

TUF VERSION CONTROL AND DEVELOPMENT STATUS

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ABSTRACT

The TUF (**T**wo-**U**nequal-**F**luid) code has been developed at Ontario Hydro as the main safety analysis tool for primary and secondary heat transport systems of CANDU nuclear reactors. The code has been used extensively for both operational and abnormal transients. The objectives of this paper are (1) to briefly outline the code capabilities, (2) to present the version control logic currently being applied and (3) to report the development status of the TUF code.

1. INTRODUCTION

The TUF code is a best estimate thermal hydraulic system code for safety analyses of CANDU reactor plants in Ontario Hydro. It is capable of modelling network thermal hydraulics, heat conduction, neutron kinetics, special components and reactor control systems. The primary objective of the TUF development effort is to provide a two-fluid tool that will enhance the capability to analyze postulated reactor accidents and to assist reactor design and operation. A general description of the TUF code has been presented elsewhere (Reference 1).

The code has been used extensively for both operational and abnormal transients in Ontario Hydro. The objectives of this paper are to briefly outline the code capabilities, to present the version control logic currently being applied and to report the development status of the TUF code.

2. TUF CAPABILITIES

TUF can be characterized as an advanced system thermal hydraulic code for CANDU reactors. The code incorporates state-of-the-art methods and models. The models in TUF are designed to yield realistic results as opposed to conservative evaluation models currently in licensing codes. The TUF code mainly differs from the SOPHT code (Reference 2) in its thermal hydraulics, channel models, piping materials capability, special components modelling and numerical methods used to solve the thermal hydraulics and fuel-pin conduction equations. It was originally developed from the SOPHT code, so it is capable of performing the same applications as SOPHT in addition to others requiring more advanced models.

The main capabilities of the TUF code are: (1) reactor network capability, (2) one-fluid and two-fluid models, (3) reactor control systems, (4) steady state scheme, (5) water hammer analysis capability, (6) multi-channel capability, (7) heat loss to moderator and (8) bundle movement simulation. These capabilities are briefly described below.

Reactor Network Capability

The TUF code has the capability to simulate the following thermal hydraulic systems of a CANDU reactor: primary heat transport, secondary heat transport, feed and bleed, emergency coolant injection, shut down cooling, D2O purification and steam generator emergency cooling systems. The only thermal hydraulic systems which are not included in the reactor network are the moderator and its auxiliary systems. These systems, together with the containment system, are simulated by using different computer codes in Ontario Hydro. In the TUF code, the moderator is treated as a lumped control volume where the transient heat transfer coefficients between calandria tube and the moderator are input values.

TUF models the reactor network as a series of interconnecting control volumes. The basic fluid model solves the mass and energy conservation equations for control volumes that are connected by links or junctions. Momentum equations are solved to obtain flow rates through the links. The code capabilities allow many complex flow systems to be analyzed.

The network and controller information for a specific nuclear station application is generated using the following information: (1) node and link types, (2) component location code numbers of nodes and links, (3) auxiliary vectors and (4) control vectors.

One-fluid and Two-fluid Models

Mixture variables such as mass, internal energy and flow rate are used as the primary variables. This choice is made because in a plant simulation, the global phenomena are typically more important than microscopic phenomena. Also the constitutive correlations obtained from experiments are in terms of the mixture variables. An additional set of differential equations used to describe the unequal phase velocity and unequal phase temperature effects is required for the two-fluid model. TUF contains both one-fluid and two-fluid models in the thermal hydraulic modelling. The two-fluid model can be reduced to a one-fluid model by imposing identical phase velocities, phase temperatures and sonic speeds. The capability of reducing the two-fluid model to a one-fluid model distinguishes the TUF code from other advanced thermal hydraulics codes.

Reactor Control Systems

The reactor control systems used in the TUF code are station dependent, and simulate the following plant control systems:

- (a) Overall Unit Control
- (b) Reactor Regulating System
- (c) Steam Generator Pressure and Level Controls
- (d) Heat Transport Pressure and Inventory Controls
- (e) Bleed Condenser Pressure and Level Controls
- (f) Pressurizer Pressure and Level Controls
- (g) Reactor Shutdown Systems (SDS1 and SDS2)

There are two unit control methods used in Ontario Hydro nuclear generating stations and their use is dictated by the station design and its intended mode of operation. These control modes are usually referred to as reactor leading (turbine following) and turbine leading (reactor following). The reactor leading mode is the basic control method employed at the Pickering reactors, and the alternative mode is employed at the Bruce and Darlington reactors.

The Unit Power Regulator controls the overall unit electric power output. The Reactor Regulating System controls the neutronic power and rate of change of neutronic power of the reactors. The Boiler Pressure Control System controls the boiler pressure where the boiler pressure setpoint is a function of reactor neutronic power in the reactor leading format. The Boiler Level Control System controls the boiler level as a function of unit thermal power. The Shutdown Systems are designed to shut down the reactor under abnormal or potentially hazardous operating conditions. The Heat Transport Pressure Controller maintains the reactor outlet header pressure at its setpoint by modulating the pressurizer steam bleed valves and the pressurizer heaters for the Bruce and Darlington reactors, or by modulating the feed and bleed valves for the Pickering reactors.

The control systems employed in the TUF code have been simplified from the actual plant control systems in the following areas: (1) The code does not simulate errors in the reactor power measurement. The measurements of ion chambers with logarithmic amplifiers, and in-core flux detectors with linear amplifiers, are simulated from the neutron kinetics model. The thermal power measurement is simulated from the heat transport flow and temperature rise. (2) Each reactor has several light water zone control absorbers. Each zone is controlled by a control valve and all zone levels may not be at the same value due to flux tilt. In TUF, all zones are assumed identical, and the individual zone and flux tilt control is not simulated. However, initial flux tilt and related transient reactivity effects are simulated in the multi-channel TUF model. (3) The code does not simulate details of the bleed cooler. The heat transfer from D2O to H2O is modelled as a simple heat sink based on steady state operating conditions, and (4) In the turbine simulation, the reheater steam flow is simplified as a function of turbine steam flow. The reheater drains flow is assumed to follow the steam flow by a time constant.

Steady State Program

TUF consists of two separate programs modules: steady state and transient. In the steady state program, the following equations are solved:

- (a) The nodal pressure
- (b) The nodal specific enthalpy
- (c) The nodal heat flux
- (d) The heat exchanger film resistance
- (e) The link flow rate
- (f) The link resistance or valve position
- (g) The link phase velocity difference

The set of simultaneous non-linear equations is solved by the Newton-Raphson iteration method. The major assumption in the steady state program is that both phases are assumed to be in thermal equilibrium. To remove this assumption and to meet further compatibility between steady state and transient programs, a zero transient run (i.e. without changing any boundary conditions and control states) is recommended. The advantage of the steady state program is in its ability to directly obtain the normal operating conditions of a specific reactor.

To match the steady state solutions with normal operation conditions, different boundary condition flags are utilized in the input data of nodes and links. The purpose of these flags is: (1) to impose a fixed boundary condition, (2) to suppress the appropriate mass, momentum or energy conservation equation, and (3) to introduce either link resistance or valve position as a variable.

Fluid-Structure Interactions

In order to simulate water-hammer phenomena where transient pressure pulse may be obtained, the interaction between piping elasticity/plasticity and the thermal hydraulics is included as an option. When the pressure pulse is low, the contribution of the effective bulk modulus of the liquid phase on the pressure wave propagation velocity is much larger than that due to piping elasticity, and therefore, the assumption of a rigid pipe is applicable. However, elasticity effects are considered in the simulation of water-hammer with a high pressure pulse.

The longitudinal movement of the pipe is assumed to be restrained in the model. Small effects such as radial fluid velocity and radial pressure gradients, as well as inertial forces in the pipe are neglected in the first-order approximation, applied in the code. In addition, dilation of the piping causes axial stress waves to propagate along the pipe wall. This effect may become important only for very thin pipes, which is not the case normally encountered.

Non-Condensible Gas

Non-condensible gas is included in the thermal hydraulic modelling to simulate the effects of hydrogen production due to metal-water reaction or annulus gas between the pressure and calandria tubes. The non-condensible gas is assumed to be indistinguishable from the vapour phase in the solution of the momentum equation, but mass and energy conservation equations are solved for the specific gas modelled.

3. TUF VERSION CONTROL

TUF contains modules dealing with thermal hydraulics (one-fluid and two-fluid models), reactor physics (point kinetics), heat conduction (piping wall, boiler tubes, pressure/calandria tubes and fuel pins), special components (pump, valve, pressurizer, etc.), special models (discharge model, level swell analysis, etc.) and station controllers. Two separate areas are created in the TUF code structure: "STANDARD" and "CONTROLLER" areas. The STANDARD area, which is about 90% of the program, contains all generic common blocks and routines used in

generic models and in control routines. The CONTROLLER area contains all common blocks and routines for specific stations (for example, Darlington, Bruce and Pickering stations). For the cases involving non-station applications (for example experimental simulations of blowdown and water-hammer phenomena and header-to-header simulations, etc.), the control routines are not activated.

When a particular version of the code substantially meets its performance objectives, it is pre-released for internal use and the user testing stage begins. There is an independent assessment stage that involves testing of the new version against a set of standard reactor specific cases. If the results of the independent assessment are satisfactory, the new version is then released for external use at Ontario Hydro. Released version numbers are assigned separately to the STANDARD and CONTROLLER areas. Hence, strict configuration management is maintained in the release of updated versions of the code.

A formal TUF users' group has been initiated to discuss user problems and to identify areas for code improvement. The TUF code is available for use outside Ontario Hydro through the TUF Users' Group.

4. TUF DEVELOPMENT STATUS

Work continued on the development of the TUF code based on the phenomena governing large LOCA, small LOCA, loss of forced circulation and other applications such as water-hammer analysis.

In the case of large LOCA, there is a continuous transient of three distinct phases for a CANDU reactor: blowdown, feeder refill and channel refill. Each of these periods is governed by different dominant physical phenomena, whose modelling details are important to the predicted behaviour of the fuel channels. There are many best-estimate models available for these phenomena in the literature. In the case of small LOCA, where the depressurization is slower and longer, the additional important parameters are the break location and orientation, pump characteristics for two-phase operations, heat transfer between primary and secondary sides, etc. In the case of water-hammer analysis, the condensation rate plays a dominant role in the prediction of the pressure surge at pipe dead-ends and/or for colliding liquid columns.

There are two areas involved in the continuous development of the TUF code: namely, numerical and physical models. In the numerical models area, work involves code sensitivity to input data, model assumptions and limitations, numerical scheme and numerical stability. The physical models area deals with empirical correlations and model improvements.

The general procedure for TUF assessment and testing involves two stages: development assessment and station data testing. The development assessment involves primarily a wide variety of thermal-hydraulic experiments and theoretical analyses. The objective is to define the limits of validity of the methods, correlations and models. In the second or station data testing stage, work involves simulations of commissioning tests and different abnormal operation data. The primary objective of this activity is to determine the overall station response predicted by the code and to compare with known station data. Some activities currently involving in the TUF development program are briefly outlined below.

Sensitivity of Input Data

There are two available approaches in the nodalization scheme : central and non-central nodalizations. In the central nodalization scheme, except for the special links, link hydraulics geometries are calculated from the geometries of the two associated nodes. In the non-central nodalization scheme, the geometric data are applied directly to the link geometric data. Therefore, TUF requires input of the geometric descriptions, such as volume, flow area, elevation and

such thermal hydraulic properties as pressure, specific enthalpy for the modules. For special links, the model requires such physical input as length, area, diameter, elevation, together with initial flow rate and minor friction loss coefficients. With the help of auxiliary vectors, the component models require geometric input (for example pump inertia), performance curves (pump curve) and other physical information.

In the reactor simulations, usually various piping sections in a given region are combined into a single module. A representative area and length must be chosen to represent this module. There are uncertainties in the required input information. These uncertainties are usually resolved with engineering judgement and sensitivity studies.

The sensitivity studies conducted involve the varying of selected input parameters (for example, those used in the control routines) to determine what effect uncertainties in these parameters would have on the predictions. Modelling studies are also performed to examine the effects of nodalization and analytical and numerical model options in the code on predictions. Due to the fact that a particular parameter or model change may have a small effect on one type of transient and a large effect on a different transient, various cases for small and large breaks are analyzed. Additional studies and verifications are planned in the near future to demonstrate code robustness and stability over a wide range of conditions.

Two-Step Method

It is well known, that for stability reasons, integration formulae of the explicit type do not allow an efficient treatment of the thermal-hydraulic equations. Implicit methods are favoured because of their excellent stability. However, in applying a one-step semi-implicit method to slow transients, the analyst may encounter an excessively long execution time due to stability limited time-step sizes in the finite difference equations. The time-step limitations result from transport terms, linearization procedure of non-linear equations, stiffness of the equations and explicit treatment of some heat source terms (for example, piping heat transfer rate). Stability of the semi-implicit method is limited by a material transport Courant limit in the energy and momentum equations. That is, the solution time step cannot be so large that material is transported all the way across a control volume in one time step. In general, the time step limitation is primarily due to considerations of both the accuracy of the solutions and numerical stability.

In the TUF code, two numerical methods are available: one-step semi-implicit and two-step implicit methods. One objective of the development of the two-step method is to make the code more dependable and faster running. The two-step implicit method is briefly described below.

In the first step, the mass and energy equations of the two-fluid equations are solved explicitly. These solutions are then used to update the link properties. However, the pressure and density are not updated since they are associated with acoustic wave propagation. In the second step, the flow rate equations resulting from the two-fluid equations are solved implicitly using a sparse matrix solver. After the flow rates are obtained, mass and energy variables are then solved by back substitution. The two-step method has proven to be an efficient numerical technique for the two-fluid model.

5. TUF APPLICATIONS

Comparison With Station Data

Recently, the TUF code has been used to predict the steady state conditions at 50% FP and 100% FP with four heat transport pumps running for Darlington Unit 1. The two-fluid model was applied in the simulation. As shown in Tables 1 and 2, the TUF predicted conditions are in good agreement with measured data. The TUF

predicted coolant flow rates per core pass (120 channels) are 2847 kg/s and 2834 kg/s for 50% and 100% FP, respectively. Although flow is measured for selected channels only, the good agreements on HT pump heads, change in coolant temperature from RIH to ROH, header-to-header pressure drops at a given power level, indicate that the predicted coolant flow rates are very well calculated.

Best Effort Analysis of ECI Effectiveness

The latest phase of the best effort analysis of ECI effectiveness for the case of a critical LOCA, was initiated in 1986. The objective was to develop analytical tools to quantify, within reasonable bounds of certainty, the effectiveness of the ECI system in Ontario Hydro nuclear generating stations. The TUF code is used in all best effort analyses and the validation of the methodology is an ongoing process. The system representation of the Darlington NGS is briefly described here.

The primary heat transport system consists of two loops with appropriate pressure and inventory control system and the emergency coolant injection system. The critical pass is represented by six separate core regions, each containing a certain number of channels with the average characteristics of the region. A variation of this multi-region model is also used to provide detailed channel thermal hydraulic conditions in any of the regions.

In the analysis, the following aspects are considered: the critical break size, thermal hydraulic response to the critical break, mechanical response or number and timing of ballooned pressure tube segments in the core, role of the ECI system in limiting temperature excursions, and impact of the integrated analysis on the moderator subcooling required to maintain fuel channel integrity.

Initially, the break flow is large due to subcooled water in the system. Within a half second, reactor trip occurs. The break flow rate exceeds the flow in the broken pass, leading to flow reversal in the core. This leads to a period of flow stagnation and critical heat flux conditions reached inside the channels. The fuel elements heat up as the fuel-to-coolant heat transfer decreases. Pressure tube temperatures increase in all regions of the broken pass. Some channels with high power eventually produce increases in the pressure tube temperature to the point that the channel will be deformed and ballooned. This results in an increase of the heat load to moderator. The system pressure continues to decrease, and when it reaches the ECI setpoint, injection of cold water into the hot primary system begins. This may cause significant oscillations due to steam condensation on cold water. The reactor core gradually cools down and is refilled.

The parameters that characterize a large LOCA are the fuel sheath and pressure tube temperatures. These parameters depend on the initial stored energy in the fuel, fuel-to-sheath gap heat transfer coefficient, fuel thermal properties, heat transfer coefficients between the sheath and the coolant and between the pressure tubes and the coolant, metal-water reaction, channel power, and radial and axial peaking factors, which depend on the fuel burnup. Other parameters that influence the system response are the break flow, interfacial mass transfers, counter-current flow limitation, condensation rate and flow regime maps. A typical pressure tube temperature transient is shown in Figure 1 together with a comparison of the results obtained from a more detailed thermal model in the SMARTT code (Reference 3).

The best-effort analysis methodology, using the TUF two-fluid model, is presently being applied in support of the generic large LOCA analysis for Ontario Hydro's CANDU reactors. The TUF code is also being extensively verified against large LOCA transients as discussed later.

Water Hammer Simulations

Extensive water-hammer experiments have been conducted at the Ontario Hydro Research Division. The objective of these experiments is to verify the TUF code

predictions of water-hammer phenomena resulting from injection of cold water into a large diameter piping system. Such scenarios have been analyzed with TUF for the Steam Generator Emergency Cooling and Boiler Emergency Cooling systems of Darlington NGS and Pickering NGS, respectively. The TUF code simulations of these experiments are an ongoing activity and will be discussed at a future date.

6. TUF VALIDATION PROGRAM

A systematic code verification program has been set up in Ontario Hydro. The objective of this program is to systematically verify the adequacy of the code to represent the physical phenomena governing thermal-hydraulic behaviour in Ontario Hydro's nuclear reactors. This program involves the progressive use of benchmark tests (for example JUICE standard problems) and experimental data from separate effects experiments (for example the Nuclear Power Demonstration pressurizer experiments, and the OHRD water-hammer experiments) and integrated tests (for example, RD-14 multi-channel experiments). The RD-14 thermosyphoning and blowdown tests have been reported in Reference 4. The simulations of the multi-channel RD-14 large inlet header break tests are being conducted in Ontario Hydro to support the methodology used in the best effort large LOCA studies.

7. CONCLUDING REMARKS

The TUF code capabilities, the version control logic and the development status have been briefly outlined in this paper. Currently, the TUF code is being extensively used to simulate CANDU reactors in Ontario Hydro under LOCA or plant upset conditions.

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REFERENCES

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Table 1. Comparison of TUF prediction with measured data of Darlington Unit 1 for the case of 50% FP with four HT pumps running

PRESSURE (MPa)	STATION	TUF PREDICTION
Headers		
RIH HD2	11.07	11.11
RIH HD4	11.10	11.10
RIH HD6	11.10	11.12
RIH HD8	11.10	11.12
ROH HD1	9.87	9.90
ROH HD3	9.84	9.89
ROH HD5	9.87	9.90
ROH HD7	9.87	9.90
HD2-HD3 Delta P	1.23	1.22
HD4-HD1 Delta P	1.23	1.20
HD6-HD7 Delta P	1.23	1.22
HD8-HD5 Delta P	1.23	1.22
Pressurizer	9.84	9.98
HT Pump Suction/Discharge		
Pump 1	9.44/11.08	9.47/11.15
Pump 2	9.39/11.13	9.46/11.14
Pump 3	9.45/11.12	9.47/11.15
Pump 4	9.38/11.09	9.47/11.15
Steam Generator		
SG 1	4.960	4.967
SG 2	4.969	4.968
SG 3	4.962	4.967
SG 4	4.967	4.967
HT PUMP SPEED (rpm)		
Pump 1	1801	1800
Pump 2	1827	1800
Pump 3	1816	1800
Pump 4	1797	1800

Table 2. Comparison of TUF prediction with measured data of Darlington Unit 1 for the case of 100% FP with four HT pumps running

TEMPERATURE (C)	STATION	TUF PREDICTIONS
Headers		
RIH HD2	264.2	263.6
RIH HD4	264.4	264.0
RIH HD6	267.5	263.6
RIH HD8	264.1	264.0
ROH HD1	309.8	309.5
ROH HD3	309.0	309.1
ROH HD5	309.8	309.4
ROH HD7	309.8	309.1
Primary Side		
SG1 Inlet	309.9	309.5
SG2 Inlet	309.8	309.1
SG3 Inlet	309.5	309.4
SG4 Inlet	309.2	309.1
SG1 Outlet	262.5	263.4
SG2 Outlet	263.0	263.5
SG3 Outlet	262.0	263.4
SG4 Outlet	262.5	263.4
Feed Water/Preheater	171.7	172.6
FLOW (kg/s)		
Feedwater		
SG1	317.5	312.9
SG2	316.9	309.1
SG3	319.1	312.0
SG4	316.6	309.4
Steam Flow		
SG1	332.2	328.3
SG2	339.7	324.5
SG3	334.0	327.4
SG4	332.2	324.8
Reheater Drains		
SG1	13.6	15.4
SG2	15.0	15.4
SG3	14.5	15.4
SG4	14.0	15.4
LEVEL (m)		
Steam Generator		
SG1	14.39	14.38
SG2	14.39	14.38
SG3	14.38	14.38
SG4	14.38	14.38
Pressurizer	6.72	6.75

Comparison of TUF and SMARTT PT Temperatures
G3 / slave / 6.5 MW - node 8 - 15 Nov 91

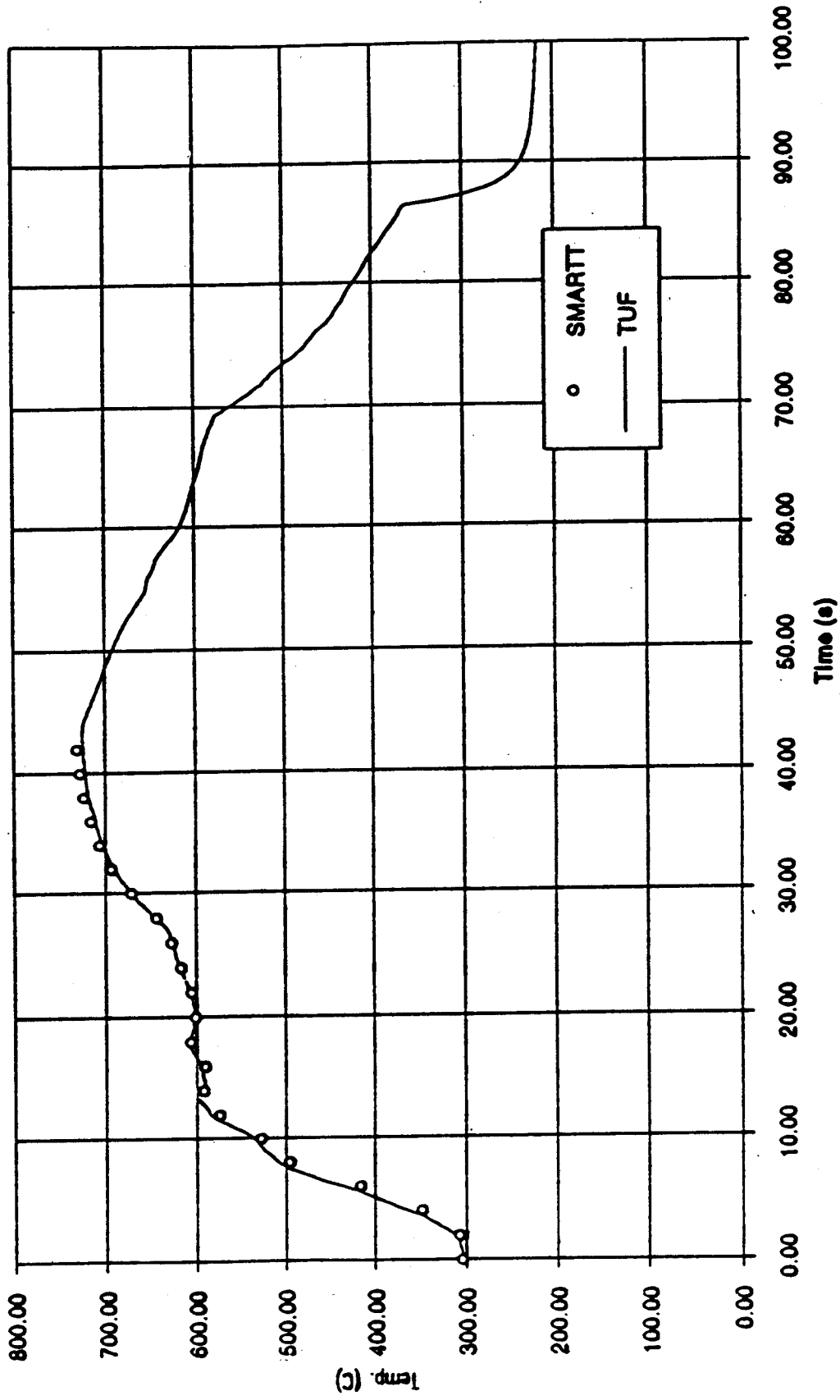


Figure 1. Transients of pressure tube temperatures predicted by TUF and SMARTT codes

