NUREG/IA-0247

International Agreement Report

RELAP5 Simulation of Darlington Nuclear Generating Station Loss of Flow Event

Prepared by:
D. Naundorf[1]
J. Yin[1]

A. Petruzzi[2]
A. Kovtonyuk[2]

[1]AMEC NSS Limited
700 University Avenue
Toronto, Ontario, Canada

[2]Nuclear Research Group San Piero A Grado
University of Pisa
Pisa, Italy

A. Calvo, NRC Project Manager

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

February 2011

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the International Code Assessment and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402-0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161-0002
   www.ntis.gov
   1-800-553-6847 or, locally, 703-605-6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: U.S. Nuclear Regulatory Commission
Office of Administration
Publications Branch
Washington, DC 20555-0001
E-mail: DISTRIBUTION.RESOURCE@NRC.GOV
Facsimile: 301-415-2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036-8002
www.ansi.org
212-642-4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG-XXXX) or agency contractors (NUREG/CR-XXXX), (2) proceedings of conferences (NUREG/CP-XXXX), (3) reports resulting from international agreements (NUREG/IA-XXXX), (4) brochures (NUREG/BR-XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG-0750).

NUREG/IA-0247

International Agreement Report

RELAP5 Simulation of Darlington Nuclear Generating Station Loss of Flow Event

Prepared by:
D. Naundorf[1]
J. Yin[1]

A. Petruzzi[2]
A. Kovtonyuk[2]

[1]AMEC NSS Limited
700 University Avenue
Toronto, Ontario, Canada

[2]Nuclear Research Group San Piero A Grado
University of Pisa
Pisa, Italy

A. Calvo, NRC *Project Manager*

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

February 2011

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the International Code Assessment and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

ABSTRACT

The Darlington NGS consists of four 940 MWe CANDU reactors. A detailed RELAP5 model of a Darlington NGS reactor has been created for use with RELAP5/MOD3.3. The model includes the primary heat transport system, the steam generator feedwater and main steam supply systems, the pressure and inventory control system, and the emergency coolant injection system. The modeling of Darlington required the creation of new subroutines for RELAP5 to describe the functionality of the process control systems and emergency control systems with sufficient accuracy. These controller subroutines are standalone modules that are compiled and linked together with the RELAP5 source to facilitate integration with future revisions to the code. The November 25, 1993, loss-of-flow event at Darlington was simulated using this RELAP5 model. Excellent agreement was obtained between the RELAP5 predictions and station measurements.

CONTENTS

Figures

Tables

Page

EXECUTIVE SUMMARY

The Canadian nuclear industry is currently considering replacing its thermal-hydraulics codes, and the USNRC codes TRACE and RELAP5, which are made available via the Code Applications and Maintenance Program (CAMP), are potential candidates. Accordingly, AMEC NSS is assessing these codes for their applicability to CANDU safety analysis. This assessment includes demonstrating that the codes have the functionality required to model features unique to CANDU reactors and their validation against data specific to CANDU reactors.

An important step towards demonstrating the applicability of the CAMP codes to CANDU reactors is showing their ability to predict known station transients. This report presents the results of such a comparison between a RELAP5 simulation and plant measurements for a loss-of-flow event at the Darlington Nuclear Generating Station.

The report first provides an overview of the CANDU process and safety systems relevant to loss-of-flow events at the Darlington Nuclear Generating Station (Section 2). A description of the November 25, 1993, loss-of-flow event at Darlington Unit 4 is then presented (Section 3). To facilitate modeling this event with the RELAP5 code, Darlington-specific control functionality was added to the RELAP5 code, and a Darlington RELAP5 input data set was created. These Darlington-specific code modifications are described in Section 4, together with testing that was performed with the modified code. Test results demonstrated excellent agreement between RELAP5 code predictions of benchmark controller responses. The Darlington RELAP5 input data set is then described in Section 5.

Results from the RELAP5 simulation of the loss-of-flow event are presented in Section 6, showing excellent agreement between predictions and measurements for the parameters of interest. Section 6 also includes results of a sensitivity simulation, which demonstrates that the header nodalization has no significant impact on the loss-of-flow simulation results.

ACKNOWLEDGMENT

The authors acknowledge the substantial contributions of several individuals. At the San Piero a Grado Nuclear Research Group (GRNSPG) of the University of Pisa, Italy: M. Cherubini, W. Giannotti, M. Kovtonyuk, and especially Prof. F. D'Auria. At AMEC NSS, Toronto, Canada: B. Mclaughlin and O. Orlov.

The authors also wish to thank Mr. M. O'Neill of Ontario Power Generation for granting access to the Darlington NGS plant data and permitting simulation of this event. The review of this report by Mr. W.K. Liauw, Ontario Power Generation, is also gratefully acknowledged.

ABBREVIATIONS

ASDV	Atmospheric Steam Discharge Valve
CAMP	Code Applications and Maintenance Program
CANDU	Canadian Deuterium Uranium
CSDV	Condenser Steam Discharge Valve
ECI	Emergency Coolant Injection
FP	Full Power
GRNSPG	Gruppo di Ricerca Nucleare San Piero a Grado
HTHP	Heat Transport High Pressure
HTLF	Heat Transport Low Flow
HTS	Heat Transport System
IWST	Injection Water Storage Tank
LOF	Loss of Flow
LRV	Liquid Relief Valve
MCA	Mechanical Control Absorber
NGS	Nuclear Generating Station
PAWCS	Post Accident Water Cooling System
P&IC	Pressure and Inventory Control
PVM	Parallel Virtual Machine
RIH	Reactor Inlet Header
ROH	Reactor Outlet Header
RRS	Reactor Regulating System
SDS1	Shutdown System 1
SDS2	Shutdown System 2
SG	Steam Generator
tc	Time constant
TUF	Two Unequal Fluids

1. INTRODUCTION

The Canadian nuclear industry is currently considering replacing its thermal-hydraulics codes, and the USNRC codes TRACE and RELAP5, which are made available via the Code Applications and Maintenance Program (CAMP), are potential candidates. Accordingly, AMEC NSS is assessing these codes for their applicability to CANDU safety analysis. This assessment includes demonstrating that the codes have the functionality required to model features unique to CANDU reactors and their validation against data specific to CANDU reactors.

An important step towards demonstrating the applicability of the CAMP codes to CANDU reactors is to show their ability to predict known station transients. This report presents the results of such a comparison between a RELAP5 simulation and plant measurements for a loss-of-flow event at the Darlington Nuclear Generating Station.

Simulation of this loss-of-flow event required creation of a RELAP5 model of a Darlington reactor, as well as the incorporation of custom controller routines in the code. The latter was required because the full functionality of the process control systems could not be captured fully with the existing version of the code.

This report includes a description of Darlington NGS, the loss-of-flow event upon which the RELAP5 simulation is based, the plant model and the resulting comparison of the code predictions with actual plant measurements.

2. DARLINGTON PLANT OVERVIEW

The Darlington Nuclear Generating Station consists of four CANDU reactors. A simplified schematic of the main process system for one CANDU reactor is shown in Figure 1. The process systems can be divided into primary side and secondary side systems. The primary side covers the nuclear side of the plant in which heavy water is circulated through the reactor core, transporting heat to the steam generators. The secondary side covers the light water, conventional side of the plant used to generate steam to supply the turbine.

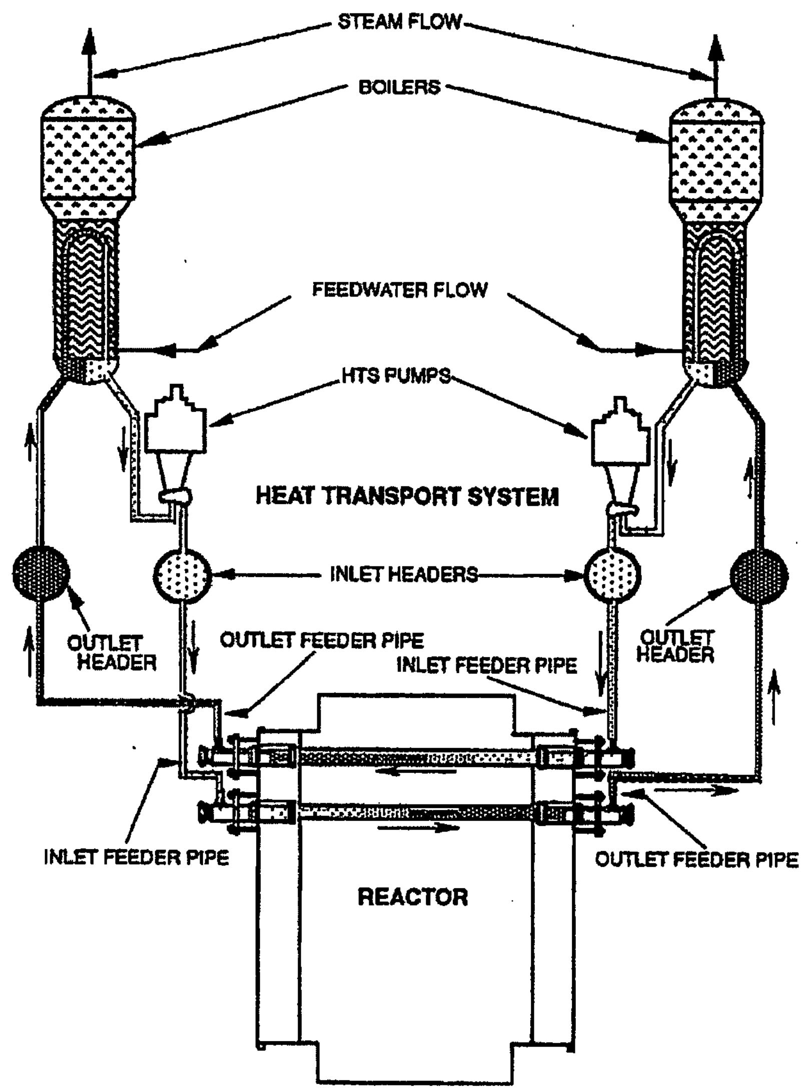

Figure 1 Simplified Schematic of the Heat Transport System

2.1 Primary Side Systems

The main system on the primary side of the plant is the Heat Transport System (HTS). The HTS circulates heavy water through 480 horizontal channels in the reactor core, each containing a pressure tube with thirteen fuel bundles. Coolant circulation is arranged in two loops, each supplying one half of the channels in the reactor core; Figure 1 is a simplified schematic of one such loop. Each loop passes through the core twice and therefore has two Reactor Inlet Headers (RIHs) and two Reactor Outlet Headers (ROHs), with the RIH in a given core pass supplying coolant to 120 channels. Circulation is provided by four heat transport pumps, two in each loop. Heat transported from the reactor core is rejected to the steam generators. Pressure on the primary side of the plant is maintained by the Pressure and Inventory Control (P&IC) system. A key component of the P&IC system is the pressurizer.

During normal operation, pressure in the HTS is controlled by adjusting the steam pressure in the pressurizer. ROH pressure measurements that exceed the pressure setpoint cause the pressurizer steam bleed valves to open. The valves discharge steam to the shell side of the bleed condenser, reducing the pressurizer steam pressure and therefore the HTS pressure. Conversely, low ROH pressure measurements cause the steam bleed valves to close, and the pressurizer heater power to increase, increasing the steam pressure in the pressurizer.

Inventory in the HTS is controlled by maintaining the pressurizer liquid level at the power dependant setpoint. A liquid level higher than the setpoint drives the bleed valves open and feed valves closed. A liquid level lower than the setpoint drives the bleed valves closed and the feed valves open.

Overpressure protection is provided by Liquid Relief Valves (LRVs). The LRVs connect the HTS to the shell side of the bleed condenser, and open if the HTS pressure exceeds the LRV opening high pressure setpoint.

2.2 Secondary Side Systems

On the secondary side of the plant, light water is supplied to the shell side of each of the four steam generators by the Steam Generator Feedwater system. The resulting steam is transported to the turbine through the piping of the Main Steam Supply system.

Steam generator pressure control is achieved through the Atmospheric Steam Discharge Valves (ASDVs) and the Condenser Steam Discharge Valves (CSDVs). The ASDVs can relieve up to 10% of the full power steam flow to the atmosphere in the event of an increase in steam generator pressure above the setpoint. The CSDVs can relieve up to 70% of the full power steam flow. The CSDVs bypass the turbine and discharge directly to the condenser and are opened automatically following a turbine trip.

2.3 Reactor Regulating System

The reactor regulating system controls the neutron power in the reactor core by manipulating either the light water zone controllers or the mechanical control absorbers.

Fourteen light water zone controllers are located in vertical columns throughout the reactor core. Neutron flux is regulated by adjusting the water level in compartments within these controllers. An increase in zone controller water level results in a higher neutron absorption rate (reduction of reactivity) and the lowering of reactor power. A decrease in zone controller water level results in a reduction in neutron absorption and an increase in reactor power.

Four Mechanical Control Absorbers (MCAs) are used for coarse control of reactor power. The four controller rods are grouped into two banks. They are used if the power error or the light water zone controller levels exceed predefined values. The MCAs are also inserted if a reactor stepback is initiated.

Stepbacks (a step change reduction in reactor power) and setbacks (a ramped reduction in reactor power) are initiated if predefined conditions are met. An example of a condition that results in a reactor stepback is a turbine trip. An example of a condition that results in a reactor setback is a high pressurizer level.

In the absence of a stepback, the rate at which the MCAs are driven into the core depends on the size of the power error. In the case of a stepback, the MCAs are dropped into the core under gravity.

2.4 Special Safety Systems

Special safety systems relevant to the loss of flow event discussed in this report are Shutdown System 1 (SDS1) and Shutdown System 2 (SDS2). Activation of SDS1 results in the insertion of negative reactivity into the reactor core through dropping of shutoff rods. Activation of SDS2 results in liquid poison injection to the moderator that surrounds the reactor core. Each shutdown system has multiple trip parameters that can initiate a reactor trip. Two trip parameters on both SDS1 and SDS2, which protect against loss of flow events, are the Heat Transport Low Flow (HTLF) and the Heat Transport High Pressure (HTHP) trips.

The SDS1 HTLF trip is based on flow measurements from three flow instrumented channels in each of the four core passes. Low flow measurements from any two channels in a given core pass result in an HTLF trip signal. The SDS2 HTLF trip logic is identical to the SDS1 trip, but it utilizes independent flow measurement channels and instrumentation.

The SDS1 and SDS2 HTHP trips are activated when the pressure measurements on any ROH exceed the HTHP trip setpoint.

3. DESCRIPTION OF LOSS OF FLOW EVENT

On November 25, 1993, a loss-of-flow event occurred at Darlington Unit 4. The event was initiated by a switchyard transformer explosion, resulting in the loss of Class IV power to Unit 4 and a consequential turbine trip. The Reactor Regulating System responded by initiating a reactor stepback to 60 %FP. The HT pumps tripped because of the loss of Class IV power, significantly reducing the coolant flow through the channels. The resulting increase in heat transport system pressure was sufficient to activate opening of the LRVs, which provided pressure relief via discharge to the shell side of the bleed condenser. The bleed condenser high pressure alarm was activated shortly thereafter.

As the HT pumps continued to run down, the measured flow in the SDS channels dropped below 20.5 kg/s, the HTLF setpoint for both SDS1 and SDS2 at reactor powers greater than 70% FP. A SDS1 trip was initiated at this time. Due to instrument sampling time, the SDS2 low flow trip was initiated shortly afterwards. A summary of the event sequence is shown in Table 1.

Event	Time	Time Relative to Loss of Class IV Power (s)
Loss of Class IV Power	15:32:31.0	0.0
Turbine Trip*	15:32:31.0	0.0
Stepback on Turbine trip*	15:32:31.5	0.5
CSDVs open	15:32:31.8	0.8
ASDVs open	15:32:32.0	1.0
SDS1 low flow trip	15:32:33.9	2.9
Liquid Relief Valves open	15:32:34.5	3.5
SDS2 low flow trip	15:32:42.6	11.6

* Timings shown are estimates.

Table 1 Darlington Loss of Flow Event Sequence

In CANDU loss-of-flow safety analysis, the focus of the full system simulations is on system response up to the time of reactor trip and shortly thereafter. This facilitates the assessment of fuel cooling up to reactor trip time, the capacity of the interim heat sink at the time of reactor trip, and heat transport system overpressure protection.

In the longer term, flow patterns in the horizontal channels can become difficult to predict in the absence of forced circulation. Flows in the channels are driven by coolant density differences around the circuit, and it becomes difficult to preclude flow reversal or temporary flow stagnation in some channels. Longer term fuel cooling is conservatively assessed assuming temporary flow stagnations in some channels.

Assessment of fuel cooling is beyond the scope of the present report, which instead focuses on the following global system parameters:

- *Core flows.* Flow rundown is a direct consequence of the loss of Class IV power, and indication of low flow resulted in a reactor trip for this event. As will be discussed later, RELAP5 single channel models of the flow instrumented safety channels are not available in the current Darlington NGS nodalization, and flow rundown in the safety channels is therefore not directly assessed. Instead, the header conditions that would be used to drive single channel analysis are qualitatively assessed.[1]

- *Reactor outlet header pressures and the pressurizer level.* Coolant swell caused by the flow reduction is reflected in the pressurization transient. The heat transport high pressure trip (based on ROH pressure measurements) provides effective back-up trip coverage for a number of loss-of-flow events.

- *Steam generator level.* The levels provide an indication of the heat transfer between the primary and secondary sides of the plant. The levels are also reflective of the available steam generator inventory (i.e., interim heat sink) following the loss of Class IV power, which is a key parameter required to mitigate the consequences of total loss of Class IV power events.

[1] The development of a 480 single-channel model is planned in the next phase of the program for validating RELAP5 against data specific to CANDU reactors, in order to assess directly the flow rundown in the safety channels.

4. DARLINGTON CONTROLLER MODELING WITH RELAP5

Simulation of the Darlington loss of flow event requires modeling of Darlington process control systems. Implementation of these controllers using built-in RELAP5 basic control components in the input data was investigated and found to be incapable of capturing all aspects of controller logic. In particular, it was found that conditional evaluation is not a function of RELAP5, nested loops (or simple loops) are extremely complex and difficult to interpret, and a maximum of 10,000 control components are permitted.

Instead, the Darlington control systems were implemented by developing new controller subroutines and compiling these with the existing RELAP5 source code. The coding was performed in a modular fashion to facilitate integration with future releases of RELAP5. To accommodate the new routines, some additional changes to the existing code were required to allow the new controller data to be read and the control variables initialized. The version of RELAP5 to which the Darlington controllers have been added is RELAP5/MOD3.3 Patch 03.

4.1 New Controller Input Data

The new Darlington controller routines cover the control functions shown in
. The list of modeled functions covers all controller responses required for simulation of the Darlington November 23, 1995, loss-of-flow event. Additional controller functions that are not required for the loss-of-flow simulation, e.g., emergency coolant injection, were also created for future analysis of other accident scenarios.

- Steam Generator Pressure Control System, comprising the following subroutines:
  - Live steam limiter
  - Steam generator pressure control
  - Condenser pressure
  - Governor valve
  - Re-heater steam flow
  - Turbine pressure
  - Turbine trip
- Normal mode pressure and inventory control, comprising the following subroutines:
  - Bleed condenser level control
  - Bleed condenser pressure spray control
  - Normal mode inventory control
  - Normal mode pressure control
- Steam Generator Feedwater Control, comprising the following subroutines:
  - Feedwater pump recirculating valves
  - Reheater return flow
- First and second order filter subroutine for simulation of the measurement lags
- Linear interpolation subroutine

- Steam Generator Level Control Emergency Coolant Injection, comprising the following subroutines:
  - High pressure bypass
  - High pressure pump startup
  - High pressure recirculation valves
  - Low pressure recirculation valves
  - Low pressure pump start
  - Steam generator cool down
  - Steam generator emergency cooling
- Reactor Regulating System, comprising the following subroutines:
  - Thermal power calculation
  - Demand power routine – slow program
  - Demand power routine – fast program
  - Light water zone controller
  - Mechanical control absorber rods
  - Shut-off rods withdrawal
- Reactor Power Stepback, comprising the following subroutines:
  - Reactor SDS1 or SDS2 trip
  - Turbine trip
  - HT pump trip
  - High HT pressure trip
  - High zone flux trip
  - High power log rate
  - Low pressurizer level trip
  - High HT pump suction pressure trip
  - Low steam drum level trip
  - High reactor power error trip
  - Absorber Rods Insertion
- Reactor Shut Down Systems, comprising the following subroutines:
  - High HT pressure trip
  - Low HT pressure trip
  - Low HT flow trip
  - Low pressurizer level trip
  - Low steam drum level trip
  - High reactor power trip
  - High reactor power log rate trip
  - Low boiler feed line pressure trip
  - Shut-off rods and poison injection

Table 2 Darlington Control Functions Implemented in RELAP5

Control variables in RELAP5 typically act as functions, i.e., following execution of the control action, the control variable is assigned a value. In contrast, the CANDU control components have been created as subroutines, thus allowing multiple outputs. The number of input/output and constant cards depends on the requirements of the controller. The general input structure of the CANDU control system components is such that the controller inputs are entered first, followed by outputs, and then constants.

A summary of the types of input and output parameters is shown below:

- Input variables
  - Input parameters originating from the thermal hydraulics code
  - Historical input parameters containing the state of a control variable from the previous program execution
  - Input parameters originating from other control systems
  - Constants that define the behavioral characteristics of the control system
- Output variables
  - Parameters returned to the thermal hydraulics code for control
  - Terms required for the next iteration of the controller, or variable demands internal to controllers

All data passed to or from a controller is included in the controller input file. This allows each controller to be activated in the code individually, provided that input from other controllers is mimicked by user supplied input.

Since the CANDU control system components produce multiple outputs, output variables are required to which the outputs are written. The CONSTANT control component can be used for this purpose. However, to prevent confusion regarding variable "constants", a CANDUOUT component has been created. The CANDUOUT component is identical to a CONSTANT component except in name. The component by itself has no function except to return one of the multiple output arguments from a CANDU control system component. The type of output is determined by its connection to the CANDU control system component.

4.2 Valve Stroking and Delay Times

An additional feature that was created to complement the controller routines is a valve stroking and delay time control component. The output from the Darlington controllers was filtered through the valve stroking and delay time control component prior to assigning a valve position to the servo-type control valves.

The valve stroking and delay time control component operates as a typical RELAP5 control component in that it is a function with a single output. It is identified by the new control component type VLVPOS.

4.3 Testing of New Controllers

The new controller routines have been tested in three stages, i.e., unit testing, system testing, and integrated testing.

Unit testing of the controllers exercised individual logic sequences of the controllers. To facilitate this, drivers were used to execute single time steps based on user supplied test data. The controller outputs were then compared to expected controller outputs, which were calculated analytically in EXCEL using the controller's logic and equations. Agreement was obtained between all code predictions and analytucal calculations.

System testing of the controllers exercised complete control systems using drivers with user supplied test data. The controller outputs were then compared to expected controller outputs, which were also calculated analytically using the controller's logic and equations. Agreement was obtained between all code predictions and analytical calculations.

Integrated testing of the controllers exercised the control systems after linking the routines with the RELAP5 executable. Controllers were tested by comparing the RELAP5 controller responses to a benchmark. The selected benchmark was the controller response predicted by the TUF code to a generic single pump trip event with a turbine trip credited and reactor shutdown not credited. The same generic transient input data was supplied to both the TUF and RELAP5 controllers, although it is likely that the controller sampling times differed marginally.

The results of the integrated testing are summarized in Figure 2 through Figure 13. Note that the only intent of these figures is to demonstrate that the new Darlington controllers yield the same response as the benchmark following integration into the RELAP5 code. As shown in these figures, there is excellent agreement between the RELAP5 and benchmark controller responses.

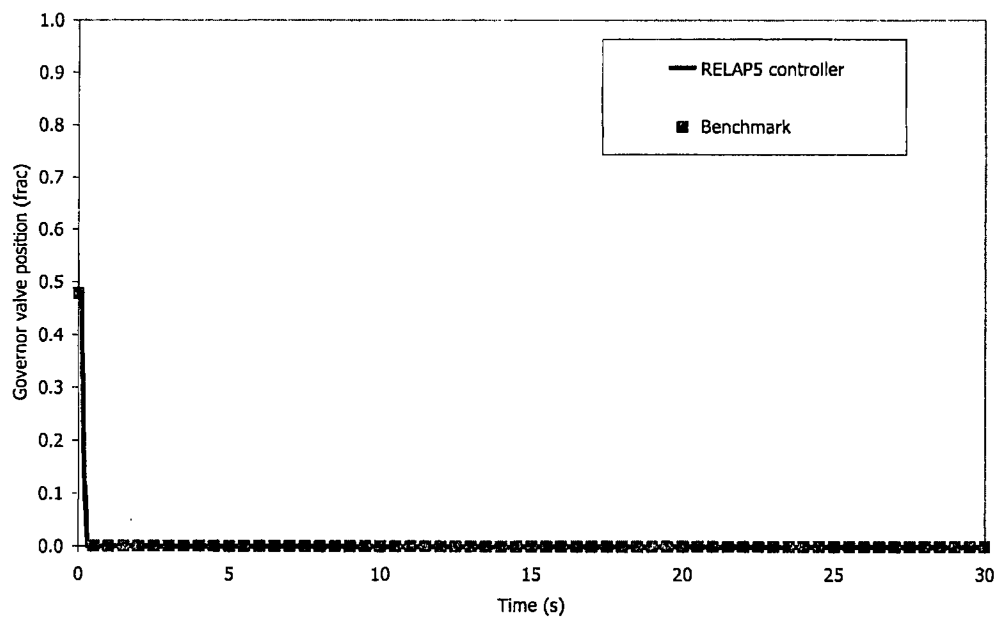

Figure 2 Integrated Testing – Governor Valve Position

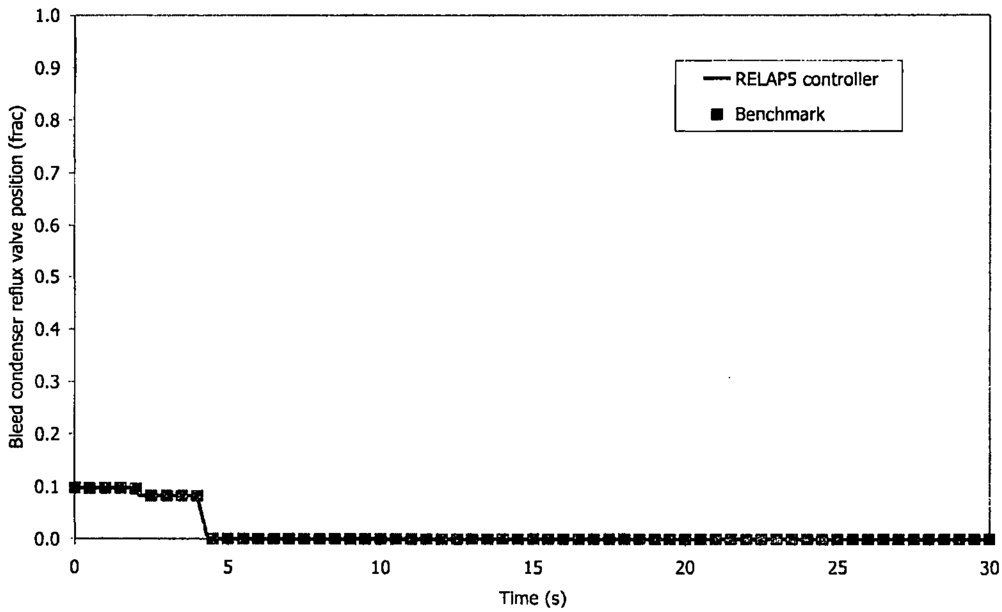

Figure 3 Integrated Testing – Bleed Condenser Reflux Valve Position

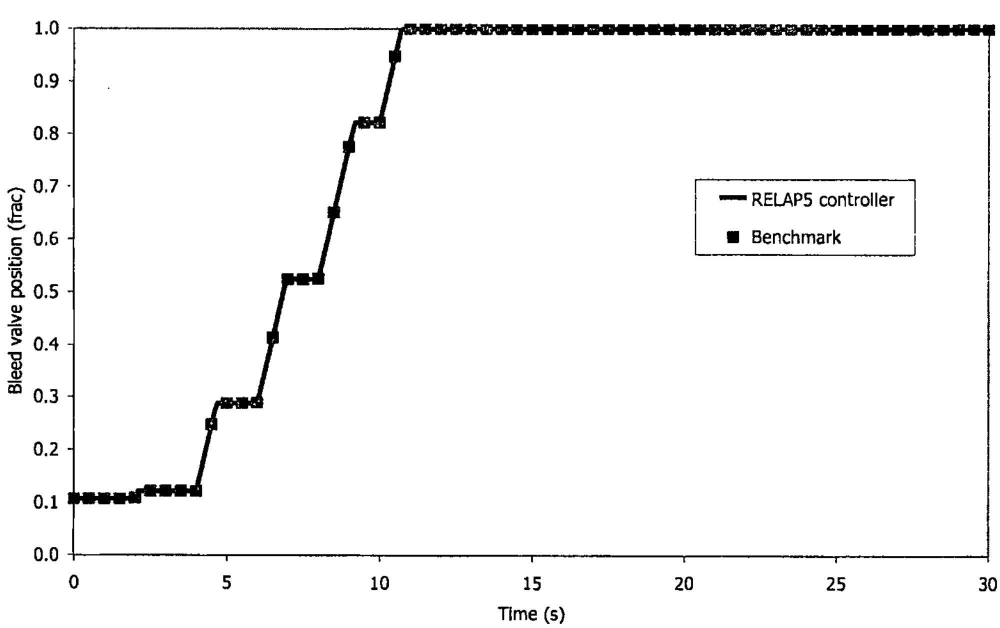

Figure 4 Integrated Testing – Bleed Valve Position

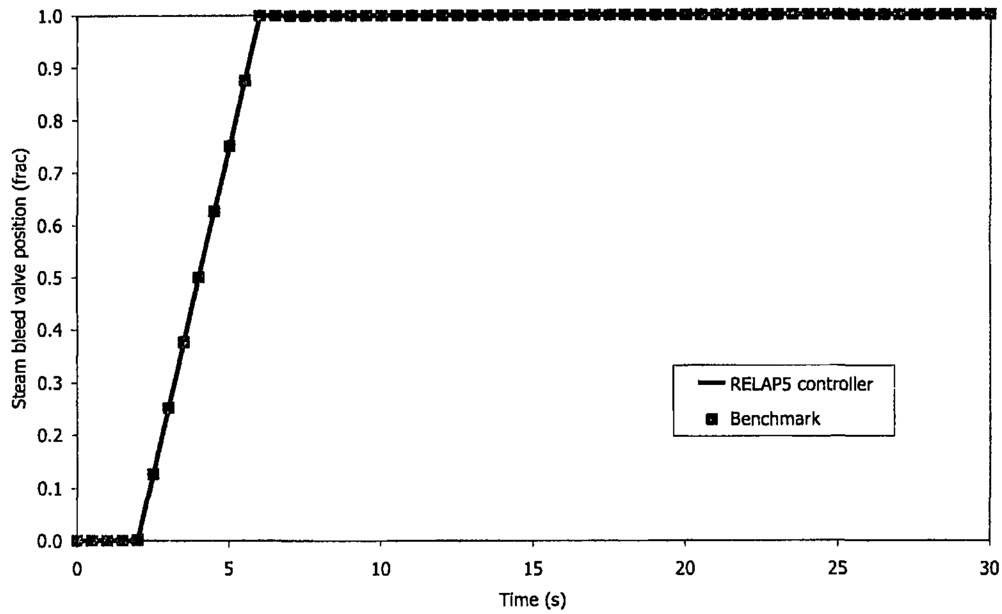

Figure 5 Integrated Testing – Pressurizer Steam Bleed Valve Position

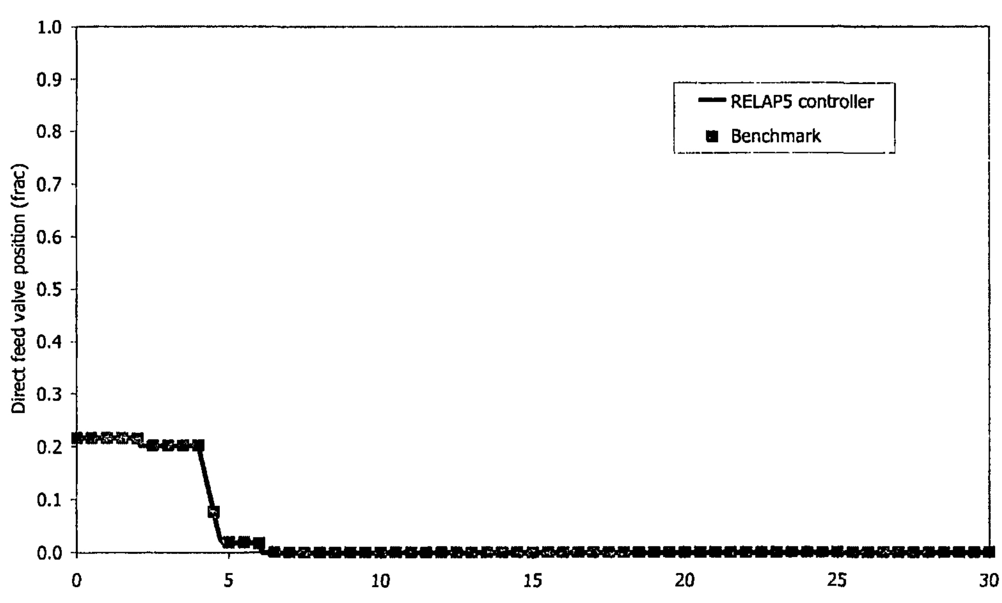

Figure 6 Integrated Testing – Direct Feed Valve Position

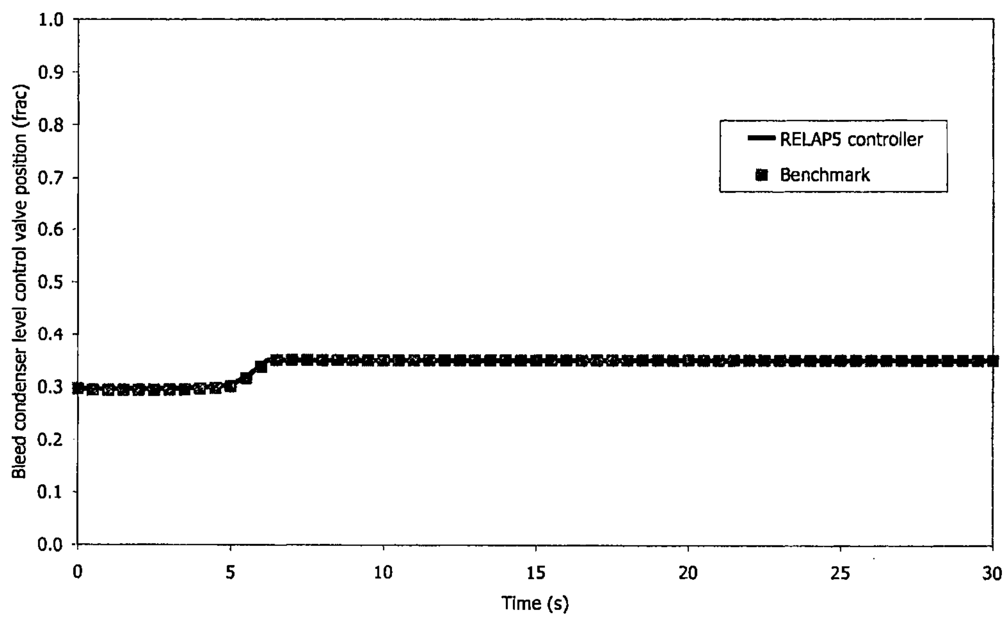

Figure 7 Integrated Testing – Bleed Condenser Level Control Valve Position

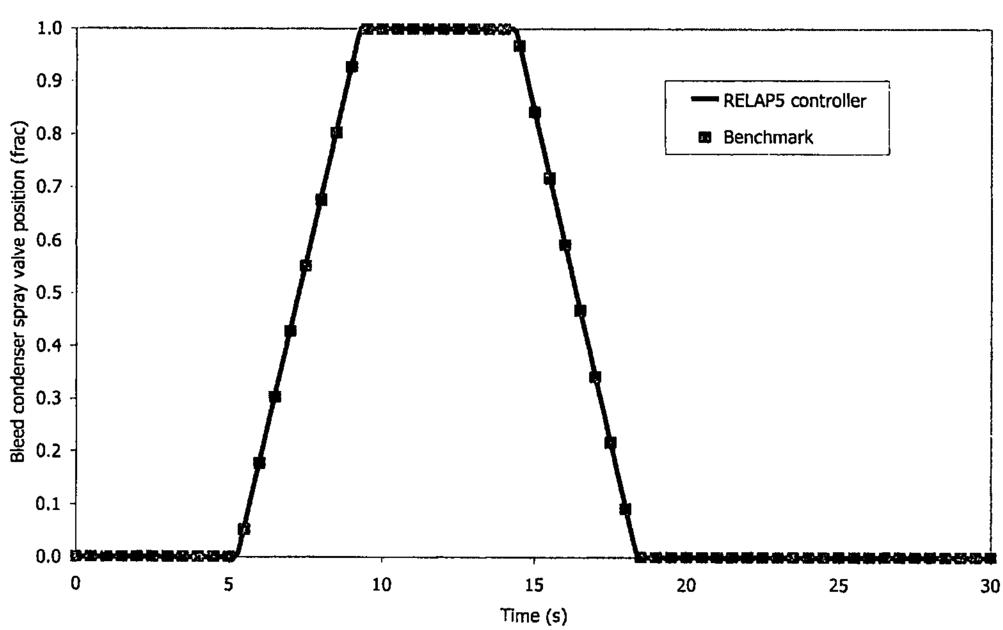

Figure 8 Integrated Testing – Bleed Condenser Spray Valve Position

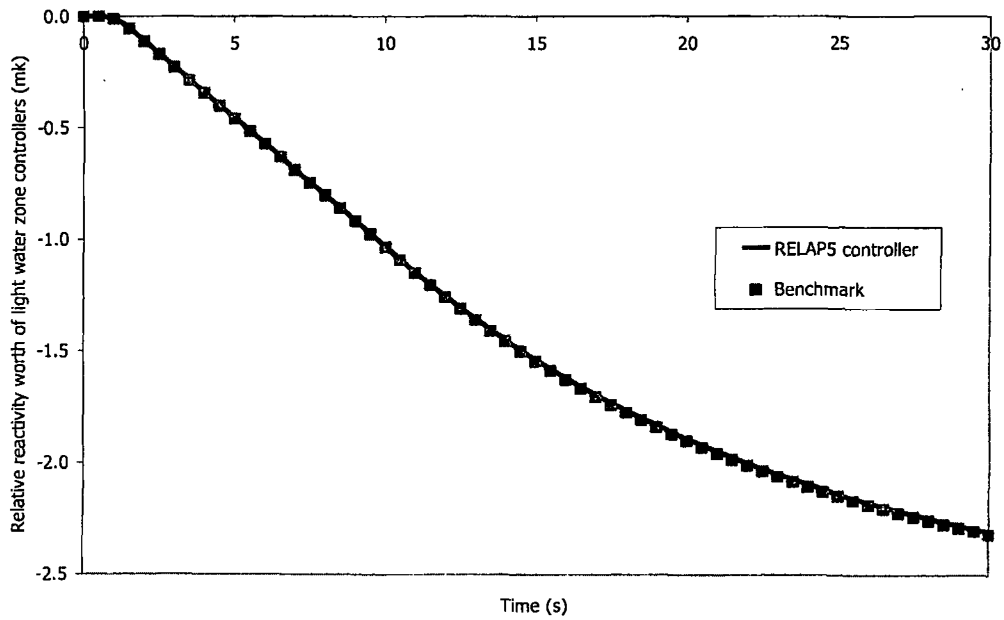

Figure 9 Integrated Testing – Reactivity Worth of Light Water Zones

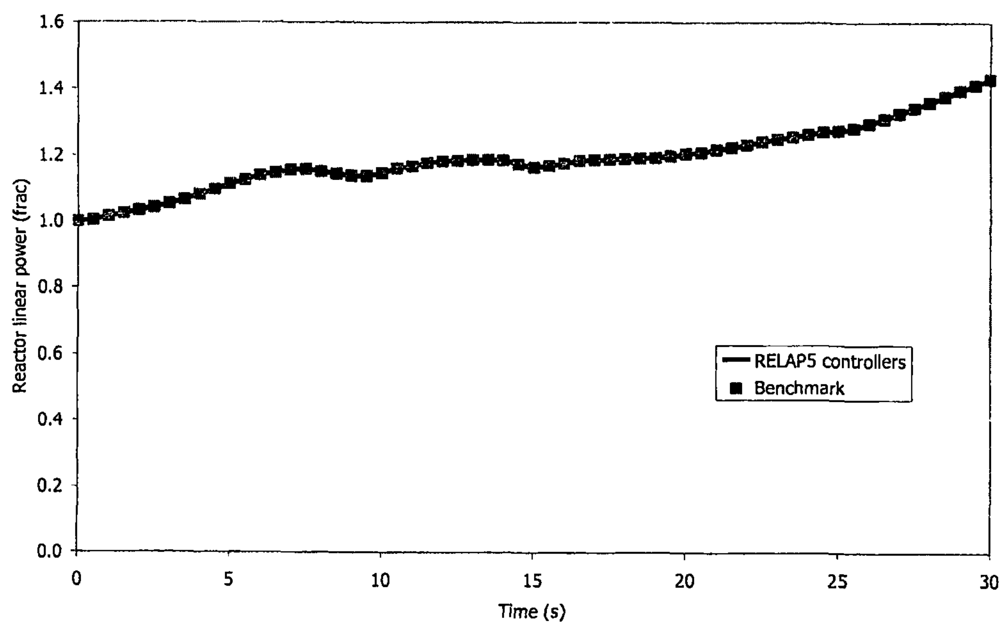

Figure 10 Integrated Testing – Linear Power Error

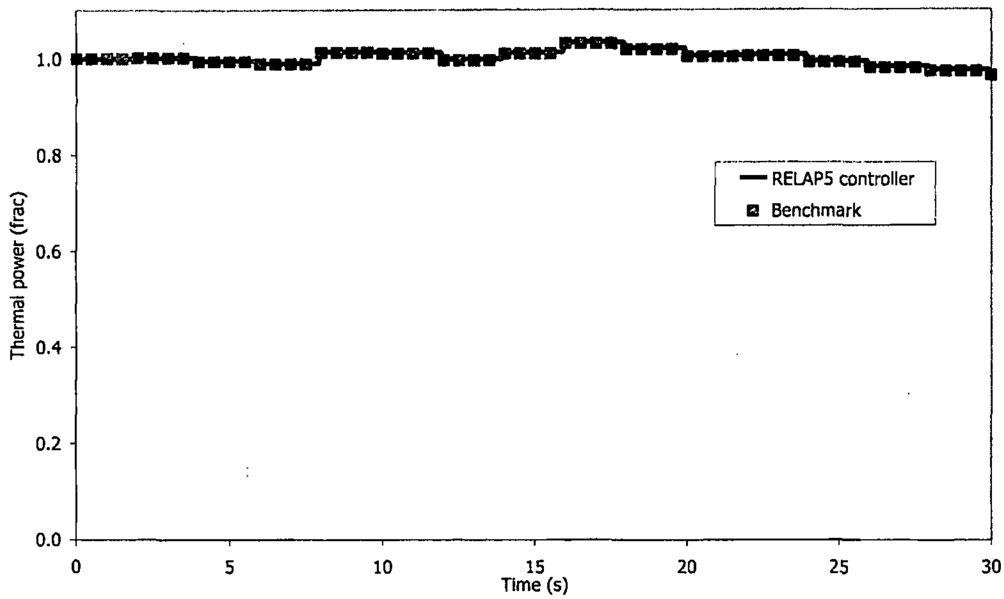

Figure 11 Integrated Testing – Calculated Thermal Power

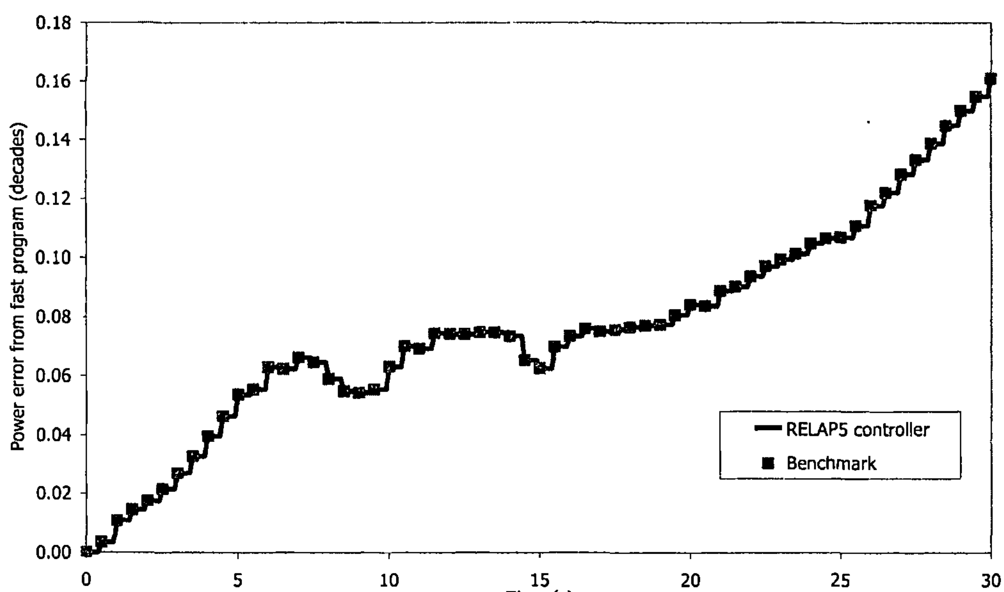

Figure 12 Integrated Testing – Power Error from Fast Program

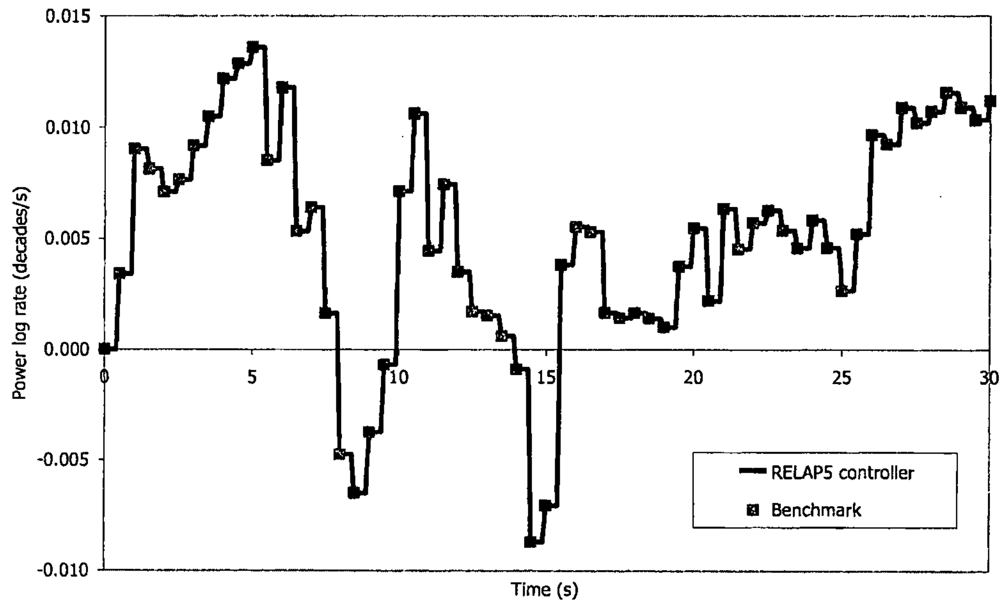

Figure 13 Integrated Testing – Log Rate Power

5. THE RELAP5 MODEL OF DARLINGTON

The Darlington RELAP5 model has been created primarily from plant reference data. The model comprises the heat transport system, the steam generator feedwater and steam supply systems, the pressure and inventory control system, and the emergency coolant injection system.

For each of these systems, standalone models were created and tested separately in null transients with appropriate boundary conditions supplied to represent interfacing systems. This permitted assessment of the response of each system without feedback from the interfacing systems. The systems were also combined in the overall Darlington RELAP5 input model and tested with a null transient.

5.1 RELAP5 Nodalization of Heat Transport System

The RELAP5 nodalization of the Darlington heat transport system (HTS) is shown in Figure 14. The model consists of the tube side of all four steam generators, the four HTS pumps, the four RIH and four ROH, feeders, channels, end-fittings and associated connecting piping.

Modeling of the channels in each core pass was done by parallel averaging of the channel flow paths in a given core pass. Therefore, each core pass is represented by one or more aggregate average channel. For three of the four core passes, one aggregate channel is used per pass representing the 120 channels comprising each pass. For the core pass originating from the Northwest (NW) RIH, the channels have been subdivided into six groups, each containing 20 channels. The core pass originating from the NW RIH therefore consists of six aggregate channels.

The RELAP5 model includes a detailed nodalization of the fuel channel end-fittings. Each channel in the core has two end-fittings, connecting the individual feeders from the headers to the channel (Figure 15). The end-fittings are designed to allow on-power fueling of the reactor.

In the reference RELAP5 nodalization, the heat transport system RIHs and ROHs have been divided into six and seven nodes respectively. In typical Darlington safety analysis, single nodes are used to represent reactor headers. The effect of modeling the headers by six nodes rather than one node was assessed through a sensitivity study to be discussed later in this report.

Darlington uses 37-element fuel bundles. Each bundle consists of a center fuel element surrounded by three concentric rings of fuel elements referred to as the inner ring, intermediate ring, and outer ring. Each Darlington fuel string is therefore represented by four heat structures for the center pin, inner, intermediate and outer rings, respectively. Each heat structure is divided into axial mesh points to allow power to be distributed along the length of the fuel string. An additional heat structure with multiple mesh points is included to simulate the presence of the pressure tube, calandria tube and the gap between the pressure and calandria tubes. The external surface of the calandria tube is in contact with the moderator fluid, which has a bulk temperature fixed at 70°C.

Heat structures are also associated with the feeders, headers, other connective piping above the headers, and steam generator U-tubes. The number of heat structures and other hydraulic components in the RELAP5 model of the heat transport system is summarized in Table 3.

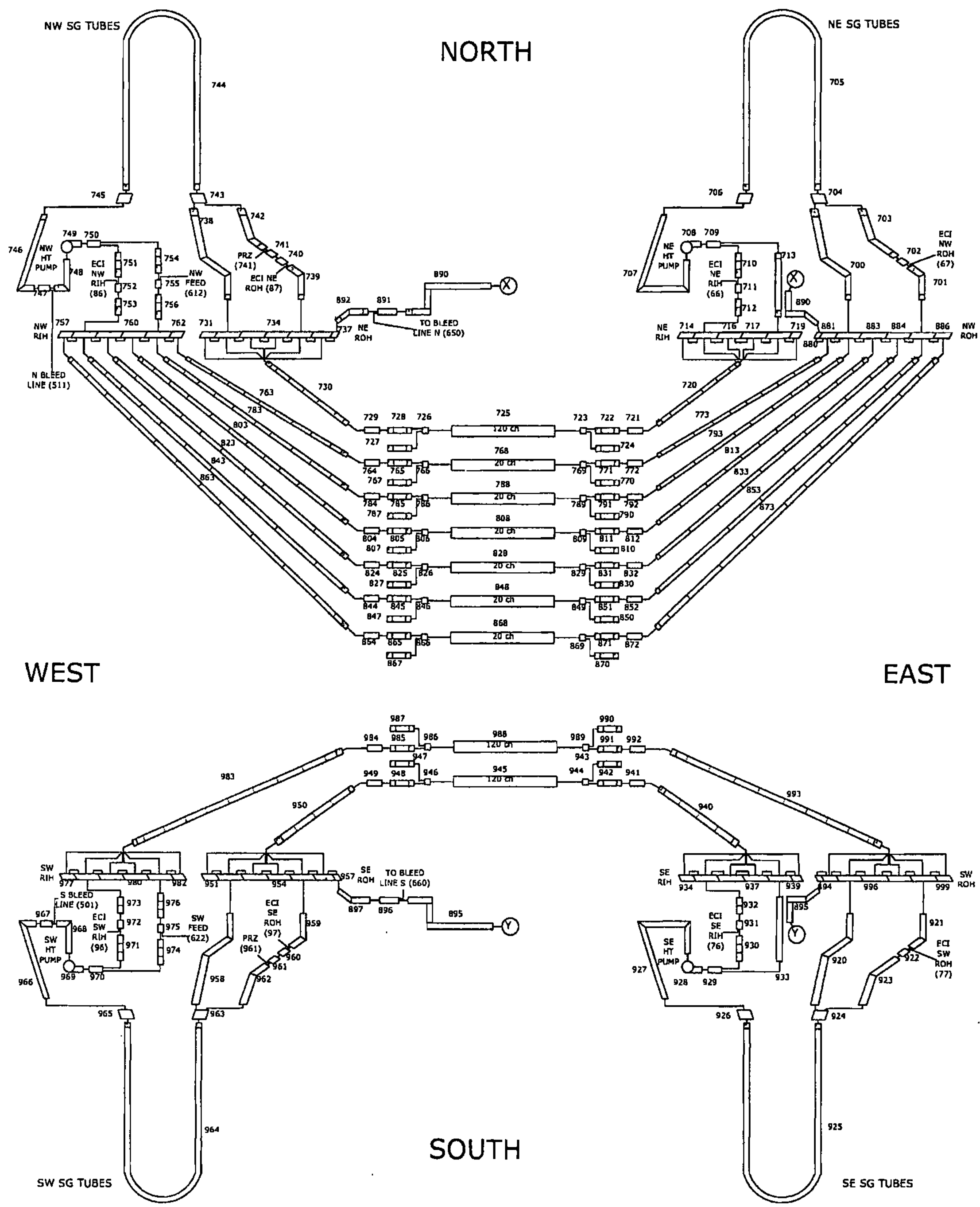

Figure 14 RELAP5 Nodalization of Darlington Heat Transport System.

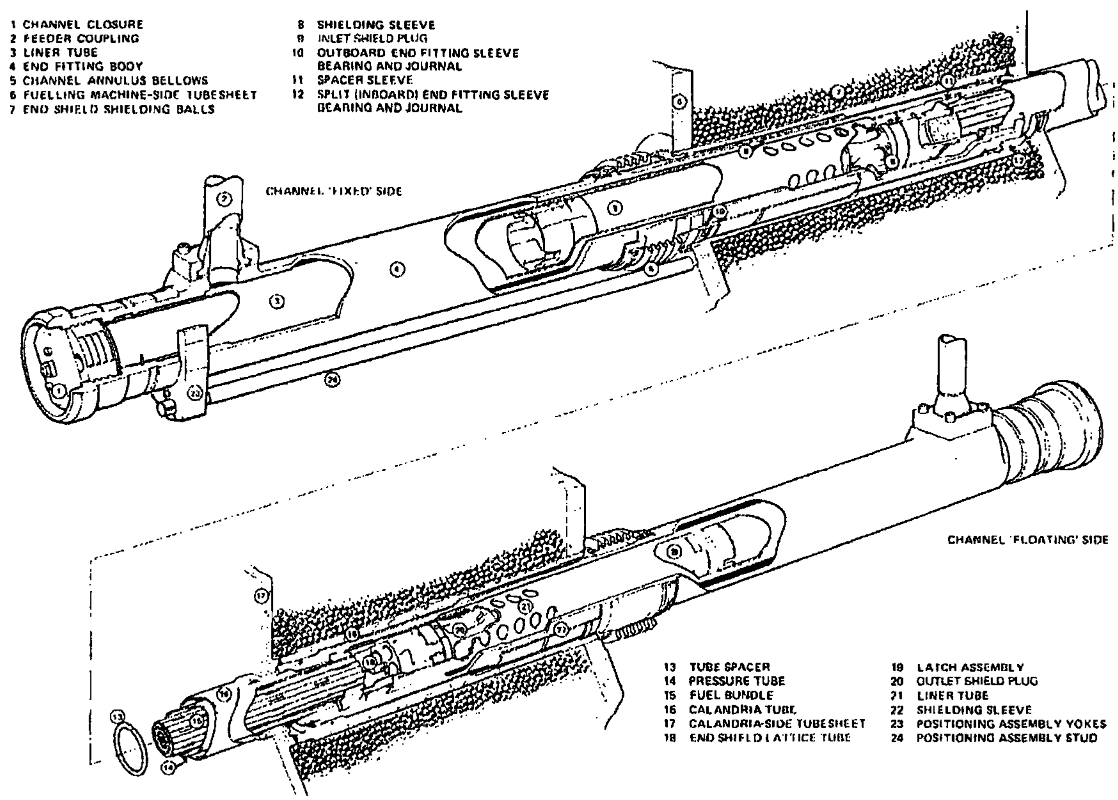

Figure 15 Darlington Fuel Channel Assembly.

Parameter	Value
No. of hydrodynamic components	250
No. of volumes	1674
No. of junctions	1721
No. of heat structures	2470
No. of mesh points	31467

Table 3 Summary of RELAP5 Components in HT System

5.2 RELAP5 Nodalization of the Secondary Side

The secondary side RELAP5 model covers the following main components:

- the steam generator feed water system from the deaerator up the shell side of the steam generators;
- the shell side of the steam generators; and
- the steam supply system from the shell side of the steam generators up to the turbine.

The RELAP5 nodalization of the secondary side is shown in Figure 16.

The starting point, i.e., the deaerator, and the ending point, i.e., the turbine, are both modeled as boundary conditions through time-dependent volume components. The main steam generator feed water pump component in the RELAP5 nodalization represents three parallel pumps. The pumps are modeled using the time-dependent junction component in RELAP5, where the flow rate-head is specified instead of the "time-dependent variable" relationship.

The High Pressure (HP) Heaters immediately downstream of the feedwater pumps are modeled using heat structures with a user defined power source to increase the feedwater temperature to approximately 177°C prior to entering the steam generators.

Feedwater flow to each steam generator (SG) is regulated by level control valves. Three such valves are shown in the RELAP5 nodalization for each SG, i.e., two large valves and one small valve. Under normal operation, only one large valve is used to regulate feedwater flow. Operation of the small valves would only occur at low reactor powers.

The shell side of the NE steam generator is shown in Figure 17. Appropriate heat structures have been added to the hydraulic part of the SG nodalization. In particular, heat transfer can occur through:

- the shroud between the downcomer and the preheater, and between the downcomer and the upper and lower boiling regions;
- the divider plate between the preheater and the lower boiler region; and
- the tubesheet between the preheater and the SG outlet bowl, and between the lower boiling region and the SG inlet bowl.

Heat losses to the environment from the external vessel of the SG are set to zero by representing the vessel as an insulated heat structure.

The number of heat structures and other hydraulic components in the RELAP5 model of the heat transport system is summarized in Table 4.

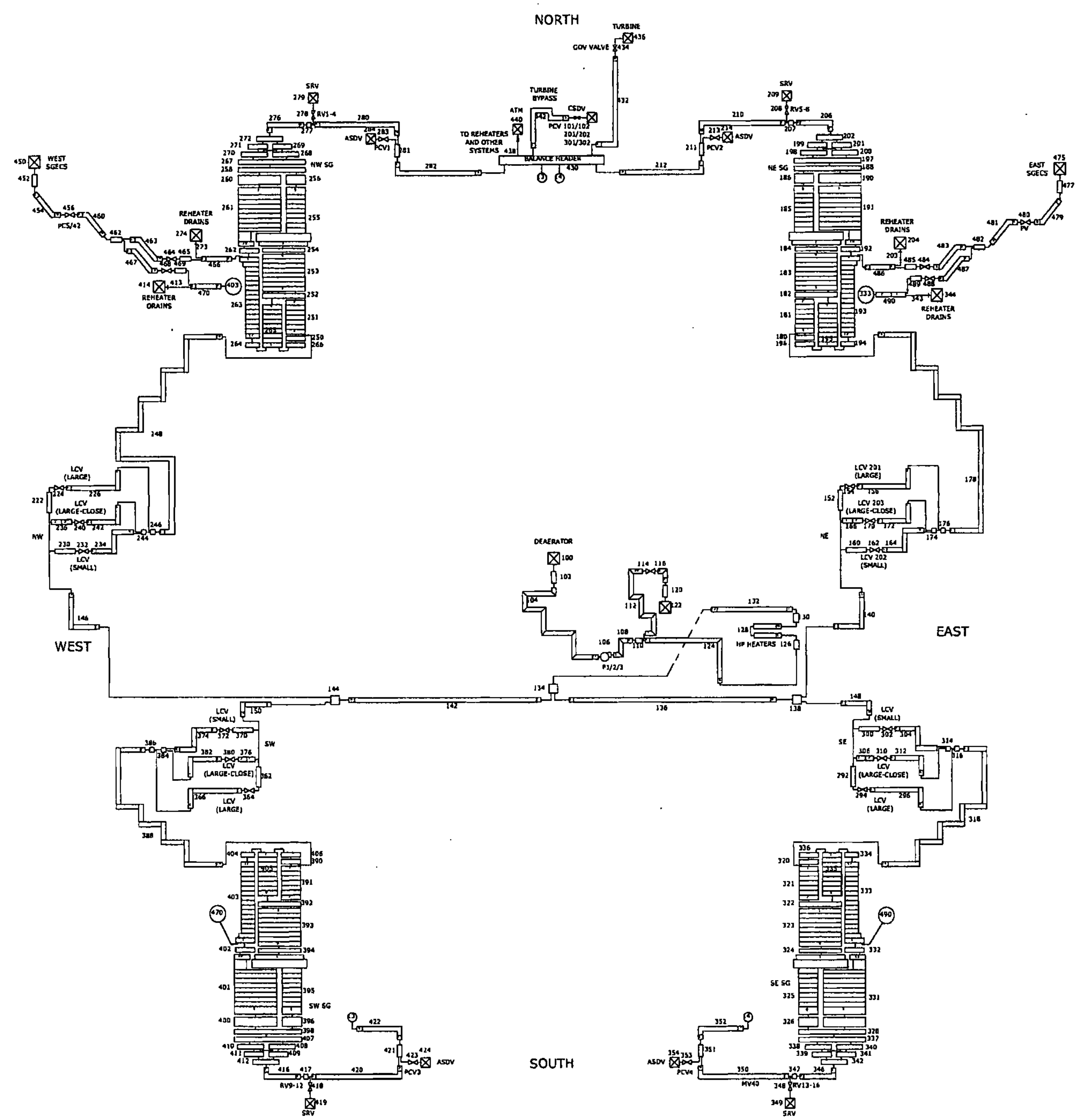

Figure 16 RELAP5 Nodalization of Darlington Secondary Side

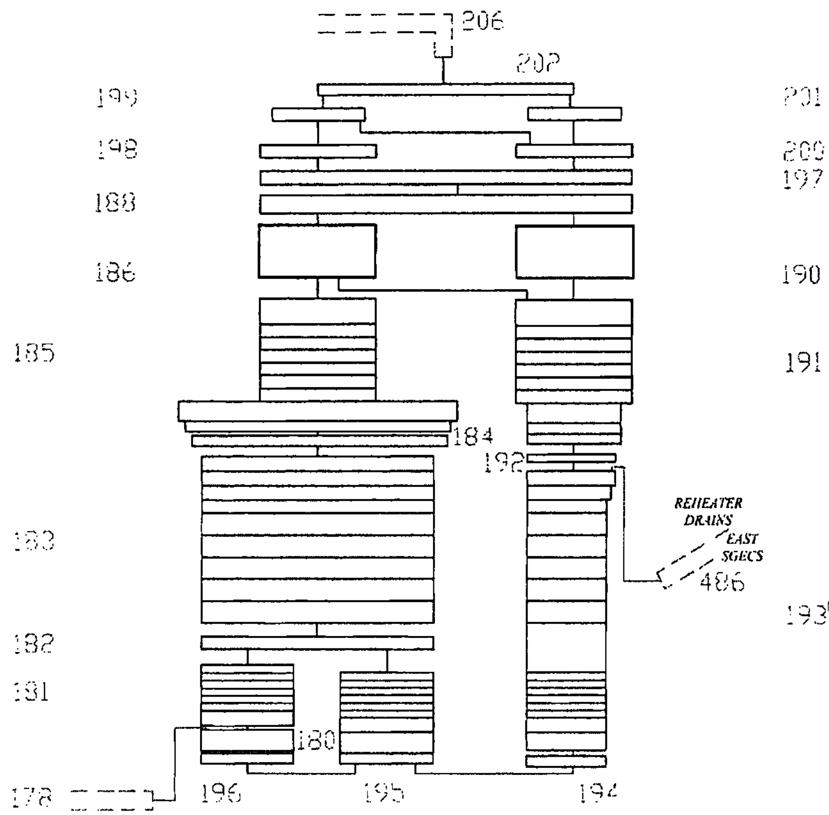

Figure 17 RELAP5 Nodalization of Darlington SG Shell Side

Parameter	Value
No. of hydrodynamic components	250
No. of volumes	1475
No. of junctions	1503
No. of heat structures	372
No. of mesh points	4986

Table 4 Summary of RELAP5 Components in Secondary Side

5.3 RELAP5 Nodalization of the Pressure and Inventory Control System

Pressure and inventory control of the heat transport system is performed through adjustments of the pressure and level in the pressurizer. Of secondary importance to heat transport system conditions is the bleed condenser. Hot bleed flow from the heat transport system is discharged to the shell side of the bleed condenser. The hot bleed flow flashes as it passes through the bleed valves, choking the flow. Due to the choked flow, variation of the conditions in the shell side of the bleed condenser does not have a significant effect on heat transport system conditions during overpressure transients such as occur for loss of flow events.

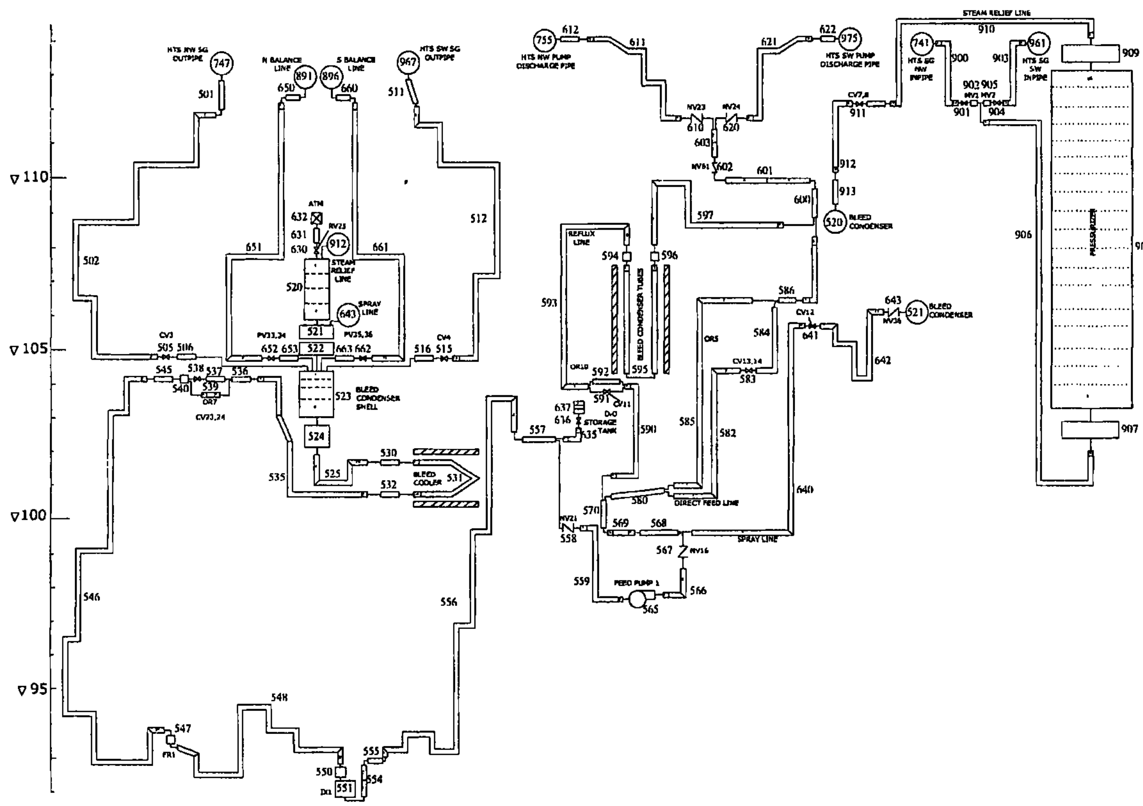

Figure 18 RELAP5 Nodalization of Darlington Pressure & Inventory Control System

The RELAP5 nodalization of the pressure and inventory control system is shown in Figure 18. The pressurizer is shown on the right side of the figure. The bleed condenser shell and tube sides are shown on the left and middle portion of the diagram, respectively. Beginning from the piping exiting the shell side of the bleed condenser, other components in the model are the bleed cooler, purification system, D_2O storage tank, D_2O feed pump(s), and the associated control and check valves.

The bleed valves are relevant to the prediction of conditions in the heat transport system during a loss-of-flow induced overpressure transient. Modeling of the choked flow through these valves has been performed based on the manufacturer supplied valve recovery coefficient.

The RELAP5 code does not support the recovery coefficient method of predicting choked flow rates in control valves. Rather the default criterion for determining choking is the Henry-Fauske correlation, which did not capture choking in the bleed valves during preliminary simulations. As a consequence, the flow rates through the bleed valves were originally overpredicted. Therefore, the valve recovery coefficient method was implemented in the input data to adjust the RELAP5 predicted flows. A series of control variables, single volumes and time dependent junctions was used for this purpose.

The number of heat structures and other hydraulic components in the RELAP5 model of the pressure and inventory control system is summarized in Table 5.

Parameter	Value
No. of hydrodynamic components	150
No. of volumes	358
No. of junctions	362
No. of heat structures	40
No. of mesh points	200

Table 5 Summary of RELAP5 Components in Pressure and Inventory Control System

5.4 RELAP5 Nodalization of the Emergency Coolant Injection System

Loss-of-flow assessment at Darlington does not require the use of the Emergency Coolant Injection (ECI) System. As such, the RELAP5 model implemented for ECI is intended for application in future analysis and will not be discussed in detail in this report. The RELAP5 nodalization of the emergency coolant injection system is shown in Figure 19.

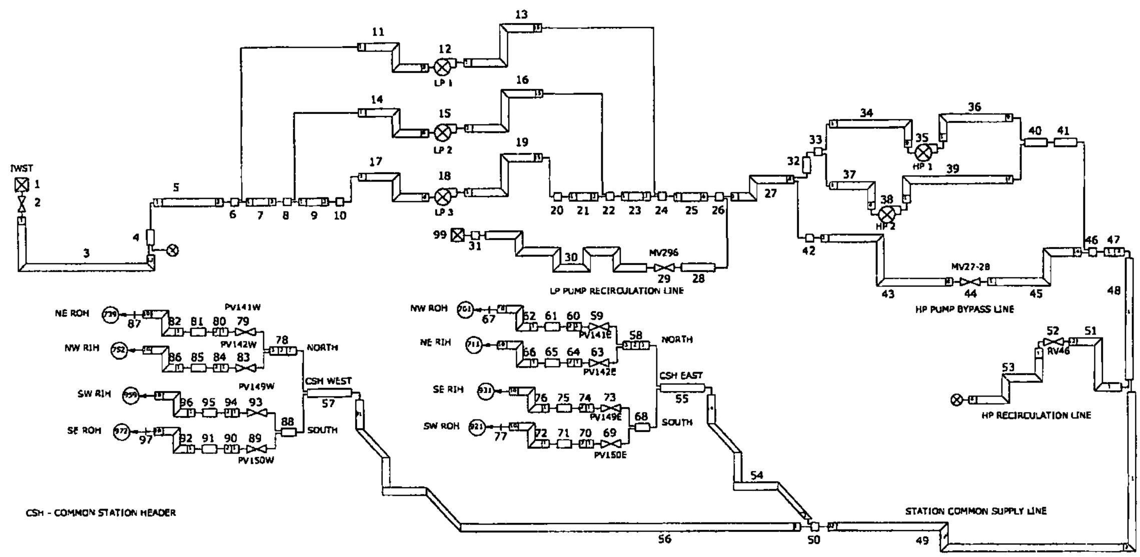

Figure 19 RELAP5 Nodalization of Darlington Emergency Coolant Injection System

In the event of ECI initiation, the coolant water supply is initially drawn from the Injection Water Storage Tank (IWST) shown on the left hand side of the figure. The flow passes through three low pressure pumps and two high pressure pumps prior to injection into each of the headers in the affected reactor.

Following depletion of the IWST inventory, ECI operation transitions to a recovery mode in which the injection water is drawn from sumps in the reactor vault. This mode of operation is not currently modeled and would require the representation of the Post Accident Water Cooling System (PAWCS).

The ECI model does not contain any heat structures. Other hydraulic components are summarized in Table 6.

Parameter	Value
No. of hydrodynamic components	98
No. of volumes	444
No. of junctions	452

Table 6 Summary of RELAP5 Components in Emergency Coolant Injection System

6. RELAP5 SIMULATION OF LOSS OF FLOW EVENT

The loss-of-flow event described in Section 3 has been simulated using the RELAP5 Darlington controllers (Section 4) and Darlington input model (Section 5). Additional assumptions specific to the loss of flow simulation, and the simulation results, are presented below.

6.1 Modeling Assumptions Specific to Loss-of-Flow Assessment

The assumptions applied in this assessment can be divided into two groups, i.e., those required to define the initial operating state, and those required to define the equipment credits following the initiating event.

The initial operating state has been defined based on the station data. To achieve this operating state, the following assumptions were applied to the holding transient:

- The reference 100% reactor thermal power to the coolant is 2651 MW. The indicated reactor power prior to the event was 99.8% full power.
- There is no flux tilt across the reactor core, i.e., the flux is distributed symmetrically.
- The steam generator pressure control is in hold mode with an average steam generator pressure of 5.05 MPa(g).
- The reactor outlet header pressure control is in normal mode with a setpoint of 9820 kPa(g).
- The initial bleed valve position has been adjusted to yield an average total bleed flow from the heat transport system of 12.2 kg/s.
- The temperature of the feedwater in the deaerator is 153°C.
- All four heat transport pumps are operating.
- Three steam generator feedwater pumps are operating.
- One D_2O feed pump is operating.
- All process systems are functioning normally.

Following the initiating event, the plant response depends largely on which pieces of equipment continue to operate following the loss of Class IV power. Class IV power is one of four classes of power in the Power Distribution System in CANDU Nuclear Plants, each with varying degrees of reliability and continuity. Class IV power is supplied from the operating unit or the bulk electrical grid. Of relevance to the present assessment, Class IV power supplies both the main heat transport pumps and the steam generator feedwater pumps. Under normal operation, Class IV power also supplies Class III power, which powers some equipment relevant to the loss-of-flow event simulation. Loads on Class III power can be interrupted for several minutes without affecting plant safety. From an examination of plant data, it appears that Class III power was restored within a few seconds of the initiating event.

The above considerations were used to inform assumptions regarding post-event equipment credits. Assumptions for simulation of the loss-of-flow event are as follows:

- A turbine trip occurs at time zero of the event.
- All four heat transport pumps trip at time zero of the event.
- All three steam generator feed water pumps trip at time zero of the event.
- The auxiliary steam generator feed pumps are started on Class III power 2 seconds after the initiating event.
- The operating D_2O feed pump is tripped at time zero of the event due to the temporary interruption in Class III power. Based on the bleed condenser spray flow trend in the station data, pump restart did not occur until about 85 s into the event. Since the period of interest in the present simulation is 0 s to 80 s, the D_2O feed pumps remain unavailable for the duration of the simulation.
- A reactor stepback occurred at 0.5 s due to the turbine trip. Previous work related to this event specified a window of 0 to 1 s within which the stepback could have occurred.
- A reactor trip occurred at 2.9 s due to low flow in the flow instrumented safety channels. Models of individual flow instrumented safety channels have not been created. As such, the reactor trip time based on station data has been imposed on the RELAP5 model.
- The ASDVs are available and function normally during the event.
- The CSDVs are available and function normally during the event until the condenser vacuum is lost.
- Based on station data, the condenser vacuum is lost causing the CSDVs to close at about 13.5 s. The condenser is not modeled in the RELAP5 input data. However, based on considerations of the CSDV control logic, the valve response was approximated by initiating forced closure of the CSDVs at 13.5 s and stroking the valves closed over a 2 second period.
- The bleed condenser level is controlled by the bleed condenser level control valve. Under normal operation, this valve is controlled only on the bleed condenser level error. However, a temperature override exists to close this valve in the event of high temperature readings downstream of the bleed cooler. Based on the station data, the temperature override was not activated during the loss-of-flow event. As such, the temperature override was masked in the RELAP5 simulation.

6.2 <u>RELAP5 Simulation Results and Comparison with Station Data</u>

The RELAP5 predictions of thermal-hydraulic parameters for the loss-of-flow event are shown in the figures below. Where available, station data is also shown for comparison. In all figures, event initiation is time zero.

Due to the loss of all four heat transport pumps, the loss-of-flow event affects each of the four heat transport system core passes equally. Similarly, the response of each of the four steam generators is essentially the same. As such, the discussion that follows focuses on the thermal-hydraulic parameters associated with one core pass, that originating from the North East (NE) RIH and the associated downstream ROH. The response of the steam generator upstream of the NE RIH will also be presented, i.e., the NE SG.

Figure 20 shows the response of the NE RIH pressure to the heat transport pump trips. Immediately following the heat transport pump trip at time zero, the header, which is downstream of the heat transport pumps, drops in pressure. However, the initial pressure drop is quickly terminated due to the mismatch between the power to the coolant and the heat transported out of the core by the coolant. Therefore, about 1 s into the transient, the header pressure trend temporarily reverses, increasing in pressure. At 2.9 s into the transient, the reactor is tripped on heat transport low flow and the neutron power drops sharply. The drop in thermal power to the coolant lags the drop in neutron power, but within the first 5 s the reduction is sufficient to depressurize again the NE RIH due to coolant shrinkage.

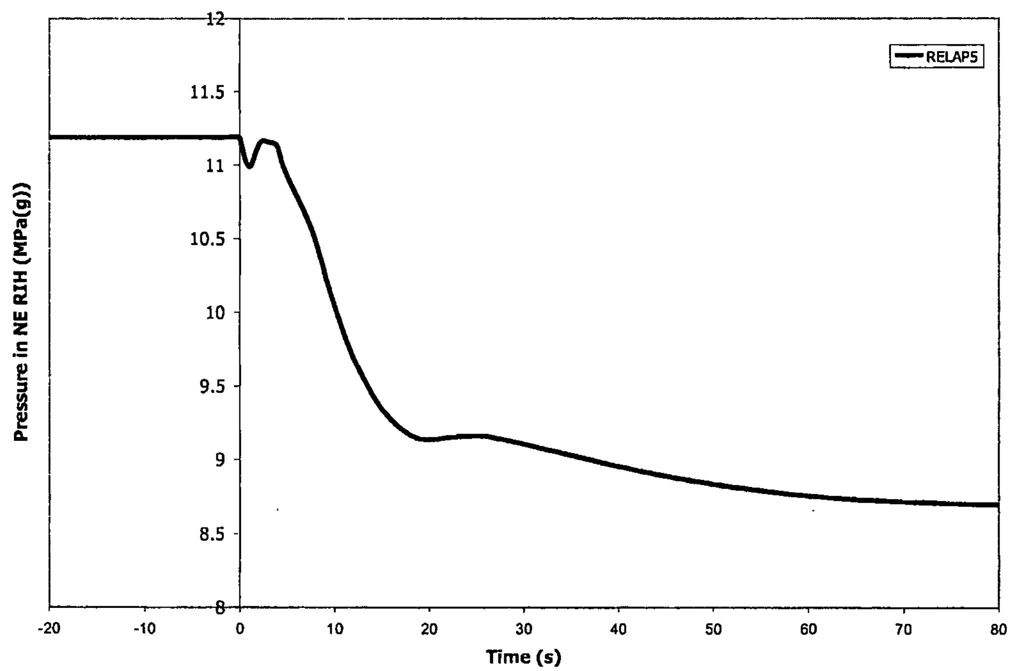

Figure 20 Darlington LOF Event – NE RIH Pressure

The corresponding temperature in the NE RIH is shown in Figure 21. The time constants associated with temperature measurement instrumentation can vary widely, depending on the instrument type and location. Quantification of instrument specific time constants for use in the temperature comparisons were not performed as part of this assessment. Instead, RELAP5 temperature predictions have been filtered using two different time constants (tc) to demonstrate the sensitivity of the measured parameter to the instrumentation time constant.

The temperature in the NE RIH is dictated in large part by the temperature on the secondary side of the upstream steam generator. For this event, there is an increase in the upstream steam generator pressure and therefore temperature. The increase in SG temperature is reflected in the increasing NE RIH temperature.

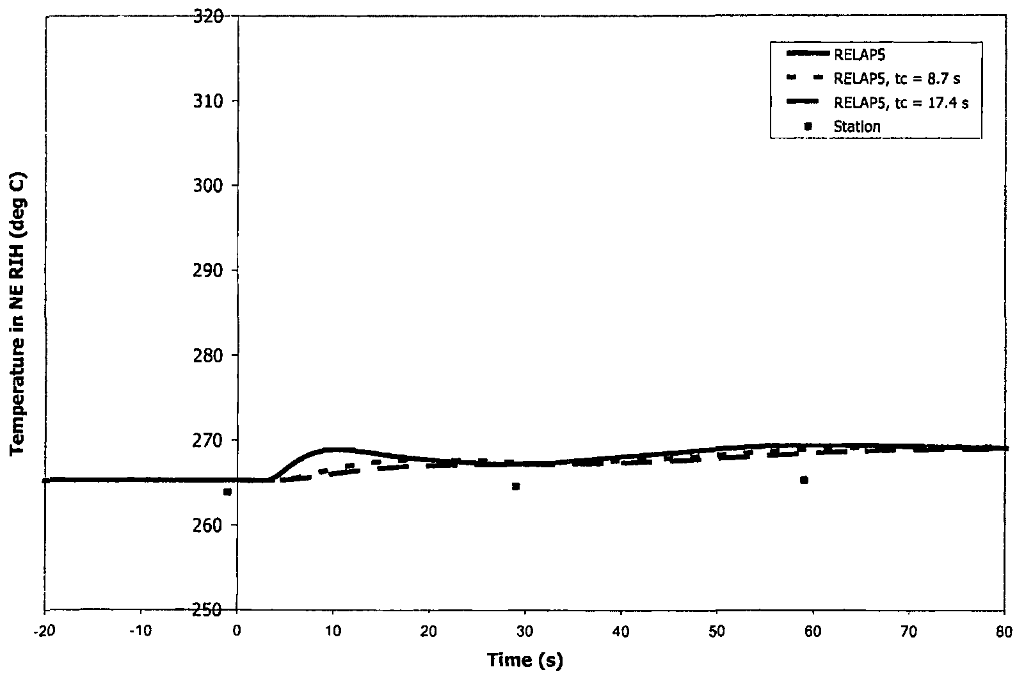

Figure 21 Darlington LOF Event – NE RIH Temperature

The ROH pressure downstream of the NE RIH is shown in Figure 22. For historical reasons, this outlet header is labeled the NE ROH. The immediate drop in pressure observed in the RIH does not occur in the ROH, due in part to the saturated coolant conditions in the header. Instead, the pressure trend in the ROH, which is directly downstream of the reactor core, is dominated by the mismatch between heat input to the core and heat transport from the core. The credited HTS low flow reactor trip at 2.9 s promptly terminates the neutron power, but because of the thermal lag, the ROH pressure continues to rise, reaching the Liquid Relief Valve opening setpoint at 3 s. Opening of the LRVs mitigates the HTS overpressure transient. The LRVs close again at approximately 6 s. Within the first 5 seconds of the transient, the combination of coolant shrinkage due to a reduction in power to the coolant, and opening of the LRVs, reverses the pressure trend and the pressure begins to decrease.

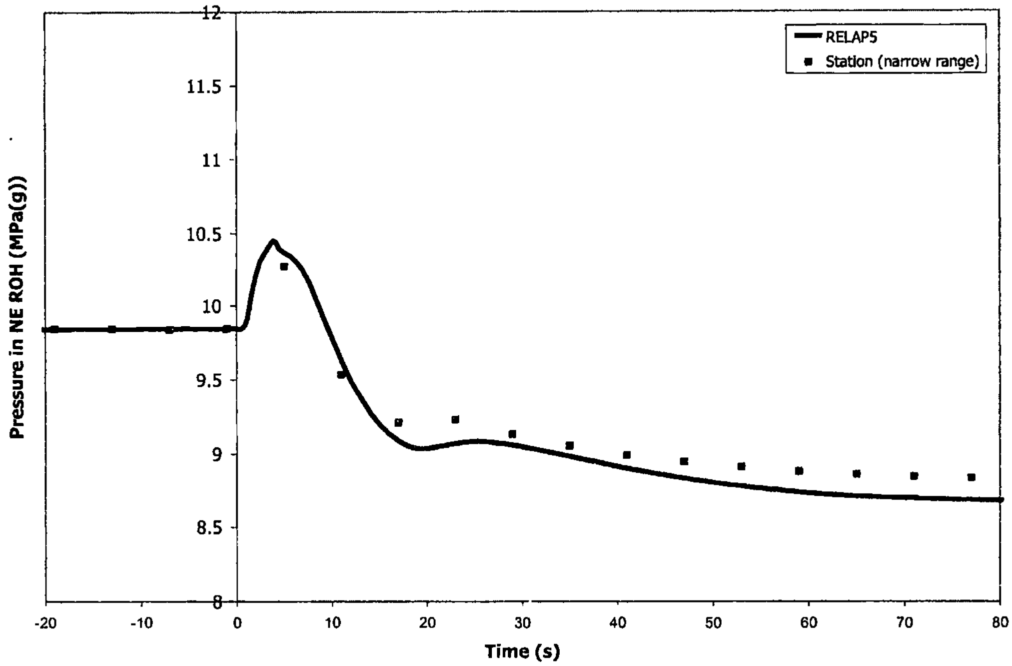

Figure 22 Darlington LOF Event – NE ROH Pressure

The drop in ROH pressure is limited by the rate at which the pressurizer depressurizes. The pressurizer is initially filled with saturated D_20 and steam at equilibrium conditions and is connected directly to the NE ROH. As the pressure drops below the initial equilibrium condition, the saturated liquid in the pressurizer vaporizes, slowing the depressurization transient. The pressure trends for the NE ROH and pressurizer are shown in Figure 23. Also shown in this figure is the void in the NE ROH. The presence of void in the HT system also slows the depressurization transient as shown between 20 and 25 s in the transient. After approximately 25 s, the void in the HT has collapsed, and the depressurization continues, driven by the vapor pressure in the pressurizer.

Qualitatively, the overall agreement between station measurements and predictions of conditions in the RIH and ROH is good. This provides confidence that single channel analysis of flow rundown in the safety channels would also be in good agreement with station data if assessed.

The temperature in the NE ROH is shown in Figure 24. Three RELAP5 temperature trends are shown, i.e., the direct RELAP5 temperature prediction, and two filtered trends which demonstrate the effect time constants associated with instrumentation can have on the measured value. The initial temperature increase shown in the RELAP5 prediction is caused by the mismatch between the power to coolant and the heat transport from the core discussed previously.

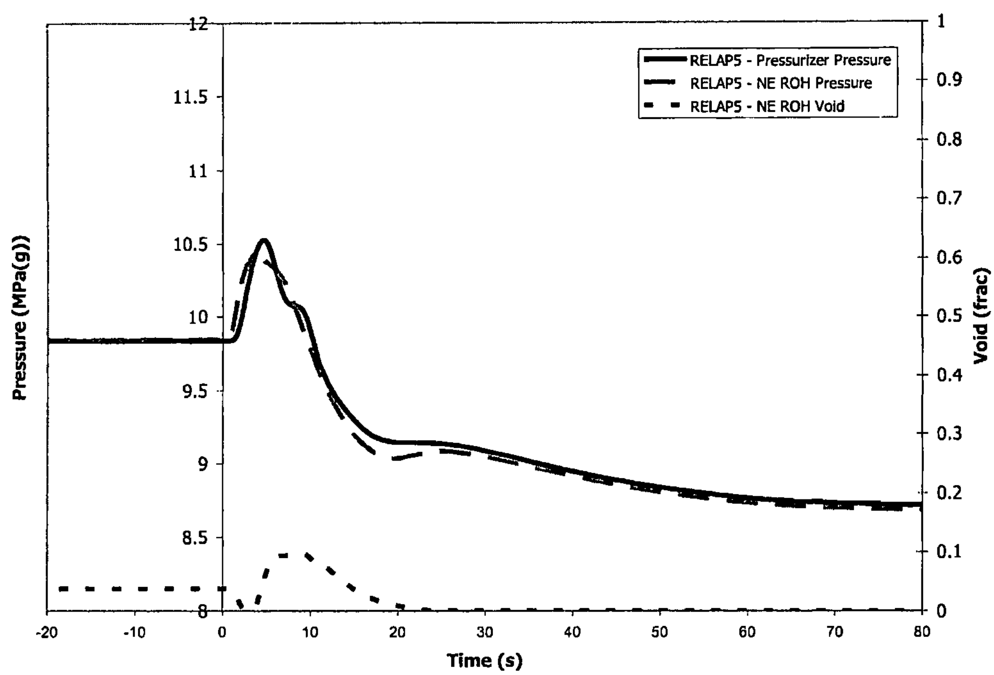

Figure 23 Darlington LOF Event – Comparison of Pressurizer and ROH Conditions

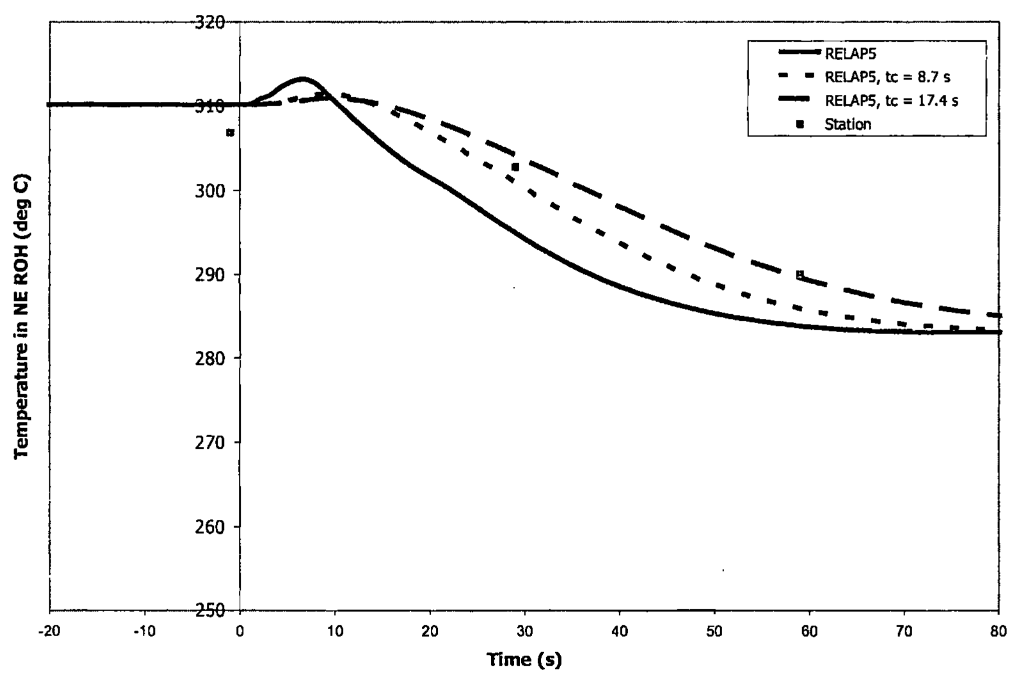

Figure 24 Darlington LOF Event – NE ROH Temperature

Comparisons of the RELAP5 predictions and station measurements for the pressurizer pressure and level are shown in Figure 25 and Figure 26. The RELAP5 pressure is underpredicted compared to station measurements in the later portion of the transient. This difference is also reflected in the ROH pressure trend presented previously in Figure 22. Quantification of measurement uncertainty was not part of this assessment and as such a quantitative assessment of the agreement between predictions and measurements was not performed. However, qualitatively, the overall agreement between the predictions and measurements is good. This provides confidence that the RELAP5 code is capable of capturing pressure transients of importance to CANDU safety analysis.

As noted, the inventory in the steam generators at the time of reactor trip is of interest in CANDU safety analysis for events in which SG feedwater is lost or interrupted. The available inventory allows the SGs to be used as an interim heat sink until a longer term heat sink, or alternate means of SG inventory make-up, are established. Although loss-of-flow events are not limiting with respect to SG inventory at time of trip, comparison of SG level predictions with measurements does provide a preliminary means of assessing the SG model. The NE SG level transient is shown in Figure 27 and is in excellent agreement with station data. The corresponding pressure in the NE SG is shown in Figure 28. The variations in SG pressure have a second-order effect on the HTS conditions, in that they change the shell-side temperature of the SG by approximately 1°C for every 0.1 MPa change in pressure. Heat transfer to the SGs is therefore temporarily reduced marginally with the increase in SG pressure. The shape of the predicted SG pressure trend within the first 13.5 s is driven in large part by the assumed response of the condenser. During this time period, steam discharge from the SG is primarily to the condenser. The condenser was not modeled in RELAP5, rather a condenser pressure response was assumed as described in Section 6.1. The SG pressure response does not have a significant impact on the HT system response.

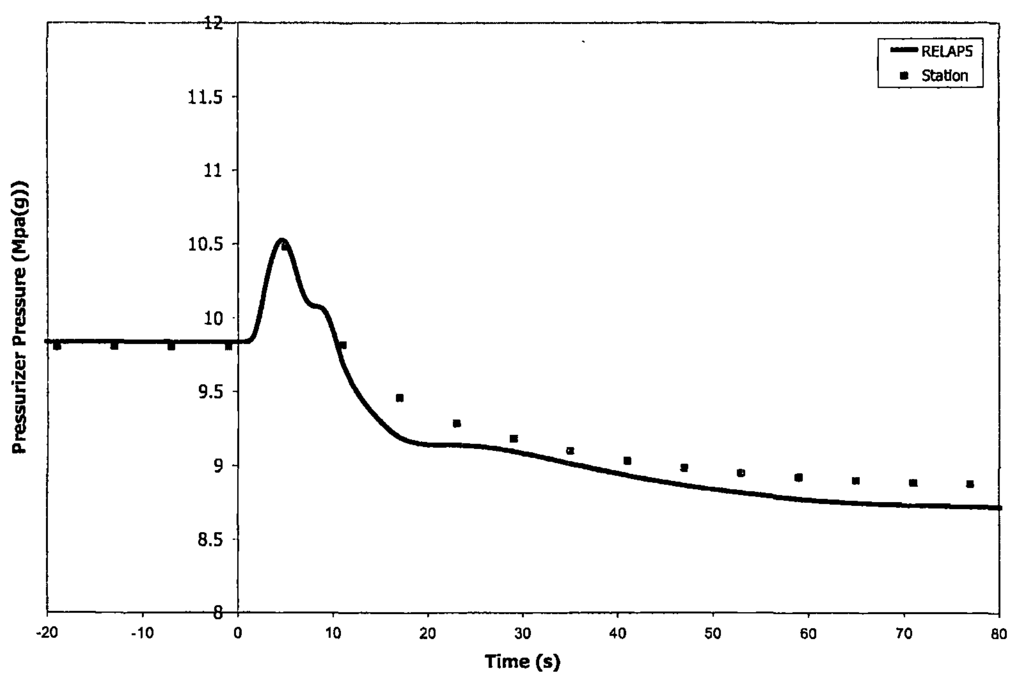

Figure 25 Darlington LOF Event – Pressurizer Pressure

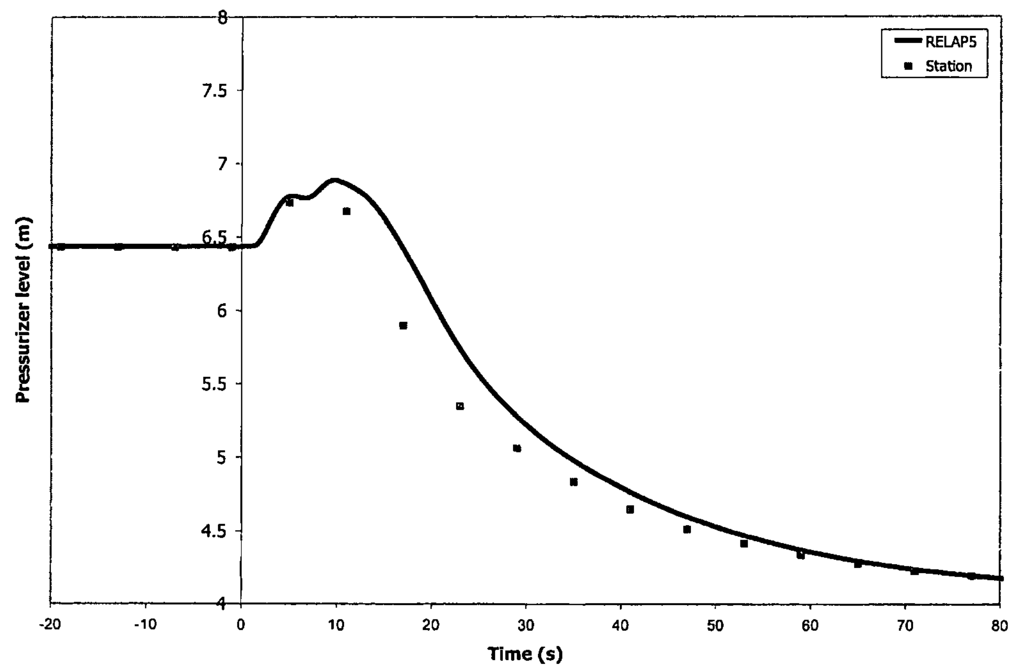

Figure 26 Darlington LOF Event – Pressurizer Level

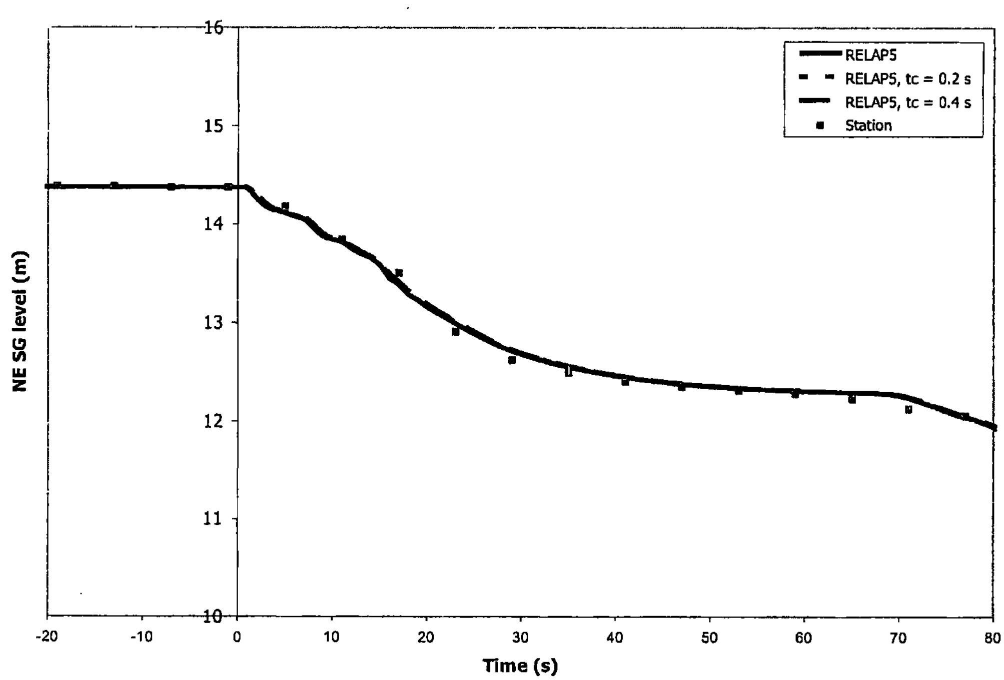

Figure 27 Darlington LOF Event – NE SG Level

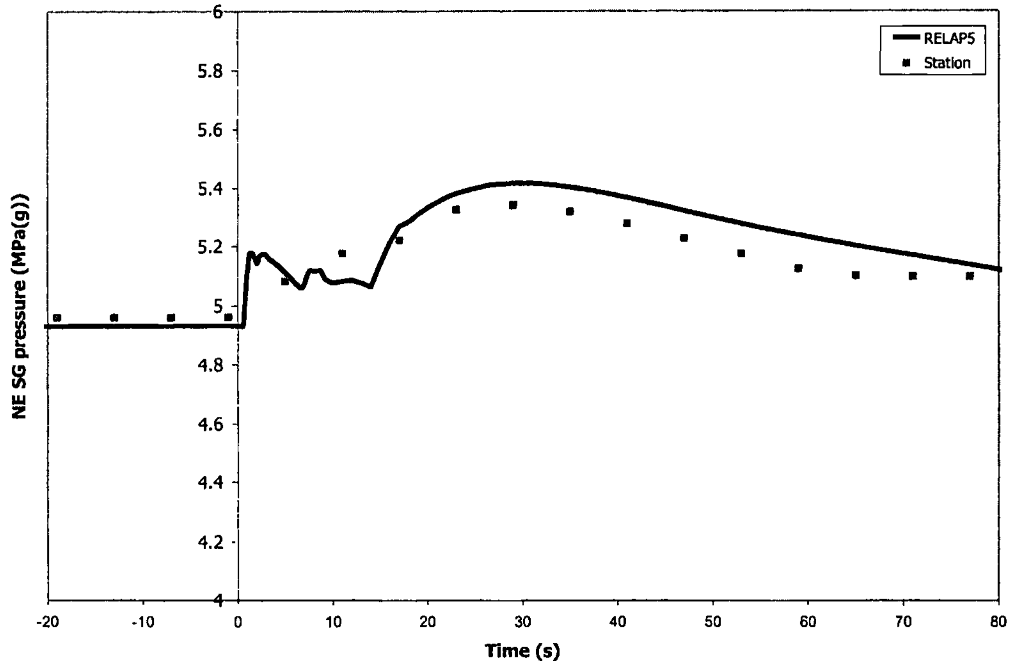

Figure 28 Darlington LOF Event – NE SG Shell Side Pressure

The response of the pressure and inventory control system is generally of secondary importance in this loss-of-flow event. One exception to this is the response of the HTS bleed valves. These valves connect the suction of the HTS pumps to the shell side of the bleed condenser. The shell side of the bleed condenser nominally operates at 1.6 MPa(g). By contrast, the pressure at the HTS pump suction is approximately 10 MPa(g) under nominal operating conditions. Therefore, the pressure drop across the bleed valves is large and the hot HTS coolant flashes, choking the flow as it discharges to the bleed condenser. Since the flow is choked, conditions in the bleed condenser shell and downstream pressure and inventory control system piping have little impact on the response of the HT system. However, the flow through the bleed lines does impact the HT system response. The bleed flows from the north and south HTS loops are shown in Figure 29 and Figure 30. The RELAP5 calculated bleed flows overpredict the measured bleed flows. Reasons for the overprediction include: use of the same controller for both bleed valves (CV3 and CV4); and the symmetry of the Darlington NGS model, in contrast with the asymmetry of station data for the measured bleed flow. This overprediction contributes to the underprediction of pressures observed in earlier figures (Figure 22 and Figure 25).

Conditions in the shell side of the bleed condenser are shown in Figure 31 and Figure 32. Consistent with the overprediction of bleed flow, the bleed condenser level and pressure are overpredicted. Because of the bleed valve choking, these conditions have no significant effect on the HTS conditions.

Overall, there is excellent agreement between RELAP5 predictions of system thermal-hydraulic conditions and station measurements for the November 25, 1993, loss-of-flow event at Darlington.

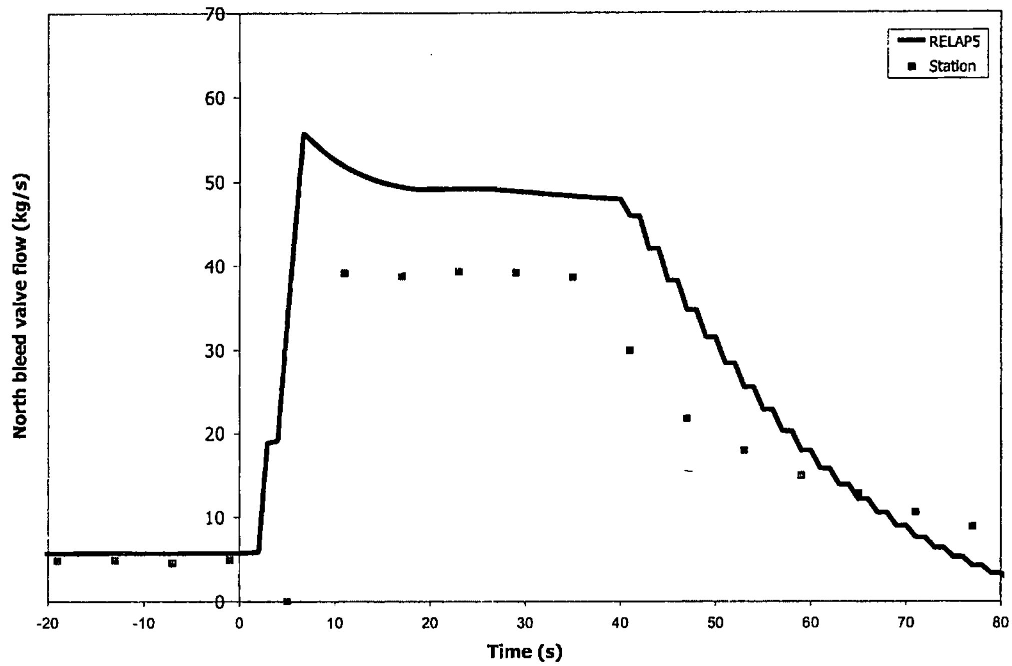

Figure 29 Darlington LOF Event – North HT Bleed Valve Flow

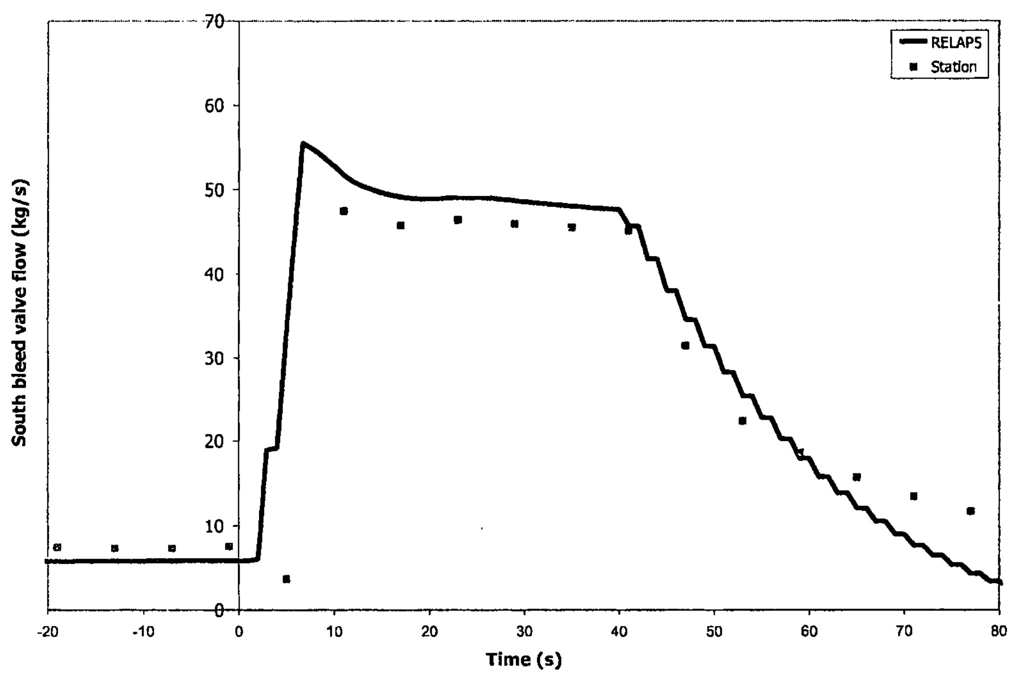

Figure 30 Darlington LOF Event – South HT Bleed Valve Flow

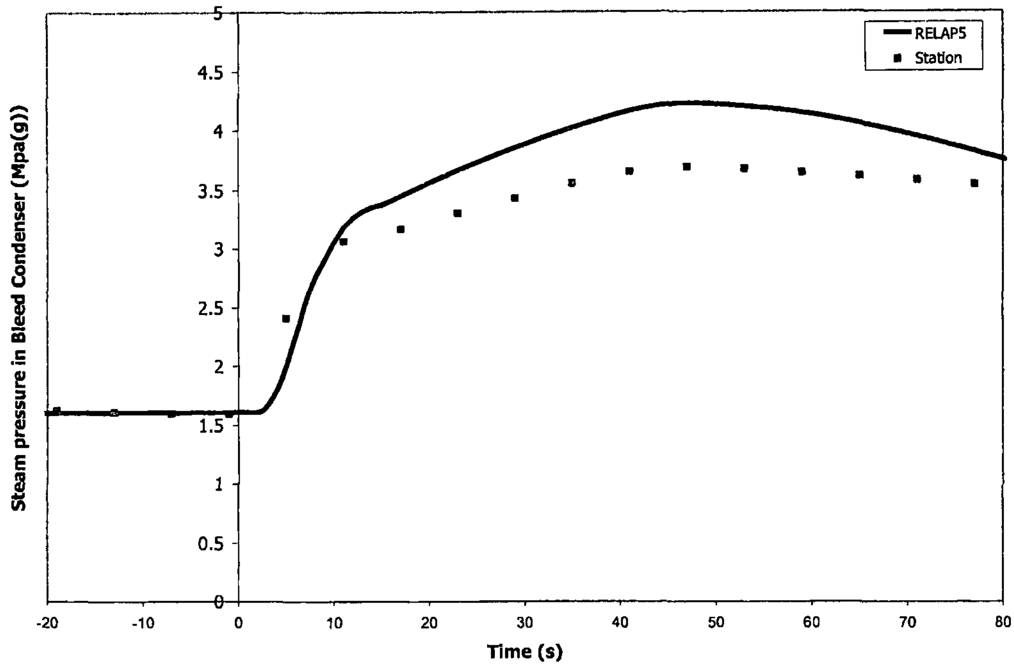

Figure 31 Darlington LOF Event – Steam Pressure in Bleed Condenser

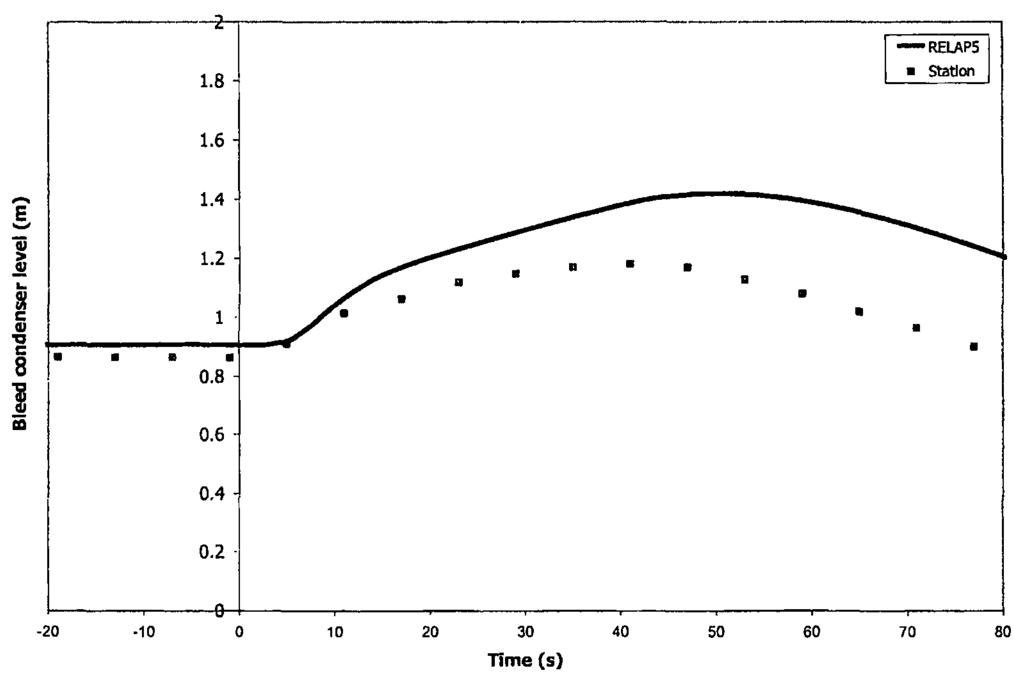

Figure 32 Darlington LOF Event – Bleed Condenser Level

6.3 Sensitivity to Header Nodalization

Sensitivity analysis was performed to assess the effect of heat transport system header nodalization on the heat transport system response. In the sensitivity case, the headers were represented by single nodes instead of the multiple node representation in the reference case. The boundary conditions for the reference and sensitivity cases were identical, with only the header nodalization changed.

The sensitivity analysis was performed on an earlier version of the simulation than that presented in Section 6.2. As such, the boundary conditions applied to both the reference and sensitivity case were not identical to those applied in the analysis documented in Section 6.2. Modeling of the flows exiting the shell side of the bleed condenser differed from the assessment in Section 6.2. Since the shell side of the bleed condenser is essentially decoupled from the heat transport system due to choking of the bleed valves, the difference has no significant impact on the heat transport system response. All other boundary conditions were identical to those used for the analysis in Section 6.2. Therefore, the predicted difference in heat transport system response between the reference and sensitivity analysis results is representative of the expected difference that header nodalization would have on the predictions presented in Section 6.2.

The results of the sensitivity and reference runs are shown in Figure 33 and Figure 34. In these figures, event initiation time is 1500 s, run D1 is the reference simulation with the multiple node headers, and run SA3 is the sensitivity simulation with the single node headers.

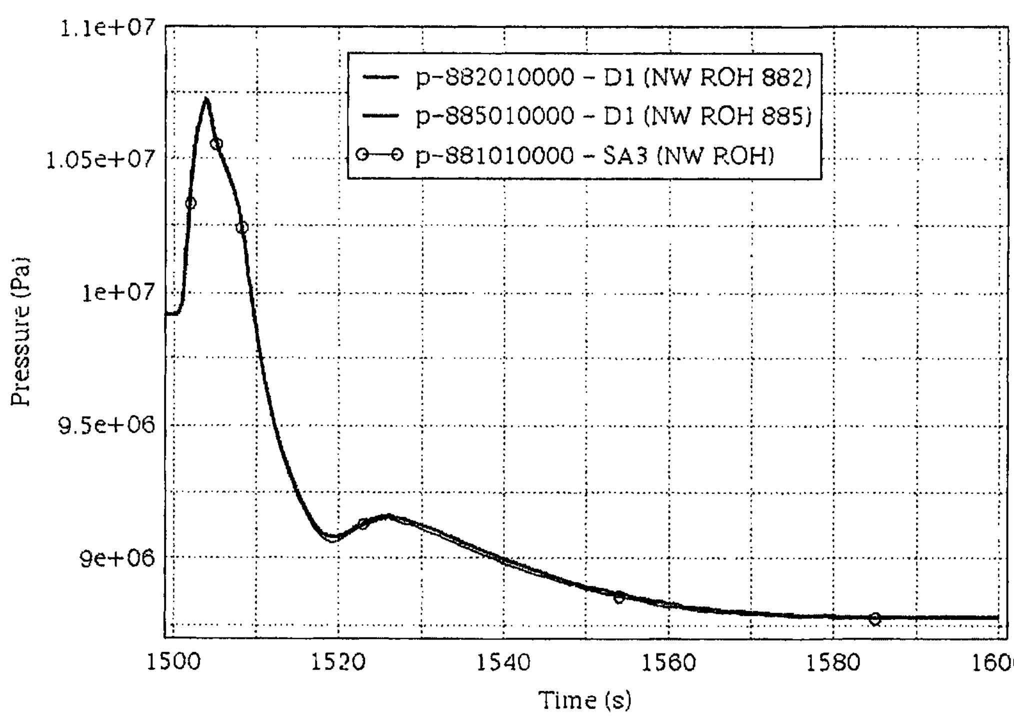

Figure 33 Sensitivity of NW ROH Pressure to Header Nodalization

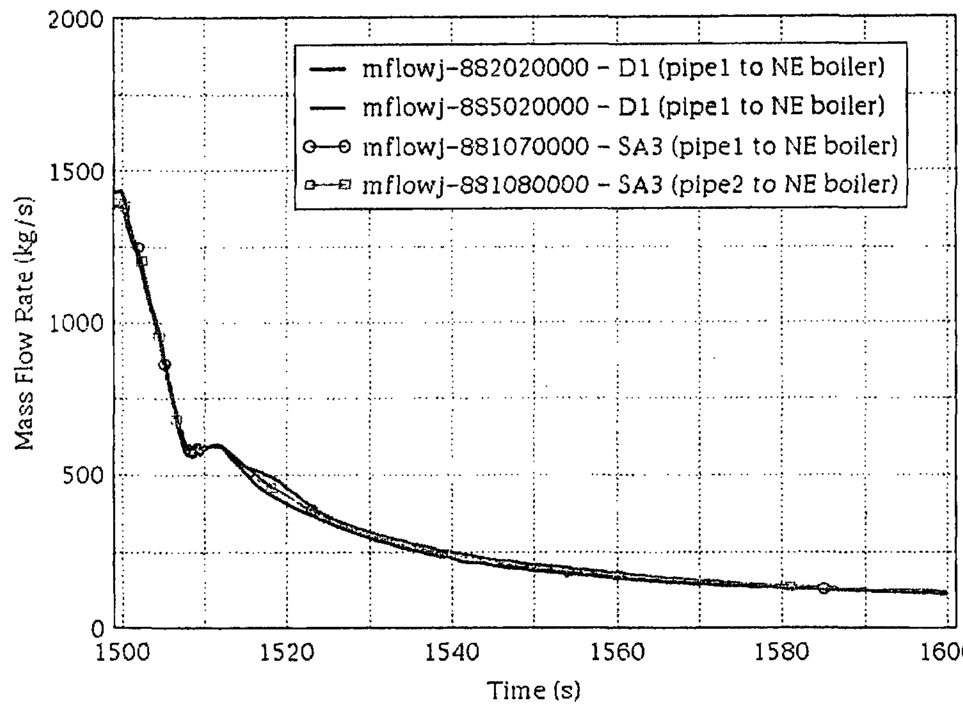

Figure 34 Sensitivity of NW ROH Outlet Mass Flow Rate to Header Nodalization

As shown in Figure 33, header nodalization has no significant effect on the predicted header pressure for the analyzed loss of flow event.

With respect to flow predictions, Figure 34 demonstrates that the results are also not significantly affected by header nodalization. Marginal differences are noticeable between approximately 0 s to 1 s, 8 s to 9 s, and 15 s to 25 s into the transient. During these times, the reference nodalization exhibits small asymmetries between flow predictions in the two parallel pipes connecting the header to the steam generator inlet. Use of a single node representation produces flow predictions that are averages of the asymmetric flows in the reference case. The differences are small and have no significant impact on the overall transient simulation results.

Therefore the loss-of-flow results are not sensitive to header nodalization.

6.4 SUMMARY OF FINDINGS

Overall excellent agreement was obtained between the RELAP5 predictions and the station measurements presented in Section 6.2. Of the three thermal-hydraulic parameters of interest to the present assessment (Section 3), the header pressures and steam generator levels compared well to station data. Assessment of core flows could not be performed due to the absence of single channel RELAP5 models. However, the overall qualitative agreement between the station measurements and the predicted system response provides confidence that core flows would likewise compare favorably if assessed.

7. SUMMARY

A detailed RELAP5 system model of one of the operating units at the Darlington Nuclear Generating Station has been developed based on reference plant data. The model includes the heat transport system, feedwater system and steam supply system (comprising the secondary side), pressure and inventory control system, and the emergency coolant injection system.

Controllers specific to Darlington have also been developed in modular fashion to allow these to be easily compiled with the base RELAP5 code. The controllers relevant to a loss-of-flow assessment have been tested against benchmark controller responses and found to be in excellent agreement.

The loss-of-flow event at Darlington on November 25, 1993, was simulated with the RELAP5 Darlington model. The simulation results are not sensitive to header nodalization. Good agreement was obtained between station measurements and the RELAP5 predictions.

NRC FORM 335 (9-2004) NRCMD 3.7 — U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
NUREG/IA-0247

2. TITLE AND SUBTITLE
RELAP5 Simulation of Darlington Nuclear Generating Station Loss of Flow Event

3. DATE REPORT PUBLISHED

MONTH	YEAR
February	2011

4. FIN OR GRANT NUMBER

5. AUTHOR(S)
Douglas Naundorf (AMEC NSS)
Jianghui Yin (AMEC NSS)
Alessandro Petruzzi (GRNSPG)
Andriy Kovtonyuk (GRNSPG)

6. TYPE OF REPORT
Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*
AMEC NSS Limited
700 University Avenue
Toronto, Ontario, Canada

Nuclear Research Group San Piero A Grado
University of Pisa
Pisa, Italy

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*
Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

10. SUPPLEMENTARY NOTES
A. Calvo, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The Darlington NGS consists of four 940 MWe CANDU reactors. A detailed RELAP5 model of a Darlington NGS reactor has been created for use with RELAP5/MOD3.3. The model includes the primary heat transport system, the steam generator feedwater and main steam supply systems, the pressure and inventory control system, and the emergency coolant injection system.

The modeling of Darlington required the creation of new subroutines for RELAP5 to describe the functionality of the process control systems and emergency control systems with sufficient accuracy. These controller subroutines are standalone modules that are compiled and linked together with the RELAP5 source to facilitate integration with future revisions to the code.

The November 25, 1993, loss-of-flow event at Darlington was simulated using this RELAP5 model. Excellent agreement was obtained between the RELAP5 predictions and station measurements.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*
Darlington NGS
CANDU reactors
RELAP5
San Piero a Grado Nuclear Research Group (GRNSPG)
University of Pisa, Italy
AMEC NSS
Ontario Power Generation
Code Application Maintenance Program (CAMP)
thermal-hydraulics codes
TRACE

13. AVAILABILITY STATEMENT
unlimited

14. SECURITY CLASSIFICATION
(This Page)
unclassified
(This Report)
unclassified

15. NUMBER OF PAGES

16. PRICE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, DC 20555-0001

OFFICIAL BUSINESS