COBRA IV-I program. The results show that this program can be used to calculate hydraulic safety parameters with sub-channel model.

Keywords: Critical heat flux, Passive LWR, MDNBR, COBRA IV-I, Cladding temperature.

1. Introduction
In order to reduce manpower in operation and maintenance and influence of human errors on reactor safety, a concept of JAERI passive safety reactor has been developed in Japan atomic energy research institute, this reactor is light water reactor type using passive safety design(LWR passive safety design)[1].

In the LWR passive safety design, canned-motor pumps are adopted as the primary coolant circulation pumps so as to eliminate the pump-seal-water feed system which is used for primary coolant circulation pumps of conventional pressurized water reactor (PWR's). As a result, the chemical and volume control system (CVCS) can be simplified. On the other hand, the coolant flow rate reduces very rapidly under pump trip accident due to the low inertia of canned-motor pump compared to the
conventional circulation pumps. Therefore, the departure from nucleate boiling ratio (DNBR) under pump trip accident is potentially one of limiting design constraints of this reactor.

The DNBR is defined as the ratio between the critical flux predicted by an applicable critical heat flux (CHF) correlation and the local heat flux of a fuel rod. In order to maintain an adequate safety margin, the DNBR in every part of the core should be larger than a minimum allowable DNBR (MDNBR) criterion which gives a 95 % probability at the 95% confidence level that no fuel rod in the core experiences DNB. The DNBR criterion has been one of the major limitations on the thermal output of a light water reactor (LWR).

In the present study, steady-state DNBR in a hottest channel of the LWR passive safety design is evaluated by using the COBRA-IV-I subchannel code [2] coupled with a CHF correlation. Sensitivity studies with respect to various such as DNB correlation, mixing coefficient and other input conditions are also performed. Furthermore, transient DNB analysis under pump trip accident condition are performed to evaluate the safety margin of the present LWR PASSIVE SAFETY DESIGN design including the density reactivity coefficient, pump inertia, and power density. The core inlet transient conditions were calculated with RETRAN-02 code [3] and then the COBRA-IV-I code was used to obtain local flow conditions and fuel rod surface heat flux for DNBR calculations.

The present report describes the results of steady-state DNBR analyses from a view point of thermal-hydraulic feasibility of the LWR passive safety design.

2. Concept of LWR Passive Safety Design

Mayor parameters of LWR passive safety design listed in Table 1. The mayor features of this reactor are as follows;

a. The nuclear steam supply system has an inherent matching nature of core heat generation and heat removal.

b. In-vessel control rod drive mechanism units, low power density core and once-through steam generators are adopted to eliminate chemical shim, to reduce the Doppler effect and to minimize the change of steam generator exit temperature, respectively.

c. In consideration of containing coolant in the primary system, a large volume pressurizer and passive residual heat removal systems are adopted.

d. A passive engineered safety injection system which does not require any auxiliary systems or components of "safety class" is adopted. The emergency diesel generator can also be eliminated.

Based on the above characteristics, the manpower in operation and maintenance and influence of human errors on reactor safety are significantly reduced. The reactor pressure vessel is longer than that the ordinary 800 MWe class PWRs in order to incorporate the in-vessel CRDMs. Two units of residual heat exchangers are installed outside of the pressure vessel and each unit has a natural circulation loop with six air cooler units.

The capacity of one unit of residual heat exchanger with five air cooler units has a heat removal capacity of 6% of full power when the primary system is in the hot conditions and 0.3% in the refuelling conditions. Each heat exchanger is connected to the upper plenum and the downcomer with normally-closed valves for isolation from the system in operating condition which are normally opened in inspection and repair works and several check valves and actively operated valves between the heat exchanger and the downcomer.

The most important point of this concept are:

a. By closing main steam line or stopping feedwater supply, the core heat generation is physically interrupted in a short time.

b. Without considering the heat removal from the main steam line or using additional residual heat removal systems residual heat from the core can be removed by the residual heat exchanger systems and/or the in-containment air cooler in emergency and in reactor shutdown system.
c. The radioactive material can be contained in the NSSS in most of load reduction events and even a tube failure event in HE SGs. Even in the case where pressure relief valves open the radioactive material can be contained in the containment.

Figure 1 and 2 show the cross section of core and fuel assembly, respectively. The core consists of 145 fuel assemblies. Four-batches refuelling cycle is planned to obtain high burnup. In order to attain the inherent matching of core heat generation and heat removal from the secondary side, the ratio between the reactivity coefficient in moderator density and the absolute value of negative reactivity coefficient on fuel temperature (Doppler effect) should be increased in comparison to a current PWR [4]. The lower core power density is effective to realize the above characteristic as well as the elimination of chemical shim control. Therefore, the core power density was reduced to about 75% of a current PWR [5].

The configuration of fuel assembly including control rod thimbles and an instrumentation thimble is the same as that of the conventional 17 X 17 type PWR fuel assembly. However, the control rod cluster should be installed to all fuel assemblies to compensate excess reactivity due to the elimination of chemical shim. Since the space of penetration holes at the top of pressure vessel is limited, in-vessel control rod drive mechanisms (CRDMs) which are being developed for a concept of a new marine reactor being designed at JAERI [6] were adopted.

Passive residual heat removal and containment cooling system which does not require any auxiliary systems or components of safety class are adopted in the LWR passive safety design. As a result, emergency diesel generators and active components such as valves and pumps can be eliminated. Preliminary evaluation [7-8] by assuming one-dimensional flow under steady-state condition indicated that one unit of residual heat exchanger has a heat removal capacity of 6 % of full power when the primary system is in the hot conditions and 0.3 % in the cold conditions. The containment cooling system can remove 1 % of full power at the pool temperature of 75°C, when the available number of air cooler units is four and air temperature is as high as 30°C. The heat imbalance between heat from the residual heat exchanger and heat to the air coolers is stored in the pool in a certain time. Therefore, the gravity coolant injection pool work as a heat reservoir.

The Passive engineering safety system consists of core make up tanks (2 units), accumulators (2 units) and gravity coolant injection system (2 units), accumulators (2 units) and gravity coolant injection systems (2 units). The high-pressure core makeup tanks with a pressure equalizing line are
actuated by actively-operated valves or by passively-operated valves. When the system pressure decreases to a certain level, the accumulators are actuated by opening check valves and water is provided by nitrogen gas pressure. The gravity coolant injection pool is used for reservoir for back-up water injection. The pool is also used for heat sink of decay heat and for absorber of heat and iodine in case of LOCA. In case of severe core damage events, water in the pool is injected to the bottom of pressure vessel in the reactor cavity by passive devices and/or active valves.

Table 1. Major parameters of LWR passive safety design

| Parameter                        | 2-loop PWR |
|----------------------------------|------------|
| Core power                       | 1853 MWt (630 MWe) |
| Control rod drive mechanism      | In-vessel CRDM |
| Chemical shim                    | Not used   |
| Steam Generator                  | Once-through type |
| Primary pump                     | Non-seal canned pump |
| Operating pressure               | 15.7 MPa   |
| Fuel assembly                    | 17 X 17 type |
| Number of fuel assembly          | 145        |
| Fuel rods/assembly               | 264        |
| Control rods/assembly            | 24         |
| Fuel cladding diameter           | 9.5 mm     |
| Core height                      | 3.66 m     |
| Linear power density             | 13.2 kW/m  |
| Core Inlet temperature           | 285 °C     |
| Core exit temperature            | 325 °C     |

3. COBRA-IV-I Input Data

3.1 Subchannel Nodding Model and Power Distribution

The whole core is represented by the 1/8 sector due to the symmetrical configuration of the LWR PASSIVE SAFETY DESIGN core. Figure 3 and 4 shows the radial nodding schematics together with the radial peaking factors. As shown in these figures, the core is divides into 83 subchannels including 8 subchannels in the hot assembly. The radial peaking factors were based on neutronic calculations [9].
The analysis indicated that the highest radial power factor appears in the peripherical at the beginning of eighth equilibrium cycle (BOEC). The radial peaking factor is assumed to be 1.616 by considering the effect of control rod insertion. The power distribution for each fuel rod within the hot assembly has not been obtained. In the present analysis it was assumed that the radial peaking factor of hot rod in the hot assembly is 1.2 with respect to the bundle average power. The engineering hot channel factor of 1.03 and the nuclear uncertainty factor of 1.05 were then taken into account. As a result, the radial peaking factor (Fr) of the hot rod is determined as follows:

\[
Fr = 1.616 \times 1.2 \times 1.03 \times 1.05 = 2.097
\]

There are two hot rods, one is located at the corner and the other is located between control rod thimbles as shown in Figure 4.
Figure 4. Subchannel nodding model and radial power factor.

Therefore, subchannels 1 and 6 are the hot channels. Subchannel 1 is named a typical cell that is surrounded by heater rods including the hot rod. Subchannel 6 is named a thimble cell that is surrounded by a control rod thimble and three heater rods, one of which is the hot rod. The enthalpy rise is larger in the typical cell due to the larger heat input and the mass velocity is lower in the thimble cell due to the smaller equivalent diameter. Therefore, the DNBR becomes minimum in either of these two cells depending on subchannel analysis. The radial power factors of other rods were determined by normalization of power distribution obtained by the neutronic calculation. Figure 5 shows axialnodding, grid spacer location and axial power distribution. The axial peaking factor is determined to be 1.29 based on the neutronic calculation at the beginning of eighth equilibrium cycle. The axial power distribution is assumed to be chopped cosine. The fuel assembly is divided into 40 axial nodes with 91.5 mm length for each node. The axial peaking factor is determined to be 1.29 based on the neutronic calculation at the beginning of eighth equilibrium cycle. The axial power distribution is assumed to be chopped cosine.
3.2 Geometrical Parameters

The LWR passive safety design geometry input parameters for the COBRA-IV-I code are listed as follows:

1. Coolant channel length
   \[ L = 3.66 \text{ m}. \]

2. Cross sectional flow area for all coolant channels
   - For typical cell
     \[ A_{c1} = (d_1)^2 - \pi/4 (d_2)^2 \]
     \[ = 8.788 \times 10^{-5} \text{ m}^2 \]
   - For thimble cell
     \[ A_{c2} = (d_1)^2 - \pi/4 (d_3)^2 \]
     \[ = 7.634 \times 10^{-5} \text{ m}^2 \]
   - For side cell
     \[ A_{c3} = D_e (d_2/2 + 2,35) - \pi/4 (d_2)^2 \]
     \[ = 5.402 \times 10^{-5} \text{ m}^2 \]
   - For fuel assembly
     \[ A_{c4} = (d_1 \times 17)^2 - \pi/4 (d_3)^2 \times 264 - \pi/4 (d_3)^2 \times 25 \]
     \[ = 2.425 \times 10^{-5} \text{ m}^2 \]

3. Heated perimeter of each channel
   - For typical cell
     \[ P_{h1} = \pi d_2 \]
     \[ = 2.984 \times 10^{-2} \text{ m} \]
   - For thimble cell
     \[ P_{h2} = \pi d_3 \times 3/4 \]
     \[ = 2.238 \times 10^{-2} \text{ m} \]
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- For side cell
  \[ P_{h3} = \pi d_1 \times 1/2 = 1.492 \times 10^{-2} \text{ m} \]
- For fuel assembly
  \[ P_{h4} = \pi d_2 \times 264 = 7.88 \times 10^{-2} \text{ m} \]

(4) Wetted perimeter of each channel
- For typical cell
  \[ P_{w1} = \pi d_2 = 2.984 \times 10^{-3} \text{ m} \]
- For thimble cell
  \[ P_{w2} = (\pi d_2 \times 1/3) + (\pi d_2 \times 1/4) = 3.197 \times 10^{-2} \text{ m} \]
- For side cell
  \[ P_{w3} = \pi d_2 \times 1/2 + d_1 = 2.752 \times 10^{-3} \text{ m} \]
- For fuel assembly
  \[ P_{w4} = \pi d_2 \times 264 + \pi d_1 \times 25 = 8.84 \times 10^{-3} \text{ m} \]

(5) Width of gaps between fuel rods and between rods and walls
Gap width between fuel rods and assembly wall
  \[ = 2.35 \times 10^{-3} \text{ m} \]
Gap width between fuel rods
  \[ = (d_1 - d_2) = 3.1 \times 10^{-3} \text{ m} \]
Gap width between fuel assembly
  \[ = (215 - 12.6 \times 17) = 0.8 \times 10^{-3} \text{ m} \]

\[ d_1, \text{ Fuel rod pitch} = 1.26 \times 10^{-2} \text{ m} \]
\[ d_2, \text{ Fuel rod outer diameter} = 9.5 \times 10^{-3} \text{ m} \]
\[ d_1, \text{ Control rod and instrumentation thimble diameter} = 1.22 \times 10^{-2} \text{ m} \]
\[ \text{No. of fuel rods per assembly} = 264 \]
\[ \text{No. of control rod guide tube} = 24 \]
\[ \text{No. of instrumentation guide tube} = 1 \]
\[ \text{Fuel assembly arrangement} = 17 \times 17 \]
\[ \text{Fuel assembly pitch} = 2.15 \times 10^{-1} \text{ m} \]

3.3 Turbulent Mixing Coefficient

Based on the sensitivity study by Reddy and Fighetti [10], the turbulent mixing coefficient which is an input parameter of the subchannel code has the most significant effect on the local flow conditions. In the present analysis, the turbulent mixing coefficient was determined to be 0.038 which is the same value as used in the safety analysis of Takahama No.3 and 4 plant [11] because the configuration of fuel assembly including grid spacers of JPSR is designed to be approximately the same as that of the power plant. An exact value of turbulent mixing coefficient should be determined by a thermal diffusion experiment using a simulated fuel assembly of JPSR. The flow mixing will be enhanced under two-phase flow condition. However, the mixing coefficient under two-phase flow is assumed to
be the same as in the case of single-phase flow because two-phase mixing data have not been available for the JPSR fuel assembly.

3.4 Heat Conduction in Fuel Rod

The heat conduction calculation by COBRA-IV-I requires following input data; specific heat of fuel, specific heat of fuel, specific heat of cladding, mass density of fuel, mass density of cladding, thermal conductivity of fuel, thermal conductivity of cladding and heat transfer coefficient in the gap between fuel and cladding. These input data for JPSR heat conduction calculation is shown in Table 2.

3.5 Initial Conditions

Based on the operational conditions of JPSR, initial conditions for the COBRA-IV-I analyses were determined as follows;

(1) System pressure
   System pressure = 15.65 MPa (2269.2 psi)

(2) Inlet temperature
   Inlet temperature = 285.2 °C (545.3 °F)

(3) Mass velocity
   Flow area of fuel assembly = 0.02459 m²
   Number of fuel assembly in core = 145
   Total flow area = 0.02459 X 145 = 3.565 m²
   Total core flow rate = 8110 kg/s
   Core inlet mass velocity = 8110/3.565= 2275 kg/s.m² (1.678 Mlb/hr.ft²)

(4) Average surface heat flux
   Cladding outer diameter = 9.5 mm
   Effective fuel length = 3.66 m
   Number of fuel rod per fuel assembly = 264
   Number of fuel assembly = 145
   Total heat transfer area = 0.1092 X 264 X 145 = 4180 m²
   Thermal power = 1853 MWt
   Average surface heat flux = 1853/4180 = 0.4433 MW/m² (0.1405 MBtu/hr.ft²)

3.6 Thermal-hydraulic Correlation

The thermal-hydraulic correlations used in the COBRA-IV-I analysis are listed in Table 3. The EPRI void correlation [12] and EPRI-Columbia CHF correlation [9] were implemented to the original COBRA-IV-I code [2]. The W-3 CHF correlation [13] in the original code was modified for the 17 X 17 fuel assembly with mixing vanes.

| Table 2. Material properties for heat conduction calculation of fuel rod |
|-----------------|-----------------|
| Parameters      | Value           |
| Geometry data   |                 |
| Rod diameter    | 9.5 mm          |
| Pellet diameter | 8.17 mm         |
| Cladding thickness | 0.57 mm      |
| Gap             |                 |
| Material properties of fuel |          |
| k               | thermal conductivity |
| Cp              | specific heat    |
| ρ               | density          |
| To              | reference temperature (1898 oF) |
Subscripts
UO2 = uranium dioxide fuel used core
c = cladding (zircaloy-4)

\[ K_{UO2}(T) = K_{UO2}(T_0) \left( 1 + C_1(T - T_0) + C_2(T - T_0)^2 + C_3(T - T_0)^3 \right) \]

\[ K_{UO2}(T_0) = 2.371 \text{ w/k.m} = 2.89 \text{ Btu/hr.}^\circ\text{F} \]

\( T = \text{Temperature (}^\circ\text{F)} \)
\( T_0 = \text{Reference temperature (}^\circ\text{F)} (1898 \text{ }^\circ\text{F}) \)
\( C1 = +3.7379 \times 10^{-4} \)
\( C2 = 2.3302 \times 10^{-7} \)
\( C3 = -2.9043 \times 10^{-11} \)

Other properties are constant with respect to temperature.
\( C_{p_{UO2}} = 324.9 \text{ j/kg.k} \) (0.0776)
\( \rho_{UO2} = 10115.0 \text{ kg/m}^3 \)
\( k_c = 16.72 \text{ w/k.m} \)
\( C_{p_{c}} = 374.4 \text{ j/kg.k} \) (0.0894 Btu/lb.\(^\circ\)F)
\( \rho_{c} = 6514.2 \text{ kg/m}^3 \)

3) Fuel-clad gap conductance : \( h_{gap} \)
\( h_{gap} \) is calculated from TRAC-PF1 model

\[ h_{gap} = h_{gas} + h_{contract} + h_{rad} \]

(1) \( h_{gas} = \frac{K_{gas}}{\Delta (r_{gap} + \delta)} \)
\( K_{gas} = 0.2727 \text{ w/k.m} \)
\( \Delta r_{gap} = 0.095 \times 10^{-3} \text{ m} \)
\( \delta = 4.4 \times 10^{-6} \text{ m} \)
\( h_{gas} = 2743.5 \text{ w/m}^2.\text{k} \)

(2) \( h_{contact} = 0.0 \text{ w/m}^2.\text{k} \)

(3) \( h_{rad} = \sigma \frac{F(T_f^4 - T_c^4)}{(T_f - T_c)^4} \)
\( F = 1/(1/\varepsilon_{f} + (R_f/R_c) (1/\varepsilon_{c} -1)) \)
\( \sigma = 5.67 \times 10^{-8} \text{ w/m}^2.\text{k}^4 \)
\( T_c = 620 \text{ k}, \varepsilon_f = 0.8707, R_f = 4.085 \times 10^{-3} \text{ m} \)
\( T_f = 900 \text{ k}, \varepsilon_c = 0.7500, R_c = 4.75 \times 10^{-3} \text{ m} \)
\( h_{rad} = 71.7 \text{ w/m}^2.\text{k} \)

There for,
\( h_{gap} = (2743.5 + 0 + 71.7) \text{ w/m}^2.\text{k} = 2815.2 \text{ w/m}^2.\text{k} = 495.8 \text{ Btu/hr.ft}^2.\degree\text{F} \)

COBRA-IV-I input = 500 Btu/hr.ft\(^2\).\degree\text{F}

### Table 3. Thermal-hydraulic correlations used in COBRA-IV-I analysis

| Parameters                  | Value            |
|-----------------------------|------------------|
| Subcooled void model        | Levy             |
| Bulk void model             | EPRI (*)         |
| Rod friction coefficient    | Blasius          |
| Spacer loss coefficient     | 1.0              |
| Heat transfer correlation   | RELAP-4 package  |
| Cross flow resistance       | 0.5              |
| Cross flow momentum factor  | 0.5              |
Turbulent momentum factor : 0.0  
Cross flow axial velocity : \( \frac{U(j)+U(i)}{2} \)  
CHF correlation : EPRI-Columbia(*) or W-3  

(*) The EPRI void correlation [12] and EPRI-Columbia CHF correlation [9] were implemented in the original COBRA-IV-I

4. Evaluation of DNBR under steady-state condition

4.1 DNBR under Steady-State Condition

The EPRI-Columbia correlation gives the MDNBR of 3.270 at 1.922 m in the thimble cell. On the other hand, the W-3 correlation gives the MDNBR of 2.439 at 1.922 m in the thimble cell. These MDNBR's are much higher than the MDNBR limit of 1.3 using the W-3 correlation for conventional PWR design. In the present study, the results from EPRI-Columbia correlation were selected as the base case because the W-3 correlation gave unreasonable result under transient condition due to the fact that the minimum flow rate was out of the applicable range of the W-3 correlation. The difference between the results with two CHF correlation under steady-state condition should be resolved by performing DNB experiments using a test section which simulates the LWR passive safety design core geometry.

Although some uncertainty remains in the calculations, it is suggested that an enough safety margin is assured under the steady-state operational condition of LWR passive safety design.

4.2 Parametric Effects

The effects of various input parameters such as axial nodding model, mixing coefficient, core inlet mass velocity and core power on the MDNBR calculation results are shown in Table 4 up to Table 7 and Figure 6 using the EPRI-Columbia and W-3 CHF correlations respectively.

(1) Effect of axial nodes
In the base case, the core was divided in to 40 axial nodes. The number of axial node was increased to 60 and the results were compared with the base in Table 4. The MDNBR decreases from 3.270 to 3.258 (0.3 %) by increasing the axial node from 40 to 60. The difference is so small that the effect of axial nodding number can be neglected.

(2) Effect of mixing coefficients
With increasing the turbulent mixing coefficient from 0.0 to 0.1, the MDNBR tends to increase because higher mixing coefficient results in the lower enthalpy and higher mass velocity in the hot channel. However, the effect of mixing coefficient on MDNBR is not significant in the calculation as shown in Table 5.

(3) Core inlet mass velocity
The MDNBR increases with the core inlet mass velocity as shown in Table 8. The MDNBR is larger than 1.3 even when the core inlet flow rate is decreased to 1000 kg/s.m\(^2\) (44 % of the nominal mass velocity), indicating that there exists enough safety margin with respect to the core inlet mass velocity. It should be noted that the mass velocity less than 1356 kg/s.m\(^2\) is not included in applicable range of W-3 correlation.

(4) Core power
The MDNBR decreases to 1.255 (PRI-Columbia correlation) or 1.267 (W-3 correlation) with increasing the core power up to 180% of operating condition as shown in Table 7. At least up to 160
% of the nominal power, the MDNBR is larger than 1.3, assuring enough safety margin under overpower transient condition.

Table 4. Effect of axial nodding

| Axial node | MDNBR results | Local condition |
|------------|---------------|-----------------|
|            | MDNBR | Elevation (mm) | cell  | Enthalpy (kJ/kg) | Mass velocity (kg/s.m^2) |
| 40         | 3.270  | 1922           | Thimble | 1414           | 2203                     |
| 60         | 3.258  | 1922           | Thimble | 1411           | 2179                     |

*base case

Table 5.a. Effect of mixing coefficient using EPRI-Columbia CHF correlation

| Mixing coefficient β | MDNBR results | Local condition |
|----------------------|---------------|-----------------|
|                      | MDNBR | Elevation (mm) | Cell  | Enthalpy (kJ/kg) | Mass velocity (kg/s.m^2) |
| 0.000                | 3.148  | 1921.6         | Typical | 1441.54         | 2244.55                 |
| 0.002                | 3.170  | 1921.6         | Typical | 1426.80         | 2194.90                 |
| 0.010                | 3.223  | 1921.6         | Typical | 1429.22         | 2252.66                 |
| 0.020                | 3.248  | 1921.6         | Thimble | 1417.54         | 2200.83                 |
| 0.038*               | 3.270  | 1921.6         | Thimble | 1413.90         | 2203.23                 |
| 0.05                 | 3.280  | 1921.6         | Thimble | 1412.14         | 2204.36                 |
| 0.10                 | 3.311  | 1921.6         | Thimble | 1407.07         | 2207.65                 |

*base case

Table 5.b. Effect of mixing coefficient using W-3 CHF correlation

| Mixing coefficient β | MDNBR results | Local condition |
|----------------------|---------------|-----------------|
|                      | MDNBR | Elevation (mm) | Cell  | Enthalpy (kJ/kg) | Mass velocity (kg/s.m^2) |
| 0.000                | 2.376  | 1921.6         | Thimble | 1429.20         | 2193.34                 |
| 0.002                | 2.386  | 1921.6         | Thimble | 1426.80         | 2194.89                 |
| 0.010                | 2.409  | 1921.6         | Thimble | 1421.10         | 2198.50                 |
| 0.020                | 2.424  | 1921.6         | Thimble | 1417.54         | 2200.83                 |
| 0.038*               | 2.439  | 1921.6         | Thimble | 1413.90         | 2203.20                 |
| 0.05                 | 2.447  | 1921.6         | Thimble | 1412.10         | 2204.40                 |
| 0.000                | 2.467  | 1921.6         | Thimble | 1407.10         | 2207.70                 |

*base case

Table 6.a. Effect of core inlet mass velocity using EPRI-Columbia CHF correlation

| Core inlet mass velocity(kg/s.m^2) | MDNBR results | Local condition |
|-----------------------------------|---------------|-----------------|
|                                   | MDNBR | Elevation (mm) | Cell  | Enthalpy (kJ/kg) | Mass velocity (kg/s.m^2) | Equilibrium quality |
| 1000                              | 1.394  | 2837.0         | Thimble | 1805.4         | 879.70 | 0.177                         |
| 1500                              | 2.303  | 2196.1         | Thimble | 1539.5         | 1430.50 | 0.000                         |
| 2000                              | 2.954  | 2196.1         | Thimble | 1471.0         | 1901.64 | 0.000                         |
| 2275*                             | 3.270  | 2196.1         | Thimble | 1413.9         | 2203.20 | 0.000                         |
| 2500                              | 3.635  | 2196.1         | Thimble | 1429.6         | 2381.63 | 0.000                         |
| 3000                              | 4.060  | 1921.6         | Thimble | 1377.7         | 2914.01 | 0.000                         |
Table 6.b. Effect of core inlet mass velocity using W-3 CHF correlation

| Core inlet mass velocity (kg/s.m²) | MDNBR results | Local condition | Equlibrium quality |
|-----------------------------------|----------------|-----------------|--------------------|
|                                   | MDNBR Elevation (mm) | Cell | Enthalpy (kJ/kg) | Mass velocity (kg/s.m²) |
| 1000                              | 1.928 2013.1 Typical | 1637.60 | 942.34 | 0.001 |
| 1500                              | 2.228 1921.6 Thimble | 1490.90 | 1450.50 | 0.000 |
| 2000                              | 2.258 1921.6 Thimble | 1434.45 | 1933.50 | 0.000 |
| 2275*                             | 2.439 1921.6 Thimble | 1413.90 | 2203.20 | 0.000 |
| 2500                              | 2.600 1921.6 Thimble | 1400.40 | 2424.00 | 0.000 |
| 3000                              | 2.984 1921.6 Thimble | 1377.70 | 2914.00 | 0.000 |

Table 7.a. Effect of core power using EPRI-Columbia CHF correlation

| Ratio of core power (%) | MDNBR results | Local condition |
|-------------------------|---------------|-----------------|
|                         | MDNBR Elevation (mm) | Cell | Enthalpy (kJ/kg) | Mass velocity (kg/s.m²) |
| 80                      | 4.301 1921.6 Thimble | 1383.8 | 2209.50 | 0.000 |
| 100*                   | 3.270 1922.0 Thimble | 1413.9 | 2203.23 | 0.000 |
| 120                    | 2.580 2013.1 Thimble | 1457.6 | 2181.14 | 0.000 |
| 140                    | 2.073 2104.6 Thimble | 1505.2 | 2159.40 | 0.000 |
| 160                    | 1.634 2379.0 Typical | 1592.1 | 2069.50 | 0.000 |
| 180                    | 1.255 2562.1 Thimble | 1667.2 | 1891.80 | 0.032 |

Table 7.b. Effect of core power using W-3 CHF correlation

| Ratio of core power (%) | MDNBR results | Local condition |
|-------------------------|---------------|-----------------|
|                         | MDNBR Elevation (mm) | Cell | Enthalpy (kJ/kg) | Mass velocity (kg/s.m²) |
| 80                      | 3.218 1921.6 Thimble | 1383.80 | 2209.50 | 0.000 |
| 100*                   | 2.439 1921.6 Thimble | 1413.90 | 2203.20 | 0.000 |
| 120                    | 1.948 1922.1 Thimble | 1444.00 | 2194.40 | 0.000 |
| 140                    | 1.805 2013.1 Thimble | 1490.10 | 2173.00 | 0.000 |
| 160                    | 1.507 2013.1 Thimble | 1522.80 | 2137.39 | 0.000 |
| 180                    | 1.267 2013.0 Thimble | 1557.50 | 1997.80 | 0.000 |

Figure 6 show the results of calculated DNBR vs. elevation for the typical cell using the EPRI-Columbia and W-3 correlations.
6. Conclusion
DNB analyses under steady-state an transient condition for LWR passive safety design were performed with COBRA-IV-I subchannel code coupled with the EPRI-Columbia CHF correlation. Under the steady-state condition, Evaluation of the results of safety calculation indicates that the minimum DNBR in the hot channel are 3.27 at elevation of 1.922 m in the thimble cell. The cladding temperature are 349°C and 2033°C, respectively. These MDNBR value are high enough from a view point of DNBR criterion for PWR's. This primary due to the lower power density and larger density reactivity coefficient in the LWR passive safety design compared with those of current PWR's. The results indicated that the LWR passive safety design fuel rod experiences no DNB under operational and accident conditions. It is concluded that the design of reactor passive LWR has met the licensing criteria of safety.

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