Application of

RAMONA-III and IRT Codes

to

BWR and PWR Analysis

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RAMONA-III: A BWR TRANSIENT ANALYSIS CODE

D. J. Diamond and M. M. Levine

RAMONA-III is a new entry in BNL's library of codes for calculating operational transients and accidents in BWRs. The code was originally obtained from Scandpower A/S. It is intended for use in supplying technical assistance to the Division of Systems Safety. In February of this year work began under the sponsorship of the Division of Water Reactor Safety Research to make the code more accurate and generally useful for that purpose. The project is also charged with the task of generating plant model input data, performing sensitivity analysis, and validating the code.

The neutronics model is the "1-1/2 group," time dependent, diffusion equation in three dimensional geometry. A course mesh space discretization is used to solve for the fast flux and the thermal flux is obtained approximately from the fast flux. Albedo boundary conditions are used.

The coolant hydraulics is based on the continuity equations for vapor and liquid mass and for mixture momentum and energy. Thermodynamic equilibrium is not assummed. Subcooled boiling, slip, and steam generation and condensation are taken into account. The model allows for parallel channels in the core, leading to a riser region and then a steam dome. The first of two downcomer regions allows for feedwater flow and the second allows for an imposed pump head. The loop is completed with two lower plenum regions.

New components that have been developed at BNL are the steam separator, steamline, and jet pump with recirculation loop. The steamline includes a representation of the safety, relief, bypass, and turbine valves. The recirculation loop model includes the governing equations for the recirculation pump. The original fuel rod heat conduction model has been modified in order to account for heat capacity of the clad and non-uniform heat generation.

RAMONA has also been improved by adding the ability to calculate critical power ratio based on the GEXL boiling length-critical quality correlation and the concept of thermal margin. Parts of a BWR control and protection system have been added. These include feedwater and recirculation pump trips as well as scram trips. In addition the cross section parameterization has been changed so that the input is consistent with other codes in use at BNL.

Plant input models have been generated for the Peach Bottom-2 reactor. Calculations of turbine trip tests have been run using the original version and modified versions of the code. These calculations are being used to determine where further improvements in the code ought to be made.

RAMONA-III: A BWR TRANSIENT ANALYSIS CODE

THERMAL-HYDRAULICS MODELLING

- CONSERVATION EQUATIONS
  - MASS: VAPOR + LIQUID
  - ENERGY: MIXTURE
  - MOMENTUM: MIXTURE
- SLIP: SOLBERG CORRELATION
- PHASE CHANGE:
  - SUBCOOLED BOILING
  - CONDENSATION
- SATURATION CONDITIONS
  - NO SUPERHEAT
  - FLUID PROPERTIES
- MULTIPLE PARALLEL CORE HYDRAULIC CHANNELS
- JENS-LOTTES & COLBURN HEAT TRANSFER CORRELATIONS
- EXPLICIT INTEGRATION SCHEME
- FUEL ROD HEAT CONDUCTION
  - UNIFORM HEAT SOURCE
  - PIECEWISE LINEAR TEMPERATURE DISTRIBUTION
  - NO CLAD HEAT CAPACITY

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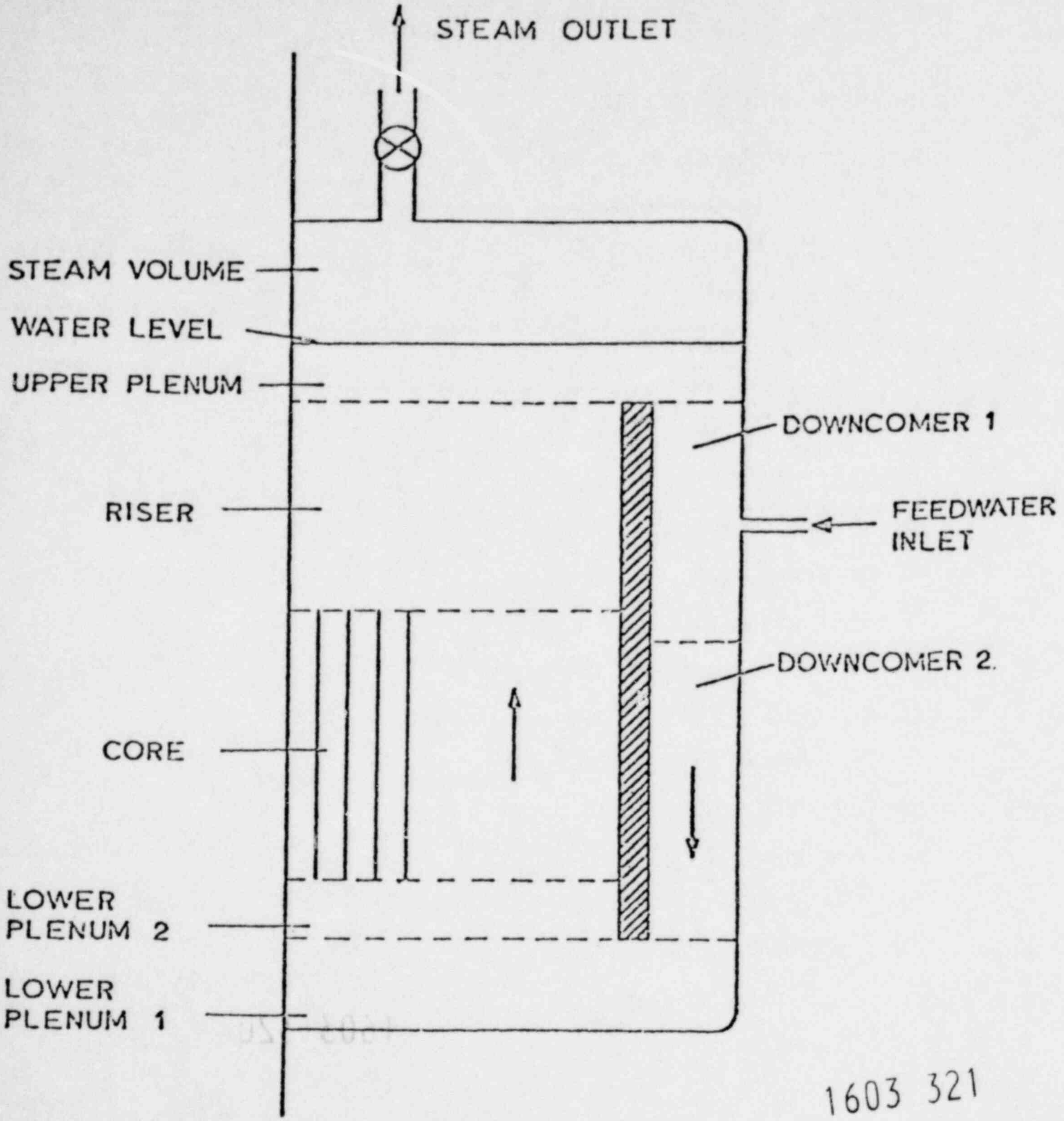

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ORIGINAL RAMONA-III MODEL

NEUTRONICS MODELLING

Diffusion Equation

Three Dimensions - Course Mesh

1-1/2 Energy Groups

- Fast Group "Exact"
- Thermal Group Approximate

Albedo Boundary Conditions

Six Delayed Neutron Precursor Groups

Decay Heat

Cross Section Parameterization

- Moderator & Fuel Temperature
- Void Fraction
- Exposure & Void History
- Control State, Xe

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NEW COMPONENT MODELLING

- Steam Separator
  - $(L/A)_{EFF} = F(x)$
  - Friction Factor
  - Carry-under
- Jet Pump/Recirculation Loop
  - Suction Flow
  - Momentum Exchange in Throat
  - Head Gain in Expanding Diffuser
  - Drive Pump
- Steamline
  - Single Phase Compressible Vapor
  - Safety, Relief, Bypass, & Turbine Valves
  - Comparison with GE Results
- Fuel Rod Heat Conduction Model
  - Clad Heat Capacity - Explicit Gap
  - Non-uniform Heat Generation

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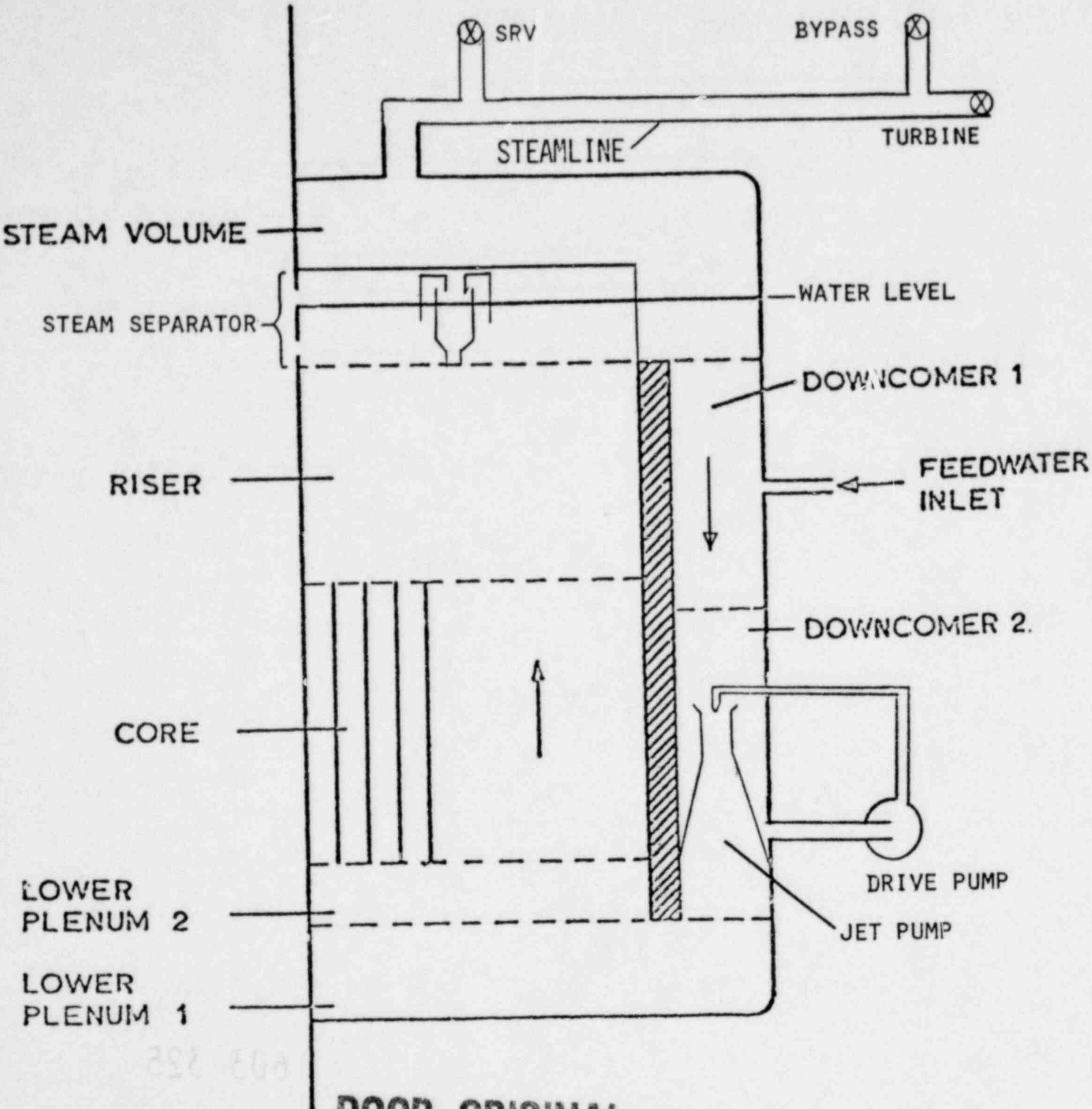

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MODIFIED RAMONA-III MODEL

New Calculational Features

- Critical Power Ratio
  - GEXL Correlation
- Cross Section Parameterization Consistent with BNL Codes
- Graphics Package
- Control and Protection System
  - Scram
    - High Power
    - High System Pressure
    - Low/High Reactor Vessel Water Level
    - Turbine Stop Valve Closure*
    - Main Steam Isolation Valve Closure*
  - Turbine Trip Signals
    - Low/High Vessel Water Level*
    - Low Turbine Inlet Pressure*
  - Recirculation Pump Trip
    - Turbine Inlet Valve Closure*
    - Low Vessel Water Level
  - Turbine Bypass Opening
    - Turbine Inlet Valve Closure*
  - Feedwater System Trip
    - High Vessel Water Level

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* Not yet implemented

PLANT MODELLING

Peach Bottom-2 End-of-Cycle 2 Turbine Trip Tests

BNL Input

ScP Input

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FUTURE WORK

- Small Break Capability
- Reactivity Edits
- Reduction to 2, 1 or 0 Dimensional Neutronics
- Additional Plant Protection and Control System
- Boron Injection
- Increased Running Speed
- Improved Correlations
- Sensitivity Studies
- Validation

IRT CODE MODIFICATION AND APPLICATION TO PWR ANALYSES

W. G. Shier and M. M. Levine

The IRT code is a fast running pressurized water reactor (PWR) systems code being modified by Brookhaven National Laboratory for the Nuclear Regulatory Commission. The code is applicable for the analysis of a number of PWR non-LOCA transients including steamline break, control rod withdrawal, loss of external load, loss of flow, loss of feedwater, and anticipated transients without scram (ATWS). In addition, the code is capable of analyzing any of these non-LOCA transients assuming concurrent steam generator tube rupture.

The IRT code solves the mass and energy equations and uses constant nodal volumes. The equation of state consists of the 1967 ASME steam tables. Since the momentum equation is not solved, main coolant flow rate is not calculated explicitly; pump volumetric flow rate is either held constant or determined from user specified values. The code models the primary system of a PWR with 17 control volumes plus a two-volume pressurizer. The primary system volumes are assumed to be homogeneous and at all the same pressure. The pressurizer is simulated with a non-equilibrium model that allows for various combinations of saturated, subcooled, and superheated conditions. Reactor coolant system pressure is determined based on the calculated pressurizer pressure and the pressure drop across the surge line.

Recent modifications to the IRT code have included the addition of a once-through steam generator model for the simulation of B&W reactors. The model represents the steam generator primary and secondary sides by twelve volumes each; the steam generator downcomer is represented by a homogeneous equilibrium model. Five modes of heat transfer are available to calculate the primary to secondary heat transfer. This model has been used to simulate the initial phase of the Three Mile Island event and the results agree reasonably well with the data.

Other modifications that have been implemented in the IRT code include the following:

- Reactor upper head region representation
- Metal heat capacity modelling
- Pressurizer level swell model
- Steam generator tube rupture model
- Decay heat for finite reactor operation
- Accumulator modelling

In addition to the analysis of the TMI event, the IRT code has also been used to analyze the effects of several plant modifications proposed by B&W to mitigate the effects of a TMI type event and to analyze an overfeed transient for a typical B&W plant design. For plant transient analysis of reactors with a U-tube steam generator design, IRT has recently been used to analyze steamline break (with concurrent tube rupture) transients, loss of flow transients and turbine trip transients. Future work includes the analysis of a number of operational transients (assuming concurrent tube rupture) for reactor designs with the once-through and U-tube steam generator designs.

IRT CODE MODIFICATION AND
APPLICATION TO PWR TRANSIENT ANALYSIS

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IRT CODE FEATURES

- PLANT TRANSIENT CODE FOR ANALYSIS OF PWRs
- ORIGINATED FROM COMBUSTION ENGINEERING CESEC CODE
- APPLICABLE TO MANY NON-LOCA PLANT TRANSIENT CALCULATIONS
- 17 FIXED PRIMARY SYSTEM NODES
- ONCE-THROUGH AND U-TUBE STEAM GENERATOR MODELLING

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IRT ASSUMPTIONS

- CONSERVES MASS AND ENERGY

- PUMP FLOW RATE USER SPECIFIED
  (NO CONSERVATION OF MOMENTUM)

- HOMOGENEOUS EQUILIBRIUM MODEL FOR
  PRIMARY SYSTEM

- NON-EQUILIBRIUM PRESSURIZER MODEL

- UNIFORM PRIMARY SYSTEM PRESSURE

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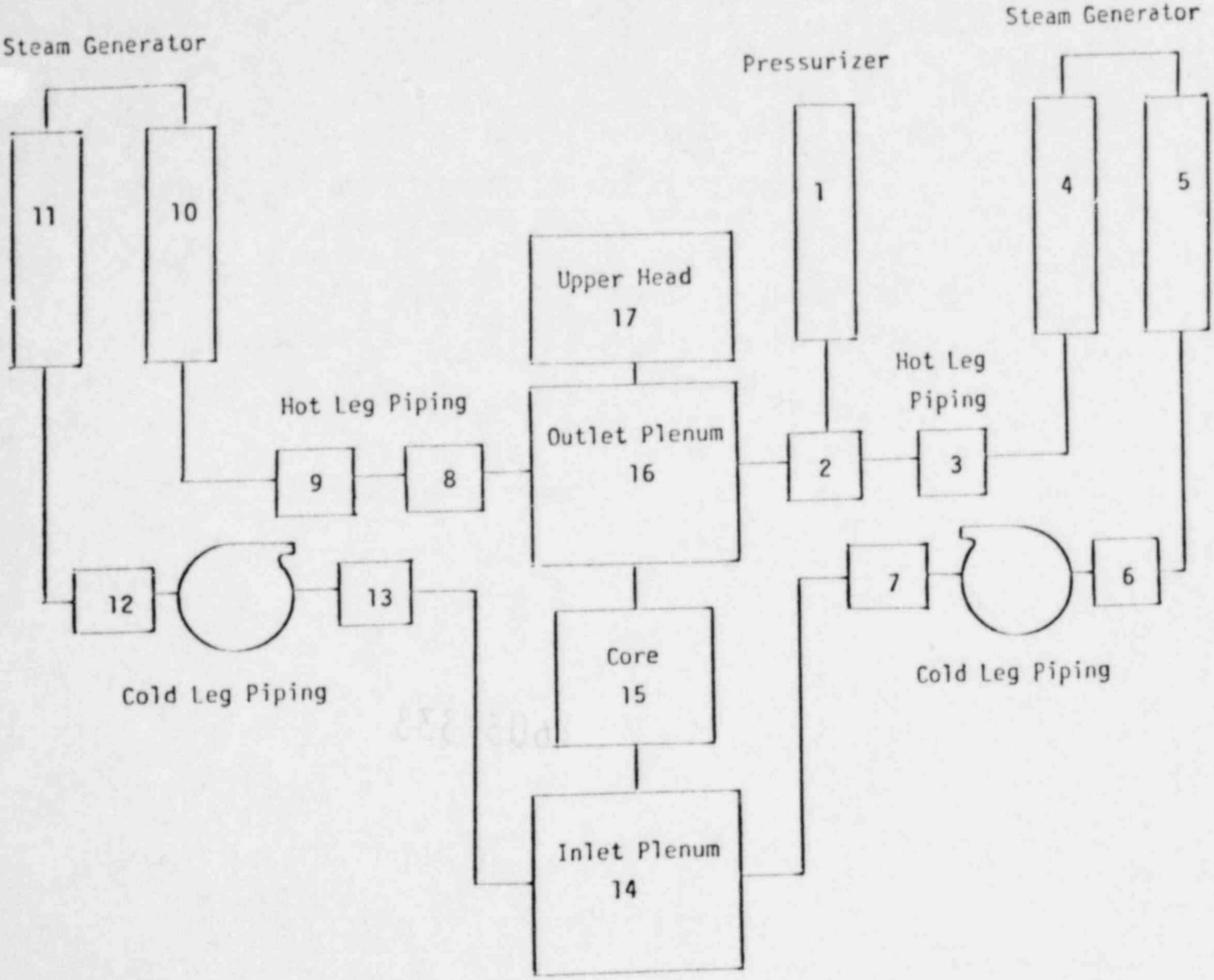
IRT PRIMARY SYSTEM NODALIZATION



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PRESSURIZER MODELLING ASSUMPTIONS

- Complete Phase Separation
  - Homogeneous Phases
  - Steam: Saturated or Superheated
  - Water: Saturated or subcooled
- Mass and Energy Transport Across Interface
  - Instantaneous Bubble Rise or Droplet Fall
  - Bulk Boiling or Bulk Condensation
  - Interface Surface Evaporation and Condensation Neglected

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PRESSURIZER MODES OF OPERATION

Steam Region	Water Region
Superheated	Subcooled
Superheated	Saturated
Saturated	Subcooled
Saturated	Saturated
Superheated	Empty
Saturated	Empty
Empty	Subcooled
Empty	Saturated

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ADDITIONAL IRT FEATURES

- Metal Heat Capacity
- Plan. Protection System
- Safety Injection System
- Accumulator Model
- Steam Generator Tube Rupture Model
- DNB Calculations
- Two Phase Core Heat Transfer
- Decay Heat for Finite Reactor Operation
- Pressurizer Level Swell Model

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IRT U-TUBE STEAM GENERATOR SECONDARY SIDE MODEL

- FOUR FIXED NODES

  DOWNCOMER (SEPARATED LIQUID AND VAPOR)

  TUBE REGION (VAPOR LIQUID MIXTURE)

  RISER (VAPOR LIQUID MIXTURE)

  STEAM DOME (ALL VAPOR)

- ALL REGIONS AT SATURATED CONDITIONS

- ADDITIONAL SYSTEMS MODELLED

  STEAM DUMP

  STEAM BYPASS

  STEAM GENERATOR RELIEF VALVES

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IRT ONCE-THROUGH STEAM GENERATOR MODEL

MOD-1

- TUBE REGION
  - 12 FIXED NODES
  - CONSERVATION OF MASS AND ENERGY
  - HOMOGENEOUS EQUILIBRIUM MODEL
- DOWNCOMER
  - SINGLE NODE
  - HOMOGENEOUS EQUILIBRIUM MODEL
- AUXILIARY FEEDWATER
  - DISTRIBUTED OVER SUPERHEATED NODES IN TUBE REGION
- STEAM GENERATOR TUBE RUPTURE MODEL
- UNIFORM SECONDARY SIDE PRESSURE

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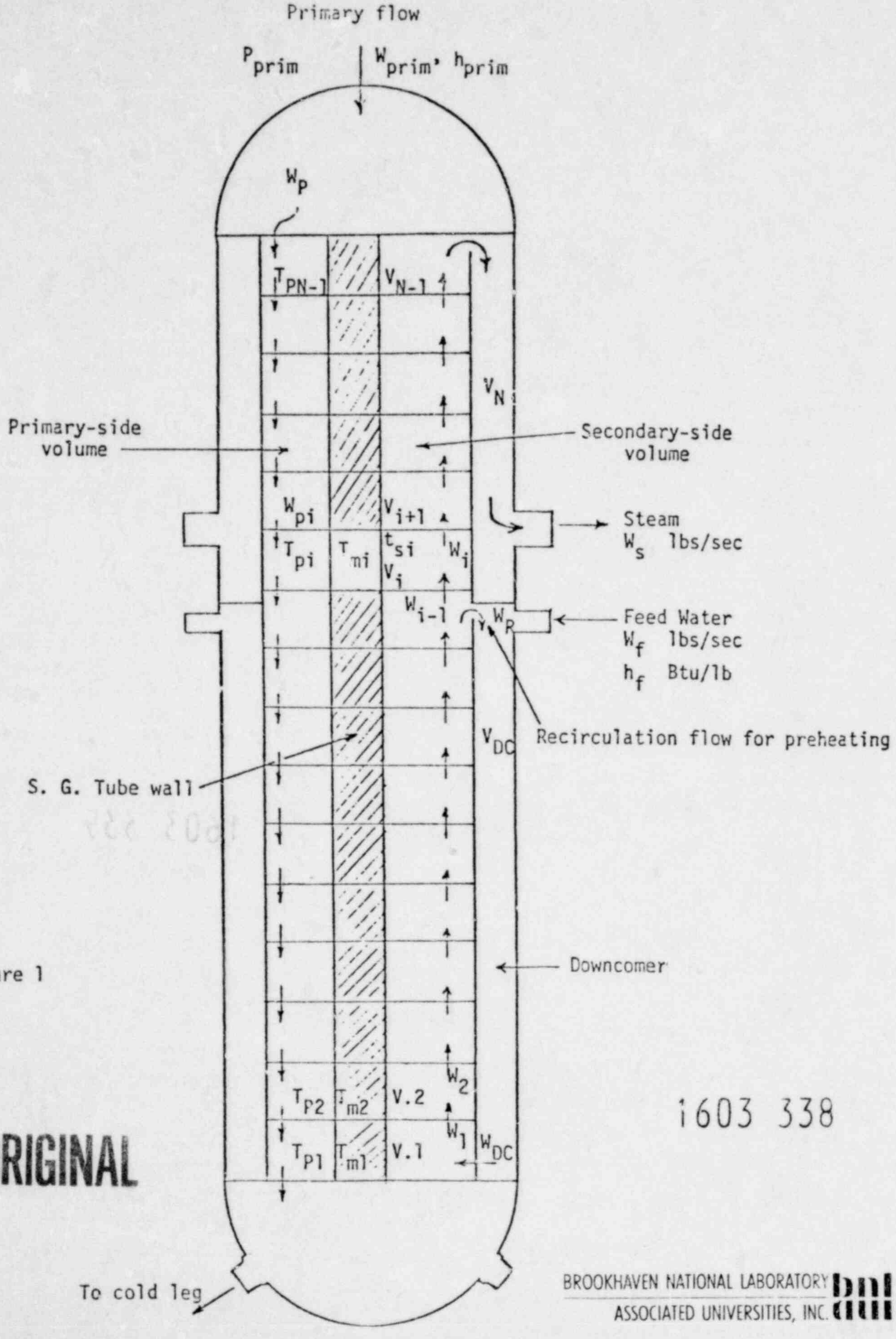

Figure 1

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MOD-2

TUBE REGION

FOUR REGIONS (TWO MOVING BOUNDARIES)

TWO FIXED BOUNDARIES AT LOCATION OF TUBE RUPTURE AND ASPIRATOR FLOW

MOVING BOUNDARIES DIVIDE LIQUID AND VAPOR REGIONS FROM TWO PHASE REGION

HOMOGENEOUS EQUILIBRIUM MODEL

SAME DOWNCOMER MODEL AS MOD-1

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TYPICAL TRANSIENTS FOR B&W DESIGN

- TMI LOSS OF FEEDWATER EVENT
- STEAM GENERATOR OVERFEED TRANSIENT
- EVALUATION OF PROPOSED B&W DESIGN CHANGES
- OPERATIONAL TRANSIENTS WITH CONCURRENT STEAM GENERATOR TUBE RUPTURE

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TYPICAL TRANSIENTS FOR U-TUBE STEAM GENERATOR PLANT DESIGN

- STEAMLINE BREAK
- LOAD REJECTION
- CONTROL ROD WITHDRAWAL
- LOSS OF FLOW
- FEEDLINE BREAK
- ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)
- STEAM GENERATOR TUBE RUPTURE
- OPERATIONAL TRANSIENTS WITH CONCURRENT STEAM GENERATOR TUBE RUPTURE

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TYPICAL RESULTS

- ONCE-THROUGH STEAM GENERATOR MODEL
  TMI LOSS OF FEEDWATER EVENT

- U-TUBE STEAM GENERATOR MODEL

  STEAMLINE BREAK

  FULL POWER ROD WITHDRAWAL

  LOSS OF MAIN COOLANT FLOW

  LOSS OF EXTERNAL LOAD

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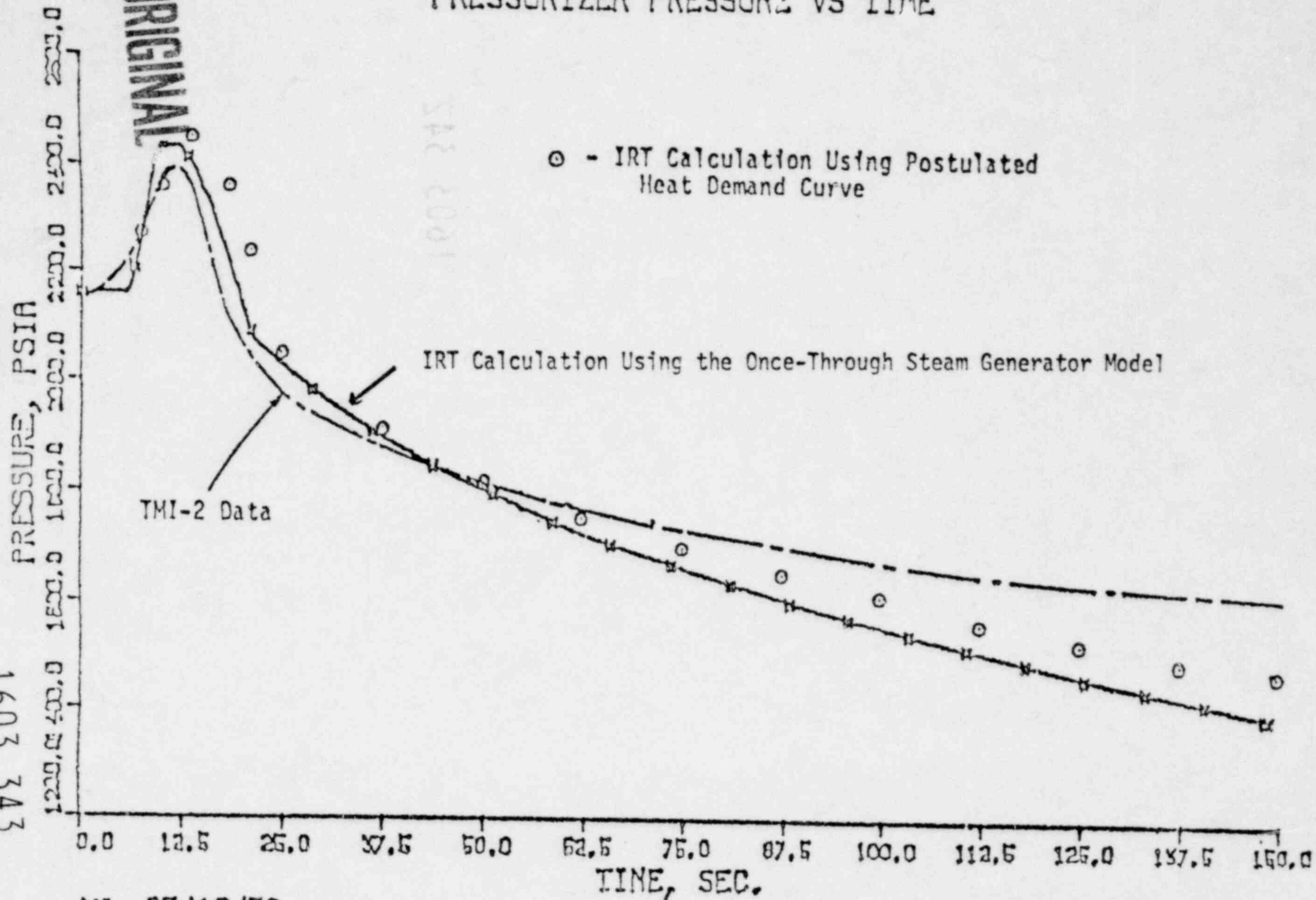
FIGURE 1: TMI LOSS OF FEEDWATER TRANSIENT

PRESSURIZER PRESSURE VS TIME



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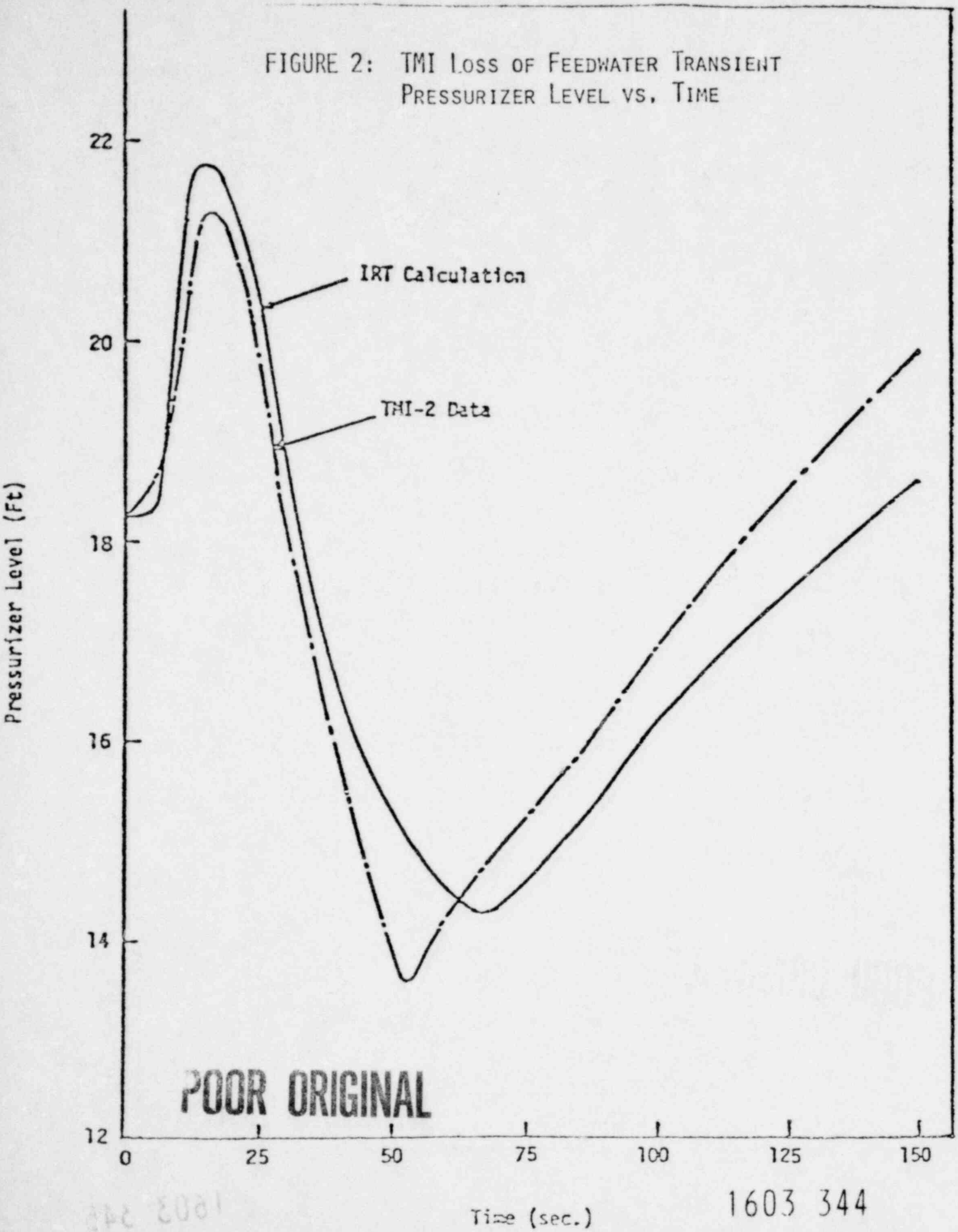

FIGURE 2: TMI LOSS OF FEEDWATER TRANSIENT
PRESSURIZER LEVEL VS. TIME

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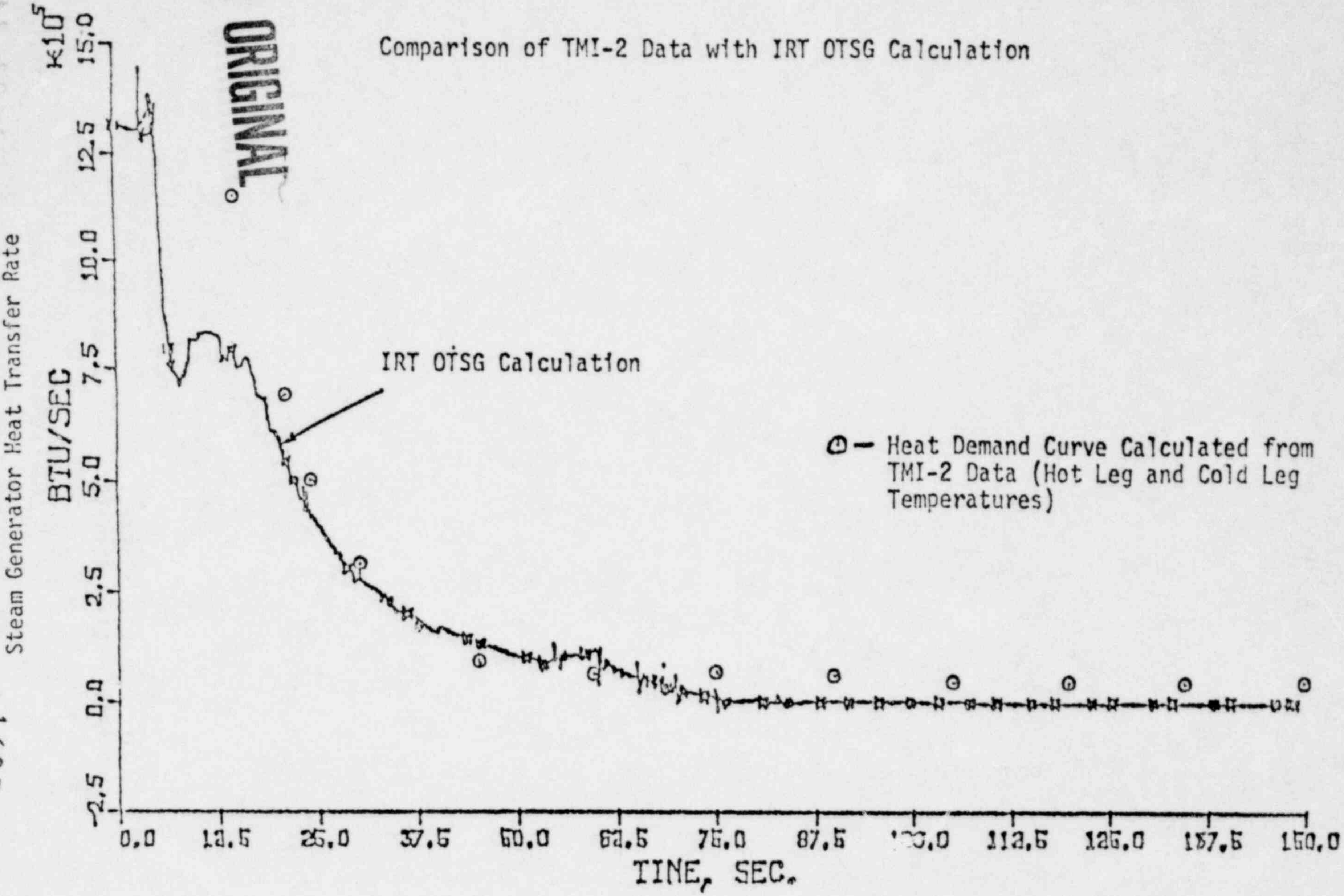
FIGURE 3: TMI LOSS OF FEEDWATER TRANSIENT

Comparison of TMI-2 Data with IRT OTSG Calculation



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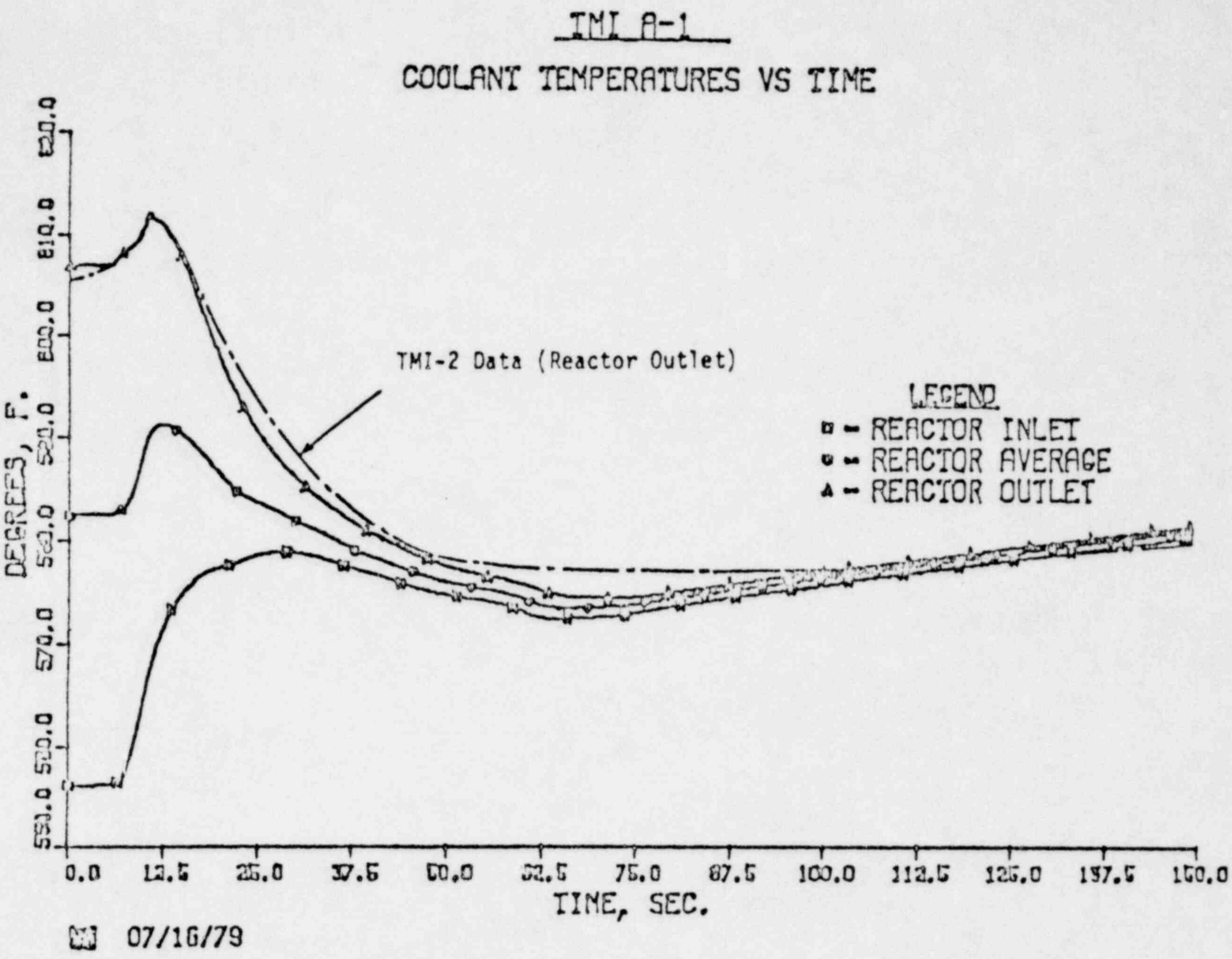

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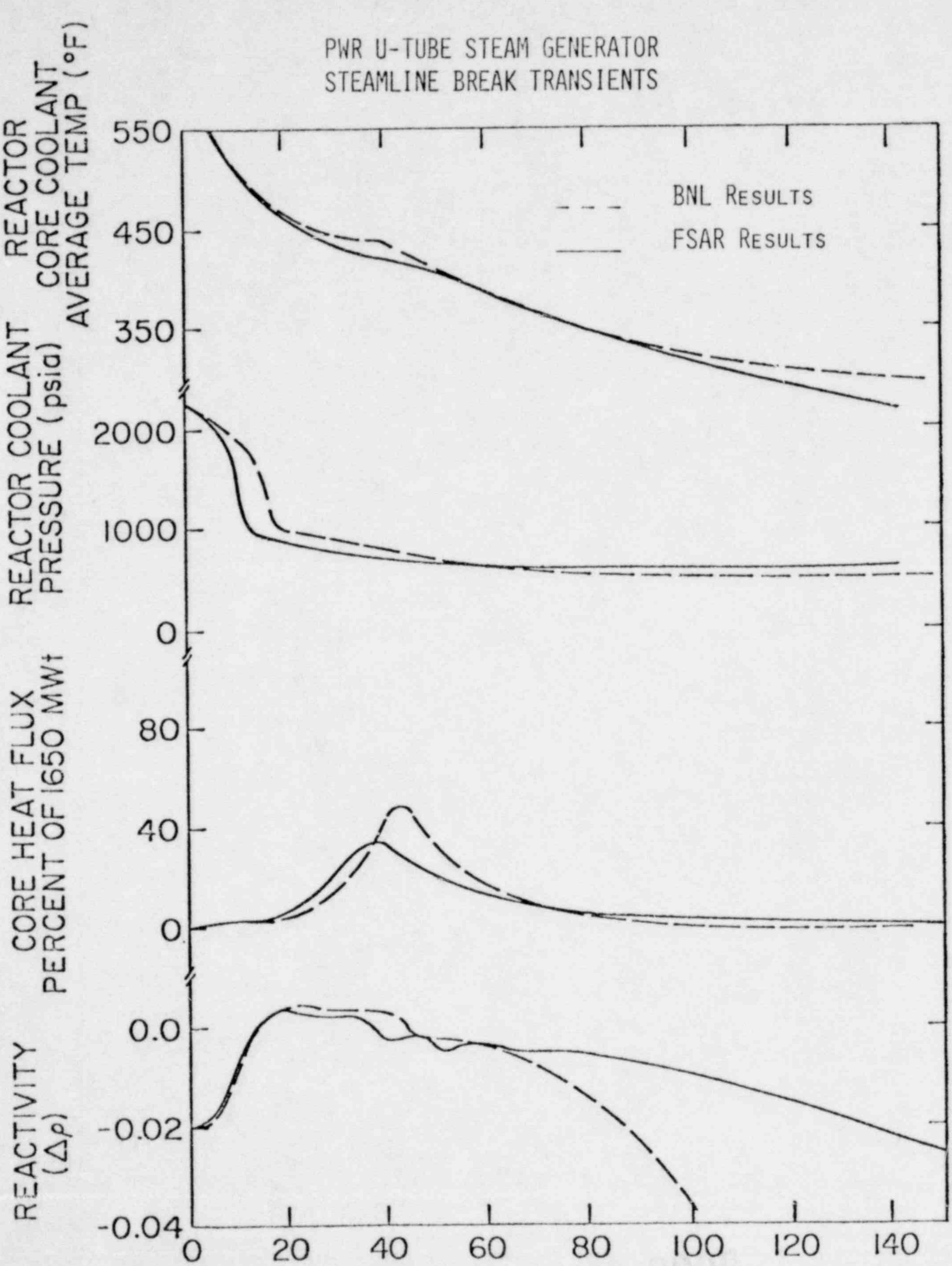
PWR U-TUBE STEAM GENERATOR
STEAMLINE BREAK TRANSIENTS



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COMPARISON OF IRT RESULTS WITH AVAILABLE FSAR RESULTS

	IRT RESULTS	FSAR RESULTS
FULL POWER ROD WITHDRAWAL TRANSIENT		
PEAK REACTOR POWER	109.3%	110. %
TIME OF PEAK REACTOR POWER	37 SECS	38 SECS
PEAK PRESSURIZER PRESSURE	1.6347×10^7 PASCALS	1.6207×10^7 PASCALS
TIME OF PEAK PRESSURIZER PRESSURE	43 SECS	42 SECS
LOSS OF MAIN COOLANT FLOW		
1. FOUR PUMP LOSS OF FLOW		
MINIMUM DNBR	1.55	1.55
TIME OF MINIMUM DNBR	10.1 SECS	10.5 SECS
2. ONE PUMP LOSS OF FLOW		
MINIMUM DNBR	1.52	1.45
TIME OF MINIMUM DNBR	4.9 SECS	5.8 SECS

COMPARISON OF IRT RESULTS WITH AVAILABLE FSAR RESULTS

	IRT RESULTS	FSAR RESULTS
LOSS OF EXTERNAL LOAD		
HIGH PRESSURE SCRAM SETPOINT	1.6694×10^7 PASCALS	1.6694×10^7 PASCALS
TIME OF SCRAM	8.7 SECS	7.0 SECS
PEAK REACTOR POWER	104.7%	106.0%
TIME OF PEAK POWER	8.8 SECS	8.0 SECS
PEAK PRESSURIZER PRESSURE	1.74458×10^7 PASCALS	1.7232×10^7 PASCALS
TIME OF PEAK PRESSURIZER PRESSURE	11.0 SECS	9.1 SECS
PEAK STEAM GENERATOR PRESSURE	$.71582 \times 10^7$ PASCALS	$.71479 \times 10^7$ PASCALS
TIME OF PEAK STEAM GENERATOR PRESSURE	15. SECS	12. SECS