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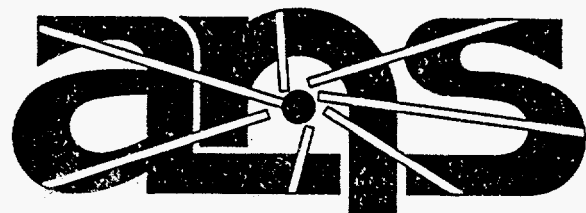
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**Description of TASHA:
Thermal Analysis of Steady-State-
Heat Transfer for the Advanced
Neutron Source Reactor**

D. G. Morris
N. C. Chen
W. R. Nelson
G. L. Yoder

October 1996

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Advanced Neutron Source

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DESCRIPTION OF TASHA:

**THERMAL ANALYSIS OF STEADY-STATE-HEAT TRANSFER FOR THE
ADVANCED NEUTRON SOURCE REACTOR**

**D. G. Morris
N. C. Chen
W. R. Nelson
G. L. Yoder**

**Prepared: January 1996
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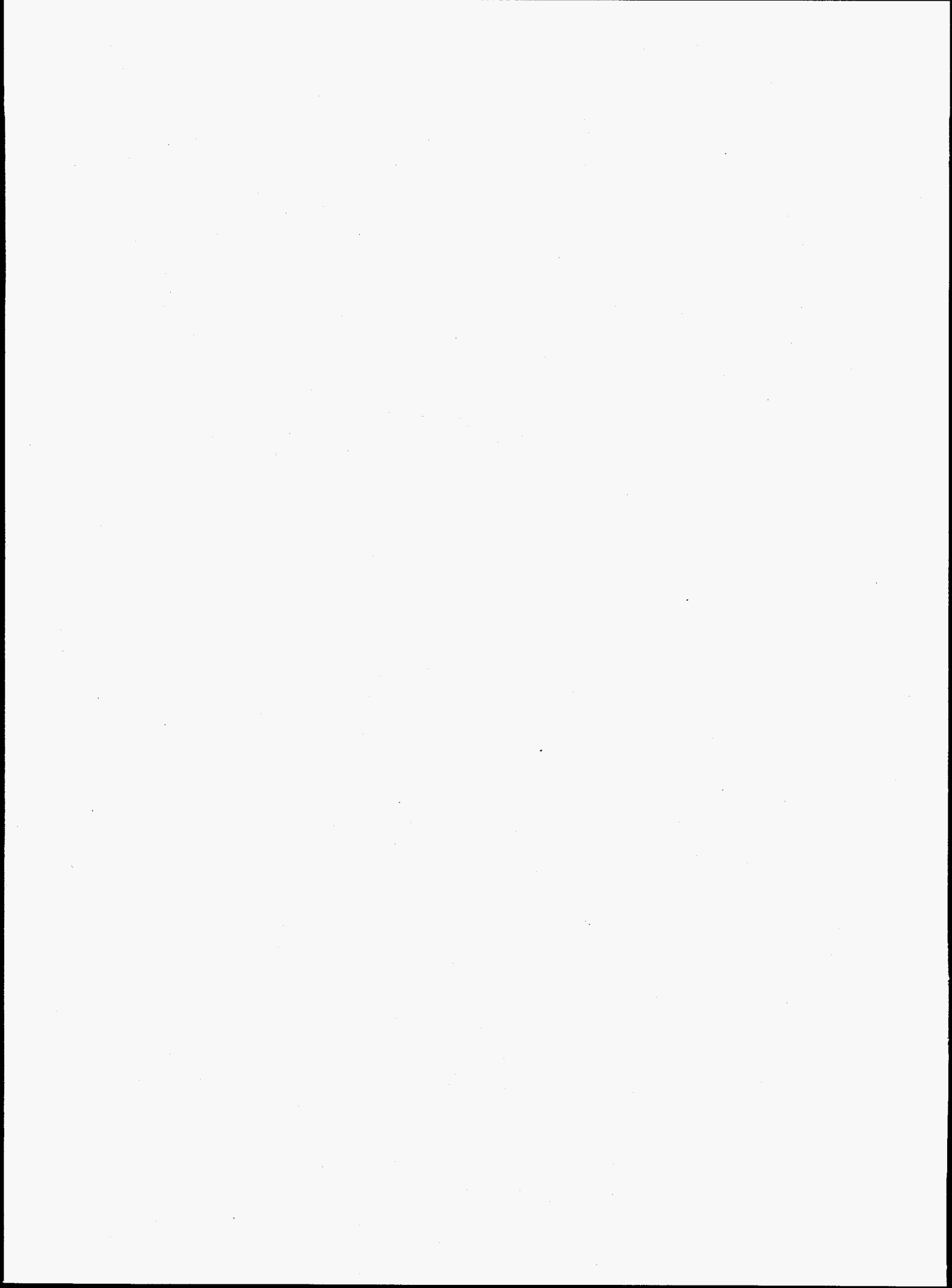
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ACRONYMS

1-D	one-dimensional
2-D	two-dimensional
3-D	three-dimensional
ANS	Advanced Neutron Source
B-R	Bergles-Rohsenow
CHF	critical heat flux
FE	flow excursion
HFIR	High Flux Isotope Reactor
IB	incipient boiling
SPF	statistical peaking factor
TASHA	<u>T</u> hermal <u>A</u> nalysis of <u>S</u> teady-State- <u>H</u> eat Transfer for the <u>A</u> NS
THTL	Thermal-Hydraulic Test Loop



NOMENCLATURE

a	constant (0.95071362)
b	constant (0.40980412)
b	gap of a rectangular channel or annulus (m)
c	constant (0.3303747)
C_p	bulk coolant specific heat (kJ/kg · K)
d	constant (-0.1605523)
D_e	equivalent channel diameter (m)
e	constant (-0.13561859)
f	constant (0.01459134)
f_D	friction factor
F_{D_2O}	correction factor for applying B-R correlation for heavy water
F_{sub}	subcooling factor
g	acceleration due to gravity (9.81 m/s ²)
g	constant (0.012468927)
h_{FC}	forced convection heat transfer coefficient (kW/m ² · K)
h_{lv}	latent heat of vaporization (kJ/kg)
k	thermal conductivity (kW/m · K)
k_b	bulk coolant thermal conductivity (kW/m · °C)
k_c	clad thermal conductivity (W/m · °C)
k_m	fuel meat thermal conductivity (kW/m · °C)
k_{ox}	oxide thermal conductivity (W/m · °C)
k_t	rate constant (μm ⁿ⁺¹ /h)
K_H	correction factor for heated liquids
n	constant (mechanism number)
P	pressure (Pa)
Pr_b	bulk coolant Prandtl number
q''	local heat flux (kW/m ²)
q'''	volumetric heat generation rate (kW/m ³)
q''_{CHF}	critical heat flux (kW/m ²)
q''_{IB}	local incipient boiling heat flux (kW/m ²)
q''_{Costa}	Costa correlation predicted flow excursion heat flux (kW/m ²)
q''_{Moritz}	Moritz correlation predicted flow excursion heat flux (kW/m ²)
$q''_{pool,sub}$	subcooled pool boiling heat flux (kW/m ²)
$q''_{conv,sub}$	subcooled convective heat flux (kW/m ²)
Re_b	bulk coolant Reynolds number
s	span of a rectangular channel (m)
t	time (h)
t_c	cladding thickness (m)
t_m	fuel meat thickness (m)

t_{ox}	oxide thickness
T_{sat}	saturation temperature (K)
$T_{sat,ex}$	saturated exit temperature (K)
T	temperature (K)
T_b	bulk coolant temperature (K)
T_c	fuel-clad interface temperature (K)
T_{cl}	centerline temperature (K)
$T_{m/o}$	oxide-clad interface temperature (K)
T_w	wall temperature (K)
$T_{x/c}$	oxide-coolant interface temperature (K)
x	oxide thickness
x_0	oxide thickness at $t = 0$ (μm)
x_t	oxide thickness at time t (μm)
V	coolant velocity (m/s)
z	distance (m)

GREEK SYMBOLS

ρ_l	liquid coolant density (kg/m^3)
ρ_v	vapor coolant density (kg/m^3)
σ	surface tension (N/m)
$\Delta\rho$	liquid to vapor density difference (kg/m^3)
ΔT_{sub}	subcooling temperature difference (K)
μ_b	bulk dynamic viscosity ($\text{Pa} \cdot \text{s}$)
μ_w	wall dynamic viscosity ($\text{Pa} \cdot \text{s}$)

ABSTRACT

This document describes the code used to perform Thermal Analysis of Steady-State-Heat-Transfer for the Advanced Neutron Source (ANS) Reactor (TASHA). More specifically, the code is designed for thermal analysis of the fuel elements. The new code reflects changes to the High Flux Isotope Reactor steady-state thermal-hydraulics code. These changes were aimed at both improving the code's predictive ability and allowing statistical thermal-hydraulic uncertainty analysis to be performed. A significant portion of the changes were aimed at improving the correlation package in the code. This involved incorporating more recent correlations for both single-phase flow and two-phase flow thermal limits, including the addition of correlations to predict the phenomenon of flow excursion. Since the code was to be used in the design of the ANS, changes were made to allow the code to predict limiting powers for a variety of thermal limits, including critical heat flux, flow excursion, incipient boiling, oxide spallation, maximum centerline temperature, and surface temperature equal to the saturation temperature. Statistical uncertainty analysis also required several changes to the code itself as well as changes to the code input format.

This report describes these changes in enough detail to allow the reader to interpret code results and also to understand where the changes were made in the code programming. This report is not intended to be a stand alone report for running the code, however, and should be used in concert with the two previous reports published on the original code. Sample input and output files are also included to help accomplish these goals.

In addition, a section is included that describes requirements for a new, more modern code that the project planned to develop.

1. INTRODUCTION

A new steady-state thermal-hydraulics code has been developed to perform core thermal-hydraulic design in support of the Advanced Neutron Source (ANS) Reactor. The code is based upon an existing one developed in support of the High Flux Isotope Reactor (HFIR).^{1,2} Space- and time-dependent relative power density profiles, generated with detailed two-dimensional (2-D) (r,z) neutronics calculations, are used as input to the new code. It is designed to calculate maximum allowable powers using user-selectable limiting criteria of maximum centerline temperature, maximum oxide temperature drop, incipient boiling (IB), critical heat flux (CHF), flow excursion (FE), and wall temperature equal to the local saturation temperature.

In the analysis, the radial coordinate within each fuel element (and the relative power density profile) is translated into the spanwise direction (along the arc length of the fuel plate). Coolant channels are then assumed to be bounded by flat plates with a width equal to the fuel plate span. Three separate channel types are analyzed, one with an average coolant gap, one with a minimum coolant gap, and one with the maximum allowable coolant gap.

Each channel is split into spanwise subchannels and axial cells, forming a grid on the fuel plate (see Fig. 1 for a diagram of the grid). An energy balance and dynamic and frictional pressure gradients are calculated across each axial cell. Mass flow through each subchannel is iteratively determined by forcing the overall pressure drop across each subchannel to be equal to the specified core pressure drop. Losses because of entrance and exit effects are accommodated in the code using standard loss coefficients, while frictional pressure drop is calculated using a friction factor correlation. All property values vary appropriately with temperature along the channel. Correlations for the properties of both light and heavy water have been developed and incorporated in the code; their errors are quantified in ref. 3. Each cell within the grid has an associated local relative power density within the fuel plate that is calculated via neutronics considerations. A forced convection heat transfer correlation is used to determine the local water/oxide interface temperature ($T_{x/c}$). Temperatures within the fuel clad and fuel meat are then calculated using the one-dimensional (1-D) heat conduction equation.

In order to update the code a series of modifications has been made to the original HFIR steady-state thermal-hydraulics code.¹ These changes have included modifications to the calculational scheme and changes to the correlations included in the code. An initial modification of this code was made by Cole et al. and was documented in ref. 2. This report presents additional changes made since that date. As with ref. 2, this document is not intended to stand alone, but is intended to be used along with both refs. 1 and 2 in order to provide a reasonably complete synopsis of the state of the code. During the ANS Project, it was found useful to develop an acronym for the code to make it more convenient to reference. The acronym TASHA (Thermal Analysis of Steady-State-Hydraulics for the ANS) is frequently used in other reports and documents to describe the code. Since it has been continuously improved and revised throughout the project, generic references to TASHA do not necessarily refer to the same version of the code. This document presents changes made up to the time that the ANS Project was terminated. It was always anticipated that the project would develop a new, better organized, steady-state thermal-hydraulics code in order to perform final design calculations, which would also be used during the operation of the reactor. Unfortunately, time and funding never permitted this development to take place. If

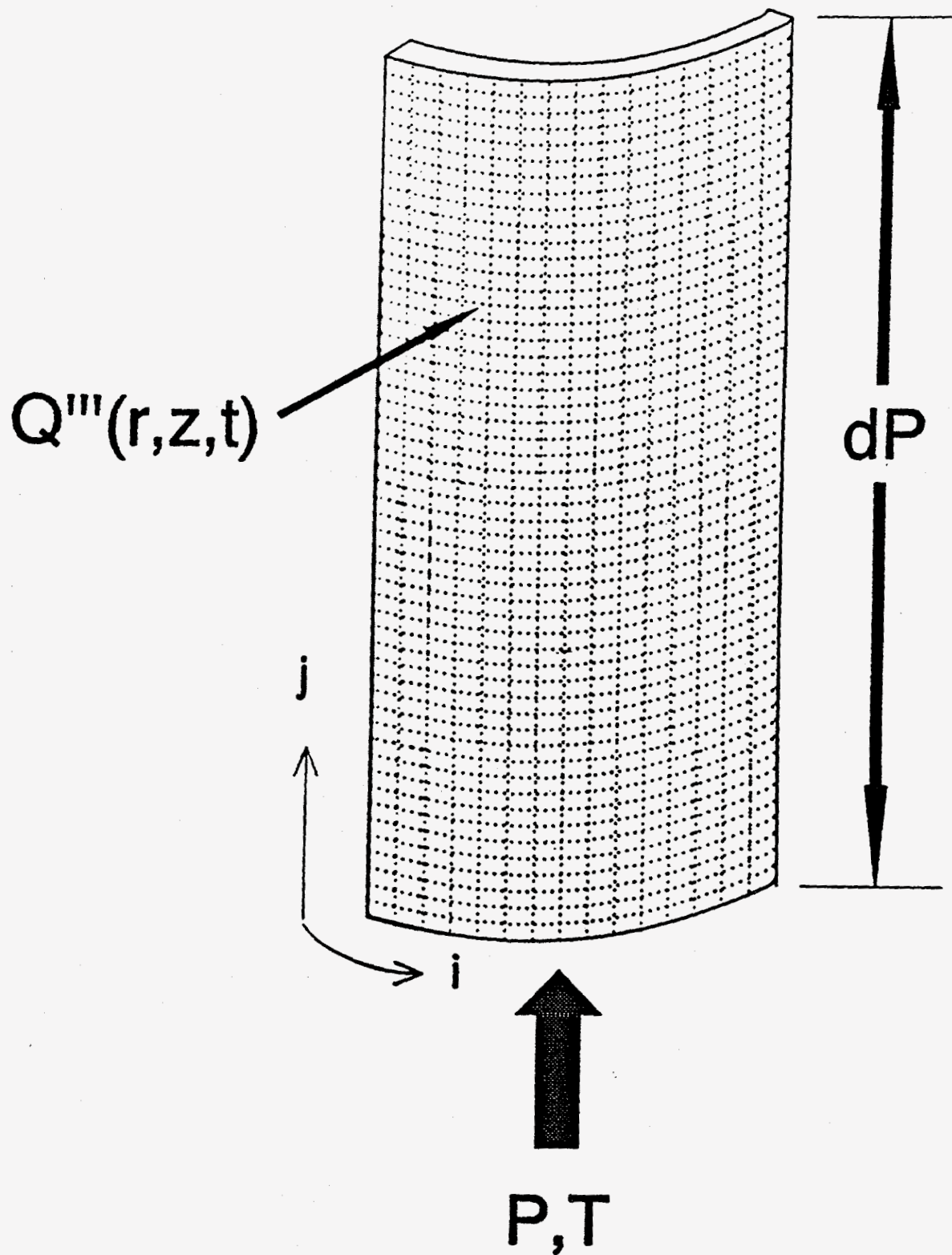


Fig. 1. Fuel plate grid geometry.

a new reactor design were to be pursued, the authors would recommend that a new code be developed. However, this documentation will provide some insight into the thought processes and the components that would be necessary to develop a new code. In addition, as it presently exists, this code could be a useful tool for calculating operating conditions and margins for existing research reactors with plate fuel design (such as HFIR), because it incorporates up-to-date thermal-hydraulic correlations.

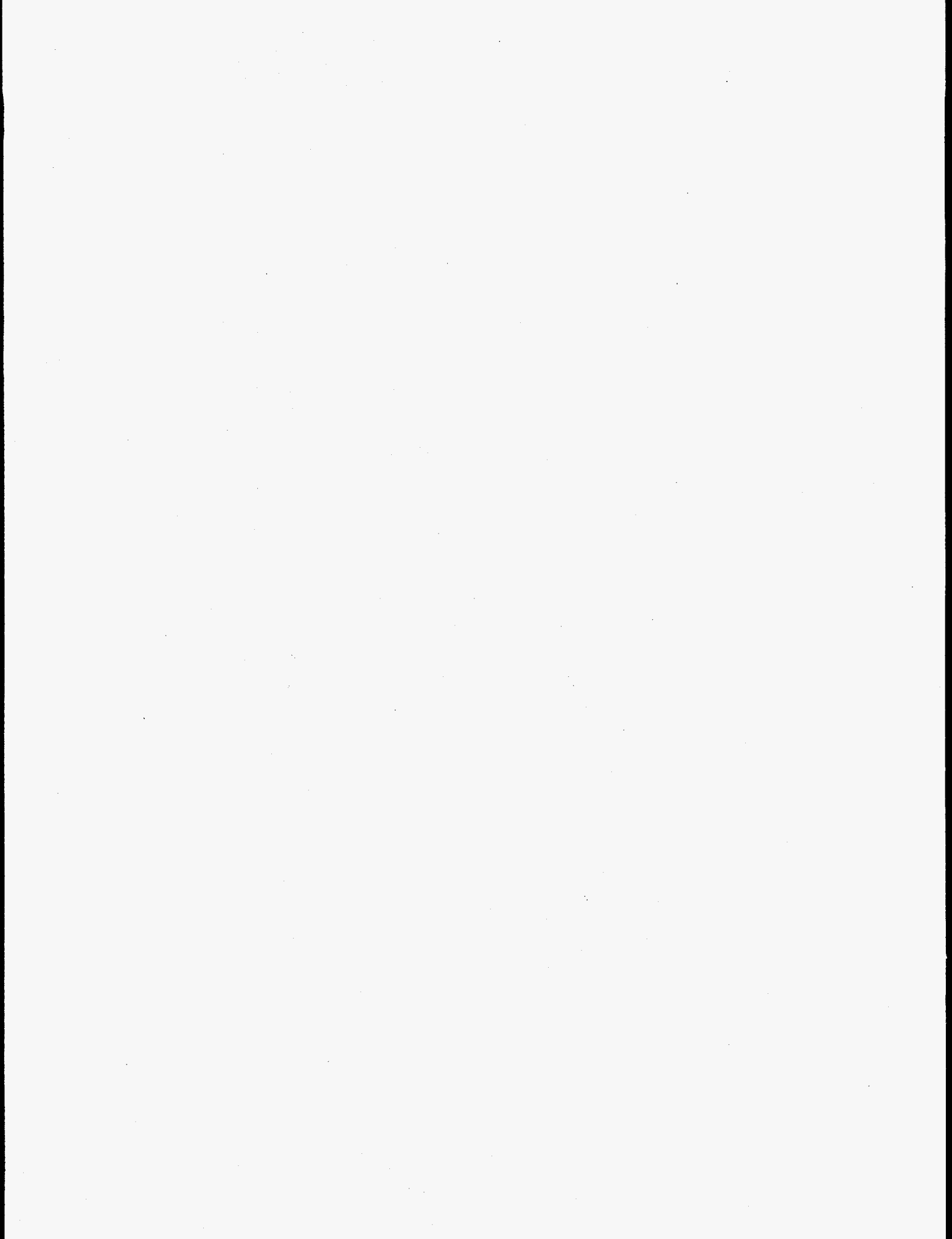
This report is loosely divided into seven sections, the first being this introduction, and the last being conclusions. Changes to the code programming are discussed in Chap. 2. A significant amount of time was devoted to selecting appropriate correlations for use in the code. New correlations have been incorporated, including those for the single-phase heat-transfer coefficient, the oxide growth rate, the CHF limit, and the single-phase friction factor relationship, as well as others. Where possible, these changes have been made in a manner to give the user the option of using these new correlations or the original ones, thereby providing a significant amount of flexibility to the user. One of the objectives of the ANS Project was to develop a defensible methodology of statistically incorporating thermal-hydraulic uncertainties in the analysis. This required evaluation of various techniques and development of uncertainty distributions for incorporation in the analysis. That work is presented in refs. 4 and 5. An important aspect of the work was the development of a code system that would allow Monte Carlo sampling of uncertainty levels using the TASHA code to evaluate thermal-hydraulic conditions within the ANS core. The SAMPLE⁶ code was used for this analysis and was combined with TASHA in order to develop statistically-based maximum reactor power limit distributions. Additionally, a series of changes to the code were made for various other purposes, including improving the user interface with the code, improving the convergence, and allowing the code to calculate conditions for a three-element core. With the exception of some of the changes made to enable the integration of TASHA and SAMPLE, these are discussed in the Chap. 2.

Chapter 3 discusses revised input requirements, highlighting differences between the present code input and that of ref. 2. Chapter 4 shows some example calculations and detailed calculational results. Chapter 5 outlines the various verification checks that have been made on the code to insure calculational fidelity.

As was stated above, it was always intended to develop a more modern, more organized code to perform steady-state thermal-hydraulic analysis. Although a task for this work existed in the task plans, funding and time limitations never permitted the work to begin. However, a series of requirements for the new code was developed and is presented in Chap. 6. This chapter should, therefore, provide some insight into a potential path to pursue in developing a thermal-hydraulic code for use in research reactor design.

It is, in fact, impossible to have such a clear delineation in the actual programming of the code, and, therefore, it is to be expected that some overlap between the chapters will exist.

The TASHA source code, example input and output files, and a brief readme file are stored in electronic form in the ANS archives. In addition, although the integrated SAMPLE/TASHA code was not documented, similar information for this code is also stored electronically in the archives.



2. MODIFICATIONS TO TASHA

2.1 THERMAL-HYDRAULIC CORRELATIONS

The original thermal-hydraulic correlation package used in the code had not been significantly updated since the original code was developed. As part of the ANS Project, an evaluation of available correlations was initiated, and, as a result, a new correlation package was incorporated in the code.⁷

2.1.1 Single-Phase Heat Transfer

The original single-phase heat-transfer coefficient was a modification of the Hausen equation.⁸ The single-phase correlation for the heat-transfer coefficient of Petukhov⁹ has been added to the code with some modifications for variable physical properties and rectangular channels:

$$h_{FC} = \frac{k_b}{D_e} \frac{(f_D/8) Re_b Pr_b (\mu_b/\mu_w)^{0.11}}{(1+3.4f_D) + (11.7 + \frac{1.8}{Pr_b^{1/3}}) (f_D/8)^{1/2} (Pr_b^{2/3} - 1)} \quad (1)$$

The Filonenko correlation¹⁰ is used for the Darcy friction factor,

$$f_D = \frac{[1.0875 - 0.1125 (b/s)]}{(1.82 \log_{10} Re_b - 1.64)^2} \quad (2)$$

All parameters and physical properties are evaluated at local bulk conditions except μ_w , which is evaluated at the wall temperature. Although the heat-transfer coefficient is applied to both hot spot and average channel analysis, the wall temperature, and thus the wall viscosity, is always evaluated at average channel conditions, since the boundary layer behavior is not expected to be impacted significantly by a 2-mm diam hot region in the fuel plate (the spot size used for hot spot analysis). Therefore, the hot spot temperature would not be appropriate for evaluation of the wall viscosity correction term. The Petukhov correlation is evaluated in subroutine PETUKH. The gap, b , and span, s , are input parameters that are passed to the PETUKH subroutine. The Filonenko friction factor used for this correlation is also included in this subroutine.

2.1.2 Single-Phase Friction Factor

A new subroutine has been added to the code in order to incorporate a new friction factor correlation for single-phase forced-convective flow. The Filonenko correlation,¹⁰ can be selected

from the input, as well as the friction factors originally included in the code. The Filonenko correlation also includes a modification for heating in rectangular channels:

$$f_D = \frac{[1.0875 - 0.1125 (b/s)]}{(1.82 \log_{10} Re_b - 1.64)^2} \left[\frac{7 - (\mu_b/\mu_w)}{6} \right]. \quad (3)$$

The correction factor for rectangular channels is based on the Bhatti and Shah¹¹ correlation using the channel span-to-gap aspect ratio. The viscosity ratio term was proposed later by Petukhov⁹ in conjunction with his forced-convection correlation,

$$K_H = \left[\frac{7 - (\mu_b/\mu_w)}{6} \right]. \quad (4)$$

Note that this last correction factor *should not* be used with Petukhov's forced-convection correlation [Eqs. (1) and (2)] because Petukhov's correlation already includes a separate correction for variable physical properties. The correlation is included in addition to the friction factors originally in the code. A selection between the different friction factor correlations is made in the input file. Subroutine FILOFRIC contains the FORTRAN for the Filonenko friction factor. The input variable, IFRIC, should be set as 2 to select the Filonenko correlation.

2.1.3 Critical Heat Flux

The method of calculating the CHF or the point of departure from nucleate boiling is now based on the Gambill additive CHF Correlation,¹² the Weatherhead correlation¹³ for CHF wall temperature, and the Petukhov correlation⁹ for calculating the forced-convection heat-transfer coefficient [see Eq. (1)]. These three correlations are iteratively solved to arrive at the appropriate CHF value and the corresponding wall temperature. This combination is designated as the Gambill-Weatherhead correlation and is implemented as follows:

$$\begin{aligned} q''_{CHF} &= q_{pool,sub} + q_{conv,sub}, \\ q''_{pool,sub} &= 0.18 h_{lv} \rho_v (\sigma g \Delta \rho / \rho_v^2)^{0.25} F_{sub}, \\ F_{sub} &= 1 + (\rho_l / \rho_v)^{0.75} [C_p \Delta T_{sub} / (9.8 h_{lv})], \\ q''_{conv,sub} &= h_{FC} [T_w - T_b], \end{aligned} \quad (5)$$

with T_w at the CHF point being calculated by iteration using Weatherhead's correlation¹³ for fully developed nucleate boiling of water:

$$T_w = [47.7 - 0.127 (T_s - 273.16)] (q''_{CHF} / 3154.6)^{0.25} + T_{sat} \quad (6)$$

and using the Petukhov correlation⁹ for the turbulent forced-convection, nonboiling, heat-transfer coefficient in combination with the Filonenko correlation¹⁰ for the friction factor [see Eqs. (3) and (4)]. The correlation is implemented in the code in addition to the scheme used by Cole et al., which used an earlier version of Gambill's additive method. When the heavy water option (ICOOL = 2) is chosen, the Gambill-Weatherhead CHF correlation is used. If the lightwater option (ICOOL = 1) is chosen, the original CHF scheme is used. The Gambill-Weatherhead correlation is implemented in subroutine EQ72. To select the CHF thermal limit, set input variables IBO and IQIC equal to 1.

2.1.4 Flow Excursion Correlations

Two flow excursion correlations have been implemented in the code, the first from Costa,¹⁴ and the second from Moritz,¹⁵

$$q''_{Costa} = \frac{V^{1/2} (T_{sat,ex} - T_b)}{0.0128} \quad (7)$$

$$q''_{Moritz} = \frac{9.995 \times 10^2 V (T_{sat,ex} - T_b)}{(21.2 + 0.7708V)} \quad (8)$$

Either of these correlations can be selected for use. In addition, the user may also decide if the calculated heat flux values are compared to either the hot streak or hot spot surface heat fluxes. The input variable IQFL is the flow excursion flag and is described later in Sect. 3.1.3. These correlations are included in subroutine LOOP8.

2.1.5 Incipient Boiling Correction Factor

The IB correlation used in the code remains the Bergles-Rohsenow (B-R) correlation;¹⁶ however, a multiplier ($F_{D,O}$) has been added to the correlation so that it can be used for heavy water (the correlation was originally developed for light water). In addition, the correlation is now used in combination with the Petukhov¹⁰ forced-convection correlation for calculating the corresponding wall temperature. The formulation now used in the code is:

$$q''_{IB} = (F_{D_2O}) \times 1.7978 \times 10^{-6} P^{1.156} [1.8 (T_w - T_{sat})]^{(2.8285/P^{0.0234})},$$

$$q''_{IB} = h_{FC} (T_w - T_b). \quad (9)$$

T_w is calculated by iteration through the Petukhov⁹ correlation for the forced-convection heat-transfer coefficient, which uses the Filonenko correlation¹⁰ for the friction factor, as modified for rectangular channels [see Eqs. (1) and (2)].

Using the Davis and Anderson¹⁷ correlation for IB, a comparison of IB limits was made between heavy water and light water. Based on this comparison, a correction factor was developed as a function of pressure for use with the B-R correlation for heavy water rather than light water,⁴

$$F_{D_2O} = (a+c \cdot P+e \cdot P^2+g \cdot P^3)/(1+b \cdot P+d \cdot P^2+f \cdot P^3). \quad (10)$$

This correlation is included in the code when calculating the IB heat flux and is found in subroutine EQ72. The IB thermal limit is selected by setting input variables IBO = 2 and IQIC = 1.

2.2 FUEL PLATE OXIDE CORRELATIONS

Oxide growth rate can now be calculated using either the Griess correlation,¹⁸ which was originally in the code, or by using one of two correlations developed specifically for the ANS reactor. Both of the latter correlations have the same form as the original Griess correlation. They are based on the general rate equation:

$$\frac{dx}{dt} = \frac{k_t}{x^n}. \quad (11)$$

Early kinetic results for the ANS steady-state experiments were consistent with the assignment of $n = 0.351$, so that the integrated form of the equation is:

$$x_t = (x_o^{1.351} + 1.351 \cdot k_t \cdot t)^{0.74}. \quad (12)$$

The shape of the growth rate curves is essentially identical to those of Griess¹⁸ (and Kritz¹⁹) who reported a growth exponent of 0.778. If the model described in Eq. (11) is accurate, Eq. (12) can be used to predict oxide layer thicknesses for any set of specified conditions, providing the rate constant, k_t , can be assigned for the conditions involved. Thus, a "correlation" essentially defines the rate constant as a function of its critical variables (in this case, mainly the thermal-hydraulic parameters that control temperatures across the reacting system). Since Eq. (12) was first presented, several additional experiments that met its requirements were added to the data base. These have reinforced its applicability. A conservative equation for k_t (except at very low rates) was developed from these data, and is given by²⁰

$$k_t = 6.992 \times 10^5 \exp [-7592/(T_b + 10^3 q^{1/2})] \mu m^{1.351}/h . \quad (13)$$

This correlation is incorporated in the code in subroutine ANSCOR and is selectable by setting input variable ICOR = 2. The experiments contributing to this data base were all conducted under conditions where the nominal coolant pH was 5; the coolant velocity was 25.6 m/s; and the coolant inlet temperature was between 39 and 49°C.

An additional correlation²¹ has also been developed from ANS data, which is not as conservative as that presented in Eq. (13):

$$k_t = 6.388 \times 10^7 \exp [-9154/(T_{x/c} + 1056. q^{1/2})] \mu m^{1.351}/h . \quad (14)$$

This correlation has also been incorporated in the code in order to provide a less conservative method of predicting the oxide growth rate. It is included in subroutine ANSCOB (set input variable ICOR = 3).

Spallation of the boehmite films was observed toward the end of some of the more aggressive loop experiments. Metallographic examinations of such specimens showed that spallation was followed by the onset of internal reactions in the metal beneath the oxide. The presence of an extensive internal reaction zone is clearly detrimental to efficient heat flow, as well as structural integrity, and should not be allowed to occur in the ANS fuel cladding. A new input variable, TSPLM, has been added to specify the spallation limit in terms of the temperature drop across the oxide film. Steady-state experiments indicate that spallation will not occur if the temperature drop across the growing oxide film is maintained at less than 119°C, and this limit is normally used as a criterion for core calculations. Additionally, spallation was not induced by moderate temperature cycling at or near the end of the tests.

The "timing" of the oxide growth calculation has been adjusted in the code so that it is conservatively accounted for in maximum power calculations. These calculations are performed at selected time points using a series of time-dependent power density distributions. Previous calculations at each time point assumed oxide only grew up to the point in time at which the power distribution was prescribed. If this distribution is assumed to apply until the next time point, then the calculated maximum power could be nonconservative. The code now accounts for the "extra" oxide growth time.

2.3 THERMOPHYSICAL PROPERTIES

The thermophysical properties correlations in the code have been changed to those developed specifically for the ANS Project. These updates were made for both light and heavy water. The applicable range for these correlations is from 35–250 °C for heavy water and from 20–300 °C for light water. The subroutines developed for these calculations, however, do not warn the user that the properties are being calculated out of the correlation range. The user must, therefore, be cautious

to ensure that calculations are not inadvertently made out of the applicable range. A complete description of these correlations is presented in ref. 3 and will not be repeated here. The thermophysical properties correlations are found in subroutines PROPS and PROP2.

2.4 THERMAL LIMITS

The original HFIR steady-state thermal-hydraulic code as well as the later Cole et al. version allowed calculation of maximum operating powers (the power where the thermal limit is first reached for at least one point in the core) using IB and CHF as the limiting thermal conditions. It was found early in the ANS program that examination of additional limits was necessary. The code presently has the capability of calculating maximum powers based on 6 user-selectable thermal limits—maximum centerline temperature, maximum oxide temperature drop (spallation), IB, CHF, FE (based on either the local hot spot or the hot streak heat flux)—and wall temperature equal to the local saturation temperature. In addition, a combination of these limits can be selected (e.g., maximum power based on the most limiting criterion of CHF, FE, maximum centerline temperature, and spallation limits).

A modification to the code was also made to enable it to calculate an oxide film thickness that is "consistent" with the calculated maximum power (i.e., the oxide thickness is calculated based on the maximum power). This is an option that can be used for calculated maximum power below the nominal (desired) level, above nominal, both, or neither. In the first case, the calculation would provide a more realistic estimate of the true maximum nominal (continuous) power level. When the calculated maximum power is above nominal and the interest is in brief power excursion margins or simply the margin from nominal conditions to the maximum power, this option should not be used. However, for continuous operation at higher than nominal power levels, this option should be used. The input variable ICONSI controls how the oxide film thickness is calculated relative to the calculation of the maximum power.

A second input modification involves NOTI, the number of time steps. In the revised code, two parameters are specified to indicate the number of time steps to reach the first and final times of interest in the fuel cycle. These parameters are NOTINP and NOTIT, respectively, and replace the single input parameter (NOTI). With the revised input, the code calculates maximum power at the first (NOTINP) through the last time steps (NOTIT). For example, a specification of NOTINP = 1 and NOTIT = 1 would cause the code to calculate the maximum power only at the first time step. If NOTINP = 1 and NOTIT = 5, then maximum power would be calculated for time steps 1 through 5.

A third modification involves the two lines input to specify the initial guesses of the maximum core power and the core pressure drop. These lines are only required (in both the original and revised codes) if MQO is input as 2. In the revised code, the second line of the two contains additional input to specify thermal limit criteria information, and both lines are required for each time step.

2.5 SOLUTION ALGORITHMS

In order to improve the calculational stability of the code and to ensure that the code arrived at the correct numerical solution after the updates were incorporated, several modifications to the numerical schemes were made.

1. In some cases, convergence problems occurred in calculating the wall temperature, T_w , using the Weatherhead correlation during a maximum core power search. The cause was a high guess (trial) of the core power that caused the calculated bulk coolant temperature (T_b) to be higher than the saturation temperature and that resulted in a negative calculated CHF. This problem was fixed simply by checking to ensure that the calculated CHF was greater than zero, and, if not, it was set to zero. In addition, a check on the number of iterations used to calculate T_w was implemented, and, if the number exceeds 100, the calculation is terminated and an error message written.
2. It was found that for some cases where all uncertainties are removed, the code will converge to a maximum power that has subchannel(s) with exit T_b greater than the saturation temperature. Since the code only accounts for single-phase flow, this is an erroneous solution. A new power limit flag, ITSB, was implemented to prevent this from occurring. Because of this change, the nomenclature for the phenomena limiting the core power was changed as follows:

Limit	Limit definition
1	Fuel centerline temperature
2	Oxide temperature drop
3	Fuel surface temperature = T_{sat}
4	$T_f = T_{sat}$
5	IB or CHF
6	Flow excursion

3. During the investigation of the problem described in item 1, it was found that the convergence algorithm for core maximum power was not working properly (nonconverging oscillations) in some cases when CHF was limiting. For this limiting phenomenon, the Newton-Raphson method,²² which had been used, was replaced by a simple method that estimates the new maximum core power as the average of the previous estimate and the current estimate.
4. A revision was made to implement the use of variable fluid density and friction factor for each node along the length of each subchannel in the pressure/flow equations. Previously, subchannel average values were used. The revision obviously

provides a more accurate calculation of pressures and flows. An additional reason for this revision involves the Filonenko friction factor correlation in which a bulk fluid-to-wall viscosity ratio is employed. If a channel average bulk fluid viscosity is used, then a corresponding plate average wall viscosity should be used. However, the use of such an averaged ratio could result in inaccurate flow and pressure calculations. In addition, logic and equations were incorporated to limit the temperature used in calculating the wall viscosity to the saturation temperature. Note that the bulk fluid-to-wall viscosity ratio is used in the Petukhov forced-convection heat-transfer correlation in addition to the Finonenko friction factor correlation. Previously, nodal pressures to calculate the saturation temperature were not calculated in all portions of the code (i.e., Parts 1 and 2).

In the process of making these revisions, the iterative solution scheme used in calculating flows and pressures was replaced. The original scheme was obscure and required an excessive number of iterations to converge. The new scheme uses simple successive substitution and has performed quite well.

2.6 U-FACTORS

Revisions were made with regard to the treatment of the hot streak. The code was initially revised to allow unique hot streak U factors for each axial position in the fuel element mesh. These U-factors are contained in an array named U24M. Only two unique input values have been used in U24M, one for the entrance (lower half of each fuel element) region of the core and one for the exit (upper half of each fuel element) region. The latter was developed by appropriately "averaging" "local" (≈ 1.27 -cm length), potentially above nominal fuel loadings as high as (+10%) along the entrance half of the fuel element. These U factors had then been used to account for hot streak uncertainty effects on T_b and heat flux. The latter is used in determining the oxide buildup and for evaluating flow excursion limits when comparison to hot streak heat fluxes is desired. Since these phenomena can be limiting on a "local" basis, it is nonconservative to use "length-averaged" values contained in U24M. Therefore, the code was revised so that oxide buildup and flow excursion uncertainties are accounted for using a separate U-factor, the "old" U(24) factor (the value or distribution used here would be "larger" than those used in the U24M array).

The accounting of uncertainties in the CHF correlation through the use of the IBO input flag has been eliminated. "If statements" checking the value of IBO in subroutine EQ72 were commented out. This method has been replaced by "reimplementing" the CHF U-factor, U (22). Thus, setting IBO equal to 1 or 3 is equivalent and results in the "straight" evaluation of the CHF correlation. The value calculated is then multiplied by the U(22) factor.

A revision to the code was made to incorporate the IB uncertainty on a heat flux basis. Previously, this uncertainty was incorporated into the code via an uncertainty on the IB correlation wall super-heat along with an uncertainty on the forced-convection heat transfer correlation.

Finally, a new U-factor, U(25), was created for the input of the uncertainty in the flow excursion correlation.

2.7 EXTENDED ANALYSIS

All of the changes discussed in this report are included in the two latest versions of the code (TASHA3ELEMENT and TASHAM2) except two options discussed in this section. The three-element core option is available in version TASHA3ELEMENT of the code, but does not include the arcuate plate option. The arcuate plate option is available in version TASHAM2 but is not available in version TASHA3ELEMENT. Eventually these changes would have been merged together; however, the project was terminated before this task was completed.

2.7.1 HFIR Analysis

The code was revised to allow analysis of the HFIR, thereby providing an updated code for HFIR analysis since TASHA contains correlations more appropriate to HFIR (e.g., modified Filonenko friction factor and Petukhov forced-convection heat-transfer correlations). The code also allows the analysis of HFIR with the additional thermal limits it provides. With the code, HFIR safety margins can be better estimated. Setting the input parameter "ITYPE" to 1 selects the HFIR option (note: need to also specify the input variable "ICOOL" as 1). With this option, HFIR specific fuel deflection analysis and nonbond treatment is invoked. Note that when the latter is invoked the localized fuel nonhomogeneity U-factor is appropriately combined with nonbond effects (modeled with "hardwired" correlations) to provide a net hot spot peaking factor that accounts for fuel meat thickness variations and variations in the forced-convection heat-transfer coefficient. For the purpose of evaluating nominal conditions in HFIR (i.e., "no uncertainties" calculations), ITYPE can be set to 2 causing this "hardwired" uncertainty treatment to be bypassed.

2.7.2 Arcuate Plate Capability

The capability of the TASHA code was upgraded to allow analysis of arcuate-shaped fuel plates. This involved the incorporation of a new subroutine that determines the geometric details (e.g., fuel plate gap width, number of plates in each fuel element) associated with the new fuel plate design. The upgrade necessitated only limited revisions to the original TASHA code.

Initial calculations used a three-point arc fit to the original involute geometry. Points were matched at the inner and outer radii and the center of the fuel annulus (as was presented in ref. 23). The coolant gap at the inner radius was maintained at 1.27 mm, which fixed the interrelationship between fuel plates. Calculations performed using this geometry are presented in Table 1, which shows the percent decrease in maximum power because of this geometry change for each thermal limit. For the IB, CHF, and FE limit, the upper core is limiting, and this geometry change causes only a 3.5%–5% decrease in maximum power.

Table 1. Power decreases because of change from involute^a to arcuate^b plates

Limit	Core	% Decrease ^c
IB	Upper	5.0
	Lower	11.1
CHF	Upper	4.0
	Lower	11.2
FE	Upper	3.8
	Lower	12.0
Oxide	Upper	3.5
	Lower	4.6

Coolant velocity variation across span:

^aInvolute 24.5 m/s – 25.5 m/s

^bArcuate 24 m/s – 26.2 m/s

$$^c \% = \frac{P_{INV} - P_{ARC}}{P_{INV}}$$

Since the arcuate geometry plate is considerably more rigid than an involute one, a calculation was also performed to examine the effect of increasing the nominal velocity from 25 to 26.5 m/s. This increase improves the arcuate plate performance to the point where these plates only present a 0.7% decrease in IB power when compared to the involute plate and a 1% increase in oxide limited power. Additional calculations examined the effect of decreasing the fuel plate thickness by 10% (increasing the number of plates and maintaining the nominal arcuate coolant gap). Results of these calculations indicate that both the IB and the oxide limits are improved to 0.6% above the nominal involute performance.

2.7.3 Three-Element Core Design

The TASHA code was modified to allow analysis of a three-element core design. The modifications maintain the current capabilities of the code, while adding options for the analysis of a third fuel element. The majority of the changes involved adjusting array sizes, formatting for output, etc. Several additional input variables were added:

1. The variable NELE was added to the end of the second input line. A setting of 2 or 3 designates a two- or three-element calculation.
2. A new line containing the inner and outer diameters for the inner and outer side plates for the middle element. This line is inserted after the line with the same quantities for the inner and outer elements.
3. Two new constants C9 and C10 are input on the same line as the other values (C5, C6, C7, and C8) for the uncertainty factor U26. This line is in the time-step dependent data section.

2.7.4 Fuel Plate Conduction Analysis

The capability of performing 1-D conduction calculations through the oxide layer and the fuel plate have also been added to the code. The steady-state 1-D heat flow equation is used:

$$\frac{d^2T}{dz^2} + \frac{q'''}{k} = 0 . \quad (15)$$

The equation is solved for three components, the oxide layer, the fuel clad, and the fuel meat. Within the fuel meat, the solution is:

$$T_{cl} - T_c = 0.125 q''' t_m^2 / k_m . \quad (16)$$

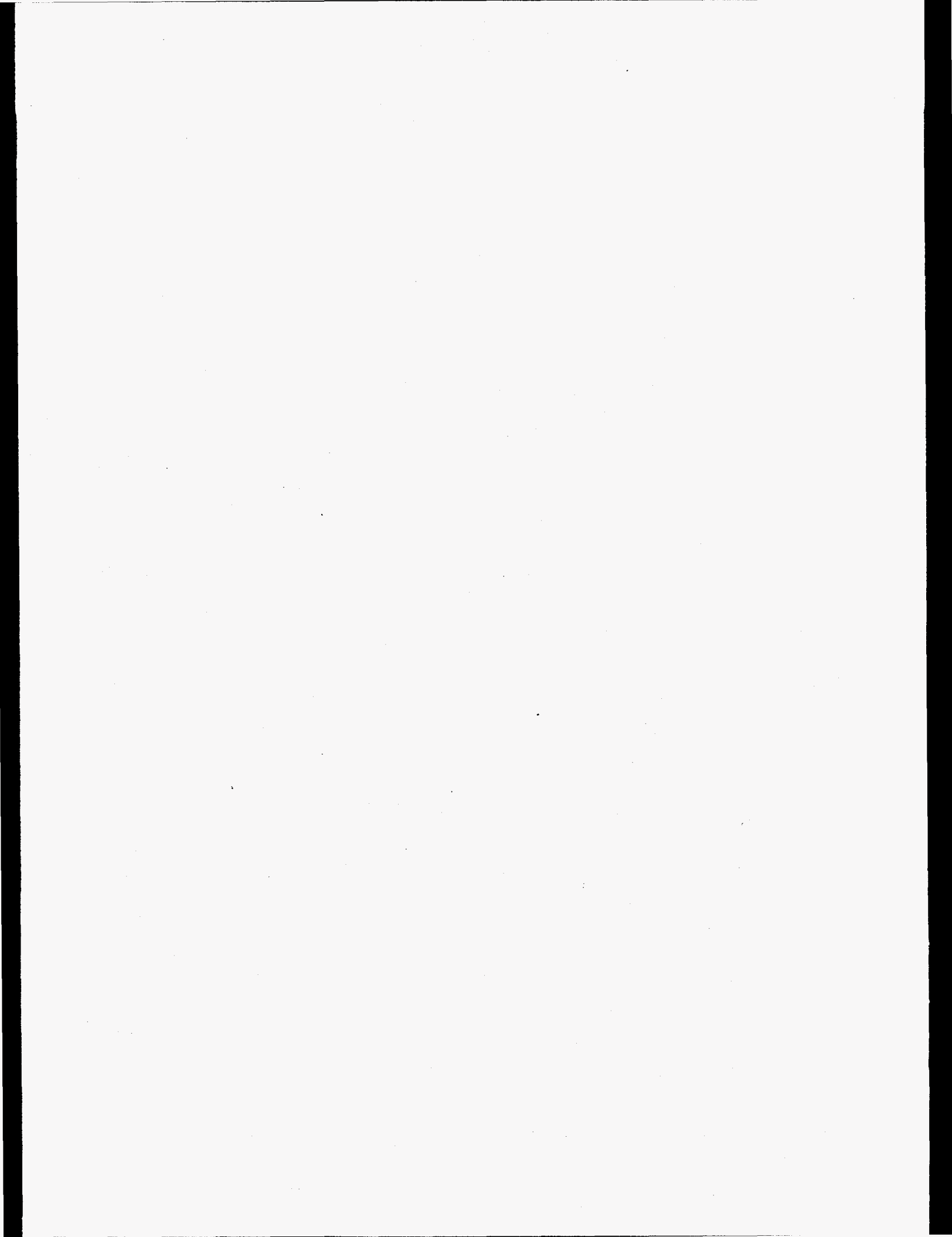
For the cladding, assuming that the heat generation in the cladding is negligible, the solution is:

$$T_c - T_{m/o} = 0.5 q''' t_c t_m / k_c . \quad (17)$$

For the oxide, again assuming that the heat generation in the oxide layer is negligible, the solution is:

$$T_{m/o} - T_{x/c} = 0.5 q''' t_m t_{ox} / k_{ox} . \quad (18)$$

The results calculated using these equations are now part of the code output (using the full output option).



3. INPUT AND OUTPUT

3.1 INPUT

This section has been adapted from ref. 2. Since the code has been converted to FORTRAN77 and list-directed READ statements have been employed throughout, certain rules need to be followed in writing input:

1. There must be no embedded blanks in numbers in the data. Blanks are used as delimiters between successive entries.
2. Each line must contain all the required information. Blanks must not be used for zeros. It is permissible to place comments on a line to the right of the last data item, if space is available.

The data may be divided into three main parts: general run information, fuel element dependent data, and time-step dependent data.

3.1.1 General Run Information

This information is read in either MAIN or Subroutine INPUT. It consists of the following:

1. Title card formatted 20A4 up to 80 character title string,
2. ICOOL ITYPE NELE

ICOOL = 1 for H₂O properties or 2 for D₂O properties,
ITYPE = 1 for HFIR or 2 for ANS,
NELE = 2 for two-element case or 3 for three-element case.

3. MTD IBO LTDIN KCO MQO MSC JUMP ICOR IFRIC ICONSI
JCALC JOXIDE JBUC JDEF JOXRAT

MTD is method option 1 or 2.

IBO = 1 for critical heat flux or 2 for incipient boiling,

LTDIN is output option; LTDIN = 1 produces maximum output or 4 for minimum output (used for maximum power results),

KCO is cold channel option for Part III: set KCO = 2 for this option, otherwise 1,

MQO is option to calculate maximum power: set MQO = 2 for maximum power or = 1 to bypass it,

MSC is stacked case option = 1 for last case, 0 if other cases follow,

JUMP is a flag to bypass printing the hot spot location: set JUMP = 1 to bypass,

ICOR is the oxide correlation option; set to 1 for Griess (HFIR), 2 for ANS corrosion loop, or 3 for modified ANS corrosion loop,

IFRIC is the friction factor option; set to 1 for HFIR or 2 for Filonenko,

ICONSI controls how the oxide is calculated relative to the calculation of the maximum power.

ICONSI = 0 oxide is never based on maximum power (q_{max}),
= 1 if q_{max} / nominal power < 1,
= 2 if q_{max} / nominal power > 1,
= 3 oxide is always based on maximum power,

JCALC is a flag that controls how the maximum core power is calculated.

JCALC = 1 for maximum core power based on the most limiting location in any fuel element.

JCALC = 2 to calculate maximum core powers based on most limiting location in each fuel element.

JOXIDE, JBUC, JDEF, and JOXRAT are options that when set to 1 will cause oxide thickness, fuel plate temperature profile, fuel plate thermal deflection, and oxide ratio, respectively, datasets to be saved to external files. A setting of 0 bypasses.

4. M N NOTINP NOTIT A ST SF QSP

M is the number of radial increments in the mesh (30 maximum).

N is the number of axial increments in the mesh (80 maximum).

NOTINP is the first time step for maximum power calculation.

NOTIT is the last time step for maximum power calculation.

(Maximum of 10 for both NOTINP and NOTIT).

A is the fuel assembly nominal heat transfer area, ft^2 .

ST is the fuel plate thickness in mils.

SF is the fraction of the heat deposited in the fuel assembly.

QSP is heat generation in the side plates, $Btu/(h \cdot in^3)$ @ 100 MW.

5. QH2O P Q DELPF F T B SPN(I), I = 1, NELE

QH2O is heat generation in the coolant, $Btu/(h \cdot in^3)$ @ 100 MW.

P is pressure upstream of the fuel elements, psia.

Q is reactor power in megawatts.

DELPF is pressure drop across the core, psi.

F is a constant used to calculate friction factor.

T is the coolant inlet temperature in degrees F.

B is the constant in the relationship for the increase in fuel plate thickness due to radiation swelling, mils/h. It has been input as 0 for ANS analyses.

SPN is the fuel plate span for the inner, outer, and middle elements in inches.

6. XI(I), I = 1, 4, is an array for selecting geometry cases:

XI(1) = 1 and other XIs = 0 for Case 1; similarly XI(2) = 1 with other XIs = 0 for Case 2 etc., for the four cases. (Only Option II, Cases 1-3, are programmed in this code version.)

7-10. U_i are the uncertainty factors for various parameters:

- U(1) Uncertainty factor for the reactor power level
- U(2) Uncertainty factor for the total heat transfer area
- U(3) Uncertainty factor for the power density distribution
- U(4) Uncertainty factor for the average fuel concentration in the hot plate (input 1.0 as detailed spatial input is specified later)
- U(5) Uncertainty factor for the average fuel concentration in the cold plate (input 1.0 as detailed spatial input is specified later)
- U(6) Uncertainty factor in the inlet water temperature
- U(7) Uncertainty factor for the friction factor correlation
- U(8) Uncertainty factor for the heat transfer coefficient correlation
- U(9) Uncertainty factor for the oxide film correlation
- U(10) Uncertainty factor for the deflection of a fuel plate due to the differential pressure across the fuel plate
- U(11) Uncertainty factor for the deflection due to the difference of the temperature of an individual fuel plate and that of an average fuel plate
- U(12) Uncertainty factor for the increase in the fuel plate thickness due to heating
- U(13) Uncertainty factor for the increase in the fuel plate thickness due to radiation swelling
- U(14) Uncertainty factor for the fuel plate deflection due to longitudinal buckling
- U(15) Uncertainty factor for the longitudinal deflection of the fuel plate due to the radiation growth of the fuel plate material
- U(16) Uncertainty factor for the side plate heat generation rate
- U(17) Uncertainty factor for the coolant heat generation rate
- U(18) Fuel segregation flux peaking factor for the hot side of the fuel plate (hot channel)
- U(19) Fuel segregation flux peaking factor for the cold side of the fuel plate (cold channel)
- U(20) Nonbond factor for the hot channel (see Eq. A-68c and A-68d on p. 140 in ref. 1)
- U(21) Nonbond factor for the cold channel (see Eq. A-68e and A-68f on p. 140 in ref. 1)

U(22) Uncertainty factor for the additive method critical heat flux correlation

U(23) Uncertainty factor for the incipient boiling relation

U(24) Hot streak factor

U(25) Uncertainty factor for the flow excursion correlation

Four lines are required to enter U(1) to U(25) in groups of 7 on the first 3 lines and 4 on the fourth line.

U24M(I), I = 1,N are modified hot streak uncertainty factors. Twelve values are entered per line for a total of N values.

The next N times 2 lines contain U4 (I,J). Two lines are required for M values. U4 is the uncertainty factor for fuel concentration in the hot plate.

The next N times 2 lines contain U5(I,J). Two lines are required for M values. U5 is the uncertainty factor for fuel concentration in the cold plate.

DSIII DSIIO DSIOI DSIOO DSOII DSOIO DSOOI DSOOO

These are the inner and outer diameters in inches for the inner and outer side plates of the inner and outer fuel elements.

DSMII DSMIO DSMOI DSMOO

These are the inner and outer diameters in inches for the inner and outer side plates of the middle fuel element. (Input only for a 3-element case.)

AMIN ALAB CONDSP ALSP

AMIN is the flow area in the annulus between the inner fuel element and the target holder, in².

ALAB is the minimum flow area in the annulus between the inner and outer fuel elements, in².

CONDSP is the thermal conductivity of the side plate material, Btu/(h·ft·°F).

ALSP is the mean coefficient of thermal expansion of the side plate material, in/(in·°F).

3.1.2 Fuel Element Dependent Data

The following statements are read for the inner fuel element.

EA(1) R(1) AMBA(1) DPNW(1) EN(1) EW(1) ENMIN(1)

EA is the average fuel element coolant channel thickness in mils prior to reactor operation.

R is the outside radius of the inner side plate; it is the radius of the generating circle for the involute curve; inches.

AMBA is the wavelength λ of the sinusoidal longitudinal buckling of the fuel plate or any sinusoidal deviation in the initial coolant channel thickness; inches.

DPNW is the average differential pressure across a fuel plate between a narrow coolant channel and a wide coolant channel; psi.

EN is the average thickness of the narrow coolant channel prior to reactor operation; mils.

EW is the average thickness of the wide coolant channel prior to reactor operation; mils.

ENMIN is the minimum thickness of the narrow coolant channel prior to reactor operation; mils.

EWMIN(1) ENE(1) EWE(1) NP(1)

EWMIN is the minimum thickness of the wide coolant channel prior to reactor operation; mils.

ENE and EWE are the thicknesses of the inlet and exit of the narrow and wide coolant channels prior to reactor operation; mils (assumed to be the average for narrow and wide channels respectively, see p. 33, in ref. 1).

NP is the number of plates in the fuel element.

Three lines. DR(I,1), 10 per line, except for the third which has I = 21, M.
DR is the radial increment for the node.

Five lines. DZ(J,1), 10 per line, except for the fifth which has J = 41, N.
DZ is the axial increment for the node.

One line. DD is the amplitude of the initial sinusoidal wave in the coolant channel thickness; mils.

Three lines. U25(I,1), 10 per line, except for the third which has I = 21, M.
U25 is the flux peaking factor for fuel extending beyond its nominal axial boundaries (see U25, p. 69, and discussion p.17 in ref. 1).

N times 3 lines. TMA(I,J,1), three lines for each axial node. Initial guess at metal temperature for each node in the average fuel plate; °F.

Fuel element dependent data are then read for the outer and middle elements with the index changed from 1 to 2 or 3, respectively.

3.1.3 Time Step Dependent Data

For each time step, the following lines must be read in: power distribution (ratio of local to average power) N times 3 lines each, for the inner, outer, and middle fuel elements. For every axial node, three lines are input (up to 10 values per line). The following line is read for each time step:

THETA RATIO C5 C6 C7 C8 C9 C10

THETA is the length of the time step; hours.

RATIO is the ratio of fuel plate densities before and after irradiation. No use is made of this in the current versions of the code.

C5, C6, C7, C8, C9, and C10 are the values for the uncertainty factor U26 for mesh points 4,1;

M-2,1; 4,2; M-2,2; 4,3; and M-2,3, respectively (i.e., the innermost and outermost fuel-containing points of the fuel plate). U26 is the flux peaking factor for fuel extending beyond its nominal radial boundaries.

The next two lines are read if the maximum power option has been selected:

Q DELPF

Q is a new value for reactor power in megawatts for the trial-and-error search for the power at the selected thermal limit.

DELPF is pressure drop across the core in psi (can be input at the user's discretion).

Q TMAX TSPLM IQIC IQFL ITCL ITSP ITST ITSB

Q is a new (larger) value for reactor power in megawatts for the trial-and-error search for the power at the selected thermal limit.

TMAX is the maximum allowable fuel plate centerline temperature in degrees F.

TSPLM is the maximum allowable temperature drop through the oxide film in degrees F.

IQIC is the CHF/IB flag; set to 1 to limit the maximum power calculation or 0 to bypass,

IQFL is the flow excursion flag; set to 1 for the Costa correlation with hot streak heat flux, 2 for the Costa correlation with hot spot heat flux, 3 for the Moritz correlation with hot streak heat flux, 4 for the Moritz correlation with hot spot heat flux, or 0 to bypass,

ITCL is the centerline temperature flag; set to 1 to limit the maximum power calculation or 0 to bypass,

ITSP is the oxide temperature drop flag; set to 1 to limit the maximum power calculation or 0 to bypass,

ITST is the fuel plate surface temperature less than the saturation temperature flag; set to 1 to limit the maximum power calculation or 0 to bypass,

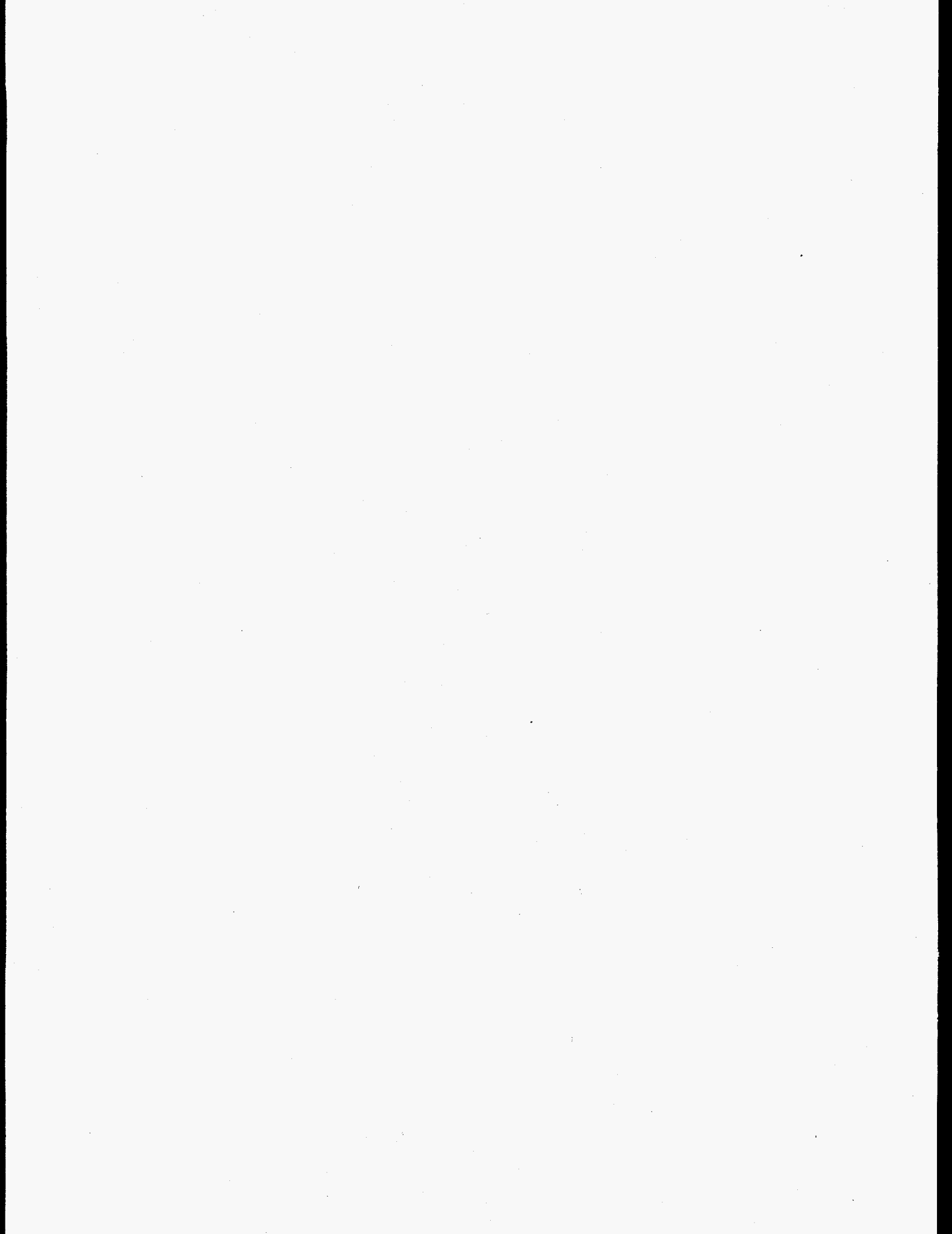
ITSB is a flag that prevents the condition of the exit bulk temperature being greater than saturation temperature; set to 1 to prevent this condition from occurring or 0 to bypass.

3.2 OUTPUT OPTIONS

Two types of output options are available. The first is a minimum output that prints only maximum power information. This option is activated by setting variable LTDIN = 4. This output is also always activated when ICALC = 2, the plate level core analysis. The second output option

is a maximum output option activated by setting variables LTDIN = 1. This output option prints out all variable information. As an example, with LTDIN =1 approximately 440 pages of output are produced for one time step in a maximum power, 3-element core calculation.

A file that contains the ratio of local temperature drop across the oxide film to the maximum allowable temperature drop across the oxide film as a function of the radial and axial position can also be output if desired by altering the input variable, JOXRAT. This feature allows a quantitative determination of the location and degree of limitation imposed by the TSPLM oxide spallation limit. As indicated in Sect. 3.1, input variables JOXIDE, JBUC, and JDEF can be set to 1 to generate other datasets, if desired.



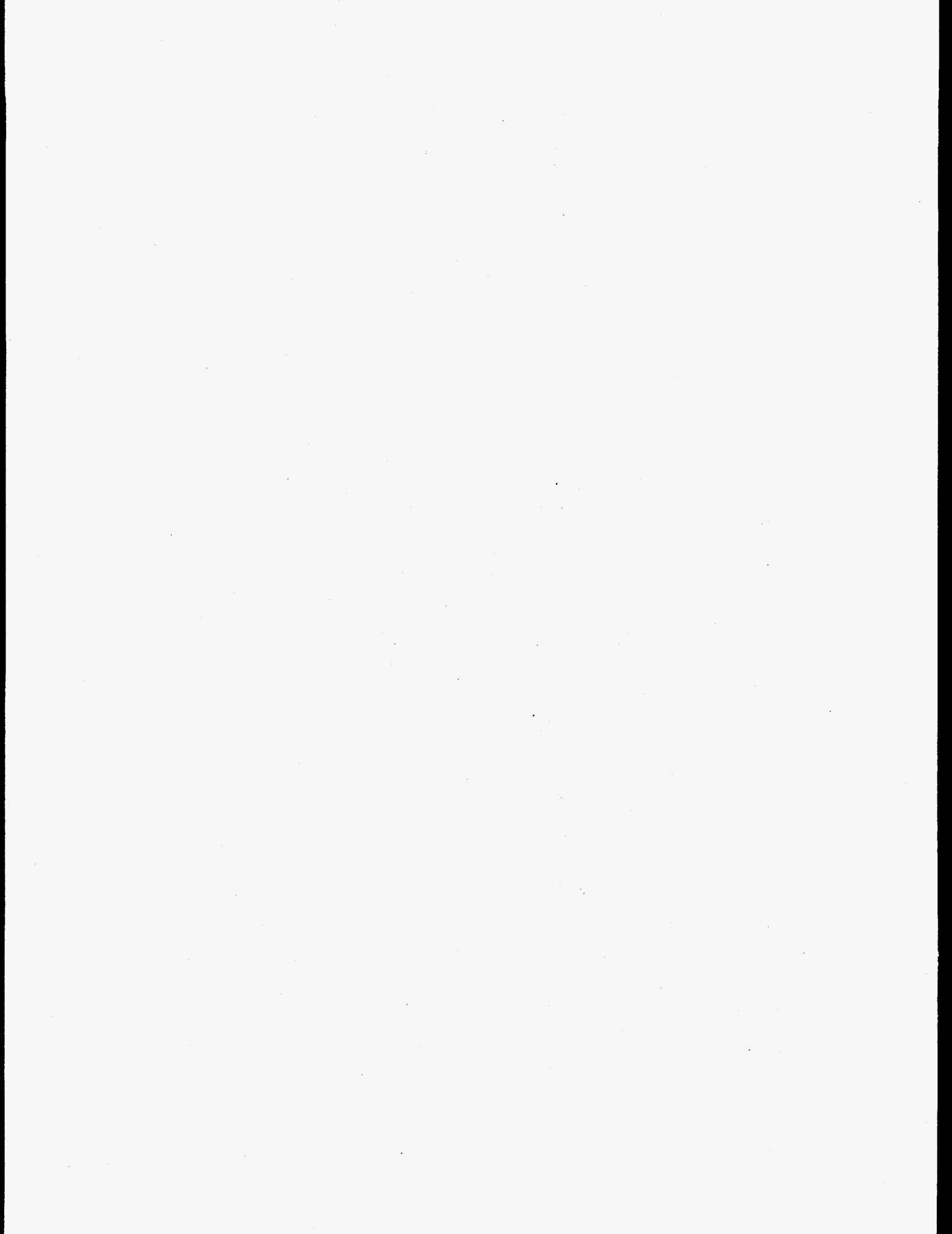
4. ANALYSIS WITH TASHA

4.1 IMPLEMENTATION OF STATISTICAL PEAKING FACTORS

The original HFIR steady-state thermal-hydraulics code¹ and the modified version developed by Cole et al.² employed the "worst-case" approach for the treatment of parameter uncertainties. In this approach, each parameter is assigned a conservative constant value reflecting the "maximum" (i.e., typically 2 or 3 standard deviations from the mean) deviation from nominal, and all of the uncertainties are assumed to exist simultaneously at their maximum value. The "worst-case" approach was judged to be unnecessarily conservative, and it does not provide any quantitative measure of the level of risk involved (i.e., the nonexceedance probability on a thermal limit). Thus, for ANS, statistical techniques were employed that were implemented in two methods, both of which used Monte Carlo sampling. The two methods are described in ref. 4. Of the two methods, the statistical peaking factor (SPF) method was used for most analyses. In contrast with the worst-case approach where the individual U-factors (see Chap. 3) are each assigned worst-case values and input to the code, when the SPF method is used, some uncertainties are statistically combined and then input to TASHA. The combined uncertainties are included in two SPFs, the hot spot factor, and the hot channel factor. Implementing this approach in TASHA requires that the U-factors for the parameters whose uncertainties were combined be set to 1.0 (except in cases where the U-factor is used to contain the combined result). Then, the calculated hot spot SPF(s) is input as U(18) and U(19) for CHF, IB, and $T_{wall} = T_{sat}$ limits, U(24) for FE limits, and both for fuel plate oxide temperature drop and centerline temperature (two hot spot SPFs are used for these limits; see ref. 4). The calculated hot channel SPFs are input as U24M as described in Sect. 2.6. Use of the SPF method is illustrated in the three-element core analysis presented in the next section.

4.2 THREE-ELEMENT CORE ANALYSIS

Appendices A through C contain examples of input for and output from the TASHA code for a three-element calculation. The 95% nonexceedance probability level peaking factors are used with the CHF thermal limit. Appendix A contains the input listing. Appendix B contains a minimum output listing with input variables LTDIN = 4 and JCALC = 2. Appendix C contains an example of a maximum output listing (variables LTDIN and JCALC set to 1).



5. QUALITY ASSURANCE

Several means of verifying or validating the code were used during the updating process. These included TASHA version-to-version results comparisons, hand calculations, and comparison of TASHA results to results with other codes.

Typically, as the code progressed from version to version, duplicate calculations were performed with both the old and new versions of the code to ensure that code changes did not affect the calculations in unexpected ways. This process also served to exercise the new code in all possible calculational modes, proving that the numerical algorithms were stable.

Hand calculations were used to check individual correlations within the code when changes were made. Heat transfer coefficients, friction factors, CHF correlations, FE correlations, etc. were all verified in this manner. This was often done in two ways. In both instances, local conditions calculated by the code (T_b , P , V , etc.) were used to evaluate the correlations by hand, including determining needed fluid properties. In the first method, the correlations used in the code were evaluated by hand and compared to the code calculated values. In the second method, hand calculated values were obtained with different correlations than used in the code (e.g., using the Dittus-Boelter correlation for heat transfer coefficient as opposed to the Petukov correlation, which is used in the code) and compared to the code calculated values. By using these methods, a check could be made on both the FORTRAN programming and the implementation of the correlation.

The third verification method used comparisons between two different codes. Most often, a subchannel code was used for this comparison. This code was developed independently of TASHA and used different numerical solution techniques to solve for local subchannel fluid and fuel plate surface conditions. A typical comparison would involve establishing inlet conditions in the subchannel code that were the same as those in the TASHA code. Results from each code were then compared to verify TASHA. Comparisons could be made on a global basis using this approach. Parameters such as local pressures, local bulk coolant temperatures, fuel surface temperatures, etc. were also compared using this approach.

Finally, the HEATING7 code was also used to validate some TASHA results. In the example that follows, TASHA local fluid and oxide conditions were input into HEATING and the conduction problem through the fuel plate solved.

The maximum normal fuel centerline temperature calculated with the TASHA and HEATING7 codes were compared for the G693 core design, 300 MW, under zero uncertainties assumptions. From the HEATING7 analyses, the highest normal fuel temperature occurs near the exit of the upper core at $r = 211$ mm and $z = 531$ mm in the last time step. The results from both codes for this location and time are provided in Table 2; as can be seen, the results compare very well.

Table 2. Comparison of results from TASHA and HEATING7 assuming zero uncertainties

Parameter	HEATING7	TASHA
Heat transfer coefficient ($\text{kW/m}^2 \cdot ^\circ\text{C}$)	173.0	173.0
Temperature at the oxide-D ₂ O interface ($^\circ\text{C}$)	145.2	145.1
Temperature at the clad-oxide interface ($^\circ\text{C}$)	152.2	152.2
Temperature at the meat-filler interface ($^\circ\text{C}$)	171.2	171.5
Temperature at the centerline ($^\circ\text{C}$)	175.8	176.1

6. FUTURE CODE

As stated in the Introduction, it was always intended to develop a new, more modern, more organized steady-state thermal-hydraulics code to perform ANS final design calculations, and also to serve as the code that would be used to perform steady-state thermal-hydraulic calculations for reactor operations. Although this task was never begun, a series of requirements for the new code was established in order to guide the development program. These requirements were developed from experience with the existing code as well as a recognition of the path that the thermal-hydraulics task within the ANS Project would need to take to achieve reactor design and safety goals.

6.1 CODE OBJECTIVES

1. The code must be capable of defining the steady-state operating power limits. These must include FE limit, CHF limit, IB limit, limit defined when the fuel plate surface temperature reaches the saturation temperature, limit imposed by a limiting centerline temperature, and limits imposed by oxide spallation.

2. The code must be structured in such a way as to make it easily understood, make it conform to ANS quality assurance requirements, and make use of state-of-the-art computing technology. Since the code will be used well into the next century, and since it will be used and probably modified by many analysts, the code structure must provide a logical platform for expansion and modification as requirements for the code change and improved models and correlations become available. Since statistical analysis and detailed fuel plate analysis both require extensive computational time, it will be necessary for the code to take advantage of the most recent computing technology.

3. The code should have the capability to refine analysis of transients within the core region beyond that possible using RELAP5 in order to enable more accurate (less conservative) analysis of transient events. Existing core calculations using RELAP5 are 1-D and do not allow detailed modeling of the core region. Detailed analysis of the fuel plate will eliminate the necessity of including some conservatism that must now be incorporated in the analysis.

4. The code should enable detailed, as-manufactured, core analysis. This would allow manufacturing data to be incorporated into the thermal-hydraulic analysis in as detailed a format as required by the project. This process should be automated to allow the thermal-hydraulic code to readily interface with the core inspection information so that thermal analysis could be performed on each core if necessary.

5. The code should provide a real-time model of operations. This capability would give reactor operators the capability of predicting core behavior during operation and could allow optimization of some operations, such as moving from one power level to another, preanalyzing anticipated operations (e.g., shutting down a pump or heat exchanger), etc.

6.2 CODE REQUIREMENTS

1. The code should use, and be updatable to incorporate, the latest ANS thermal-hydraulic correlation package. In order to ensure consistent calculations across the project, this code should incorporate correlation and model sets consistent with those used in other parts of the project.

2. The code should incorporate global multidimensional fuel plate conduction as necessary. Preliminary evaluation of these global multidimensional effects should be performed using existing codes to determine the importance of their inclusion in the final core code. Localized hot spot conditions (because of fuel defects) such as local heat fluxes and centerline temperatures should be coupled to detailed conduction calculations (as is presently being performed using the HEATING code) in order to determine the local limiting points and oxide conditions at the hot spot locations. This capability will allow determination of hot spot characteristics as an integral part of the thermal-hydraulic calculations and will allow a relaxation of some conservatism presently in the calculations.

3. The code will likely need the capability to execute multiple cases (with varying input parameter values) for sampling in Monte Carlo statistical uncertainty analysis. Precise requirements await selection of a methodology for statistical analysis; candidate methodologies include simple Monte Carlo sampling, Latin Hypercube Sampling, and Response Surface Model development, among others. The code's speed of execution requirement will depend upon the statistical methodology selected and the nonexceedance probabilities required. It may be desirable to develop a "quick running option" for the code in which less detailed (but more conservative) submodels are used (in place of some of the "normal" submodels), providing enhanced computational speed.

4. Detailed thermal-hydraulic analysis of the core will require considerable modeling input, and a large volume of output results will be produced. At a minimum, the code will contain a large 2-D mesh representing a fuel plate and cooling channel. Calculations will likely be performed at each mesh point and will include the evaluation of nominal, cold, and hot fuel plates associated in various ways with nominal, wide, and narrow cooling channels. The code needs to be designed to allow efficient entry of model input to the code that will include detailed neutronics (power distribution) information for multiple time points in the fuel cycle (output from a neutronics code) and likely information derived from fuel element structural analysis (output from a structures code). A convenient interface between the code and a graphics package is crucial. An efficient means to develop simple "x-y plots" to sophisticated 3-D representations of output results is required.

5. Fuel plate swelling occurs as fuel is burned up and is exacerbated with high fuel temperature. Swelling can significantly affect thermal-hydraulic behavior. First, swelling affects the fuel plate configuration for conduction analysis; the plate thickness can increase, and its thermal conductivity is affected. In addition, the channel gap width decreases as a result of the swelling. The extent of swelling (and its associated effects) will vary from point to point over the plate.

6. The code should include the capability to predict two-phase pressure drop within the fuel channels. This requirement is necessary to predict accurately coolant flow rates within the channel under two-phase conditions that would exist between the IB point and the CHF or FE limit. Thermal-Hydraulic Test Loop (THTL) experiments indicate that this pressure drop could be significant at the FE limit (up to 30% higher than the single-phase pressure drop). This implies that calculations based on single-phase pressure drops are nonconservative in terms of how the code would predict coolant channel flows.

7. Because of a number of causes, fuel plates deform to some extent during operation. Causes include thermal expansion of fuel plates (relative to side plates and other fuel plates), initial fuel element manufacturing variations in channel gap width and fuel plate thickness, and fuel plate swelling, which, in turn, all cause velocity and pressure variations that can either act to make deformations more or less severe. It is readily apparent that thermal-hydraulic and structural behavior of the fuel element are closely coupled. Thus, an interface to a structural analysis code is needed, or suitable "structural correlations" need to be developed and incorporated into the core code. Obviously, from a speed of execution standpoint, the latter is preferable if acceptable accuracy can be obtained.

8. As minimum, side-plate temperatures are needed in order to evaluate fuel plate deformation because of differential thermal expansion. However, it is also desirable to incorporate the potentially significant benefit that cooler side plates provide to the fuel plates via conduction from the outer span regions. Analysis will need to include the effects of reduced convection in the corners formed at the intersection of fuel plates and the sidewall, "contact" resistance between the plates and sidewall, as well as include the influence of welded joints.

9. The code should have the capability of predicting natural circulation and laminar flow thermal-hydraulics. In order to use the code under refueling conditions, the code must include the buoyancy terms associated with natural circulation flows driven by the core-heated channels. Flow conditions will eventually be low enough to necessitate the use of laminar flow thermal and fluid correlations.

10. The previous HFIR and ANS core codes have conservatively assumed that a hot fuel plate is cooled by two narrow cooling channels. A more realistic analysis would account for additional cooling that would be provided by neighboring cooling channels if they possessed "cooler" plates and/or wider channels. The coupling between channels would be via fuel plate conduction; for example, a hot plate with a narrow channel on one side and a wide channel on the other would transfer more heat to the wide channel. Initial scoping analysis indicates a ~1% benefit can be obtained with such analysis. To do this analysis, a more realistic "unit cell" model incorporating multiple plates and/or cooling channels would be required.

11. The code should include the capability of calculating multidimensional fluid behavior. Channel gap variations because of both flow and thermal stress variation across the fuel plate will cause cross flows between coolant subchannels that would not be accounted for in one dimensional fluid analysis. Local flow rates will influence the local temperatures and therefore local oxide conditions and thermal stresses.

12. Detailed core analysis during transient conditions [with less conservative modeling consumptions than used in global transient codes (e.g., RELAP5)] would be useful in evaluating transients in which the global analysis indicates thermal limits are exceeded. Thermal-hydraulic boundary conditions (e.g., some combination of pressure, temperature, and flow at the core inlet and outlet) for analysis would be obtained from global code results. Detailed analysis could be used to guide core model refinements in the global code.

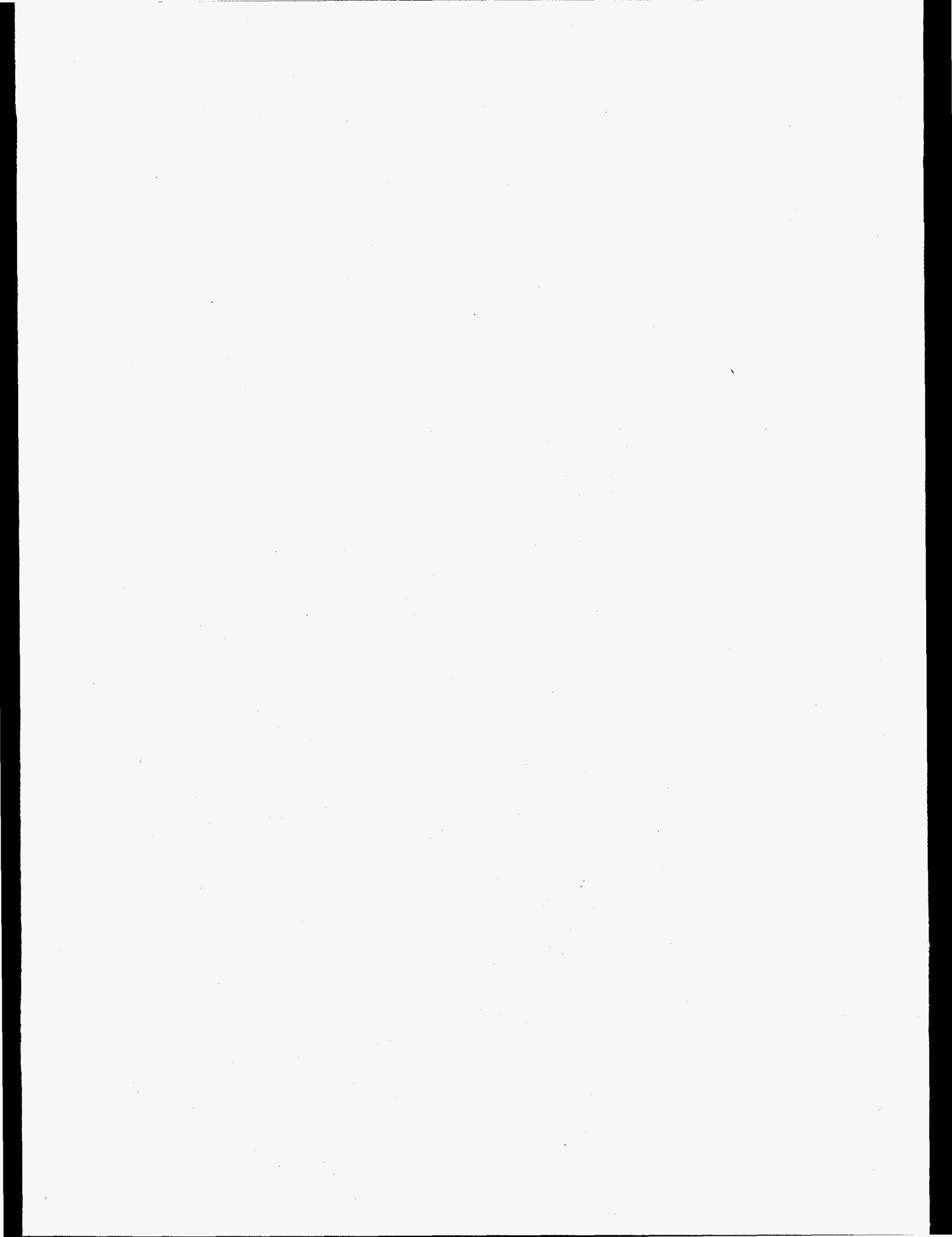
13. The necessity of including models of the rest of the core (outside fuel elements) in the core code depends on whether the behavior of the fuel element and the rest of the core are closely coupled and whether the modeling is to be done elsewhere (for both transient and steady-state analysis). For transients, it may be necessary to include nonfuel element components, but with less

detail than in the fuel element. These regions would be expected to be less limiting thermally and can be modeled acceptably with relatively simple, conservative models.

14. The code should incorporate neutronics calculations in order to facilitate core design as well as provide a method of including fuel conditions within the thermal-hydraulic calculations. This feature would also allow inclusion of reactivity feedback during transient calculations.

7. CONCLUSIONS

The TASHA code has been developed from the HFIR steady-state thermal-hydraulics code and includes significant improvements in both predictive ability and numerics. The code has been developed in support of the ANS Project, but can be a useful tool for other plate-type research reactor systems (such as HFIR) where more modern and up-to-date thermal-hydraulic predictions are necessary or useful. The code has also been upgraded to perform statistical uncertainty analysis allowing quantitative evaluation of nonexceedance probability levels. User selectable thermal limit analysis allows maximum power calculation based on a chosen set of one or more of the six available thermal limit criteria. The ANS Project had always intended to develop a replacement for this code to perform final thermal-hydraulic safety analysis and serve as the thermal-hydraulic analysis code for use during reactor operations. It is still recommended that this be done for any new research reactor design. However, the TASHA code in its present state can serve as a reference for those wishing to identify the important phenomena, and the code does incorporate the latest thermal-hydraulic correlations making it useful for performing plate-type reactor thermal-hydraulic design.



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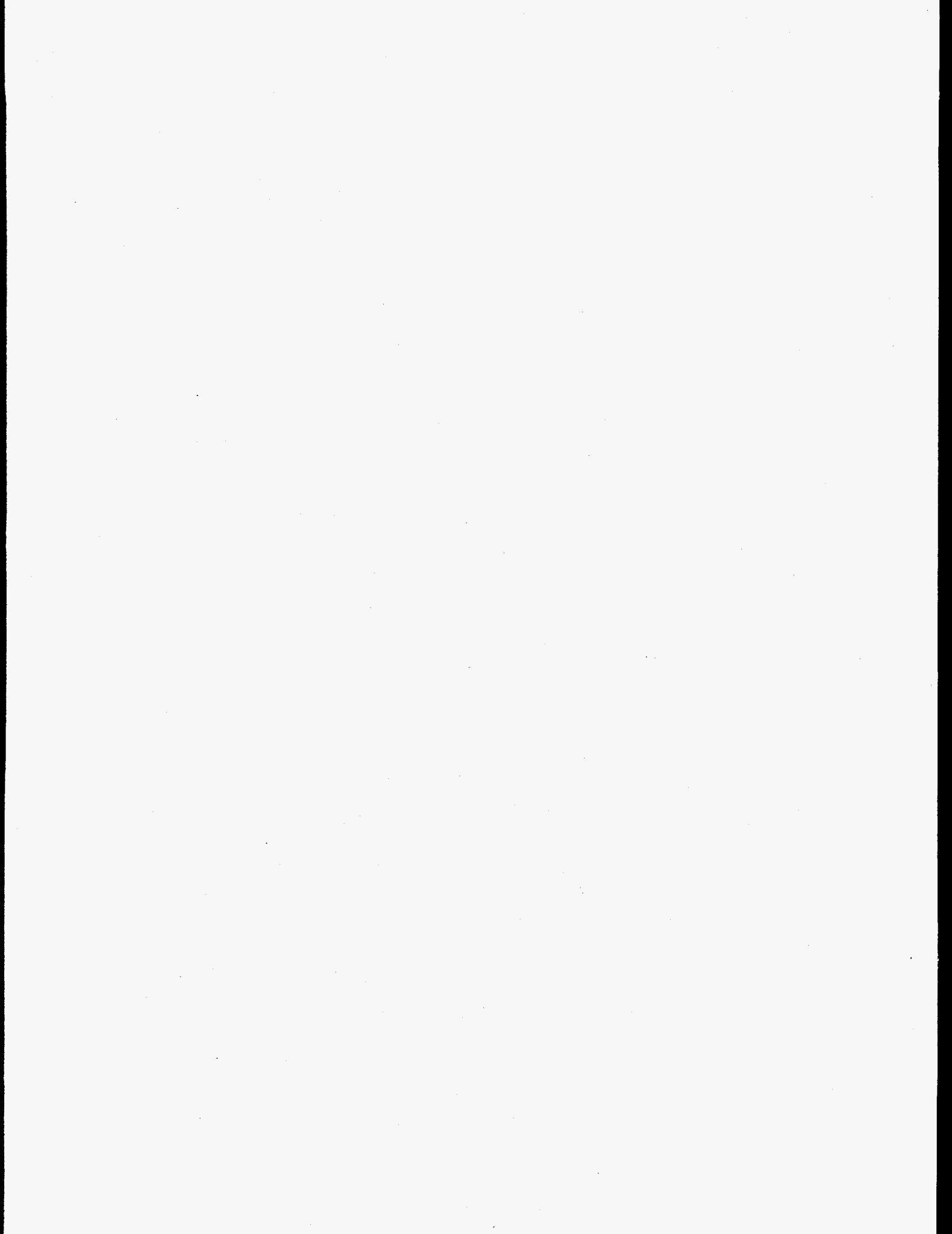
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Appendix A

Example TASHA Input Listing



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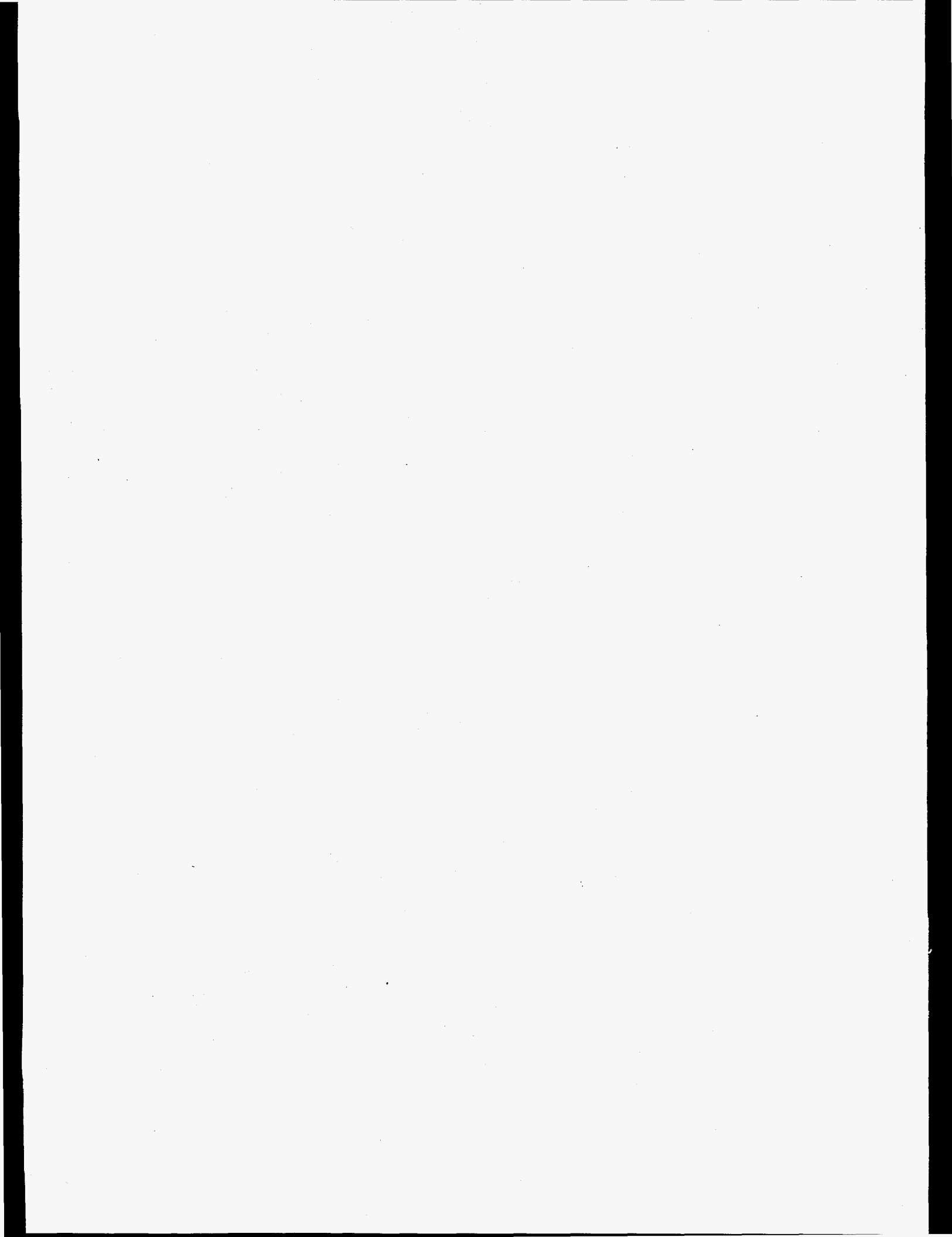
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1.03097 1.02117 1.02601 1.04425 1.07451 1.11683 1.16977 1.23166 1.30712 1.39133
1.44838 1.43688 1.40551 .00000 .00000
.00000 .00000 .00000 1.53218 1.40138 1.29961 1.21320 1.14656 1.09542 1.05814
1.03450 1.02514 1.03041 1.04900 1.07957 1.12223 1.17556 1.23766 1.31266 1.39470
1.44671 1.42806 1.38985 .00000 .00000
.00000 .00000 .00000 1.51838 1.39303 1.29564 1.21234 1.14750 1.09742 1.06085
1.03775 1.02888 1.03452 1.05341 1.08421 1.12714 1.18079 1.24305 1.31754 1.39737
1.44431 1.41853 1.37357 .00000 .00000
.00000 .00000 .00000 1.50431 1.38432 1.29133 1.21121 1.14825 1.09928 1.06343
1.04088 1.03245 1.03848 1.05766 1.08867 1.13182 1.18571 1.24807 1.32199 1.39955
1.44143 1.40859 1.35699 .00000 .00000
.00000 .00000 .00000 1.49032 1.37555 1.28693 1.21004 1.14902 1.10121 1.06612
1.04411 1.03613 1.04253 1.06198 1.09316 1.13645 1.19052 1.25289 1.32617 1.40143
1.43827 1.39848 1.34041 .00000 .00000
.00000 .00000 .00000 1.47641 1.36668 1.28241 1.20876 1.14975 1.10318 1.06889
1.04748 1.03996 1.04670 1.06635 1.09761 1.14096 1.19511 1.25743 1.33003 1.40296
1.43483 1.38822 1.32384 .00000 .00000
.00000 .00000 .00000 1.46255 1.35770 1.27771 1.20731 1.15036 1.10510 1.07168
1.05088 1.04382 1.05087 1.07068 1.10195 1.14527 1.19944 1.26165 1.33351 1.40411
1.43107 1.37780 1.30732 .00000 .00000
.00000 .00000 .00000 1.44872 1.34858 1.27280 1.20564 1.15077 1.10682 1.07427
1.05409 1.04747 1.05483 1.07477 1.10603 1.14930 1.20343 1.26549 1.33658 1.40485
1.42700 1.36725 1.29087 .00000 .00000
.00000 .00000 .00000 1.43495 1.33934 1.26769 1.20372 1.15091 1.10827 1.07659

1.05699 1.05081 1.05847 1.07855 1.10979 1.15299 1.20708 1.26895 1.33925 1.40522
1.42268 1.35665 1.27458 .00000 .00000
.00000 .00000 .00000 1.42133 1.33006 1.26244 1.20161 1.15086 1.10954 1.07871
1.05971 1.05397 1.06192 1.08213 1.11334 1.15645 1.21045 1.27210 1.34159 1.40527
1.41817 1.34609 1.25858 .00000 .00000
.00000 .00000 .00000 1.40775 1.32065 1.25697 1.19926 1.15057 1.11060 1.08067
1.06228 1.05698 1.06522 1.08553 1.11667 1.15964 1.21350 1.27489 1.34354 1.40496
1.41344 1.33556 1.24288 .00000 .00000
.00000 .00000 .00000 1.39403 1.31097 1.25114 1.19650 1.14988 1.11128 1.08227
1.06453 1.05968 1.06819 1.08858 1.11963 1.16242 1.21611 1.27721 1.34501 1.40421
1.40843 1.32499 1.22737 .00000 .00000
.00000 .00000 .00000 1.38002 1.30089 1.24484 1.19324 1.14869 1.11148 1.08341
1.06632 1.06193 1.07071 1.09117 1.12211 1.16471 1.21819 1.27897 1.34590 1.40294
1.40304 1.31430 1.21203 .00000 .00000
.00000 .00000 .00000 1.36574 1.29043 1.23809 1.18950 1.14702 1.11121 1.08410
1.06767 1.06374 1.07280 1.09333 1.12415 1.16654 1.21978 1.28020 1.34623 1.40113
1.39728 1.30354 1.19691 .00000 .00000
.00000 .00000 .00000 1.35120 1.27963 1.23094 1.18532 1.14492 1.11053 1.08440
1.06866 1.06521 1.07455 1.09515 1.12583 1.16797 1.22094 1.28096 1.34609 1.39889
1.39125 1.29275 1.18201 .00000 .00000
.00000 .00000 .00000 1.33604 1.26816 1.22308 1.18043 1.14212 1.10920 1.08410
1.06909 1.06614 1.07576 1.09640 1.12692 1.16877 1.22141 1.28098 1.34517 1.39591
1.38465 1.28172 1.16722 .00000 .00000
.00000 .00000 .00000 1.31994 1.25576 1.21427 1.17458 1.13838 1.10694 1.08290
1.06864 1.06620 1.07610 1.09678 1.12711 1.16866 1.22094 1.28003 1.34325 1.39196
1.37727 1.27028 1.15241 .00000 .00000
.00000 .00000 .00000 1.30313 1.24266 1.20472 1.16795 1.13382 1.10384 1.08082
1.06730 1.06536 1.07556 1.09629 1.12647 1.16774 1.21969 1.27829 1.34057 1.38734
1.36937 1.25854 1.13753 .00000 .00000
.00000 .00000 .00000 1.28521 1.22850 1.19409 1.16022 1.12812 1.09954 1.07751
1.06469 1.06324 1.07372 1.09452 1.12457 1.16558 1.21719 1.27530 1.33661 1.38147
1.36042 1.24616 1.12245 .00000 .00000
.00000 .00000 .00000 1.26589 1.21302 1.18214 1.15115 1.12105 1.09384 1.07275
1.06057 1.05957 1.07031 1.09117 1.12108 1.16182 1.21306 1.27063 1.33092 1.37386
1.34995 1.23282 1.10702 .00000 .00000
.00000 .00000 .00000 1.24689 1.19791 1.17056 1.14241 1.11422 1.08829 1.06803
1.05642 1.05580 1.06676 1.08767 1.11746 1.15797 1.20890 1.26595 1.32531 1.36647
1.33974 1.21952 1.09136 .00000 .00000
.00000 .00000 .00000 1.22564 1.18089 1.15730 1.13217 1.10601 1.08138 1.06194
1.05082 1.05049 1.06157 1.08242 1.11200 1.15218 1.20266 1.25904 1.31725 1.35647
1.32708 1.20453 1.07493 .00000 .00000
.00000 .00000 .00000 1.18856 1.14886 1.12948 1.10796 1.08440 1.06145 1.04297
1.03234 1.03214 1.04303 1.06341 1.09228 1.13146 1.18070 1.23555 1.29130 1.32739
1.29714 1.17888 1.05547 .00000 .00000
.00000 .00000 .00000 1.12525 1.09079 1.07516 1.05710 1.03619 1.01491 .99723

.98678 .98629 .99640 1.01564 1.04307 1.08031 1.12732 1.17976 1.23162 1.26352
1.23667 1.13491 1.03113 .00000 .00000
.00000 .00000 .00000 1.05063 1.01864 1.00350 .98598 .96553 .94428 .92616
.91492 .91330 .92180 .93912 .96447 .99925 1.04383 1.09426 1.14305 1.17271
1.15465 1.07854 1.00315 .00000 .00000
.00000 .00000 .00000 .99388 .95443 .93092 .90717 .88317 .86003 .84075
.82841 .82517 .83117 .84519 .86670 .89717 .93779 .98569 1.03341 1.06612
1.06355 1.01747 .97271 .00000 .00000
.00000 .00000 .00000 .00000 .00000 .00000 .00000 .00000 .00000 .00000
.00000 .00000 .00000 .00000 .00000 .00000 .00000 .00000 .00000 .00000
.00000 .00000 .00000 .00000 .00000
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.00000 .00000 .00000 .00000 .00000
0. 1. 1.02 1.02 1.02 1.02 1.02 1.02
305.00 114.10
310.00 752.0 214.2 1 0 0 0 1
1
0. 1. 1.02 1.02 1.02 1.02 1.02 1.02

Appendix B

Example TASHA Minimum Output Listing



g1294-2d grading - three element core (17 day cycle), 95% prob. unc, chf

core limits are: inner element= 1011.398 at i, j= 4 39; noti= 1; limit= 5
outer element= 670.706 at i, j= 19 39; noti= 1; limit= 5
middle element= 607.793 at i, j= 21 40; noti= 1; limit= 5

inner element limits at all selected time steps:

time step	qmax	i	j	limit
1	1011.398	4	39	5

outer element limits at all selected time steps:

time step	qmax	i	j	limit
1	670.706	19	39	5

middle element limits at all selected time steps:

time step	qmax	i	j	limit
1	607.793	21	40	5

STOP

Real 3141.1
User 2706.9
System 0.0