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COOLOD-N2 : A COMPUTER CODE, FOR THE ANALYSES OF STEADY-STATE THERMAL-HYDRAULICS IN RESEARCH REACTORS

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日 本 原 子 力 研 究 所
Japan Atomic Energy Research Institute

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COOLOD-N2: A Computer Code, for the Analyses of Steady-state Thermal-hydraulics in Research Reactors

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The COOLOD-N2 code provides a capability for the analyses of the steady-state thermal-hydraulics of research reactors. This code is revised version of the COOLOD-N code, and is applicable not only for research reactors in which plate-type fuel is adopted, but also for research reactors in which rod-type fuel is adopted. In the code, subroutines to calculate temperature distribution in rod-type fuel have been newly added to the COOLOD-N code. The COOLOD-N2 code can calculate fuel temperatures under both forced convection cooling mode and natural convection cooling mode as well as COOLOD-N code. In the COOLOD-N2 code, a "Heat Transfer package" is used for calculating heat transfer coefficient, DNB heat flux etc. The "Heat Transfer package" is subroutine program and is especially developed for research reactors in which plate-type fuel is adopted. In case of rod-type fuel, DNB heat flux is calculated by both the "Heat Transfer package" and Lund DNB heat flux correlation which is popular for TRIGA reactor. The COOLOD-N2 code also has a capability of calculating ONB temperature, the heat flux at onset of flow instability as well as DNB heat flux.

Keywords: COOLOD-N, COOLOD-N2, DNB, Flow Instability, Forced Convection, Natural Convection, ONB, Plate-type Fuel, Research Reactor, Rod-type Fuel, Steady-state, Thermal-hydraulic

研究炉の定常熱水力解析コードCOOLOD-N2

日本原子力研究所東海研究所研究炉部
神永　雅紀

（1994年2月15日受理）

本報告書は、研究炉の定常熱水力解析コードCOOLOD-N2について述べたものである。本コードは、板状燃料を使用する研究炉の定常熱水力解析コードCOOLOD-Nの改良版であり、棒状燃料を使用した研究炉の解析が行えるように棒状燃料の温度計算サブルーチンを新たにCOOLOD-Nコードに組込んだものである。COOLOD-N2は、強制循環冷却及び自然循環冷却のいずれの場合にも適用可能である。本コードにおいてもCOOLOD-Nコードと同様に板状燃料を用いた研究炉用に開発された熱伝達相関式、DNB熱流束相関式等からなる「熱伝達パッケージ」が組込まれているが、棒状燃料のDNB熱流束計算では、熱伝達パッケージの他にTRIGA炉の解析で用いられているLundの相関式による値も合わせて計算するようにした。この他には、熱水力設計限界等の判定に重要な、ONB温度、流動不安定（Flow Instability）などの計算機能も有している。

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Nomenclature

A	: Flow area (m^2)
A_H	: Heated area (m^2)
Cp	: Specific heat (kJ/kg K)
D_H	: Equivalent heated diameter (m)
De	: Equivalent hydraulic diameter (m)
F, f	: Friction loss coefficient (-)
F_b	: Bulk temperature rising factor (-)
F_B	: Bond temperature rising factor (-)
F_f	: Film temperature rising factor (-)
F_U	: Fuel meat temperature rising factor (-)
F_W	: Clad temperature rising factor (-)
G	: Mass flow rate (kg/m^2s)
G^*	: Dimensionless mass flow rate $=\dfrac{G}{\sqrt{\lambda \rho_g g(\rho_\ell - \rho_g)}}$
g	: Acceleration of gravity (m/s^2)
h	: Heat transfer coefficient (kW/m^2 K)
h_{fg}	: Heat of vaporization (kJ/kg)
Δh_i	: Inlet subcooled enthalpy (kJ/kg)
k	: Thermal conductivity (kW/m K)
L	: Flow channel length (m)
L_H	: Heated length (m)
Nu	: Nusselt number
P	: Pressure (kg/cm^2 abs.)
Pc	: Critical pressure (kg/cm^2 abs.)
Pe	: Peclet number (-)
P_H	: Heated perimeter (m)
Pr	: Prandtl number (-)
q	: Heat flux (kW/m^2)
q^*	: Dimensionless heat flux $=\dfrac{q}{h_{fg}\sqrt{\lambda \rho_g g(\rho_\ell - \rho_g)}}$
$\dot{q}$	: Heat generation rate (kW)
Re	: Reynolds number (-)
T	: Temperature (°C)
V, v	: Velocity (m/s)
W	: Width of channel (m)
x	: Quality (-)
y	: Thickness (m)
Z	: Distance from inlet of channel (m)
β	: Volmetric expansion coefficient (1/K)
ε	: Surface roughness (m)
σ	: Surface tension (N/m)
λ	: Characteristic length $=\sqrt{\dfrac{\sigma}{(\rho_\ell - \rho_g)g}}$ (m)
μ	: Dynamic viscosity (Ps·s)
ν	: Kinematic viscosity (m^2/s)
ρ	: Density (kg/m^3)

ζ	: Resistance coefficient due to geometry change (-)
η	: Bubble detachment parameter (-)

Subscript

b	: Bulk
B	: Bond
c	: critical heat flux cvondition
DNB	: Departure from Nucleate Boiling
f	: Film
g	: Steam
l	: Liquid
in	: Inlet
out	: Outlet
ONB	: Onset of Nucleate Boiling
s	: Saturated
sub	: Subcooled
U	: Fuel meat
W	: Clad or wall

1. Introduction

In Japan Atomic Energy Research Institute (JAERI), COOLOD-N code was developed for steady state thermal-hydraulic analysis of research reactors in which plate type fuel is employed[1], especially for steady state thermal-hydraulic analysis under natural convection cooling, based on COOLOD code[2]. Thermal-hydraulic analyses of the JRR-3M[3], RSG-GAS in Indonesia[4], MEX-15 planning in Mexico[5], etc. have been performed, using the COOLOD-N code. COOLOD-N2 code is a revised version of the COOLOD-N code. The COOLOD-N2 is developed based on the COOLOD-N code and provides a capability for the analysis of the steady-state thermal-hydraulics of research reactors. The COOLOD-N2 is applicable not only for research reactors in which plate-type fuel is adopted, but also for research reactors in which rod-type (pin-type) fuel is adopted. In the code, subroutines to calculate temperature distribution in rod-type fuel have been newly added to the COOLOD-N code. The COOLOD-N2 code can calculate fuel temperatures under both forced convection cooling mode and natural convection cooling mode as well as COOLOD-N code. In the COOLOD-N2 code, a "Heat Transfer Package[6]" which is subroutine program to calculate heat transfer coefficient and DNB heat flux etc., and was especially developed for research reactors in which plate type fuel is adopted, is also adopted, but in case of rod-type fuel, DNB heat flux is also calculated by Lund[7] correlation which is popular for TRIGA type fuels. The COOLOD-N2 code also has a capability of calculating ONB temperature, the heat flux at onset of flow instability (for plate-type fuel only) as well as DNB heat flux.

2. Description of the COOLOD-N2 code

2.1 Fuel plate temperature calculation

Fuel plate temperatures are calculated by assuming that the heat generation in fuel meat is constant along the radial direction and considering one dimensional heat conduction. An axial fuel plate temperature distribution is calculated from local bulk temperatures of the coolant and axial peaking factors. In case of some kinds of fuel plates which have different heat generation rate one another, exist in a fuel element, or right-hand side and left-hand side of the fuel plate cooling conditions are different due to different configuration of coolant channels or different coolant velocities, the code can calculate temperature distribution of each fuel plate. In case of some kinds of fuel elements exist in a core, the code is also able to calculate temperature distribution of each fuel element by using power distribution factors etc..

Given the fuel meat material (choice U-Al-alloy, U-Al$_x$-Al) and the uranium density, the code calculates thermal conductivities of the fuel meat. Thermal conductivities of the fuel meat can be also inputted by data table. The properties of light water, heavy water and aluminum alloy are already given in the code.

2.2 Fuel rod temperature calculation

Fuel rod (pin) temperatures are calculated by assuming that the heat generation in fuel meat (pellet) is constant along the radial direction and considering one dimensional heat conduction. An axial fuel rod (pin) temperature distribution is calculated from local bulk temperatures of the coolant and axial peaking factors. In case of some kinds of fuel rods (pins) which have different heat generation rate one another, exist in a fuel element, the code can calculate temperature distribution of each fuel rods (pins). In case of some kinds of fuel elements exist in a core, the code is also able to calculate temperature distribution of each fuel element by using power distribution factors etc..

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Thermal conductivities of the fuel meat (pellet) must be inputted by data table. Thermal conductivities of the cladding must be also inputted by data table. The properties of light water, heavy water are already given in the code as described above.

2.3 Cooling system temperature calculation

In addition to the fuel plate temperature calculation, coolant temperatures of the primary and the secondary cooling system can be calculated by the COOLOD-N2 code. In this calculation, heat loss from the surface of piping, heat exchanger and so on are neglected.

Counter flow type cooling tower, and heat exchangers of counter flow type, parallel flow type and shell & tube type are treated in the code.

2.4 ONB temperature, Flow instability, DNB heat flux and Pressure drop

The code has capabilities of calculating the ONB temperature, heat flux at onset of flow instability and DNB heat flux which are important to confirm safety of the reactor. The code also has a capability of calculating pressure drops and local pressures in the core which are required to calculate above value. As flow direction in the core, downward flow, upward flow and horizontal flow are treated in the code.

2.5 Natural convection cooling

In general, pool type research reactors have a natural convection cooling mode as well as a forced convection cooling mode. In the natural convection cooling mode, the core flow is an upward flow, which is supplied by the downflow through a natural circulation valve, through a core bypass and so on. The driving force for the natural circulation is calculated by the difference between the outlet water density of the core flow heated by core power and the inlet water density through a core bypass or through a natural circulation valve, in the COOLOD-N2 code. See section 3.6

2.6 Heat transfer package

A "Heat Transfer Package" is a sub-program for calculating heat transfer coefficient, ONB temperature, heat flux at onset of flow instability and DNB heat flux. The "Heat transfer package" was especially developed for research reactors which are operated under low pressure and low temperature conditions using plate-type fuel, just like as the JRR-3M[6]. Heat transfer correlations adopted in the "Heat Transfer Package" were obtained or estimated based on the heat transfer experiments in which thermal-hydraulic features of the upgraded JRR-3 core were properly reflected. The "Heat Transfer Package" is applicable to, not only upward flow, but also downward flow. See section 3.3.

3. Calculation models

3.1 Calculation model for temperature distribution in fuel plates

Assuming that the heat generation in fuel meat is constant along the radial (thickness) direction ($\dot{q}_U = q_U / y_U =$ constant), and considering one dimensional heat conduction, temperature distribution in fuel plates are calculated as follows. Figure 1 shows calculation model of temperature distribution in fuel plates.

(1) Coolant bulk temperature : T_b

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(1) Coolant bulk temperature : T_b

$$T_b = T_{in} + F_b \frac{1}{G\,A\,Cp}\int_0^L Q(Z)dZ \tag{3.1.1}$$

(2) Clad outer surface temperature : T_W

$$T_W = T_b + F_f\,\frac{q_W}{h_W} \tag{3.1.2}$$

$$q_W = q_U$$

(3) Clad inner surface temperature : T_{WB}

$$T_{WB} = T_W + F_W\,\frac{q_U\,y_W}{k_W} \tag{3.1.3}$$

(4) Fuel meat surface temperature : T_{BU}

$$T_{BU} = T_{WB} + F_B\,\frac{q_U\,y_B}{k_B} \tag{3.1.4}$$

(5) Fuel meat maximum temperature : T_{U0}

$$T_{U0} = T_{BU} + F_U\,\frac{\dot{q}_U}{2k_U}\,{y_U}^2 \tag{3.1.5}$$

$$q_U = \dot{q}_U\,y_U$$

If the cooling condition of right hand side and left hand side of the fuel plate are different, the COOLOD-N2 code calculates a fuel meat maximum temperature until the fuel meat maximum temperature of right hand side and left hand side are equal by changing the location of maximum temperature point. If the cooling conditions of right hand side and left hand side of the fuel plate are equal, then the fuel maximum temperature appears center of the fuel meat.

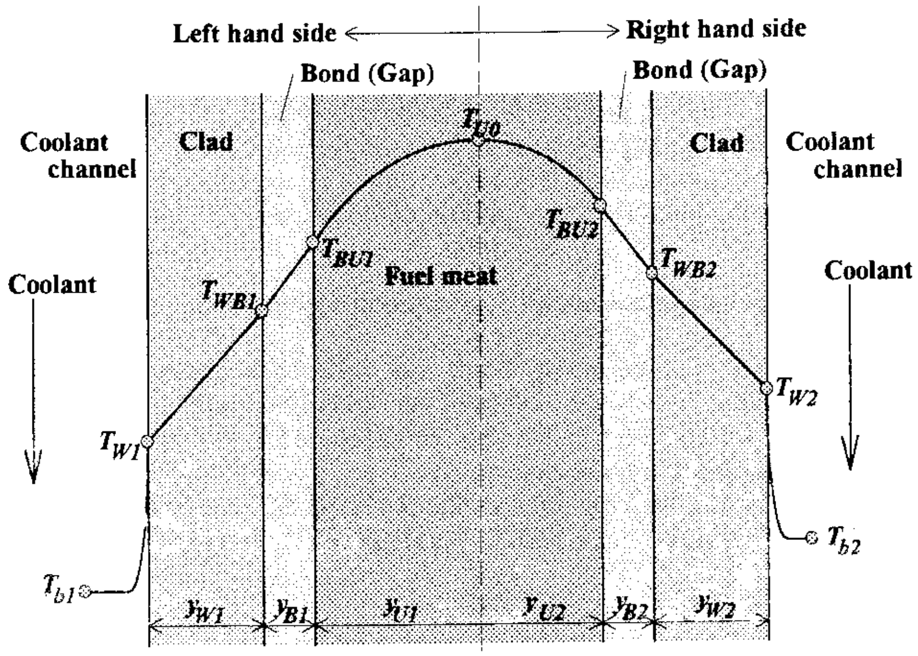

Fig. 1 Fuel plate temperature calculation model

3.2 Calculation model for temperature distribution in fuel rods

Assuming that the heat generation in fuel meat is constant along the radial direction ($\dot{q}_U = 2q_U / y_U$ = constant), and considering one dimensional heat conduction, temperature distribution in fuel rods are calculated as follows. Figure 2 shows calculation model of temperature distribution in fuel rods.

(1) Coolant bulk temperature : T_b

$$T_b = T_{in} + F_b \frac{1}{G A \; Cp} \int_0^L Q(Z) dZ \tag{3.2.1}$$

(2) Clad outer surface temperature : T_W

$$T_W = T_b + F_f \frac{q_W}{h_W}$$
$$q_W = q_U \frac{2\pi y_U}{2\pi y_W} \tag{3.2.2}$$

(3) Clad inner surface temperature : T_{WB}

$$T_{WB} = T_W + F_W \frac{q_U \, y_U}{k_W} \ln \frac{y_W}{y_B}$$
$$= T_W + F_W \frac{\dot{q}_U \, y_U^2}{2k_W} \ln \frac{y_W}{y_B} \tag{3.2.3}$$

(4) Fuel meat surface temperature : T_{BU}

$$T_{BU} = T_{WB} + F_B \frac{q_U}{h_{WB}} \tag{3.2.4}$$

(5) Fuel meat maximum temperature : T_{U0}

$$T_{U0} = T_{BU} + F_U \frac{\dot{q}}{4k_U} {y_U}^2$$
$$q_U = \frac{\dot{q}_U \, y_U}{2} \tag{3.2.5}$$

3.3 Heat transfer calculation model (Heat transfer correlations)

In the COOLOD-N2 code, the COOLOD code original heat transfer correlations as well as the "Heat Transfer Package" which was developed for thermal-hydraulic analysis of research nuclear reactors in which plate-type fuel is employed, can be selected by the input data. Table 1 shows the COOLOD code original heat transfer correlations.

The "Heat Transfer Package" used in the COOLOD-N2 code has been modified taking into account of the characteristics of rod type fuels and is shown as follows.

(1) Single-phase forced-convection flow
Downward flow (G < 0)

$$\mathrm{Nu} = \frac{h\,De}{k} = 4.0 \qquad \text{for laminar flow (Re < 2000)} \qquad (3.3.1)$$

$$\mathrm{Nu} = 0.023\,\mathrm{Re}_b^{0.8}\,\mathrm{Pr}_b^{0.4} \qquad \text{for turbulent flow (Re} \geq 2500\text{)} \qquad (3.3.2)$$

(Dittus-Boelter correlation[8])

Nusselt number is evaluated by interpolation with Eq.(3.3.1) and (3.3.2) for transition region ($2000 \leq \mathrm{Re} < 2500$).

Upward flow (G > 0)

Nu = max[Eq.(3.3.1), Collier correlation] for laminar flow (Re < 2000) (3.3.3)
where Collier correlation[9] is given as follows.

$$\mathrm{Nu} = 0.17\,\mathrm{Re}_f^{0.33}\,\mathrm{Pr}_f^{0.43}\left\{\frac{(\mathrm{Pr}_\ell)_f}{(\mathrm{Pr}_\ell)_w}\right\}^{0.25}\left\{\frac{\rho^2\beta De^3 g_c (T_w - T_\ell)}{\mu^2}\right\}_f^{0.1} \qquad (3.3.4)$$

Nusselt number is evaluated by Eq.(3.3.2) for turbulent flow region ($\mathrm{Re} \geq 2500$).

Nusselt number is evaluated by interpolation with Eq.(3.3.4) and (3.3.2) for transition region ($2000 \leq \mathrm{Re} < 2500$).

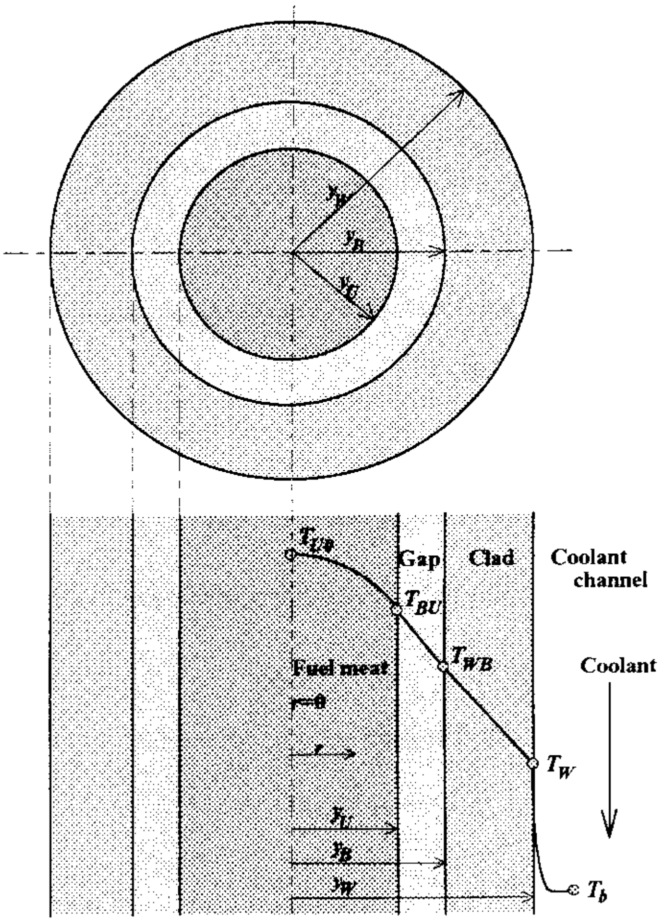

Fig. 2 Fuel rod temperature calculation model

Table 1 COOLOD code original heat transfer correlations

CGS unit

<table>
<tr><th colspan="2">Heat transfer mode</th><th>Heat transfer correlation</th><th>Note</th></tr>
<tr><td colspan="2">Single-phase forced-convection</td><td>$\mathrm{Nu} = H_1 \times \left(\mathrm{Re}^A - H_2\right) \times \mathrm{Pr}^B \left[1.0 + H_3 \left(\frac{De}{Z}\right)^C\right] \times \left(\frac{\mu_b}{\mu_w}\right)^D$</td><td>$A$, B, C, D and H_1, H_2, H_3 are given by input data.</td></tr>
<tr><td colspan="2">ONB temperature</td><td>$q = 0.025293\, P^{1.156} \left[\frac{9}{5}\left(T_{ONB} - T_{sat}\right)\right]^{\frac{2.1615}{P^{0.0234}}}$</td><td>Bergles-Rosenow</td></tr>
<tr><td rowspan="2">Nucleate boiling</td><td>Subcooled</td><td>$q_t = q_c + q_b$
$q_c = 0.023 \frac{k}{De} \mathrm{Re}^{0.8}\, \mathrm{Pr}^{0.4} \left(\Delta T_{sat} + \Delta T_{sub}\right)$
$q_b = 4.50 e^{\frac{P}{20}} \frac{\Delta T_{sat}^{3.6}}{36000}$</td><td>Sato-Matsumura</td></tr>
<tr><td>Saturated</td><td>$\Delta T_{sat} = 11.2951\, q_t^{0.25} e^{\frac{-P}{63.0}}$</td><td>Jens-Lottes</td></tr>
<tr><td colspan="2" rowspan="3">DNB heat flux</td><td>$q_{DNB} = 478800(1 + 0.0365\, v)(1 + 0.00507 \Delta T_{sub})$
$\times (1 + 0.0131 P)$</td><td>Mirshak, Durant and Towell
IHTC = 2 (CARD G1)</td></tr>
<tr><td>$q_{DNB} = \left(10890 \frac{De}{De + \frac{P_H}{\pi}} + 48 \frac{v}{De^{0.6}}\right)$
$\times \left(102.6 \ln P - 97.2 \frac{P}{P+15} - \frac{v}{2.22} + 32 - (T_b)_{DNB}\right)$</td><td>Bernath
IHTC = 3 (CARD G1)
De ; (ft)
P_H ; (ft)
$(T_b)_{DNB}$;(°F)</td></tr>
<tr><td>$q_{DNB} = 145.4\, \theta_{(P)} \left[\frac{1 + 2.5 v^2}{\theta_{(P)}}\right]^{\frac{1}{4}} \times \left(1 + 15.1 \frac{Cp \Delta T_{sub}}{\lambda \sqrt{P}}\right)$
$\theta_{(P)} = 0.99531 P^{\frac{1}{3}} \left(1 - \frac{P}{P_c}\right)^{\frac{3}{4}}$</td><td>Labuntsov
IHTC = 1 (CARD G1)
q_{DNB}; (w/cm^2)
P ; (bar)
P_c ; (bar)
v ; (m/s)
Cp ; (kJ/kg K)</td></tr>
</table>

(2) Nucleate boiling heat transfer

ONB Temperature (Bergles-Rohsenow correlation[10])

$$q = 911 P^{1.156} \left\{\frac{9}{5}\left(T_{ONB} - T_S\right)\right\}^{\frac{2.16}{P^{0.0234}}} \times \frac{1.163}{1000} \tag{3.3.5}$$

Subcooled nucleate boiling (Modified Chen correlation[9],[11])

$$\begin{aligned} q &= 0.023 \mathrm{Re}_b^{0.8}\, \mathrm{Pr}_b^{0.4} \frac{k}{De}\left(T_W - T_\ell\right) \\ &+ S\, 0.00122 \frac{k_f^{\,0.79} Cp_f^{\,0.45} \rho_f^{0.49} \left(T_W - T_S\right)^{1.24} \left(P_W - P\right)^{0.75}}{\sigma^{0.5} \mu_f^{0.29} h_{fg}^{\,0.24} \rho_g^{0.24}} \end{aligned} \tag{3.3.6}$$

where

$$S=\frac{1}{1+0.12\,\mathrm{Re}'^{1.14}} \qquad \mathrm{Re}' < 32.5$$

$$S=\frac{1}{1+0.42\,\mathrm{Re}'^{0.78}} \qquad 32.5 \le \mathrm{Re}' < 70.0 \qquad \mathrm{Re}'=\frac{|G|De}{\mu_f}\times 10^{-4}$$

$$S=0.1 \qquad 70.0 \le \mathrm{Re}'$$

Saturated nucleate boiling (Chen correlation[9],[11])

$$\begin{aligned} q &= F\,0.023\left\{\mathrm{Re}_f(1-x)\right\}^{0.8}\mathrm{Pr}_f^{0.4}\frac{k_f}{De}(T_W-T_S) \\ &+ S\,0.00122\frac{k_f^{\,0.79}Cp_f^{\,0.45}\rho_f^{0.49}(T_W-T_S)^{1.24}(P_W-P)^{0.75}}{\sigma^{0.5}\mu_f^{0.29}h_{fg}^{\,0.24}\rho_g^{0.24}} \end{aligned} \tag{3.3.7}$$

where

$$F=1.0 \qquad \frac{1}{X_{tt}} \le 0.1$$

$$F=2.35\left(\frac{1}{X_{tt}}+0.213\right)^{0.736} \qquad \frac{1}{X_{tt}} > 0.1$$

$$\frac{1}{X_{tt}}=\left(\frac{x}{1-x}\right)^{0.9}\left(\frac{\rho_f}{\rho_g}\right)^{0.5}\left(\frac{\mu_f}{\mu_g}\right)^{0.1}$$

$$S=\frac{1}{1+0.12\,\mathrm{Re}'^{1.14}} \qquad \mathrm{Re}' < 32.5$$

$$S=\frac{1}{1+0.42\,\mathrm{Re}'^{0.78}} \qquad 32.5 \le \mathrm{Re}' < 70.0 \qquad \mathrm{Re}'=\frac{|G(1-x)|De}{\mu_f}\times F^{1.25}\times 10^{-4}$$

$$S=0.1 \qquad 70.0 \le \mathrm{Re}'$$

(3) DNB heat flux[12],[13]

$$q^*_{DNB,1}=0.005\left|G^*\right|^{0.611} \tag{3.3.8}$$

$$q^*_{DNB,2}=\frac{A}{A_H}\frac{\Delta h_i}{h_{fg}}\left|G^*\right| \tag{3.3.9}$$

$$q^*_3=0.7\left(\frac{A}{A_H}\right)\frac{\sqrt{W/\lambda}}{\left\{1+(\rho_g/\rho_\ell)^{1/4}\right\}^2} \tag{3.3.10}$$

Downward flow (G < 0)
DNB heat flux is evaluated by min[Eq.(3.3.8), max[Eq.(3.3.9), Eq.(3.3.10)]

Upward flow (G > 0)
DNB heat flux is evaluated by max[Eq.(3.3.8), Eq.(3.3.10)]

DNB heat flux for rod type fuels

In subcooled boiling, the DNB heat flux is a function of the coolant velocity, the degree of subcooling, and the pressure. The correlation used to predict DNB is Lund which was developed from empirical data gathered from an experiment conducted on a test assembly that confirmed to actual fuel bundle in terms of dimension, flow and heat flux. The critical heat flux is given by[7] ;

$$q_{Lund} = 0.5 f_c \, \rho V_{inter} Cp \, (T_c - T_{out}) \tag{3.3.11}$$

where
f_c : Friction factor for the channel between fuel rods (-).
$= 0.55 \mathrm{Re}_{inter}^{-0.37}$
Re_{inter} : Reynolds number for the interrod channel (-).
$= 2 \rho V_{inter} Dr(S-1) / \mu_{sat}$
V_{inter} : Interrod channel velocity (m/s).
$= V\left[1.0 - 0.98 \, e^{-2.2(S-1)}\right]$
S : Pitch-to-diameter ratio (-)
Dr : Rod diameter (m)
V : Average velocity (m/s)
ρ : Density (kg/m^3)
μ_{sat} : Viscosity at saturation temperature (Pa·s)
Cp : Constant pressure specific heat (kJ/kg)
T_{out} : Temperature at outlet of cooling channel (°C)
T_c : Critical wall temperature (°C)

The critical wall temperature is given by

$$T_c = T_{sat}\left(1 + 6\sqrt{\theta_c}\right)$$

where
T_{sat} : Saturation temperature (°C)
θ_c : $q_c \sigma_{sat} / p \mu_{sat} g h_{fg}$
σ_{sat} : Saturation surface tension (N/m)
p : Absolute pressure (kg/m^{2}abs.)
h_{fg} : Heat of vaporization (kJ/kg)

(4) Heat flux at Onset of Flow Instability

The criterion for the onset of flow instability (flow excursion) has been obtained for rectangular channels by Whittle and Forgan[14].

$$\frac{T_{out} - T_{in}}{T_S - T_{in}} = \frac{1}{1 + \eta \dfrac{D_H}{L_H}} \tag{3.3.12}$$

Energy balance is given by

$$q A_H = Cp \, (T_{out} - T_{in}) \, |G| \tag{3.3.13}$$

From Eq.(3.3.12) and (3.3.13), a following correlation was obtained.

$$q = \frac{1}{A_H} \frac{Cp \, (T_S - T_{in})}{1 + \eta \dfrac{D_H}{L_H}} |G| = \frac{Cp \, \Delta T_{sub}}{A_H + 4 \eta A} |G| \tag{3.3.14}$$

The bubble detachment parameter η was determined empirically to be 25[14].

3.4 Pressure drop calculation model

3.4.1 Friction loss coefficient[15]

(1) Friction loss coefficient for laminar flow ($\mathrm{Re} \leq 2500$)

$$F = \frac{Cb}{\mathrm{Re}} \tag{3.4.1}$$

where Cb is a factor which depends on the configuration of the channel.

$Cb = 64.0$	for tube
$Cb = 56.9$	for square
$Cb = 96.0$	for rectangular

(2) Friction loss coefficient for turbulent flow ($\mathrm{Re} > 2500$)
Following correlations can be selected.

Blasius correlation

$$F = 0.3164\,\mathrm{Re}^{-0.25} \tag{3.4.2}$$

Kärman-Nikuradse correlation

$$\frac{1}{\sqrt{F}} = 2.0\log_{10}\left(\mathrm{Re}\sqrt{F}\right) - 0.8 \tag{3.4.3}$$

Cole-Brook correlation

$$\frac{1}{\sqrt{F}} = 2.0\log_{10}\left[\frac{\varepsilon / De}{3.71} + \frac{2.51}{\mathrm{Re}\sqrt{F}}\right] \tag{3.4.4}$$

3.4.2 Pressure drop calculation model

A pressure drop calculation model for the COOLOD-N2 code is shown in Figure 3. In this calculation model, a pressure drop due to friction loss is calculated as a pressure drop inside the segment. A a pressure drop due to geometry change is calculated as a pressure drop between segment n and segment n+1. A local pressure $P_{n,1}$ and $P_{n,2}$ of n-th segment is calculated as follows by using Bernoulli's theorem.

$$P_{n,1} = P_{n-1,2} + \frac{1}{2g}\left(\bar{\rho}_{n-1} v_{n-1}^2 - \bar{\rho}_n v_n^2 - \rho_n \zeta_n \bar{v}_{n+1}^2\right) \tag{3.4.5}$$

$$P_{n,2} = P_{n,1} + \bar{\rho}_n\left(L \Delta Z_n - F_n \frac{\Delta Z_n}{De_n}\frac{\bar{v}_n^2}{2g}\right) \tag{3.4.6}$$

where

$\bar{\rho} = \dfrac{\rho_n + \rho_{n+1}}{2}$: Average density of the segment n

L : Flow direction flag = -1 : Upward flow
= 0 : Horizontal flow
= 1 : Downward flow

$$\bar{v} = \max(v_n,\ v_{n+1})$$

and $P_{0,2} = P_{in}$ is given by input data. In the non-heated channel, $\rho_n = \rho_{n+1} = \bar{\rho}_n$.

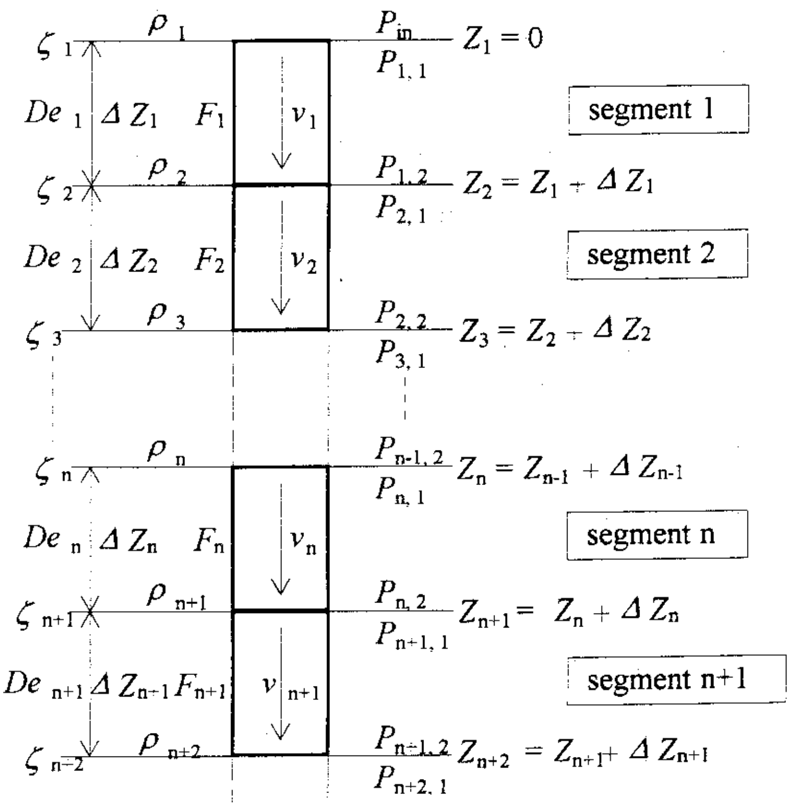

Fig. 3 Pressure drop calculation model for COOLOD-N2 code

3.5 Cooling tower and heat exchanger calculation model

3.5.1 Cooling tower temperature calculation model[14]

In case of considering a heat exchange between air and water at the cooling tower, transfer unit U of the cooling tower is expressed as follows.

$$U = \frac{K_a}{G} V \tag{3.5.1}$$

where

K_a : Overall volumetric heat transfer coefficient based on enthalpy difference (kcal/m^{3}hΔi)

G : Air flow rate (kg'/h)*

V : Volume of the cooling tower (m^3)

Transfer unit U is also expressed as follows.

$$U = N \int_{in}^{out} \frac{-dT_b}{h_b - h} \tag{3.5.2}$$

where

N : Water air ratio
h : Enthalpy of air (kcal/kg')*
h_b : Enthalpy of saturated air at water temperature T_b
T_b : Temperature at the cooling tower (°C)

* kg' means weight of dry air in the wet air

Inlet and outlet temperatures of the cooling tower are calculated from Eq. (3.5.1) and Eq. (3.5.2) by using a wet-bulb temperature, a water-air ratio N and dummy inlet and outlet temperature of the cooling tower $T_{b,in}$, $T_{b,out}$ until Eq. (3.5.1) equal to Eq. (3.5.2) where dT_b ($T_{b,in}$ - $T_{b,out}$) = constant.

3.5.2 Heat exchanger temperature calculation model[17]

Inlet and outlet temperatures of the primary coolant in a heat exchanger are calculated from the temperature T'_1 of the secondary coolant.

$$T_{in} = T'_1 + \frac{\Delta T}{E_A} \tag{3.5.3}$$

$$T_{out} = T_{in} - \Delta T \tag{3.5.4}$$

where

ΔT : Temperature difference between inlet and outlet temperature of primary coolant (°C)
T_{in} : Inlet temperature of primary coolant (°C)
T_{out} : Outlet temperature of primary coolant (°C)
E_A : Exchanger effectiveness

If a heat exchanger type is different, then E_A has also different value. E_A is calculated as follows.

(1) Counter flow type heat exchanger

$$E_A = \frac{1-\exp\left(-(\mathrm{NTU})_A(1-R_A)\right)}{1-R_A\exp\left(-(\mathrm{NTU})_A(1-R_A)\right)} \tag{3.5.5}$$

(2) Parallel flow type heat exchanger

$$E_A = \frac{1-\exp\left(-(\mathrm{NTU})_A(1-R_A)\right)}{1+R_A} \tag{3.5.6}$$

(3) Sell and tube type heat exchanger (Shell side m pass, tube side $2m$ pass)
1) $m = 1$

$$E_A = \frac{2}{(1+R_A)+\sqrt{1+R_A^2}\,\dfrac{1+\exp(-\Gamma)}{1-\exp(-\Gamma)}} \tag{3.5.7}$$

where

$$\Gamma = (\mathrm{NTU})_A\sqrt{1+R_A}$$

2) $m > 2$

$$E_A = \frac{\left(\dfrac{1-E_a R_A}{1-E_a}\right)^m - 1}{\left(\dfrac{1-E_a R_A}{1-E_a}\right)^m - R_A} \tag{3.5.8}$$

where

R_A : Capacity rate ratio of primary coolant and secondary coolant $= \dfrac{W_1}{W_2}$

$(NTU)_A$: Number of transfer unit $= \dfrac{U A_H}{W}$

U : Overall heat transfer coefficient (kcal/m^{2}h°C)

W : Heat capacity $= G\,A\,Cp$ (kcal/h°C)

A_H : Heat transfer area of the heat exchanger (m^2)

3.6 Natural convection cooling calculation model

In the natural convection cooling model, m kinds of heated channels and n kinds of core bypass channels (non-heated channel) are considered in the COOLOD-N2 code. A basic equation used in this calculation model is a equation of conservation of mass between heated channels and non-heated channels.

A sum of mass flow rates G_j for core bypass channels is equal to a sum of mass flow rates G_i for heated channels.

$$\sum_{i=1} G_i = \sum_{j=1} G_j = G_0 \tag{3.6.1}$$

On the other hand, the relation between a pressure drop of the heated channel in the core ΔP_{ci} (i=1 to i_{max}) , a pressure drop of the non-heated channel (core bypass) ΔP_{bj} (j=1 to j_{max}), and a driving force ΔP_{di} (i=1 to i_{max}) are expressed as shown below.

$$\Delta P_{ci}(G_i) + \Delta P_{bj}(G_j) = \Delta P_{di}(G_i) \tag{3.6.2}$$

$$\Delta P_{bj}(G_j) = \Delta P_b \quad \text{(constant)} \tag{3.6.3}$$

The driving force ΔP_{di} for the natural circulation is expressed with the difference between the water density ρ'_i of heated channel and the water density ρ through non-heated channel (core bypass), and is shown below.

$$\begin{aligned} \Delta P_{di} &= \int_0^{L_i} (\rho - \rho'_i)\,dx \\ &= \sum_{m=1}^{m_{max}} (\rho \ell_{im} - \rho'_{im} \ell_{im}) \\ &= \rho L_i - \sum_{m=1}^{m_{max}} \rho'_{im} \ell_{im} \end{aligned} \tag{3.6.4}$$

where

L_i : Heated length of i-th channel (m) $= \sum_{m=1}^{m_{max}} \ell_{im}$

ℓ_{im} : Heated length of m-th segment of i-th channel (m)

The driving force is calculated by the coolant temperature distribution of the heated channel which depends on the core power.

If nucleate boiling would occur in the core, the right hand side of Eq.(3.6.4) will be replaced by following equation.

$$\rho'_{im}\ \ell_{im} = (1-\alpha_{im})\rho_{\ell im}\ \ell_{im} \tag{3.6.5}$$

where

$\rho_{\ell im}$: Saturated water density of m-th segment of i-th heated channel (kg/m^3)

α_{im} : Void fraction of m-th segment of i-th heated channel

In this calculation model, the condition of onset of nucleate boiling is defined as follows[18].

$$\mathrm{Nu}_B = \frac{q\,De}{K_b(T_S - T_b)} \geq 455 \quad ;\ \mathrm{Pe} \leq 7000 \tag{3.6.6}$$

$$\mathrm{St}_B = \frac{q}{GCp_b(T_S - T_b)} \geq 0.0065 \ ;\ \mathrm{Pe} > 7000 \tag{3.6.7}$$

The void fraction is calculated by following correlation.

(1) Void fraction under subcooled boiling region (AHMAD correlation[19])

$$\alpha = \frac{x}{x + s(1-x)\rho_g / \rho_\ell} \tag{3.6.8}$$

$$s = \left(\frac{\rho_\ell}{\rho_g}\right)^{0.205} \left(\frac{G\,D}{\mu_\ell}\right)^{-0.016} \tag{3.6.9}$$

(2) Void fraction under subcooled boiling region (Zuber correlation[20])

$$\alpha = \frac{x}{1.13\left(x\dfrac{\rho_\ell - \rho_g}{\rho_\ell} + \dfrac{\rho_g}{\rho_\ell}\right) + C_b \dfrac{\rho_g}{G}\left[\dfrac{\sigma(\rho_\ell - \rho_g)g}{\rho_\ell^2}\right]^{1/4}} \tag{3.6.10}$$

(3) Void fraction under subcooled boiling region (Combination of Eq.(3.6.9) and Eq.(3.6.10)).

$G < G_{LIM}$ then Eq.(3.6.9)

$G \geq G_{LIM}$ then Eq.(3.6.10)

where, G_{LIM} (kg/s) is given by input data. The range of G_{LIM} is 500 to 1500 (kg/m^{2}s).

(4) Void fraction under nucleate boiling region (Zuber correlation[21])

$$\alpha = \frac{x}{1.13\left(x_{eq}\dfrac{\rho_\ell - \rho_g}{\rho_\ell} + \dfrac{\rho_g}{\rho_\ell}\right) + C_b \dfrac{\rho_g}{G}\left[\dfrac{\sigma(\rho_\ell - \rho_g)g}{\rho_\ell^2}\right]^{1/4}} \tag{3.6.11}$$

where

$$x = \frac{x_{eq} - x_{eqB}\, e^{x_{eq}/x_{eqB}-1}}{1 - x_{eqB}\, e^{x_{eq}/x_{eqB}-1}}$$

$$x_{eq} = \frac{q_w P_h Z/(GA) - Cp_b(T_S - T_{bB})}{h_{fg}} \tag{3.6.12}$$

x_{eqB} : Quality at the point of onset of nucleate boiling
T_{bB} : Coolant temperature at the point of onset of nucleate boiling (°C)
Z : Distance from the point of onset of nucleate boiling (m)
P_h : Heated perimeter (m)
C_b : Zuber's coefficient = 1.18 or 1.41

4. Properties used in the code

4.1 Thermal conductivities of fuel meat (plate-type-fuel)

Given the fuel meat material (choice U-Al-alloy, U-Al -Al) and the uranium density, the thermal conductivity of the fuel meat is calculated by the code. Thermal conductivities used in the code are shown below.

(1) Thermal conductivity of U-Al alloy[22] :k_{u0}

$k_{u0} = 0.415 - 1.0 \times 10^{-4}\, T_u$; ($20 < T_u < 640$ °C)
$k_{u0} = 0.135$; ($T_u > 640$ °C)

where

k_{u0} : Thermal conductivity of U-Al alloy (cal/s cm°C)
T_u : Temperature of U-Al alloy (°C)

(2) Thermal conductivity of U-Al_x dispersion fuel[23] :k_{u1}

$k_{u1} = k'_{u1}\,(1-P)^{3/2}$
$k'_{u1} = 2.16546 - 2.765x$

where

k_{u1} : Thermal conductivity of U-Al_x dispersion fuel (W/cm°C)
x : Weight fraction of uranium in the fuel meat = $\dfrac{\rho}{0.8\,\rho + 2.7(1-P)}$
ρ : Uranium density of U-Al_x dispersion fuel
P : Porosity

where, G_{LIM} (kg/s) is given by input data. The range of G_{LIM} is 500 to 1500 (kg/m^{2}s).

(4) Void fraction under nucleate boiling region (Zuber correlation[21])

$$\alpha = \frac{x}{1.13\left(x_{eq}\frac{\rho_\ell - \rho_g}{\rho_\ell} + \frac{\rho_g}{\rho_\ell} \right) + C_b \frac{\rho_g}{G}\left[\frac{\sigma(\rho_\ell - \rho_g)g}{\rho_\ell^2} \right]^{1/4}} \qquad (3.6.11)$$

where

$$x = \frac{x_{eq} - x_{eqB}\, e^{x_{eq}/x_{eqB} - 1}}{1 - x_{eqB}\, e^{x_{eq}/x_{eqB} - 1}},$$

$$x_{eq} = \frac{q_w P_h Z/(GA) - Cp_b(T_S - T_{bB})}{h_{fg}} \qquad (3.6.12)$$

x_{eqB}	: Quality at the point of onset of nucleate boiling
T_{bB}	: Coolant temperature at the point of onset of nucleate boiling (°C)
Z	: Distance from the point of onset of nucleate boiling (m)
P_h	: Heated perimeter (m)
C_b	: Zuber's coefficient = 1.18 or 1.41

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$$k_{u0} = 0.135 \qquad ; (T_u > 640\ °C)$$

where

k_{u0}	: Thermal conductivity of U-Al alloy (cal/s cm°C)
T_u	: Temperature of U-Al alloy (°C)

(2) Thermal conductivity of U-Al$_x$ dispersion fuel[23] :k_{u1}

$$k_{u1} = k'_{u1}\,(1-P)^{3/2}$$
$$k'_{u1} = 2.16546 - 2.765x$$

where

k_{u1}	: Thermal conductivity of U-Al$_x$ dispersion fuel (W/cm°C)
x	: Weight fraction of uranium in the fuel meat = $\dfrac{\rho}{0.8\,\rho + 2.7(1-P)}$
ρ	: Uranium density of U-Al$_x$ dispersion fuel
P	: Porosity

Thermal conductivities of the fuel meat can be also inputted by data table.

4.2 Thermal conductivities of aluminum[24] (plate-type-fuel) :k_{Al}

$$k_{Al} = 0.390 + 2.22 \times 10^{-4}\, T_{Al} - 3.79 \times 10^{-7}\, T_{Al}^2 + 2.42 \times 10^{-10}\, T_{Al}^3 \quad ; (20 < T_{Al} < 649\ °C)$$

$$k_{Al} = 0.170 \quad ; (T_{Al} > 649\ °C)$$

where

T_{Al} : Temperature of aluminum clad (°C)

4.3 Thermal conductivities of bond layer[25] (plate-type-fuel) :k_B

$$k_B = 0.123804 \times 10^{-4} - 0.593896 \times 10^{-7} \;\; - 0.37228 \times 10^{-10}\, T_B \quad ; (18 < T_B < 520\ °C)$$

where

T_B : Temperature of bond layer (°C)

As for the thermal conductivity of bond layer, the thermal conductivity of Xe is used in the code.

4.4 Properties of light water and heavy water[26],[27]

The properties of light water, heavy water used in the code are listed in Table 2 and Table 3.

Table 2 Properties of Light Water

Temp (°C)	Specific weight (kg/m^3)	Specific heat (kcal/kg°C)	Kinematic viscosity (m^2/s) x 10^{-6}	Thermal Conductivity (kcal/mh°C)	Thermal diffusivity (m^2/h) x 10^{-4}	Dynamic viscosity (kg s/m^2) x 10^{-4}	Surface tension (kg/m) x 10^{-3}	Saturated pressure (kg/cm^2)	Enthalpy (kcal/kg) Saturated water	Enthalpy (kcal/kg) Saturated vapor
0	999.9	1.008	1.79	0.489	4.85	1.829	7.72	0.006228	0.00[*1]	597.49[*1]
10	999.7	1.002	1.31	0.505	5.04	1.336	7.56	0.012512	10.030	601.87
20	998.2	0.999	1.00	0.518	5.08	1.022	7.39	0.023826	20.030	606.23
30	995.7	0.998	0.803	0.531	5.34	0.816	7.24	0.043251	30.014	610.57
40	992.3	0.998	0.668	0.543	5.48	0.676	7.08	0.075204	39.995	614.88
50	988.1	0.999	0.555	0.552	5.59	0.559	6.90	0.12578	49.980	619.13
60	983.2	1.000	0.480	0.562	5.72	0.482	6.74	0.20313	59.972	623.32
70	977.8	1.001	0.417	0.571	5.85	0.416	6.55	0.31776	69.975	627.43
80	971.8	1.003	0.368	0.578	5.93	0.365	6.37	0.48294	79.993	631.45
90	965.3	1.005	0.328	0.583	6.01	0.323	6.19	0.71491	90.031	635.36
100	958.4	1.007	0.297	0.586	6.08	0.290	6.00	1.03323	100.092	639.15
120	943.1	1.014	0.247	0.589	6.16	0.238	5.55	2.0246	120.311	646.31
140	926.1	1.023	0.209	0.588	6.21	0.197	5.10	3.6850	140.705	652.78
160	907.3	1.037	0.186	0.585	6.22	0.172	4.65	6.3025	161.334	658.43
180	886.9	1.054	0.168	0.578	6.25	0.152	4.17	10.224	182.267	663.10
200	864.7	1.075	0.155	0.568	6.11	0.137	3.70	15.855	203.585	666.60
220	840.3	1.102	0.146	0.544	5.98	0.125	3.24	23.656	225.393	668.75
240	814	1.136	0.139	0.537	5.81	0.115	2.78	34.138	247.827	669.30
260	784	1.183	0.133	0.517	5.57	0.106	2.32	47.869	271.076	667.91
280	751	1.250	0.128	0.493	5.25	0.098	1.85	65.486	295.414	664.09
300	712	1.36	0.13	0.462	4.77	0.091	1.40	87.621	321.261	657.07
320	667	1.54	0.13	0.423	4.12	0.083	0.95	115.12	349.337	645.76

*1) 0.01°C

Table 3 Properties of Heavy Water

Temp (°C)	Specific weight (kg/m^3)	Specific heat (kcal/kg°C)	Kinematic viscosity (m^2/s) x 10^{-6}	Thermal Conductivity (kcal/mh°C)	Thermal diffusivity (m^2/h) x 10^{-4}	Dynamic viscosity (kg s/m^2) x 10^{-4}	Surface tension (kg/m) x 10^{-3}	Saturated pressure (kg/cm^2)	Enthalpy (kcal/kg) Saturated water	Enthalpy (kcal/kg) Saturated vapor
0	1105	1.015	0.7444	0.4782	4.266	0.7556	7.72	0.006954	1.201[*1]	554.65[*1]
10	1105	1.012	1.278	0.4882	4.364	1.319	7.56	0.01063	6.270	556.73
20	1105	1.009	1.135	0.5031	4.515	1.168	7.39	0.02067	16.373	560.83
30	1103	1.006	0.9044	0.5159	4.651	0.928	7.24	0.03827	26.483	564.87
40	1100	1.003	0.7297	0.5268	4.774	0.746	7.08	0.06780	36.477	568.86
50	1196	1.001	0.6034	0.5360	4.885	0.616	6.90	0.1153	46.494	572.78
60	1091	0.9991	0.5105	0.5434	4.985	0.520	6.74	0.1890	56.492	576.60
70	1085	0.9974	0.4405	0.5493	5.076	0.448	6.55	0.2994	66.473	580.37
80	1078	0.9959	0.3864	0.5537	5.157	0.392	6.37	0.4600	76.440	584.07
90	1071	0.9946	0.3438	0.5568	5.229	0.349	6.19	0.6871	86.393	587.68
100	1062	0.9937	0.3095	0.5586	5.291	0.314	6.00	1.001	96.331	591.22
120	1044	0.9932	0.2584	0.5587	5.387	0.262	5.55	1.984	116.189	598.00
140	1024	0.9959	0.2225	0.5547	5.438	0.226	5.10	3.641	136.061	604.38
160	1003	1.003	0.1963	0.5470	5.343	0.201	4.65	6.266	156.036	610.28
180	981.5	1.018	0.1765	0.5359	5.361	0.183	4.17	10.22	176.237	615.65
200	959.6	1.044	0.1611	0.5216	5.209	0.172	3.70	15.92	196.833	620.35
220	938.1	1.083	0.1489	0.5044	4.966	0.164	3.24	23.84	219.016	623.27
240	917.1	1.140	0.1390	0.4841	4.630	0.144	2.78	34.51	239.985	623.46
260	897.0	1.220	0.1368	0.4607	4.210	0.163	2.32	48.47	262.938	623.01
280	878.0	1.328	0.1339	0.4339	3.722	0.168	1.85	66.31	287.036	619.85
300	860.3	1.470	0.1180	0.4034	3.191	0.177	1.40	88.72	312.704	612.14
320	844.1	1.652	0.1129	0.3688	2.645	0.190	0.95	116.50	340.860	598.43

*1) 5.0°C

5. Input data information for COOLOD-N2

<CARD A> Title card (A72)
TITL : Title for the calculation

<CARD B1> Control card (Free format)
INFORM : Index for input data format (I)
= 0 : COOLOD original type input data (Plate-type-fuel only)
= 1 : COOLOD-N, N2 original type input data
FZ (**CARD F4**) are defined as points.
= 2 : COOLOD-N, N2 original type input data
FZ (**CARD F4**) are defined as segments.

<CARD B2> Control card (Free format)
IAMAX : Number of calculation cases (I) (1=<IAMAX<=10)
(Number of <Card C>
IMAX : Number of calculation points in fuel meat radial direction (I) (1=<IMAX=<5)
JMAX : Number of calculation points for fuel plate axial direction (I)
(1=<JMAX<=21, INFORM = 0, 1 (**CARD B1**))
(1=<JMAX<=20, INFORM = 2 (**CARD B1**))
(Number of **CARD F4**)
NMAX : Number of different fuel elements in the core (I) (1=<NMAX=<5)
NPLOT : Plot option of calculation results (I)
= 0 : No plot
= 1 : Plot of calculation results
KEY(1) : Option for coolant temperature calculation (I)
= 0 : Cooling Tower, Heat Exchanger and Fuel temperature calculation
= 1 : Fuel temperature calculation only (Input data 'Tin' (primary coolant core inlet temperature) is required for calculation)
= -1 : Fuel temperature calculation skip
KEY(2) : Index for flow direction in the core (I)
= -1 : Upflow
= 0 : Horizontal flow
= 1 : Downflow
= 5 : Natural circulation cooling mode
KEY(3) : Index for coolant (I)
= 0 : Light water (H_2O)
= 1 : Heavy water (D_2O)
IDMAX ;Number of division in cladding region **(If IDMAX is positive value, rod type fuel calculation model will be selected.)** *<New>*

<CARD C> Thermal-hydraulic parameter (Free format)
QRR : Reactor thermal power (MW) (R)
PFLOW : Primary coolant flow rate or average coolant velocity in the core (R)
* If KVELO (**CARD G1**)=0, then PFLOW is Volumetric flow rate (m^3/min)
* If KVELO (**CARD G1**)=1, then PFLOW is Mass flow rate (kg/s)
* If KVELO (**CARD G1**)=2, then PFLOW is Average coolant velocity in the core (cm/s)
* If INFORM (**CARD B1**)=0, then PFLOW is Volumetric flow rate (m^3/min)

TIN : If KEY(1)=1 then the Primary coolant core inlet temperature (°C) (R)
If KEY(1)=0 or -1 then the Wet bulb temperature (°C) (R)
DT : Increment of inlet temperature "TIN" (°C) (R)
JAMX : Number of calculation cases for "DT" (I) (Normally : =1)

<CARD D> Cooling Tower and Heat Exchanger data (Free format)
SFLOW : Secondary coolant flow rate (m^3/min) (R)
AFLOW : Air flow rate of the cooling tower (m^3/min) (R)
CTKI : Overall heat transfer coefficient of the cooling tower (kcal/m^3h Δh) (R)
HEKI : Heat transportation coefficient of the heat exchanger (kcal/m^2h°C) (R)
SSCT : Cross sectional area of the cooling tower (m^2) (R)
ZCT : Effective height of the cooling tower (m) (R)
SSHE : Heat transfer area of the heat exchanger (m^2) (R)
IHE : Heat exchanger type (I)
= -1 : Counter flow type
= 0 : Parallel flow type
= m : Shell side m pass and tube side 2*m pass type
* **CARD D** is only used in case of KEY(1)<1(**CARD B1**).

<CARD E1> Heat transfer correlation (Free format)
$H_1, H_2, H_3, A, B, C, D$, ITWC
H_1 -H_3 and A-D (R) and ITWC (I) are shown below.

$$\mathrm{Nu} = <H_1> \times \left(\mathrm{Re}^{<A>} - <H_2>\right) \times \mathrm{Pr}^{<B>} \left[1.0 + <H_3> \left(\frac{De}{Z}\right)^{<C>}\right] \times \left(\frac{\mu_b}{\mu_w}\right)^{<D>}$$

(Single phase heat transfer correlation)
Nu : Nusselt number (-)
Re : Reynolds number (-)
Pr : Prandtl number (-)
De : Equivalent hydraulic diameter (cm)
Z : Distance from inlet of channel (cm)
μ_b : Dynamic viscosity at bulk water temperature (dynes/cm^2)
μ_w : Dynamic viscosity at wall water temperature
(Surface temperature of fuel plate) (dynes/cm^2)
ITWC : Standard temperature for property (I)
= 0 : Properties are evaluated by TWC(0)
TWC(0) = (Core inlet temperature + core outlet temperature)/2.0
= 1 : Properties are evaluated by TWC(1)
TWC(1) = Bulk coolant temperature at Z.
= 2 : Properties are evaluated by TWC(2)
TWC(2) = (TWC(0) + Fuel surface temperature at Z)/2.0
= 3 : Properties are evaluated by TWC(3)
TWC(3) = (TWC(1) + Fuel surface temperature at Z)/2.0
* **CARD E1** is only used for the case of IHTC = 1-3 (**CARD G1**), if IHTC = 4, then **CARD E1** is not used in the calculation, but dummy data are required even in the case of IHTC = 4.

<CARD E2> Core flow condition (Free format)
FRATE : FRATE = (Effective flow rate for fuel plates cooling)
/(Primary coolant flow rate) (-) (R)
VIN : Coolant velocity in the inlet plenum (cm/s) (R)

VOUT : Coolant velocity in the outlet plenum (cm/s) (R)
PRESSIN: Core inlet pressure (kg/cm^{2}abs) (R)
RAMF : Index for straight pipe friction loss for turbulent flow (R)
= -1.0 : Blasius correlation
= 0.0 : Karman-Nikuradse correlation
= ε/De: Cole-Brook correlation. ε/De is a relative roughness.

<CARD F1> Fuel element title card (A40)
TITLN : Title for fuel element

<CARD F2> Fuel element data (Free format)
NPMX : Number of different fuel plates in this kind of fuel element (I)
(Different cooling condition, different configuration) (1=<NPMX=<15)
(Number of **CARD F51-CARD F53**)
NFUEL : Number of this kind of fuel elements in the core (R)
MA : Index for fuel meat material (I)
= 0 : U-Al alloy
= 1 : U-Al_x dispersion type
= 2 : Fuel meat properties are inputted by data table (**CARD F22**)
UDENST: Uranium density in meat (g/cm^3) (R) (For U-Al and U-Alx dispersion type fuel)
POROTY: Porosity (-) (R) (For U-Al_x dispersion type fuel)
IDPMX : Number of different configuration fuel plates in this kind of fuel element (I)
(1=<IDPMX=<5) (Number of **CARD F6**)
IDCMX : Number of different configuration flow channels in this kind of fuel element (I)
(1=<IDCMX=<5) (Number of **CARD F70**, **CRAD F74** or **CARD F76**)
EAREA : Effective flow area for this kind of fuel element (cm) (R)
FRATEN: Flow rate distribution factor for this kind of fuel element (-) (R)
FRATEN = (Flow rate of this kind of fuel element)/(Average flow rate of fuel element)

<CARD FNEW> Rod type fuel element equivalent hydraulic diameter, rod-pitch ratio data card (Free format) *<New>*
WID0 : Equivalent hydraulic diameter or D (diameter)/2.0 (cm) (R)
(for DNB heat flux calculation)
RD0 : Rod diameter (cm) (R)
SSR0 : Rod-pitch ratio (Rod-pitch/Rod diameter) (-) (R)
(for DNB heat flux calculation for Lund correlation)

<CARD F221> Fuel pellet thermal conductivity data card. (If rod-type fuel is selected (IDMAX on **CARD B2** > 0 and MA =2 on **CARD F2**, then this card is required.) *<New>*
N221 : Number of data points (-) (I)
T221 : Temperature (°C) (R)
K221 : Thermal conductivity for fuel pellet ((W/cm K) (R)

<CARD F222> Gap heat transfer data card. (If rod-type fuel is selected (IDMAX on **CARD B2** > 0 and MA =2 on **CARD F2**, then this card is required.) *<New>*
N222 : Number of data points (-) (I)
T222 : Temperature (°C) (R)
K222 : Thermal conductivity for fuel pellet ((W/cm^{2}K)

<CARD F223> Cladding thermal conductivity data card. (If rod-type fuel is selected (IDMAX on **CARD B2** > 0 and MA =2 on **CARD F2**, then this card is required.) *<New>*

N223 : Number of data points (-) (I)
T223 : Temperature (°C) (R)
K223 : Thermal conductivity for fuel pellet ((W/cm K) (R)

<CARD F22> Fuel meat data table (Free format)
NUAL : Number of data sets (I)
TUAL : Temperature (°C) (R)
UAL : Thermal conductivity of the fuel meat (W/cm K)
* If MA<>2(**CARD F21**), then this card is not required.

<CARD F3> Hot channel factors (Free format)
FR : Radial peaking factor (F_R (radial) x F_E (uncertainty)) (R)
FCCOL : Engineering peaking factor for bulk coolant temperature rise (R) (F_b)
FHFLX : Engineering peaking sub-factor for heat flux (R)
(This sub-factor is used in the calculation of DNBR)
FFILM : Engineering peaking factor for film temperature rise (R) (F_f)
FCLAD : Engineering peaking factor for clad temperature rise (R)
FBOND : Engineering peaking factor for bond temperature rise (R)
FMEAT : Engineering peaking factor for fuel meat temperature rise (R)

<CARD F4> Axial peaking factors (Free format)
FZ : Axial peaking factor (R)
* If INFORM = 0 or 1 (**CARD B1**), then FZ is defined as a point ($f(M_i)$).
* If INFORM = 2 (**CARD B1**), then FZ is defined as a segment ($f(S_i)$).
* If INFORM = 0 (**CARD B1**), then following data are not required.
In this case, DDZ is calculated as follows.
DDZ = HB / (JMAX-1) HB : **CARD F6**
DDZ : Distance from point$_i$ (M_i) to point$_{i+1}$ (M_{i+1}) or a segment length (R)
* If INFORM = 1 (**CARD B1**), then DDZ is distance from M_i to M_{i+1} (DDZ = ΔZ_i). In this case DDZ_{JMAX} (ΔZ_{JMAX}) is dummy data.
* If INFORM = 2 (**CARD B1**), then DDZ is a segment length (DDZ = ΔZ_i)

$$\text{* INFORM} = 1 : \sum_{j=1}^{JMAX-1} DDZ_j = HB \qquad \text{HB : CARD F6}$$

$$\text{* INFORM} = 2 : \sum_{j=1}^{JMAX} DDZ_j = HB$$

ZET : Resistance coefficient at point$_i$ (M_i). (R) (Normally : = 0.0)
* If INFORM = 2 (**CARD B1**), then $f(M_i)$ are calculated as follows, using $f(S_i)$.

$$f(M_1) = 2f(S_1) - f(M_2)$$

$$f(M_2) = f(S_1) + \frac{\Delta Z_1}{\Delta Z_1 + \Delta Z_2}\left[f(S_2) - f(S_1)\right]$$

$$f(M_3) = f(S_2) + \frac{\Delta Z_2}{\Delta Z_2 + \Delta Z_3}\left[f(S_3) - f(S_2)\right]$$

$$f(M_n) = f(S_{n-1}) + \frac{\Delta Z_{n-1}}{\Delta Z_{n-1} + \Delta Z_n}\left[f(S_n) - f(S_{n-1})\right]$$

$$f(M_{n,\ max}) = 2f(S_{n,\ max-1}) - f(M_{n,\ max-1})$$

<CARD F51> Fuel plate title card (A20)

TITLP : Title for fuel plate

<CARD F52> Fuel plate data (Free format)

NPLATE : Number of this kind of fuel plates in this kind of fuel element (R)

FLOCL : Local peaking factor (R)

IDPL : Identity number of fuel plate configuration (I) (See **CARD F6**)

KMX : Index for cooling condition of fuel plate (I)

= 1 : Right hand side of fuel plate cooling condition and left hand side of fuel plate cooling condition are equal

= 2 : Right hand side of fuel plate cooling condition and left hand side of fuel plate cooling condition are not equal

IPLOT : Plot option for the calculation results (I)

= 0 : No plot

= 1 : Channel No.1 side calculation results are plotted

= 2 : Channel No.2 side calculation results are plotted

= 3 : Both of channel No.1 and No.2 sides calculation results are plotted

* Channel No. means ICHL of **CARD F53**.

IOUT : Print out option for pressure, ONB, DNB and Heat flux at onset of Flow instability calculation results (I)

= 0 : No print

= 1 : Print out of pressure, ONB and DNB calculation results

* If INFORM = 0 (**CARD B1**), then meaning of IOUT is as follows.

= 0 : No print

= 1 : Print out of pressure, ONB, DNB and Heat flux at onset of Flow instability calculation results, DNB heat flux is calculated by LABNTSOV correlation

= 2 : Print out of pressure, ONB, DNB and Heat flux at onset of Flow instability calculation results, DNB heat flux is calculated by MIRSHAK correlation

= 3 : Print out of pressure, ONB, DNB and Heat flux at onset of Flow instability calculation results, DNB heat flux is calculated by BERNATH correlation

<CARD F53> Coolant channel data (Free format)

ICHL : Identity number of channel configuration (I)
(See **CARD F70**, **CARD F74** or **CARD F76**)

NHEAT : Coolant condition (R)

= 1.0 : Coolant is heated from one side

= 2.0 : Coolant is heated from both sides

FRATEC : Flow rate distribution factor for this kind of channel (R)

FRATEC = (Flow rate of this kind of channel)/(Average flow rate of channel in this kind of fuel element)

* This card is required KMX (**CARD F52**) sets.

* **CARD F51-CARD F53** are required NPMX (**CARD F21**) sets.

<CARD F6> Fuel plate configuration data (Free format)

XA : Half thickness of fuel meat for plate-type fuel (cm) (R)
: Radius of fuel pellet for rod-type fuel (cm) (R)

XB : Distance between fuel meat center and clad inner surface for plate-type fuel (cm) (R) (For plate-type fuel, normally : XA = XB)
: Gap thickness between pellet and cladding for rod-type fuel (cm) (R)

XC : Distance between fuel meat center and clad outer surface for plate-type fuel (cm) (R) (Half thickness of fuel plate)
: Cladding thickness for rod-type fuel (cm) (R)
YA : Width of fuel meat for plate-type fuel (cm) (R)
: = 0.0 for rod-type fuel (R)
HA : Distance between inlet of channel and top(bottom) of fuel meat (pellet) (cm) (R)
HB : Length of fuel meat (fueled region) (cm) (R)
HC : Distance between outlet of channel and bottom(top) of fuel meat (pellet) (cm) (R)

<CARD F70> Coolant channel configuration data (Free format) (If INFORM = 0 (**CARD B1**), then this card is required.)
YCHI : Gap(thickness) of coolant channel (cm) (R)
XCHI : Width of coolant channel (cm) (R)

<CARD F71> Pressure loss calculation data (Fuel element entrance - plate entrance) (Free format) (If INFORM = 0 (**CARD B1**), then this card is required.)
ZETA(1) : Resistance coefficient of fuel element entrance (STRETCH(1)) (R)
DH(1) : Distance between fuel element entrance and fuel plate entrance (cm) (R)
HDE(1) : Equivalent hydraulic diameter of this region (cm) (R)
AR(1) : Cross sectional area of this region (Flow area) (cm^2) (R)

<CARD F72> Pressure loss calculation data (Fuel plate exit - fuel element plug entrance) (Free format) (If INFORM = 0 (**CARD B1**), then this card is required.)
ZETA(2) : Resistance coefficient of fuel element plug entrance (STRETCH(3)) (R)
DH(2) : Distance between fuel plate exit and fuel element plug entrance (cm) (R)
HDE(2) : Equivalent hydraulic diameter of this region (cm) (R)
AR(2) : Cross sectional area of this region (Flow area) (cm^2) (R)

<CARD F73> Pressure loss calculation data (Fuel element plug entrance - fuel element exit) (Free format) (If INFORM = 0 (**CARD B1**), then this card is required.)
ZETA(3) : Resistance coefficient of fuel element plug exit (STRETCH(3)) (R)
DH(3) : Distance between fuel element plug entrance and fuel element exit (cm) (R)
HDE(3) : Equivalent hydraulic diameter of this region (cm) (R)
AR(3) : Cross sectional area of this region (Flow area) (cm^2) (R)

* **CARD F70 - CARD F73** are required IDCMX (**CARD F21**) sets.
* **CARD F1 - CARD F73** are required NMAX (**CARD B2**) sets.

<CARD F74> Coolant channel configuration data (Free format) (If INFORM<>0 (**CARD B1**) and KEY(2)<>5 (**CARD B2**), then this card is required.)
YCHI : Gap (thickness) of coolant channel for plate-type fuel (cm) (R)
: Equivalent hydraulic diameter for rod-type fuel (cm) (R)
XCHI : Width of coolant channel for plate-type fuel (cm) (R)
: Effective flow area for one fuel rod for rod-type fuel (cm^2) (R)
MSFLW : Number of segments, except fuel plate region[1]. (Number of **CARD F75**)

<CARD F75> Pressure loss calculation data (Free format) (If INFORM<>0 (**CARD B1**) and KEY(2)<>5 (**CARD B2**), then this card is required.)
ZETA : Resistance coefficient of this region entrance (R)
DH : Length of flow area (cm) (R)

ZLAM : Friction loss coefficient for laminar flow Cb[2)] (R)
HDE : Equivalent hydraulic diameter of this region (cm) (R)
AR : Cross sectional area of this region (Flow area) (cm^2) (R)
* **CARD F74 - CARD F75** are required IDCMX (**CARD F21**) sets.
* **CARD F1 - CARD F75** are required NMAX (**CARD B2**) sets.

<CARD F76> Coolant channel configuration data (Free format) (If INFORM<>0 (**CARD B1**) and KEY(2)=5 (**CARD B2**), then this card is required.)

YCHI : Gap (thickness) of coolant channel for plate-type fuel (cm) (R)
: Equivalent hydraulic diameter for rod-type fuel (cm) (R)
XCHI : Width of coolant channel for plate-type fuel (cm) (R)
: Effective flow area for one fuel rod for rod-type fuel (cm^2) (R)
MSFLW : Number of segments, include fuel plate region[1)]. (In this case number of fuel plate region must be 1) (Number of **CARD F77**)
MSFUEL : Fuel plate region segment number (From top of segment) (I)

<CARD F77> Pressure loss calculation data (Free format) (If INFORM<>0(**CARD B1**) and KEY(2)=5 (**CARD B2**), then this card is required.)

ZETA : Resistance coefficient of this region entrance (R)
DH : Length of flow area (cm) (R)
ZLAM : Friction loss coefficient for laminar flow Cb[2)] (R)
HDE : Equivalent hydraulic diameter of this region (cm) (R)
AR : Cross sectional area of this region (Flow area) (cm^2) (R)
* **CARD F76 - CARD F77** are required IDCMX (**CARD F21**) sets.
* **CARD F1 - CARD F77** are required NMAX (**CARD B2**) sets.

<CARD G1> Control card (Free format) (If INFORM<>0 (**CARD B1**), then this card is required.)

KVELO : Index for primary coolant flow rate (I)
= 0 : Volumetric flow rate (m^3/min)
= 1 : Mass flow rate (kg/s)
= 2 : Average coolant velocity in the core (cm/s)
JUMAX : Number of non-heated flow segment of channel inlet side (I)
JLMAX : Number of non-heated flow segment of channel outlet side (I)
* If KEY(2)<>5 (**CARD B2**), then JUMAX + JLMAX = MSFLW (**CARD F74**)
* If KEY(2)=5 (**CARD B2**), then JUMAX + JLMAX + 1 = MSFLW (**CARD F76**)
IHTC : Index for heat transfer correlation (I)
= 1-3 : COOLOD code original heat transfer correlation. See Table 1. (Single-phase heat transfer correlation is defined by **CARD E1**.)
=1 : DNB heat flux is calculated by LABUNTSOV correlation
=2 : DNB heat flux is calculated by MIRSHAK correlation
=3 : DNB heat flux is calculated by BERNATH correlation
= 4 : "Heat Transfer Package"
KBFLG : Index for void fraction calculation in the natural circulation cooling mode (I)
= 0 : Void fraction is calculated in only nucleate boiling region (Zuber correlation)
> 0 : Void fraction is calculated in both nucleate boiling and subcooled boiling region. In subcooled boiling region, void fraction correlation is as follows.
= 1 : AHMAD correlation
= 2 : Zuber correlation

= 3 : If flow rate in the core G(kg/s) < GLIM (**CARD G5**), then AHMAD correlation.
If flow rate in the core G(kg/s) >= GLIM (**CARD G5**), then Zuber correlation.
* If forced convection cooling mode, then KBFLG = 0

NCMAX : Number of non-heated channel(Core bypass) (I)
* If KEY(2)<>5 (**CARD B2**), then NCMAX must be 0.

NATIP : Option for flow rate calculation in the natural circulation cooling mode (I)
= 0 : Hot channel factors are not used in the calculation of flow rate in the natural circulation cooling mode.
= 1 : Hot channel factors are used in the calculation of flow rate in the natural circulation cooling mode.
* If KEY(2)<>5 (**CARD B2**), then NATIP must be 0.

<CARD G2> Core bypass data (1) (Free format) (If INFORM<>0 (**CARD B1**) and KEY(2) =5 (**CARD B2**), then this card is required.)

MSFLOW: Number of core bypass segments (I)

<CARD G3> Core bypass data (2) (Free format) (If INFORM<>0 (**CARD B1**) and KEY(2) =5 (**CARD B2**), then this card is required.)

ZETA : Resistance coefficient of this region entrance (R)
DH : Length of flow area (cm) (R)
ZLAM : Friction loss coefficient for laminar flow Cb[2)] (R)
HDE : Equivalent hydraulic diameter of this region (cm) (R)
AR : Cross sectional area of this region (Flow area) (cm^2) (R)
* This card is required MSFLOW (**CARD G2**) sets.
* **CARD G2** and **CARD G3** are required NCMAC (**CARD G1**) sets.

<CARD G4> Coolant channel configuration identity data (Free format) (If INFORM<>0 (**CARD B1**), then this card is required.)

JMSH : Flag for channel configuration (I)
* ((JMSH(NP, k), NP = 1, NPMX), K = 1, KMX)
* If KEY(2) <> 5 (**CARD B2**), then this card is required MSFLW x NMAX (**CARD B2**) sets. (MSFLW (**CARD F74**)= JUMAX (**CARD G1**) + JLMAX (**CARD G1**))
* If KEY(2) = 5 (**CARD B2**), then this card is required MSFLW x (NMAX (**CARD B2**) + NCMAX (**CARD G1**)) sets. (MSFLW (**CARD F76**) = JUMAX (**CARD G1**) + JLMAX (**CARD G1**) + 1)

<CARD G5> Void fraction calculation data (Free format) (If INFORM<>0 (**CARD B1**) and KEY(2) =5 (**CARD B2**), then this card is required.)

CB : Zuber constant (R)
* You had better to use CB = 1.18 or 1.41.

GLIM : Standard flow rate for void fraction calculation (kg/s) (R)
* GLIM is used only in the case of KBFLG = 3 (**CARD G1**).
* You had better to use GLIM = 500 - 1500 (kg/m^2s).

<CARD G6> Debug control card (I)

IDBG(I), I = 1, 25
IDBG(I), I = 26, 50

IDBG : If you need debug the subroutine I, please input IDBG >= 5
See Table 4. (Normally : =0)

<CARD P1> Plot control card (1)

WITHX : Length of X axial (Maximum 200 mm) (mm) (R)
WITHY : Length of Y axial (Maximum 230 mm) (mm) (R)
TMIN : Minimum value of temperature scale (°C) (R)
TMAX : Maximum value of temperature scale (°C) (R)
PMIN : Minimum value of pressure scale (kg/cm^{2}abs.) (R)
PMAX : Maximum value of pressure scale (kg/cm^{2}abs.) (R)
HMIN : Minimum value of heat flux scale (W/cm^2) (R)
HMAX : Maximum value of heat flux scale (W/cm^2) (R)

<CARD P2> Plot control card (2) (A4)

NEWI : = "NEW" Plot on new page
= "OLD" Plot on same page
* In the first figure NEWI must be "NEW".

<CARD P3> Figure title card (A40)

TITLE : Title of figure
* If NEWI="OLD", then this card is not required.

<CARD P4> Plot control card (3) (I)

IDPLOT(1)-(7), NSMBL(1)-(7)

Plot items are listed as follows.

(1) Coolant temperature
(2) Clad surface temperature
(3) Meat maximum temperature
(4) Saturation temperature
(5) ONB temperature
(6) Pressure
(7) Clad surface heat flux

IDPLOT(I) : = 0 No plot
= 11-15 Solid line is used
= 21-25 Doted line is used

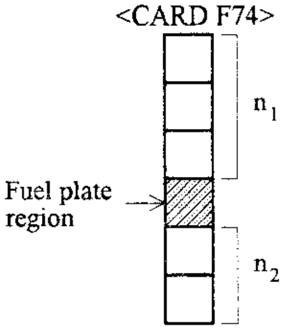
NSMBL(I) : = 0 No symbol
= 1 ○ is plotted on the line
= 2 △ is plotted on the line
= 3 + is plotted on the line
= 4 x is plotted on the line
= 11 ＊ is plotted on the line

Table 4 Debug flag for each subroutine

Subroutine name	IDGB No.	Subroutine name	IDBG No.
	1		26
	2	ONBTE	27
CALCTL	3	CLADTE	28
	4	BONDTE	29
	5	FUELTE	30
INITLZ	6	HEATBL	31
POWER	7	QHFPKG	32
TMPINL	8		33
	9	PRESS	34
DISPWZ	10	QDNB (=>8)	35
QRATE	11		36
TMPCAL	12		37
	13		38
	14		39
	15	NATURE	40
VELOC, VELOC2	16	FLWGO	41
	17	DLTPD	42
	18	LOSTL	43
	19		44
NEWTON (=>8)	20	G1CAL	45
	21		46
	22	PBPH	47
	23	REN (=1)	48
COOLTE	24	UNITI	49
PRESDRP	25	UNITO	50

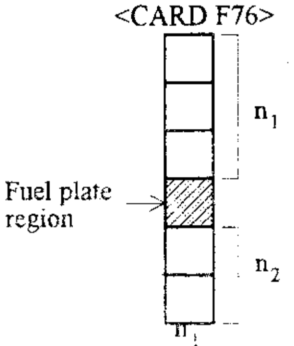
1)

<CARD F74>



$\text{MSFLW} = n_1 + n_2$

<CARD F76>



$\text{MSFLW} = n_1 + n_2 + 1$ (Fuel plate region)

2)

$$F = \frac{Cb}{\text{Re}}$$

F	: Friction loss coefficient	
Re	: Reynolds number	
Cb	: Tube	Cb = 64.0
	Square	Cb = 56.9
	Rectangular	Cb = 96.0
	(Channel of fuel element)	

6. Concluding Remarks

In this report, information required for the COOLOD-N2 code users has been described. The COOLOD-N2 was developed based on the COOLOD-N code and provides a capability for the analysis of the steady-state thermal-hydraulics of research reactors. The COOLOD-N2 is applicable not only for research reactors in which plate-type fuel is adopted, but also for research reactors in which rod-type (pin-type) fuel is adopted. This work has been done as a part of the thermal-hydraulic analysis of the JRR-4 TRIGA fueled core. Thermal-hydraulic calculations for the JRR-4 TRIGA core were successfully conducted using COOLOD-N2 code.

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References

[1] M. Kaminaga, "COOLOD-N : A Computer Code, for the Analyses of Steady-State Thermal-Hydraulics in Plate-Type Research Reactors", JAERI-M 90-021, 1990.

[2] S. Watanabe, "COOLOD : Thermal and Hydraulic Analysis Code for Research Reactors with Plate Type Fuel Elements (in Japanese)", JAERI-M 84-162, 1984.

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[5] M.A. Lucatero, M. Kaminaga, "Thermal-Hydraulic Conceptual Design of the Multiple Purpose Research Reactor MEX-15", JAERI-M 94-006, 1994.

[6] Y. Sudo, H. Ikawa and M. Kaminaga, "Development of Heat Transfer Package for Core Thermal-hydraulic Design and Analysis of Upgraded JRR-3", in Proceeding of the International Meeting of Reduced Enrichment for Research and Test Reactors, Petten, The Netherlands, October 14-16, 1985.

[7] General Atomics, "10 MW TRIGA-LEU Fuel and Reactor Design Description", UZR-14 (Rev.), 1979.

[8] F. W. Dittus and L. M. K. Boelter, Univ. Calif. Pubs. Eng., 2, 443, 1930.

[9] J. G. Collier, "Convective Boiling and Condensation", McGraw-Hill Book Co., New York, 1972.

[10] A. E. Bergles and W. H. Rohsenow, "The determination of forced-convection surface-boiling heat transfer", ASME, Ser.C, 86, 365-372, 1964.

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[12] Y. Sudo, K. Miyata, H. Ikawa, M. Kaminaga and M. Ohkawara, "Experimental Study of Differences in DNB heat Flux between Upflow and Downflow in Vertical Rectangular Channel", J. Nucl. Sci. Technol., Vol.22, No.8, 604-618, 1985.

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[6] Y. Sudo, H. Ikawa and M. Kaminaga, "Development of Heat Transfer Package for Core Thermal-hydraulic Design and Analysis of Upgraded JRR-3", in Proceeding of the International Meeting of Reduced Enrichment for Research and Test Reactors, Petten, The Netherlands, October 14-16, 1985.

[7] General Atomics, "10 MW TRIGA-LEU Fuel and Reactor Design Description", UZR-14 (Rev.), 1979.

[8] F. W. Dittus and L. M. K. Boelter, Univ. Calif. Pubs. Eng., 2, 443, 1930.

[9] J. G. Collier, "Convective Boiling and Condensation", McGraw-Hill Book Co., New York, 1972.

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[13] K. Mishima, "Boiling burnout at low flow rate and low pressure conditions", Dissertation Thesis, Kyoto Univ., 1984.

[14] R. H. Whittle and R. Forgan, "A correlation for the minima in the Pressure Drop versus Flow-rate Curves for Sub-cooled Water Flowing in Narrow Heated Channels", Nucl. Eng. Design, 6,

89-99, 1967.
[15] The Japan Society of Mechanical Engineers, "Handbook of Mechnical Engineering (6th Edition) (in Japanese)", Section 8, 13-15.
[16] H. Uchida, "Wet air and Cooling tower (in Japanese)", Shoukado, 1963.
[17] H. Obana, "Handbook of Heat exchanger Design (in Japanese)", Kougaku-Tosho, 1977.
[18] Saha, Zuber, "Point of net vapor generation and vapor void fraction in subcooled boiling", Proc. 5th Int. Heat Trans. Conf., Tokyo, 4, 175-179, 1974.
[19] S. Y. Ahmad, "Axial distribution of bulk temperature and void fraction in a heated channel with inlet subcooling", Trans. ASME, Ser.C, 92-4, 595-609, 1970.
[20] N. Zuber, F. W. Stanb and G. Bijwaard, "Vapor void fraction in subcooled boiling system", 3rd Int. Heat Transfer Conf., 5, 24-38, 1966.
[21] P. G. Kroeger and N. Zuber, "An analysis of the effects of various parameters on the average void fractions in subcooled boiling", Int. J. Heat Mass Transfer, 11, 211-233, 1968.
[22] J. E. Houghtaling, Alain Sola and A. H. Spano, "Transient Temperature Distribution in the SPERT1D-12/25 Fuel Plates during Short-Period Power Excursions", IDO-16884, 1964.
[23] S. Nazare, G. Ondracek and F. Thummler, "Investigation on UAl_x-Al Dispersion Fuels for High-Flux Reactors", J. Nucl. Materials, 56, 251-259, 1975.
[24] Y. S. Touloukian et al., "Thermophysical Properties of Matter, Vol.1, Thermal Conductivity", 1970.
[25] Y. S. Touloukian et al., "Thermophysical Properties of Matter, Vol.3, Thermal Conductivity", 288-289, 1970.
[26] The Japan Society of Mechanical Engineers, "Dennetsu-kougaku shiryo (3rd Edition) (in Japanese)", 1976.
[27] J. N. Elliott, "Tables of the Thermodynamic Properties of Heavy Water", AECL-1673, 1963.
[28] M. Kaminaga "Steady-State GA Benchmark calculation for JRR-4 TRIGA-16 Fueled Core (Steady-State Thermal-Hydraulic Analysis), (in Japanese)", private communication, 1993.
[29] General Atomics, private communication.
[30] M. Kaminaga "Steady-State Thermal-Hydraulic Analysis for JRR-4 TRIGA-25 Fueled Core (in Japanese)", private communication, 1993.
[31] M. Kaminaga "Steady-State Thermal-Hydraulic Analysis for JRR-4 TRIGA-16 Fueled Core (in Japanese)", private communication, 1993.
[32] General Atomics, private communication.

Appendix A Sample calculation results

A. 1 JRR-4 TRIGA-16 core GA (General Atomics) benchmark calculation, input data description

Thermal hydraulic analysis for the JRR-4 TRIGA-16 core[28] was carried out to verify the fuel rod temperature calculation model and Lund DNB heat flux correlation by comparing analysis results calculated by General Atomics using TIGER code[29]. This analysis has been done as a part of the thermal-hydraulic analysis of the JRR-4 TRIGA fueled core[28],[30],[31].

Following input data were used for JRR-4 TRIGA-16 core thermal hydraulic analysis. Almost all of following input data used in the analysis were as same as those used by GA calculation[29].

a. Primary coolant flow rate through TRIGA fuel element is 81.12% (5.68 m^3/min) of the total primary coolant flow rate of 7 m^3/min.
b. Hot pin factor (radial peaking factor x local peaking factor) is 1.7.
c. Engineering hot cannel factors were not considered.
d. Equivalent hydraulic diameter for TRIGA fuel element is calculated for a channel which is surrounded by 4 fuel rods.
e. Form loss coefficient at the fuel element inlet is taken from "10 MW TRIGA-LEU Fuel and Reactor Design Description", General Atomics[7].
f. An axial power distribution is calculated by the following correlation which was used for the thermal hydraulic design of 14 MW TRIGA reactor, Romania[32]. Figure A.1 shows the axial power distribution used in the analysis.

$$APF = 1.35(1+1.275e^{-39.96(1-\xi)})\cos(1.325\xi) \qquad \text{(A-1)}$$
$$\xi = |1-2Z/L|$$

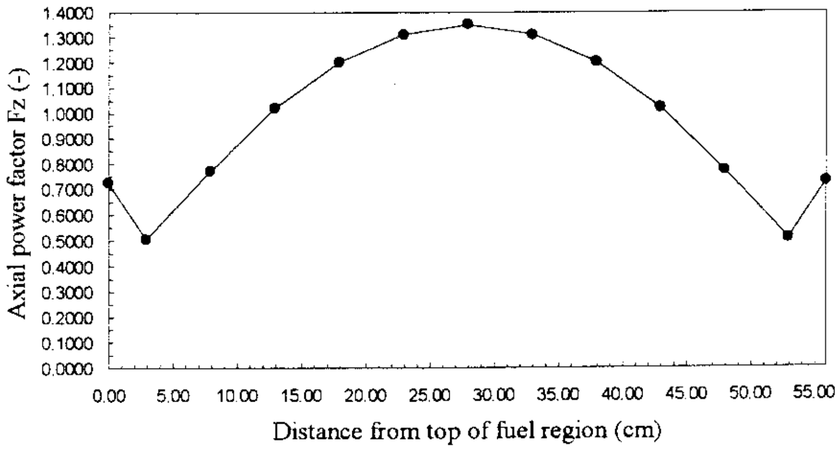


Figure A.1 Axial power distribution for JRR-4 TRIGA-16 core

g. Thermal conductivities for fuel pellet and cladding, gap conductance are shown in Table A.1.

Table A.1 Thermal conductivities and gap heat transfer coefficient used in the analysis

Fuel pellet thermal conductivity (W/m K)	0.196
Cladding thermal conductivity (W/m K)	0.162
Gap conductance (W/m^2K)	0.804

h. Fuel pellet diameter : 1.2903 cm
i. Fuel rod diameter : 1.3716 cm
j. Equivalent hydraulic diameter : 0.0406 cm
k. Sub-channel flow area : 1.1898 cm^2
(flow area per one fuel rod) Figure A.2 shows cross-sectional area of the channel.
l. Pressure drop calculation model for TRIGA fuel element. Figure A.3 shows a pressure drop calculation model for JRR-4 TRIGA-16 fuel element.

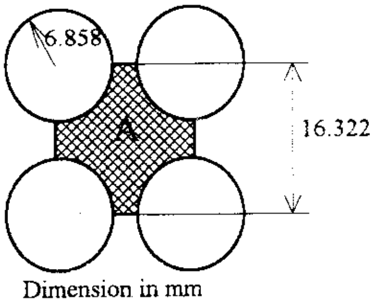
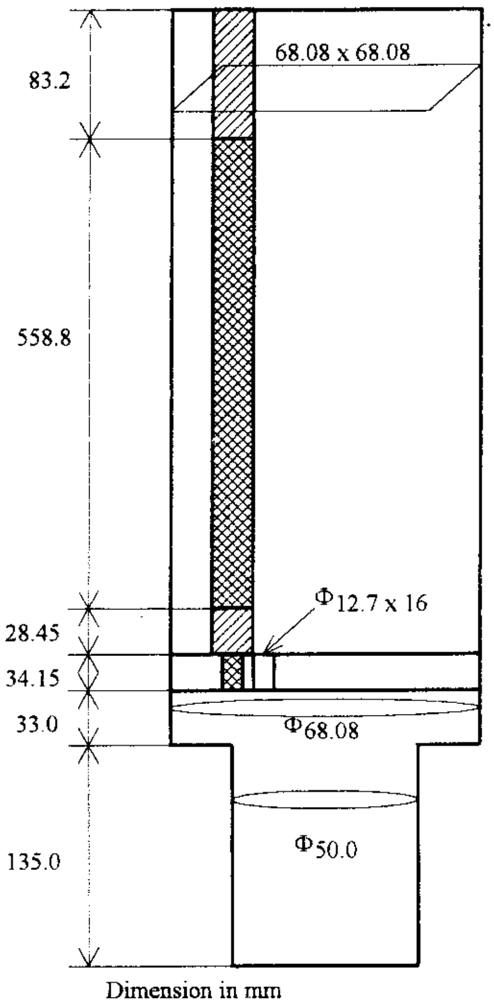


Figure A.2 Cross sectional area of the sub-channel



Form loss coefficient at the fuel element inlet

K = 3.2

Flow area and an equivalent hydraulic diameter of the sub-channel

$$A = 1.63322^2 - 0.6858^2 \pi$$
$$= 1.189849 cm^2$$
$$De = \frac{4x1.189849}{2x0.6858\pi}$$
$$= 1.1045219 cm$$

Form loss coefficient of spacers (2 locations)

K = 0.4

Flow area of fueled region

$$A = 1.189849x16 = 19.03758 cm^2$$

Grid plate

K = 0.5

$$A = \frac{1.27\pi}{4}x16 = 20.2683 cm^2$$
$$De = 1.27 cm$$

Flow channel under the grid plate

K = 1.0

$$A = \frac{6.808\pi}{4} = 36.3809 cm^2$$
$$De = 6.808 cm$$

Nozzle section

Kin = 0.5

$$A = \frac{5.0\pi}{4} = 19.6350 cm^2$$
$$De = 5.00 cm$$

Kout = 1.0

Figure A.3 Pressure drop calculation model for JRR-4 TRIGA fuel element

A. 2 JRR-4 TRIGA-16 core GA benchmark calculation, calculation results

Table A.2 shows calculation results of JRR-4 TRIGA-16 core including major input data. In the table calculation results calculated by Tiger code are also shown for a comparison. Figure A.4 shows axial temperature, pressure and heat flux distributions of the hot channel.

Table A.2 Comparison of calculation results between COOLOD-N2 and Tiger code

	Thermal power = 3.5 MW	Calculation results, Tiger code Hot channel	Calculation results, COOLOD-N2 code Hot channel
Fuel	Number of fuel elements	20	20
	Number of rod / element	16	16
	Pin diameter (m)	0.013716	0.013716
	Flow area / element (m^2)	1.904×10^{-3}	1.904×10^{-3}
Peaking factor	Fr (Radial peaking factor) (Hot pin factor)	1.7	1.7
	Fz (Axial peaking factor)	1.34	1.35
	Total	2.278	2.295
Coolant	Core inlet pressure (kg/cm^2)	2.074	2.074
	Core flow rate (m^3/min)	7.00	7.00
	Effective flow rate for fuel cooling (m^2/min)	5.68	5.68
	Average coolant velocity (m/s)	2.50	2.50
	Inlet temperature (°C)	37.0	37.0
	Outlet temperature (°C)	51.9	52.1
Results	Maximum fuel (cladding) surface temperature (°C)	129.7	127.3
	Maximum fuel (pellet) temperature (°C)	476.3	474.2
	Maximum fuel (cladding) surface heat flux (W/m^2)	1.041×10^6	1.042×10^6
	DNB heat flux (W/m^2)	2.507×10^6	2.657×10^6
	DNBR	2.41	2.55

From the calculation results, calculation results calculated by COOLOD-N2 code show good agreement with those calculated by Tiger code while there is a little difference of fuel temperatures between the two codes. The calculation results calculated by COOLOD-N2 code (including DNB heat flux) are a little bit higher than those calculated by Tiger code. Heat transfer calculation model of COOLOD-N2 code is considered to be the same with Tiger code, but at this moment, we do not have enough information of Tiger code and its input data. Therefore, it is difficult to investigate the reason of these difference furthermore.

The calculation model for rod type fuel of COOLOD-N2 code was successfully verified by comparing the calculation results calculated by Tiger code.

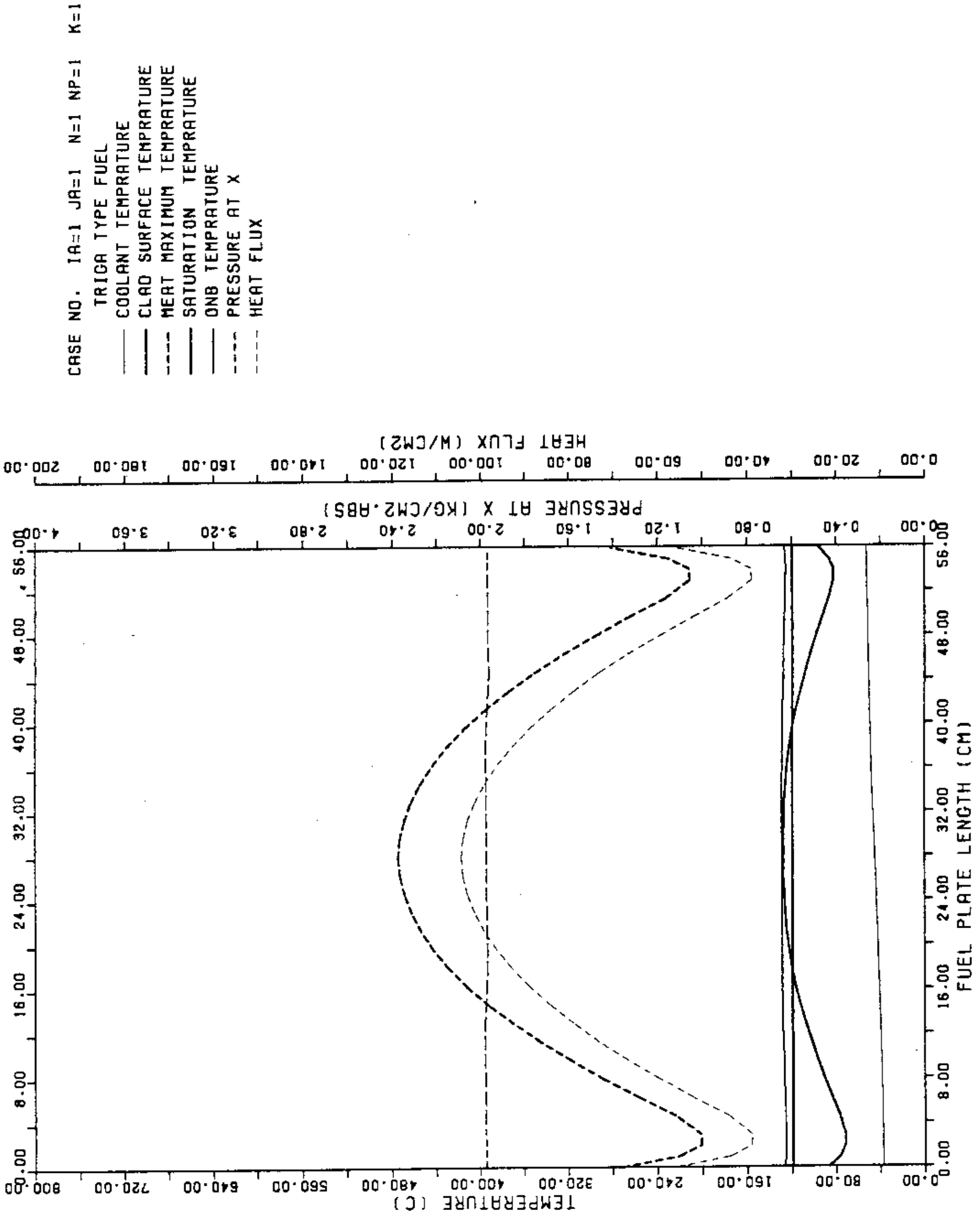


Figure A.4 Axial temperature, heat flux and pressure distribution in the hot channel of JRR-4 TRIGA-16 core

A. 3 Input data for COOLOD-N2, JRR-4 TRIGA-16 core thermal-hydraulic analysis

```
                  INPUT DATA CARD IMAGE
 COOLOD-N THERMAL HYDRAULIC CALC. (JRR-4 TRIGA-16 FUEL CORE ) 08/10/93  HECK
C   CORE FLOW RATE 7.0 M3/MIN GA002 GA BENCHMARK CALCULATION
C   <CARD B1> INFORM
             1
C   <CARD B2> IAMAX IMAX  JMAX  NMAX  NPLOT KEY(1) KEY(2) KEY(3) IDMAX
             1    5   13     1     1     1      1      0      3
C   <CARD C > QRR(MW)  PFLOW(KG/SEC)  TIN(DEG)  DT(DEG)  JAMX
             3.5     115.887        37.0       0.0      1
C   <CARD E1>  H1      H2     H3     A      B      C    D    ITWC
            0.023   0.0     0.0  0.800  0.400  0.0  0.0    1
C   <CARD E2> FRATE   VIN   VOUT     PRESIN   RAMF
          0.811159  0.0     0.0    2.0740565 0.0
C   <CARD F1> FUEL ELEMENT TITLE
 TRIGA TYPE FUEL 16PIN
C   <CARD F21> NPMX NFUEL   MA  UDENST  POROTY IDPMX IDCMX EAREA FRATEN
             1   20.0    2   19.07   0.030    1      1 19.03758 1.000
C   <CARD FNEW>    DE(CM)   D(CM)  PITCH/D (-)
             1.1045219 1.3716  1.190741        .
C   <CARD F221> ZR-H W/CM.C
             3
    0.00         0.1963317
  378.00         0.1963317
 2000.00         0.1963317
C   <CARD F222> GAP CON  W/CM2.C DUMMY (GAP = 0.00230 CM)
C   1.8467E-03 (W/CM.C) / 0.00230 (CM) = 0.80353 W/CM2.C HOT PIN
             2
    0.0          0.80353
 2000.0          0.80353
C   <CARD F223> CLAD INCOLOY W/CM.C
             2
    0.0          0.16154
 2000.0          0.16154
C   <CARD F3 > FR    FCOOL  FFILM  FHFLX  FCLAD  FBOND  FMEAT
          1.70000 1.000  1.000  1.000  1.000  1.000  1.000
C   <CARD F4 >  FZ       DDZ     ZET    (FROM CALC. RESULTS OF CITATION)
         0.72320    2.9400   0.0
         0.50300    5.0000   0.0
         0.77100    5.0000   0.0
         1.02000    5.0000   0.4
         1.20000    5.0000   0.0
         1.31000  - 5.0000   0.0
         1.35000    5.0000   0.0
         1.31000    5.0000   0.0
         1.20000    5.0000   0.4
         1.02000    5.0000   0.0
         0.77100    5.0000   0.0
         0.50300    2.9400   0.0
         0.72320    0.0000   0.0
C   <CARD F  > NPMX GROUPS
C   <CARD F51> FUEL PLATE TITLE
 TRIGA TYPE FUEL
C   <CARD F52> NPROD   FLOC  IDPL  KMX  IPLOT  IOUT
              16.0  1.000    1    1     1     1
C   <CARD F53> ICHL(1) NHEAT(1) FRATEC(1)    ICHL(2) NHEAT(2) FRATEC(2)
                1      1.0     1.000
C   <CARD F6 >   XA      XB        XC      YA        HA      HB      HC
            0.64516  0.00000  0.04064 0.0      8.320  55.880 2.845
C   <CARD F7 >  XCHI(DE) YCHI(AREA) MSFLW  MSFUEL
           1.1045219 1.189849     5      0
C   <CARD F74>    ZETA       DH      ZLAM     HDE      AR
              3.20      0.000  0.0     0.932985 22.4942
              0.50      3.415 64.0     1.27000  20.2683
              1.00      3.300 64.0     6.80600  36.3809
              0.50     13.500 64.0     5.00000  19.6350
              1.00      0.000  0.0     5.00000  19.6350
C   IF INFORM.NE.0, SET <CARD G> AS FOLLOWS
C      <CARD G1> = ONE SHIET
C      <CARD G2>-<CARD G3> = 'NCMAX' GROUPS
C      <CARD G4> ='NMAX+NCMAX' SHIET
C   <CARD G1 > KVELO  JUMAX  JLMAX  IHTC  KBFLG  NCMAX  NATIP
C              1      2      4     4      0      0      0
               1      1      4     4      0      0      0
C   <CARD G4 > ((JMSH(NN,NNP,JJ,KK),NNP=1,NPMAX),KK=1,KMAX)
    1
    2
    3
    4
    5
C   <CARD G6 > IDBG
    0 0 0 0 0  0 0 0 0 0  0 0 0 0 0  0 0 0 0 0  0 0 0 0 0
    0 0 0 0 0  0 0 0 0 0  0 0 0 0 0  0 0 0 0 0  0 0 0 0 0
C   <CARD P1>  WITHX  WITHY  TMIN  TMAX  PMIN  PMAX  HMIN  HMAX
             140.0  200.0   0.0  800.0  0.0   4.0   0.0  200.0
C   <CARD P2>  NEWI
 NEW
C   <CARD P3>  TITLE(A40)
 JRR-4 TRIGA-16 CORE CASE:GA BENCHMARK
C   <CARD P4>  ID1 ID2 ID3 ID4 ID5 ID6 ID7 NS1 NS2 NS3 NS4 NS5 NS6 NS7
              11  13  23  14  12  22  21   0   0   0   0   0   0   0
```

A. 4 Output lists of COOLOD-N2, JRR-4 TRIGA-16 core thermal-hydraulic analysis

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                  COOLOD-N THERMAL HYDRAULIC CALCULATION                                  CALCULATION DATE 94-01-11    PAGE  1

                        ********************************************************************************
                        **** COOLOD-N THERMAL HYDRAULIC CALC. (JRR-4 TRIGA-16 FUEL CORE ) 08/10/93  ****
                        ****                       CALCULATION DATE    94-01-11                     ****
                        ****                          INITIAL INPUT DATA                            ****
                        ********************************************************************************

     INPUT   CARD B1        INFORM
                             1
     INPUT   CARD B2         IAMAX      IMAX      JMAXN      NMAX     NPLOT    KEY(1)    KEY(2)    KEY(3)    IDMAX
                                 1         5         13         1         1         1         1         0        3
NI ***    5    6    9
     INPUT   CARD  C
             QRR       PFLOW      TIN         DT   JAMX
     CASE 1     3.500  115.887    37.000      0.000   1
     INPUT   CARD E1          H1        H2        H3         A         B         C         D       ITWC
                            0.023     0.000     0.000     0.800     0.400     0.000     0.000        1
     INPUT   CARD E2        FRATE       VIN      VOUT   ' PRESIN      RAMF
                           0.8112    0.0000    0.0000    2.0741    0.0000
     INPUT   CARD F1     TRIGA TYPE FUEL 16PIN

     INPUT   CARD F21         NPMX     NFUEL        MA    UDENST    POROTY     IDPMX     IDCMX     EAREA    FRATEN
                                 1      20.0         2    19.070     0.030         1         1    19.038     1.000
     INPUT   CARD F3            FR     FCOOL     FFILM     FHFLX     FCLAD     FBOND     FMEAT
                            1.700     1.000     1.000     1.000     1.000     1.000     1.000
     INPUT   CARD F4     J        FZ        DDZ        ZET
                          1    0.7232     2.940     0.000
                          2    0.5030     5.000     0.000
                          3    0.7710     5.000     0.000
                          4    1.0200     5.000     0,400
                          5    1.2000     5.000     0.000
                          6    1.3100     5.000     0.000
                          7    1.3500     5.000     0.000
                          8    1.3100     5.000     0.000
                          9    1.2000     5.000     0.400
                         10    1.0200     5.000     0.000
                         11    0.7710     5.000     0.000
                         12    0.5030     2.940     0.000
                         13    0.7232     0.000     0.000
          INPUT   CARB F51         INPUT   CARD F52                                             INPUT   CARD F53

          PLATE NAME               NPLATE    FLOCL     IDPL      KMX    IPLOT     IOUT            K  ICHL   NHEAT  FRATEC
NP= 1   TRIGA TYPE FUEL              16.0    1.000        1        1        1        1
                                                                                                1     1     1.0  1.0000
     INPUT   CARD F6           IDP       XAI       XBI       XCI       YAI       HAI       HBI       HCI
                                 1     0.645     0.000     0.041     0.000     8.320    55.880     2.845
     INPUT   CARD F74                                                INPUT   CARD F75
       IDC      XCHI      YCHI      MSFLW                            IS      ZETA        DH      ZLAM        DE      AREA
         1     1.105     1.190         5
                                                                      1     3.200     0.000     0.000     0.933    22.494
                                                                      2     0.500     3.415    64.000     1.270    20.268
                                                                      3     1.000     3.300    64.000     6.806    36.381
                                                                      4     0.500    13.500    64.000     5.000    19.635
                                                                      5     1.000     0.000     0.000     5.000    19.635

INPUT CARD <G1>       KVELO   JUMAX   JLMAX    IHTC   KBFLG   NCMAX   NATIP
                          1       1       4       4       0       0       0

INPUT DATA FORMAT => COOLOD ORIGINAL

VELOCITY(=0) MASS FLOW RATE(=1)=>   1

UPPER PRENUM MESH=   1 FUEL PLATE MESH=  13 LOWER PRENUM MESH=    4

JMSH   NN=    1 MSFLW=    5 NPMAX=    1 KMAX=    1 KEY(2)=    1
  1    1
  2    2
  3    3
  4    4
  5    5
NN=    1 JFMAX=    5 JMSH=    1    2    3    4    5

IDBG

   1   2   3   4   5   6   7   8   9  10  11  12  13  14  15  16  17  18  19  20  21  22  23  24  25
   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0

  26  27  28  29  30  31  32  33  34  35  36  37  38  39  40  41  42  43  44  45  46  47  48  49  50
   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0   0
```

```
COOLOD-N THERMAL HYDRAULIC CALCULATION  CASE = (IA- 1 JA- 1)                 CALCULATION DATE 94-01-11     PAGE  2

        **************************************************************************
        **                                                                      **
        **   COOLOD-N THERMAL HYDRAULIC CALC. (JRR-4 TRIGA-16 FUEL CORE ) 08/10/93   **
        **                    RESULTS OF CALCULATION AND USED VALUES            **
        **                                                                      **
        **************************************************************************
                     *** PRIMARY    COOLANT ***
REACTOR  INLET   TEMPERATURE =   37.00  C
REACTOR  OUTLET  TEMPERATURE =   44.23  C
PRIMRY TEMPERATURE DIFFERENCE=    7.23  C
PRIMARY COOLANT FLOW RATE    =  115.89  KG/S
                     ***  REACTOR    CORE   ****
REACTOR   THERMAL  POWER     =    3.50  MW
AREA OF TOTAL FUEL CHANNELS  =  360.75  CM2

NUMBER  OF  FUEL  ELEMENTS   =    20.0  ELEMNTS

      TRIGA TYPE FUEL 16PIN                   =     .20.0  (ELEMNTS)
        AVERAGE  HEAT  GENERATION             =    149.68  (W/CM3)
        AVERAGE  MASS  FLUX                   =  2468.872  (KG/M2 SEC)

COOLANT TEMP. --(SEPARATED MODEL)  KITE = 0
```

Average heat flux for plate type fuel

= AVERAGE HEAT GENERATION x XA

(XA : Half thickness of fuel meat (**CARD F6**))

Average heat flux for rod type fuel

$$= \text{AVERAGE HEAT GENERATION} \times \frac{XA}{2} \times \frac{2\pi XA}{2\pi(XA+XB+XC)}$$

(XA : Radius of fuel pellet (**CARD F6**))

(XB : Gap thickness between pellet and cladding (**CARD F6**))

(XC : Cladding thickness (**CARD F6**))

COOLOD-N THERMAL HYDRAULIC CALCULATION CASE = (IA- 1 JA- 1) CALCULATION DATE 94-01-11 PAGE 3

```
*****************************************************************************
**                       TRIGA TYPE FUEL 16PIN                             **
*****************************************************************************
```

AVERAGE CHANNEL TEMPERATURE DISTRIBUTION

FLOW CHANNEL AREA = 19.04 CM2
NUMBER OF FUEL PLATES
TRIGA TYPE FUEL = 16.0

TEMPERATURE DISTRIBUTION

J	COOLANT (DEG.C)	CLADDING SURFACE (DEG.C)	CLADDING INNER (DEG.C)	FUEL MEAT OUTER (DEG.C)	FUEL MEAT MAXIMUM (DEG.C)
1	37.00	65.94	74.46	117.90	175.25
2	37.29	57.36	63.29	93.50	133.39
3	37.80	68.42	77.50	123.81	184.96
4	38.51	78.76	90.77	152.04	232.93
5	39.39	86.36	100.49	172.57	267.73
6	40.39	91.21	106.63	185.32	289.20
7	41.45	93.37	109.27	190.36	297.41
8	42.51	92.46	107.88	186.57	290.45
9	43.51	88.88	103.01	175.09	270.26
10	44.40	82.68	94.69	155.95	236.84
11	45.11	73.87	82.95	129.26	190.40
12	45.62	64.30	70.22	100.43	140.32
13	45.91	72.70	81.21	124.65	182.00

** HOT CHANNEL FACTORS (EXCEPT FZ) **
F(COOLANT)= 1.000 F(FILM)= 1.000 F(CLAD)= 1.000 F(BOND)= 1.000 F(MEAT)= 1.000

HEAT TRANSFER CONDITION

J	FZ	TRANSFER COEFICIENT (W/CM2.C)	HEAT FLUX IN PLATE SURFACE (W/CM2)	HEAT FLUX IN PLATE SURFACE (KC/M2.HR)	XAA (CM)	HEAT GENERATION (W/CM3)
1	0.723	1.1350	32.851	0.28241E+06	0.000	108.252
2	0.503	1.1380	22.848	0.19642E+06	0.000	75.291
3	0.771	1.1434	35.022	0.30108E+06	0.000	115.407
4	1.020	1.1510	46.332	0.39831E+06	0.000	152.678
5	1.200	1.1607	54.509	0.46860E+06	0.000	179.622
6	1.310	1.1711	59.505	0.51156E+06	0.000	196.087
7	1.350	1.1811	61.322	0.52718E+06	0.000	202.074
8	1.310	1.1915	59.505	0.51156E+06	0.000	196.087
9	1.200	1.2014	54.509	0.46860E+06	0.000	179.622
10	1.020	1.2105	46.332	0.39831E+06	0.000	152.678
11	0.771	1.2179	35.022	0.30108E+06	0.000	115.407
12	0.503	1.2233	22.848	0.19642E+06	0.000	75.291
13	0.723	1.2263	32.851	0.28241E+06	0.000	108.252

```
**** ITERATION COUNT =     5**** CONVERGED.
**** ITERATION COUNT =     5**** CONVERGED.
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**** ITERATION COUNT =     5**** CONVERGED.
```

H E A T F L U X = Average heat flux x FZ
(FZ : Axial peaking factor (CARD F4))

F(COOLANT) = FR x FLOCL x FCOOL (**CARD F3**)
F(FILM) = FR x FLOCL x FFILM (**CARD F3**)
F(CLAD) = FR x FLOCL x FCLAD (**CARD F3**)
F(BOND) = FR x FLOCL x FBOND (**CARD F3**)
F(MEAT) = FR x FLOCL x FMEAT (**CARD F3**)

COOLOD-N THERMAL HYDRAULIC CALCULATION CASE = (IA- 1 JA- 1) CALCULATION DATE 94-01-11 PAGE 4

TRIGA TYPE FUEL (TRIGA TYPE FUEL 16PIN)

CHANNEL DIMENSION = 1.105 * 1.190 (CM)
CHANNEL VELOCITY = 249.30 (CM/SEC)

TEMPERATURE DISTRIBUTION

J	COOLANT (DEG.C)	CLADDING SURFACE (DEG.C)	CLADDING INNER (DEG.C)	FUEL MEAT OUTER (DEG.C)	FUEL MEAT MAXIMUM (DEG.C)
1	37.00	86.20	100.68	174.53	272.02
2	37.49	71.56	81.62	132.99	200.80
3	38.35	90.15	105.59	184.32	288.26
4	39.57	107.32	127.73	231.89	369.40
5	41.07	119.74	143.76	266.29	428.06
6	42.77	125.49	151.71	285.48	462.08
7	44.57	127.33	154.35	292.20	474.20
8	46.37	126.19	152.41	286.18	462.78
9	48.07	121.84	145.86	268.40	430.17
10	49.58	111.71	132.13	236.29	373.79
11	50.79	97.33	112.76	191.49	295.43
12	51.65	81.84	91.91	143.27	211.08
13	52.14	95.40	109.87	183.72	281.22

** HOT CHANNEL FACTORS (EXCEPT FZ) **
F(COOLANT) = 1.700 F(FILM) = 1.700 F(CLAD) = 1.700 F(BOND) = 1.700 F(MEAT) = 1.700

HEAT TRANSFER CONDITION

J	FZ	TRANSFER COEFICIENT (W/CM2.C)	HEAT FLUX IN PLATE SURFACE (W/CM2)	HEAT FLUX IN PLATE SURFACE (KC/M2.HR)	XAA (CM)	HEAT GENERATION (W/CM3)
1	0.723	1.1351	55.846	0.48010E+06	0.000	184.028
2	0.503	1.1402	38.842	0.33392E+06	0.000	127.995
3	0.771	1.1493	59.537	0.51183E+06	0.000	196.191
4	1.020	1.1626	78.765	0.67713E+06	0.000	259.553
5	1.200	1.1780	92.665	0.79662E+06	0.000	305.356
6	1.310	1.2229	101.159	0.86964E+06	0.000	333.347
7	1.350	1.2597	104.248	0.89620E+06	0.000	343.526
8	1.310	1.2675	101.159	0.86964E+06	0.000	333.347
9	1.200	1.2561	92.665	0.79662E+06	0.000	305.356
10	1.020	1.2676	78.765	0.67713E+06	0.000	259.553
11	0.771	1.2793	59.537	0.51183E+06	0.000	196.191
12	0.503	1.2868	38.842	0.33392E+06	0.000	127.995
13	0.723	1.2910	55.846	0.48010E+06	0.000	184.028

HEAT FLUX = Average heat flux x FR x FFILM x FZ
(FR : Radial peaking factor (CARD F4))
(FFILM : Engineering factor for film temperature rise (CARD F4))
(FZ : Axial peaking factor (CARD F4))

COOLOD-N THERMAL HYDRAULIC CALCULATION CASE = (IA- 1 JA- 1) CALCULATION DATE 94-01-11 PAGE 5

TRIGA TYPE FUEL (PRESSURE , ONB & DNB CONDITION)

	PRESSURE AT Z (KG/CM2A)	PRESSURE LOSS (KG/CM2)	TOTAL LOSS (KG/CM2)	COOLANT VELOCITY (CM/SEC)	TSAT (C)	TONB (C)	TCLAD (C)	DTONB	HEAT FLUX (W/CM2) CLAD	QONB	QDNB	DNBR	DNB ID
INLET PLENUM	2.074			0.00									
STRETCH(1) INLET	1.980	0.07176	0.07176	210.35									
STRETCH(1) OUT	1.980	0.00000	0.07176	210.35									
PLATE ENTRANCE	1.969	0.00241	0.07417	248.55									
FUEL PLATE ZONE 1	1.972	0.00522	0.07939	248.55	119.16	126.49	86.20	40.28	55.85	* 0.00	248.54	4.45	1.0
FUEL PLATE ZONE 2	1.973	0.00184	0.08122	248.59	119.18	125.35	71.56	53.80	38.84	* 0.00	248.54	6.40	1.0
FUEL PLATE ZONE 3	1.975	0.00313	0.08434	248.66	119.21	126.75	90.15	36.60	59.54	* 0.00	248.54	4.17	1.0
FUEL PLATE ZONE 4	1.976	0.00312	0.08744	248.77	119.24	127.84	107.32	20.52	78.77	* 0.00	248.52	3.16	1.0
FUEL PLATE ZONE 5	1.966	0.00311	0.09052	248.92	119.07	128.37	119.74	8.64	92.66	0.34	248.17	2.68	1.0
FUEL PLATE ZONE 6	1.968	0.00311	0.09358	249.10	119.10	128.68	125.49	3.19	101.16	41.73	248.11	2.45	1.0
FUEL PLATE ZONE 7	1.970	0.00310	0.09663	249.29	119.13	128.78	127.33	1.45	104.25	70.88	248.05	2.38	1.0
FUEL PLATE ZONE 8	1.971	0.00309	0.09966	249.48	119.16	128.71	126.19	2.53	101.16	51.12	247.98	2.45	1.0
FUEL PLATE ZONE 9	1.973	0.00308	0.10268	249.66	119.19	128.46	121.84	6.62	92.66	6.45	247.92	2.68	1.0
FUEL PLATE ZONE 10	1.963	0.00308	0.10570	249.82	119.02	127.65	111.71	15.93	78.77	* 0.00	247.57	3.14	[illegible]
FUEL PLATE ZONE 11	1.965	0.00307	0.10871	249.96	119.05	126.61	97.33	29.28	59.54	* 0.00	247.54	4.16	1.0
FUEL PLATE ZONE 12	1.967	0.00307	0.11174	250.07	119.08	125.26	81.84	43.43	38.84	* 0.00	247.54	6.37	1.0
FUEL PLATE ZONE 13	1.968	0.00180	0.11351	250.13	119.10	126.43	95.40	31.03	55.85	* 0.00	247.54	4.43	[illegible]
WORST CONDITION	1.963				119.02	126.36			104.25	* 0.00	247.54	2.37	1.0
PLATE EXIT	1.969	0.00174	0.11526	250.13									
STRETCH(2) INLET	1.972	0.00012	0.11537	234.94									
STRETCH(2) OUT	1.974	0.00153	0.11690	234.94									
STRETCH(3) INLET	1.979	0.01390	0.13080	130.89									
STRETCH(3) OUT	1.983	0.00007	0.13087	130.89									
STRETCH(4) INLET	1.932	0.02962	0.16048	242.52									
STRETCH(4) OUT	1.944	0.00122	0.16171	242.52									
STRETCH(5) INLET	1.929	0.01481	0.17652	242.52									
STRETCH(5) OUT	1.929	0.00000	0.17652	242.52									
OUTLET PLENUM	1.929	0.02962	0.20613	0.00									

DNBID=1 Q1=0.005*G**0.611 ---- DNBID=2 Q2=(A/AH)(DHI/HFG)*G ---- DNBID=3 Q3=0.7(A/AH)RT(W/R)/RT(1+(RS/RL)**0.25)

*--- TCLAD < TSAT

HEAT FLUX OF CLAD --- Q*FR*FH*FL

(KARMAN - NIKURADSE EQUATION WAS USED FOR WALL LOSS CALCULATION)

H E A T F L U X (CLUD) = Average heat flux x FR x FFILM x FLOCL x FZ x (FHFLX/FFILM)
= Average heat flux x FR x FHFLX x FLOCL x FZ

COOLOD-N THERMAL HYDRAULIC CALCULATION CASE = (IA- 1 JA- 1) CALCULATION DATE 94-01-11 PAGE 6

TRIGA TYPE FUEL (PRESSURE , ONB & DNB CONDITION)

	PRESSURE AT Z (KG/CM2A)	PRESSURE LOSS (KG/CM2)	TOTAL LOSS (KG/CM2)	COOLANT VELOCITY (CM/SEC)	TSAT (C)	TONB (C)	TCLAD (C)	DTONB	HEAT FLUX (W/CM2) CLAD	QONB	QLUND	DNBR
INLET PLENUM	2.074			0.00								
STRETCH(1) INLET	1.980	0.07176	0.07176	210.35								
STRETCH(1) OUT	1.980	0.00000	0.07176	210.35								
PLATE ENTRANCE	1.969	0.00241	0.07417	248.55								
FUEL PLATE ZONE 1	1.972	0.00522	0.07939	248.55	119.16	126.49	86.20	40.28	55.85	* 0.00	288.37	5.16
FUEL PLATE ZONE 2	1.973	0.00184	0.08122	248.59	119.18	125.35	71.56	53.80	38.84	* 0.00	286.93	7.39
FUEL PLATE ZONE 3	1.975	0.00313	0.08434	248.66	119.21	126.75	90.15	36.60	59.54	* 0.00	284.38	4.78
FUEL PLATE ZONE 4	1.976	0.00312	0.08744	248.77	119.24	127.84	107.32	20.52	78.77	* 0.00	280.79	3.56
FUEL PLATE ZONE 5	1.966	0.00311	0.09052	248.92	119.07	128.37	119.74	8.64	92.66	0.34	276.06	2.98
FUEL PLATE ZONE 6	1.968	0.00311	0.09358	249.10	119.10	128.68	125.49	3.19	101.16	41.73	271.04	2.68
FUEL PLATE ZONE 7	1.970	0.00310	0.09663	249.29	119.13	128.78	127.33	1.45	104.25	70.88	265.72	2.55
FUEL PLATE ZONE 8	1.971	0.00309	0.09966	249.48	119.16	128.71	126.19	2.53	101.16	51.12	260.39	2.57
FUEL PLATE ZONE 9	1.973	0.00308	0.10268	249.66	119.19	128.46	121.84	6.62	92.66	6.45	255.35	2.76
FUEL PLATE ZONE 10	1.963	0.00308	0.10570	249.82	119.02	127.65	111.71	15.93	78.77	* 0.00	250.57	3.18
FUEL PLATE ZONE 11	1.965	0.00307	0.10871	249.96	119.05	126.61	97.33	29.28	59.54	* 0.00	246.98	4.15
FUEL PLATE ZONE 12	1.967	0.00307	0.11174	250.07	119.08	125.26	81.84	43.43	38.84	* 0.00	244.44	6.29
FUEL PLATE ZONE 13	1.968	0.00180	0.11351	250.13	119.10	126.43	95.40	31.03	55.85	* 0.00	243.00	4.35
WORST CONDITION	1.963				119.02	126.36			104.25	* 0.00	242.87	2.33
PLATE EXIT	1.969	0.00174	0.11526	250.13								
STRETCH(2) INLET	1.972	0.00012	0.11537	234.94								
STRETCH(2) OUT	1.974	0.00153	0.11690	234.94								
STRETCH(3) INLET	1.979	0.01390	0.13080	130.89								
STRETCH(3) OUT	1.983	0.00007	0.13087	130.89								
STRETCH(4) INLET	1.932	0.02962	0.16048	242.52								
STRETCH(4) OUT	1.944	0.00122	0.16171	242.52								
STRETCH(5) INLET	1.929	0.01481	0.17652	242.52								
STRETCH(5) OUT	1.929	0.00000	0.17652	242.52								
OUTLET PLENUM	1.929	0.02962	0.20613	0.00								

DNB HEAT FLUX CALCULATED BY LUND

*--- TCLAD < TSAT

HEAT FLUX OF CLAD --- Q*FR*FH

(KARMAN - NIKURADSE EQUATION WAS USED FOR WALL LOSS CALCULATION)

```
----*--PLOT INPUT DATA FROM FT05 -*----4----*----5----*----6----*----7--

  WITHX     WITHY      TMIN      TMAX      PMIN      PMAX      HMIN      HMAX
140.000   200.000     0.000   800.000     0.000     4.000     0.000   200.000

NEWI= NEW

TITLE= JRR-4 TRIGA-16 CORE CASE:GA BENCHMARK

 IDPLOT= 1 IDPLOT= 2 IDPLOT= 3 IDPLOT= 4 IDPLOT= 5 IDPLOT= 6 IDPLOT= 7
       11        13        23        14        12        22        21
 NSMBL = 1 NSMBL = 2 NSMBL = 3 NSMBL = 4 NSMBL = 5 NSMBL = 6 NSMBL = 7
        0         0         0         0         0         0         0


----*--PLOT INPUT DATA FROM FT11 -*----4----*----5----*----6----*----7--

IA= 1  JA= 1  N= 1  NP= 1  K= 1

   J    X(J)  TCOOLANT     TCLAD     TMEAT      TSAT      TONB     PRESS HEAT FLUX
   1   0.000     37.00     86.20    272.02    119.16    126.49   1.97162     55.84
   2   2.940     37.49     71.56    200.80    119.18    125.35   1.97270     38.84
   3   7.940     38.35     90.15    288.26    119.21    126.75   1.97455     59.53
   4  12.940     39.57    107.32    369.40    119.24    127.84   1.97641     78.75
   5  17.940     41.07    119.74    428.06    119.07    128.37   1.96573     92.65
   6  22.940     42.77    125.49    462.08    119.10    128.68   1.96763    101.14
   7  27.940     44.57    127.33    474.20    119.13    128.78   1.96953    104.23
   8  32.940     46.37    126.19    462.78    119.16    128.71   1.97144    101.14
   9  37.940     48.07    121.84    430.17    119.19    128.46   1.97336     92.65
  10  42.940     49.58    111.71    373.79    119.02    127.65   1.96267     78.75
  11  47.940     50.79     97.33    295.43    119.05    126.61   1.96460     59.53
  12  52.940     51.65     81.84    211.08    119.08    125.26   1.96651     38.84
  13  55.880     52.14     95.40    281.22    119.10    126.43   1.96764     55.84

PLOT START,  JMAX3=  16
II = 1  PLOT END
II = 2  PLOT END
II = 3  PLOT END
II = 4  PLOT END
II = 5  PLOT END
II = 6  PLOT END
II = 7  PLOT END

NORMAL END
```

Appendix B Sample JCL for COOLOD-N2

```
   T(01) W(03) I(03) C(03) GRP
// EXEC      LMGOEX,LM='J3907.CLDN9305',Q='.LOAD',
//           PRN=TEMPNAME,A='ERRCUT=0',GOSYSIN='DDNAME=SYSIN'
//FT05F001 DD DSN=J3907.JRR4ROD.DATA(GA002),DISP=SHR
// EXPAND    DISK,DDN=FT11F001
// EXPAND    GRNLPLIM,SYSOUT=U,OTLIM=300000
//
```

The COOLOD-N2 code is also available for NEC-PC9801 series and IBM PC-AT compatibles (PC version, there are some restrictions).