~~Impact of Pressure Relief Holes on Core Coolability for a PWR During a Large-Break Loss of Coolant Accident With Core Blockage Using RELAP5-3D~~

Effect of LOCA Holes on PWR Core Coolability

Timothy Crook
Department of Nuclear Engineering
Texas A&M University

2014 IRUG Meeting
12 September 2014
Idaho Falls, ID, USA

TEXAS A&M UNIVERSITY

Presentation Overview

Introduction

Input Model Description

- Model without LOCA Holes
- Model with LOCA Holes
- Blockage Simulation

Simulation Results

- Simulation Approach
- Results without LOCA Holes
- Impact of LOCA Holes

Conclusions

Effect of LOCA Holes on PWR Core Coolability

INTRODUCTION

Introduction: Background

GSI-19

Conc LOCA

When t ation can bec

Debr ance

Debr enter the

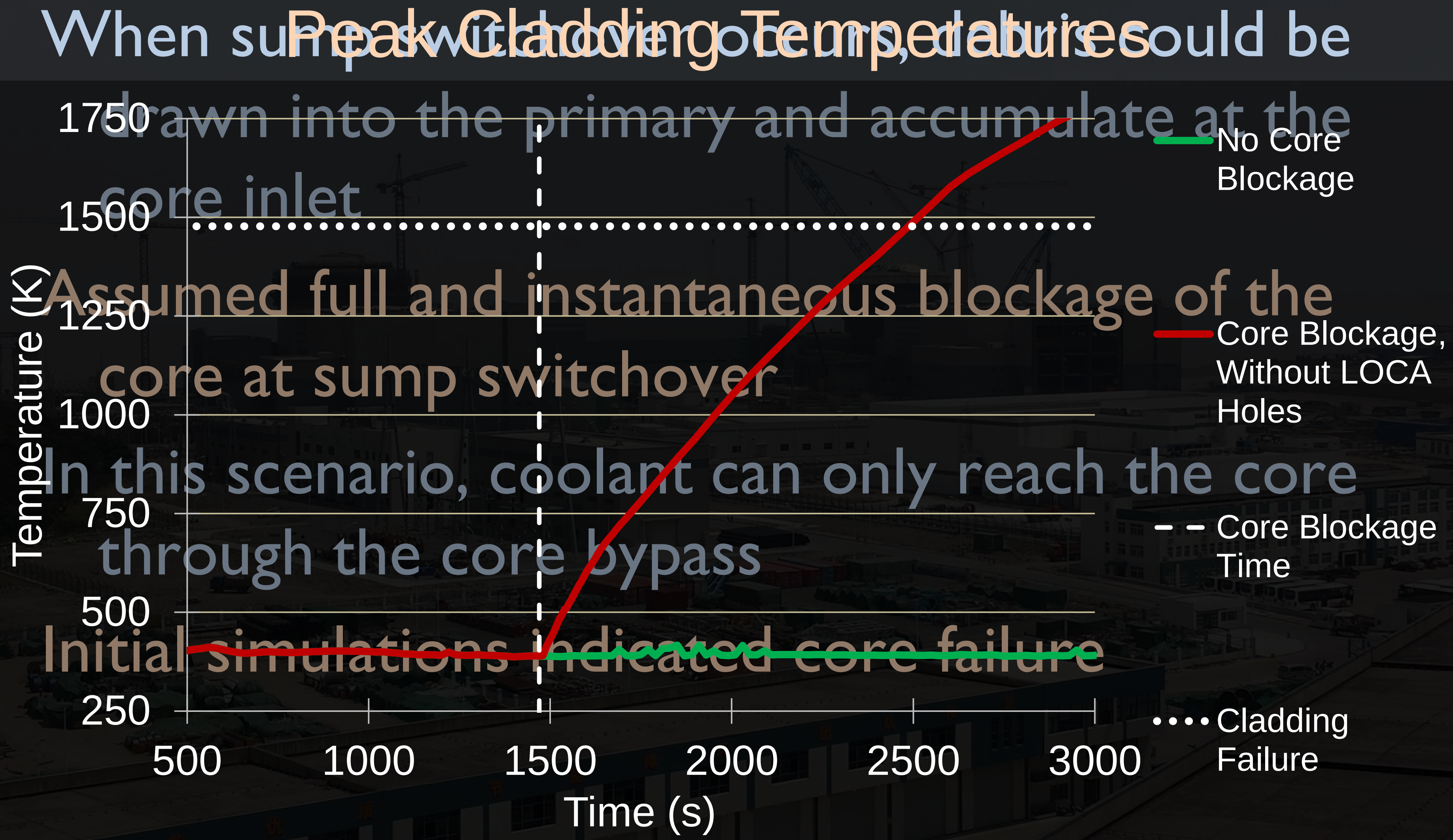
ECCS F

Cool y

Then pumps draw water from the containment sump

Introduction: Study Motivation

- When sump switchover occurs, debris could be drawn into the primary and accumulate at the core inlet
- Assumed full and instantaneous blockage of the core at sump switchover
- In this scenario, coolant can only reach the core through the core bypass
- Initial simulations indicated core failure



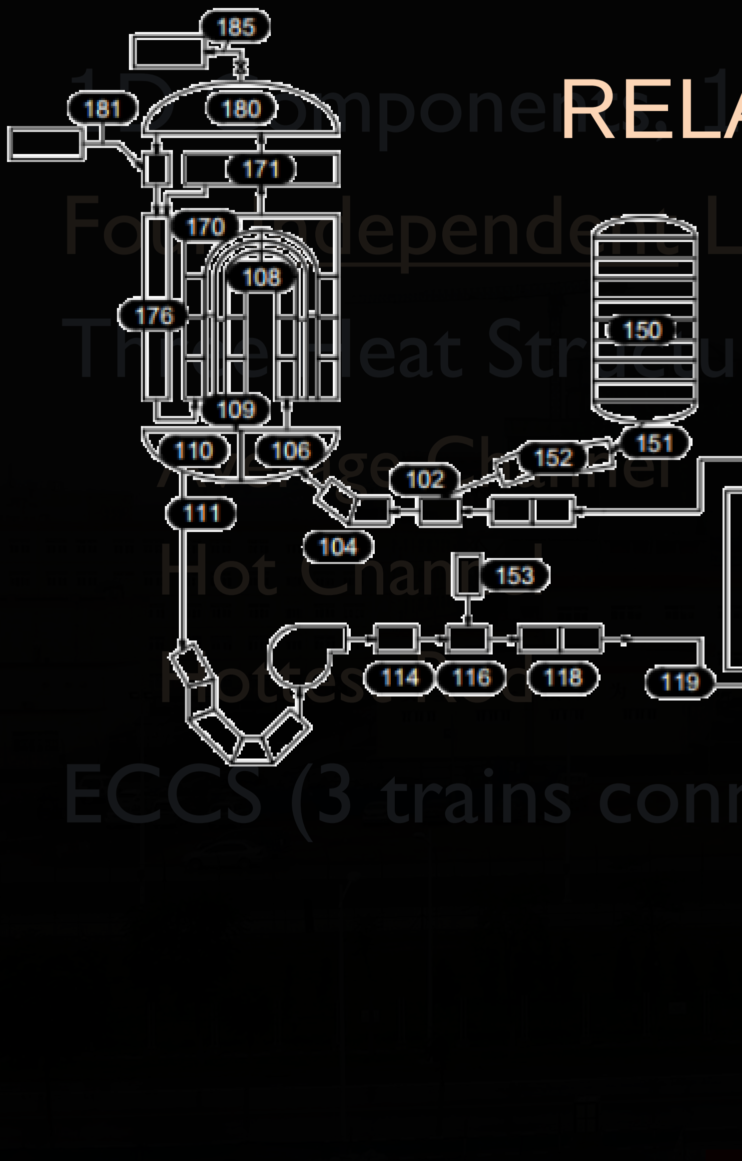
Effect of LOCA Holes on PWR Core Coolability

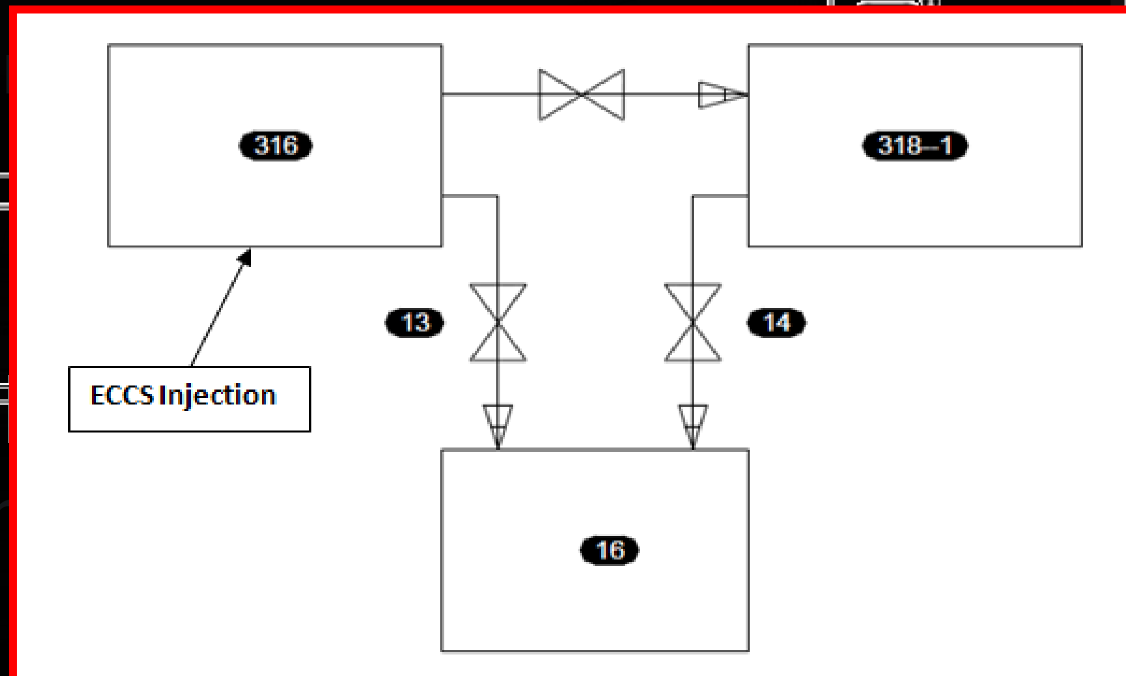
INPUT MODEL DESCRIPTION

TEXAS A&M UNIVERSITY

Model without LOCA Holes

RELAP5-3D Model



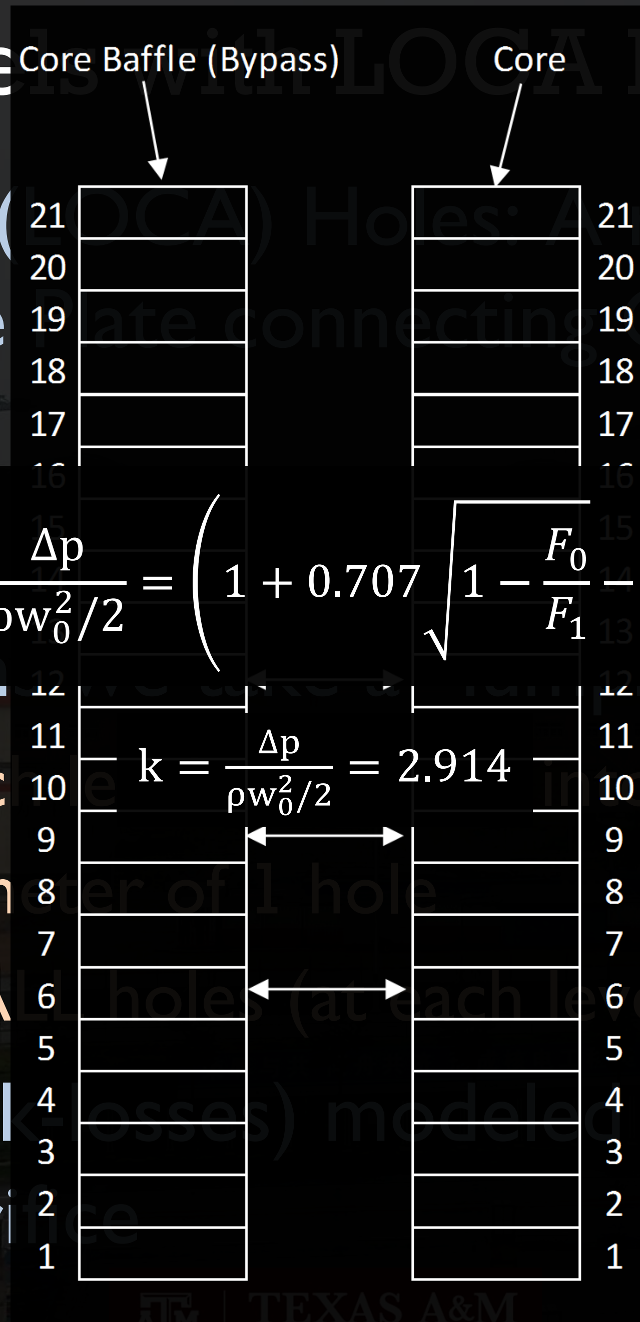


DEG Break Model

Models with LOCA Holes

- Pressure Relief (LOCA) Holes: A ring of holes through Baffle Plate connecting Core and Core Bypass
- 1-3 Levels
- 1D Model means we take a "lumping" approach
  - All holes at each le… into 1 hole
  - Hydraulic Diameter of 1 hole
  - Flow Area of ALL holes (at each level)
- Friction losses (k-losses) modeled as flow through an orifice

$$k = \frac{\Delta p}{\rho w_0^2/2} = \left(1 + 0.707\sqrt{1 - \frac{F_0}{F_1}} - \frac{F_0}{F_1}\right)^2$$ [7]



Blockage Simulation

Full and instantaneous blockage of core inlet at sump switchover

Simulated by increasing the forward k-loss at core inlet to prevent flow (to 1.0E6)

Bypass was left free (unblocked)

Effect of LOCA Holes on PWR Core Coolability

SIMULATION RESULTS

Simulation Approach

8 total simulations

Parameters of Interest:

- Peak Cladding Temperature (1478 K limit)
- Core Collapsed Liquid Level
- Bypass Flow

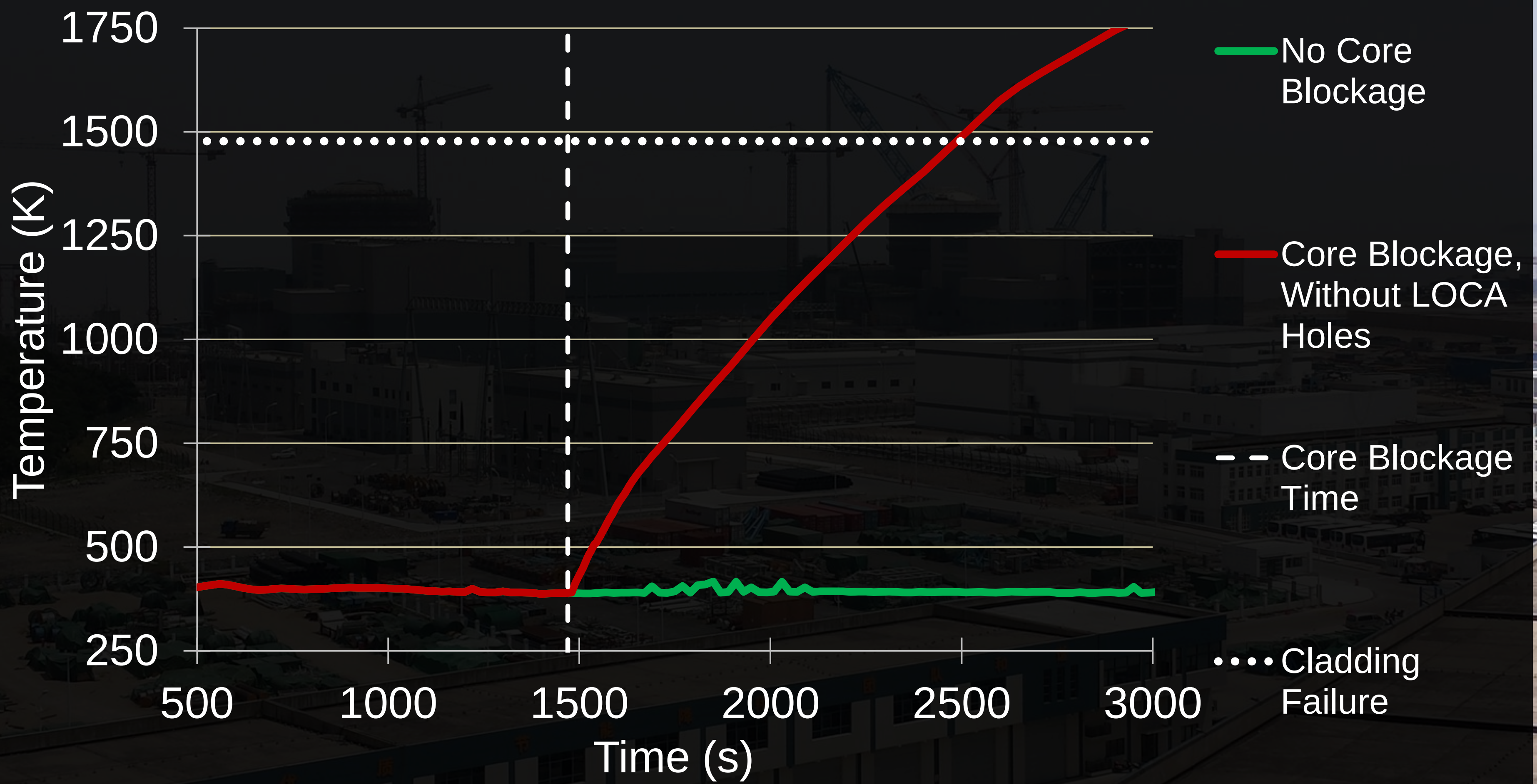
Simulation Approach

(-300 to 0 seconds) → (0 to 1470 seconds) → (1470 to 5000+ seconds)

	Simulation Phase		
	Steady-State	Safety Injection	Long-Term Cooling
Without LOCA Holes	Null Transient	DEG CL Transient	Core Blockage
			No Core Blockage
With 1 LOCA Hole			Core Blockage
			No Core Blockage
With 2 LOCA Holes			Core Blockage
			No Core Blockage
With 3 LOCA Holes			Core Blockage
			No Core Blockage

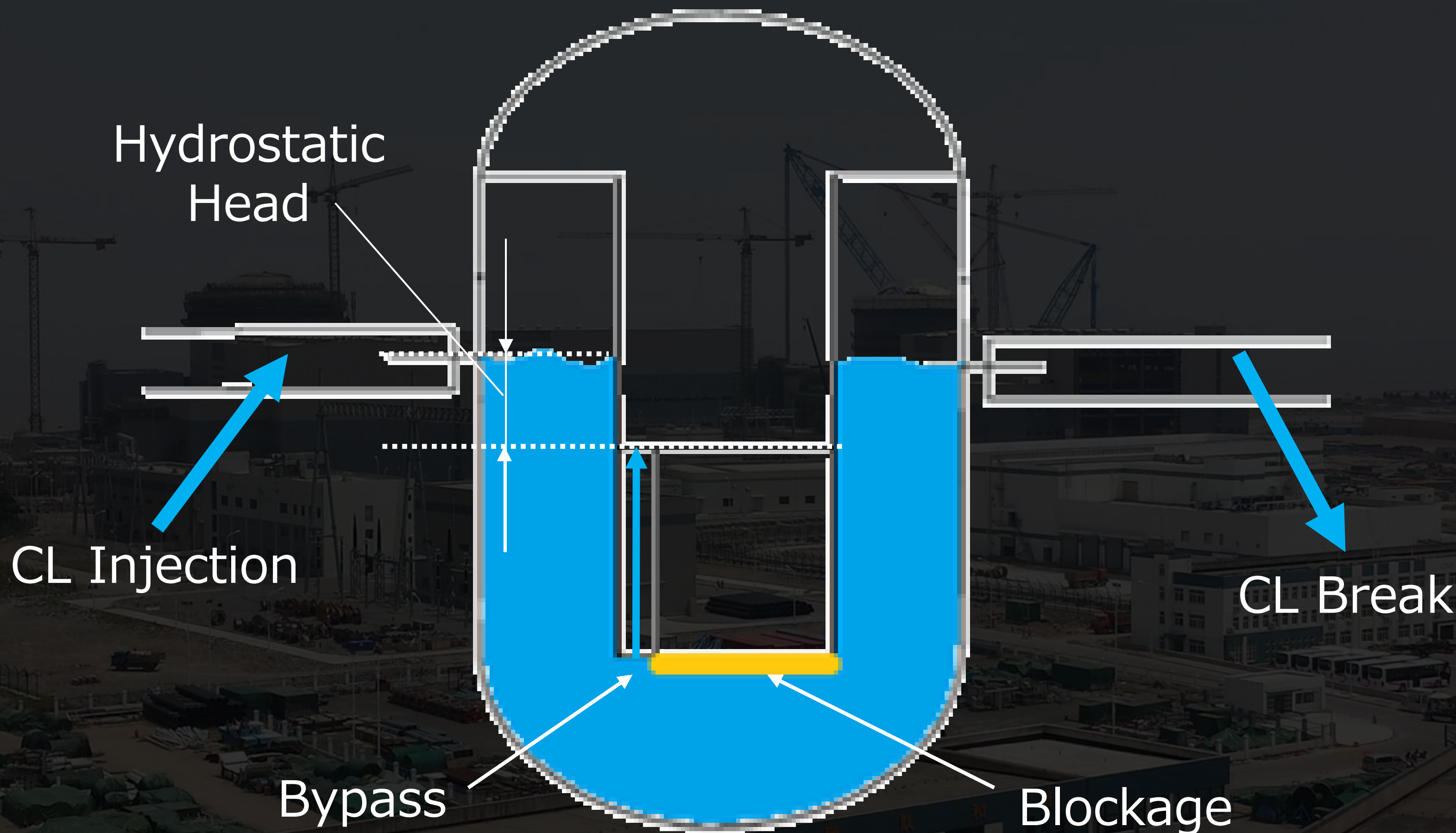
Results without LOCA Holes

Peak Cladding Temperatures



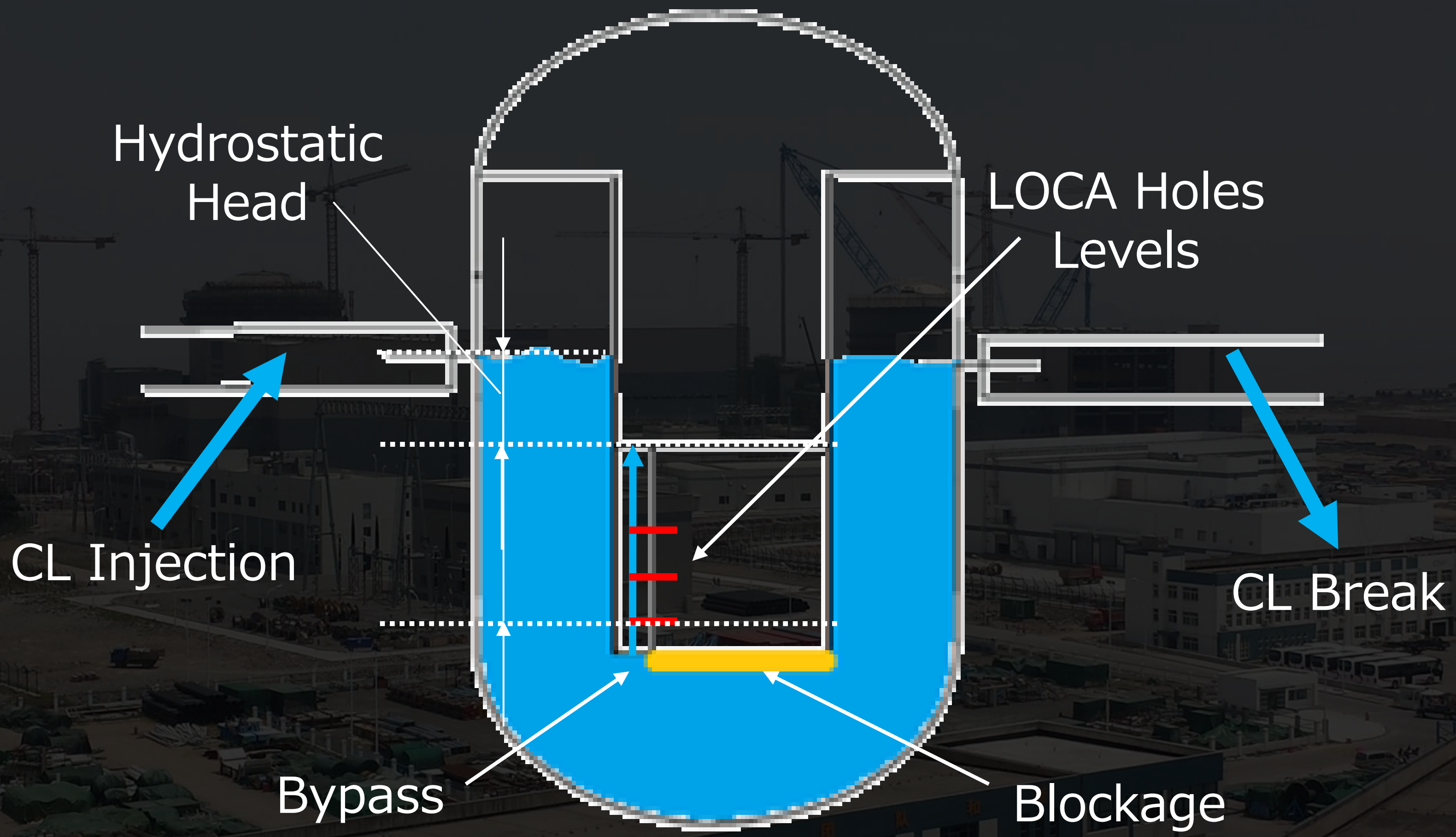
TEXAS A&M UNIVERSITY

Results without LOCA Holes



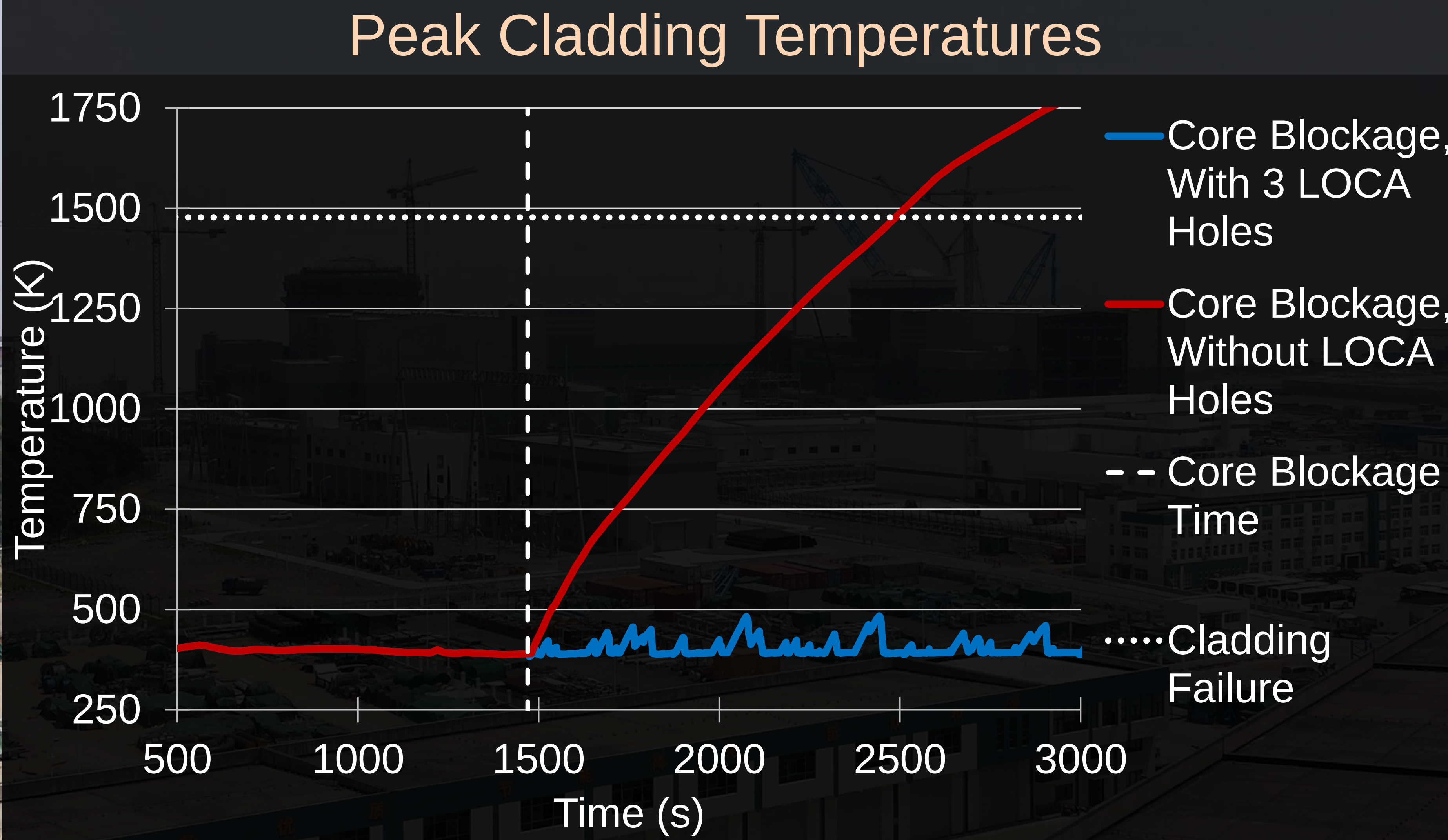
[8]

Results with LOCA Holes



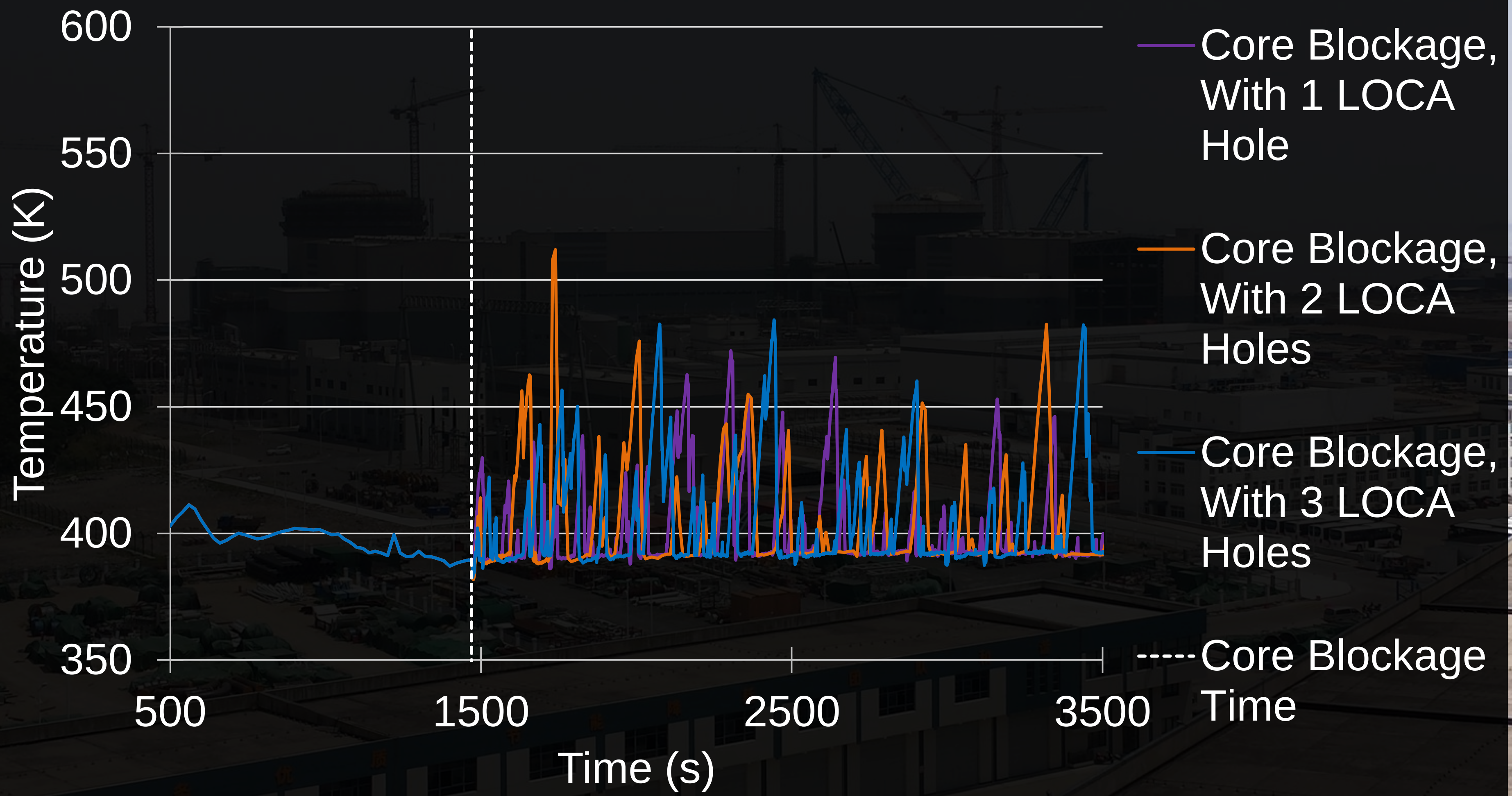
[8]

Results with LOCA Holes



Results with LOCA Holes

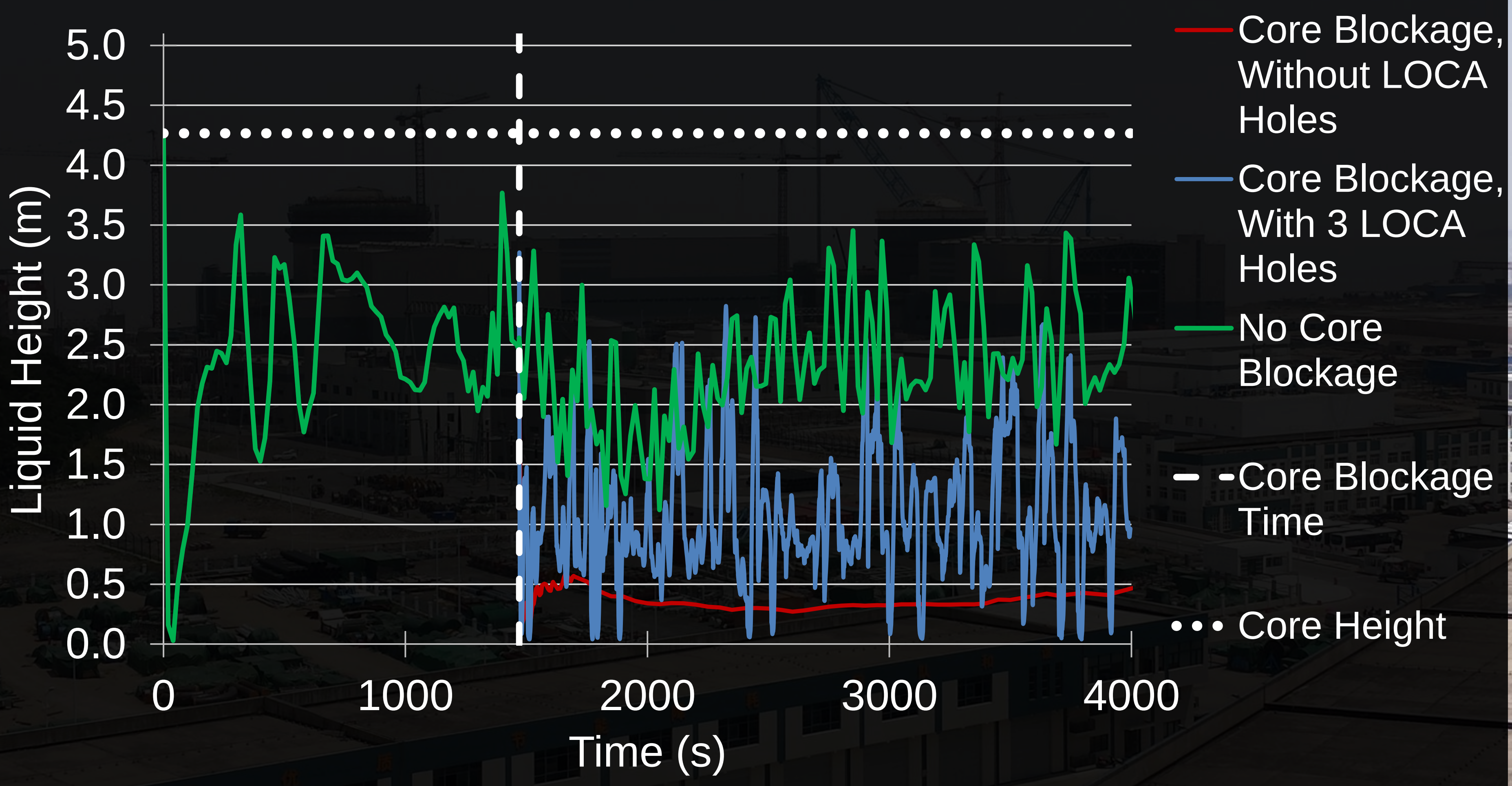
Peak Cladding Temperatures



TEXAS A&M UNIVERSITY

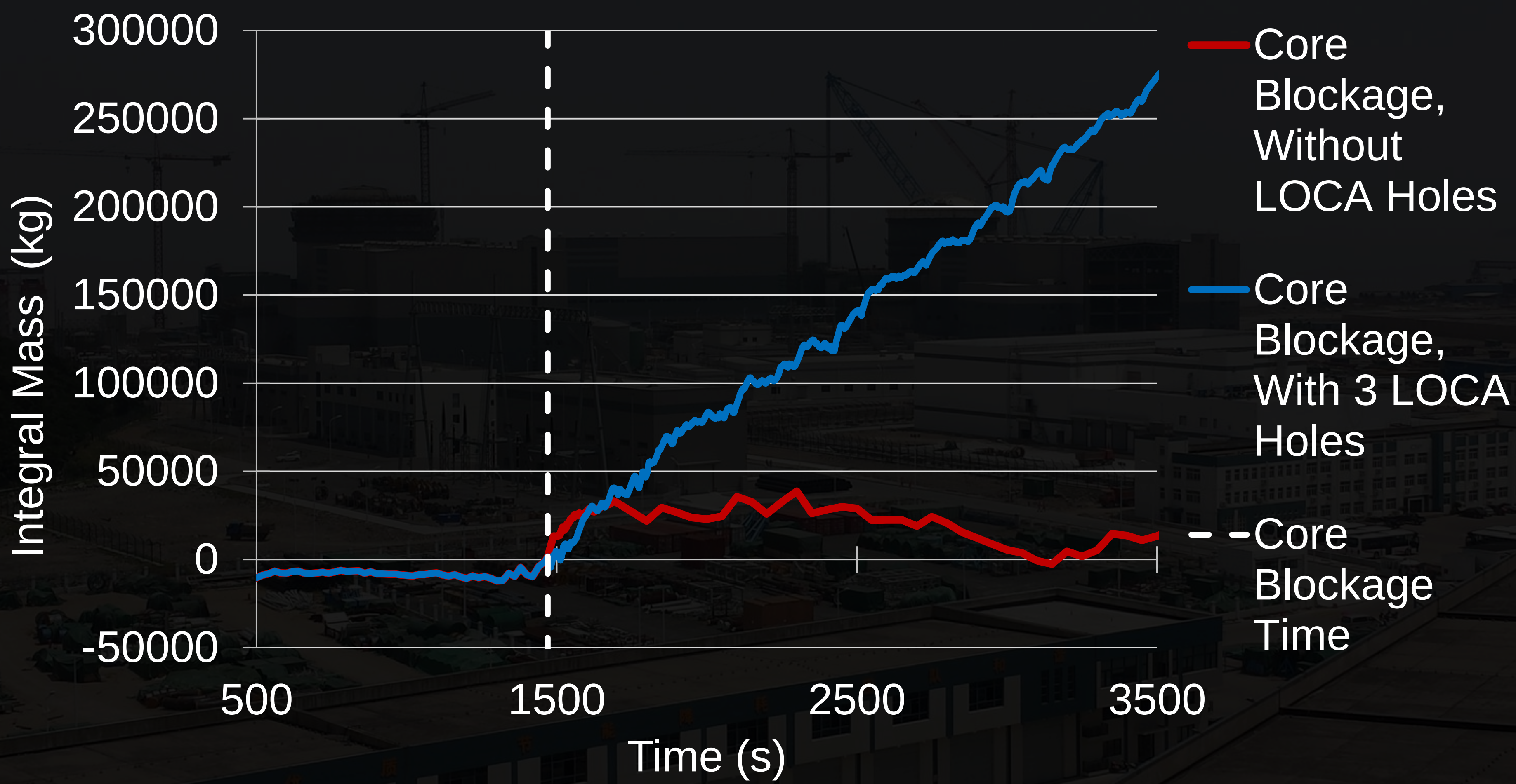
Results with LOCA Holes

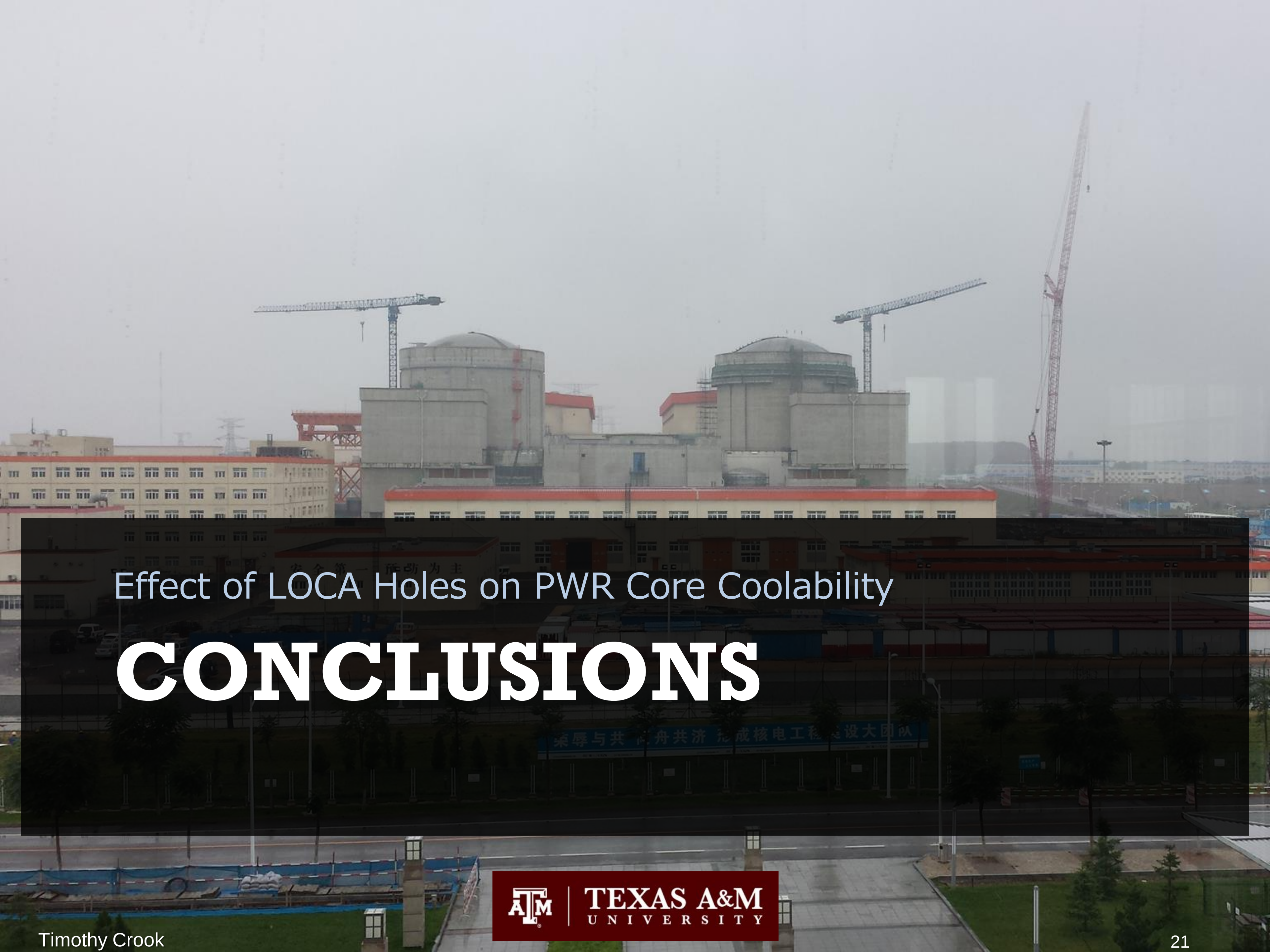
Core Collapsed Liquid Level



Results with LOCA Holes

Core Bypass Integral Flow





Effect of LOCA Holes on PWR Core Coolability

CONCLUSIONS

Conclusions

- Performed RELAP5-3D simulations
  - Cold-leg DEG LOCA with full core blockage
  - Three simulations included LOCA holes, one did not
- Determine LOCA hole effect on core flow and coolability by examining:
  - Peak Cladding Temperature
  - Core Collapsed Liquid Level
  - Core Bypass Integral Flow

Conclusions

No LOCA Holes

- Substantially less coolant supplied to core
- Cladding temperature increased to failure

With LOCA Holes

- More coolant flowed into the bypass (Bypass Integral Flow plot)
- More coolant reached the core itself (Collapse Liquid Level plot)
- Core Coolability was improved and no failure occurred (Peak Cladding Temperature plots)

References

1. M. R. FARD, "Resolution of Generic Safety Issues," NUREG-0933, U.S. Nuclear Regulatory Commission, (2011).

2. D. V. RAO, C.J. SHAFFER, M.T. LEONARD, K.W. ROSS, B.P. JAIN, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance," NUREG/CR-6808, LA-UR-03-0880, U.S. Nuclear Regulatory Commission (2003).

3. RELAP5-3D User's Manual, INEEL-EXT-98-00834 (2005).

4. MELCOR computer code manuals vol. 1: Primer and users' guide, version 2.1. Technical Report NUREG/CR -6119, Sandia National Laboratories, September 2008 (2008).

5. R. VAGHETTO, B.A. BEENY, Y.A. HASSAN, K. VIEROW, "Analysis of Long-Term Cooling of a LOCA by Coupling RELAP5-3D and MELCOR," ANS Annual Meeting, Chicago, IL, June 24-28 (2012).

6. R. VAGHETTO, B.A. BEENY, Y.A. HASSAN, K.VIEROW, "Sensitivity Analysis of a Typical Large, Dry Containment Response during a Loss of Coolant Accident using RELAP5-3D and MELCOR," ANS Annual Meeting, Atlanta, GA, June 16-20 (2013).

7. IDELCHIK, I. E. Handbook of Hydraulic Resistance. 2nd Ed. Washington (1986).

8. R. VAGHETTO, A. FRANKLIN, Y.A. HASSAN, "Sensitivity Study of Hypothetical Debris-Generated Core Blockage Scenarios," ANS Annual Meeting, Washington, DC, November 10-14 (2013).

Thank You!
Any Questions?