PROCEEDINGS OF THE WORKSHOP ON SEVERE ACCIDENT RESEARCH HELD IN JAPAN (SARJ-97)
OCTOBER 6-8, 1997, YOKOHAMA, JAPAN

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(Ed.) Jun SUGIMOTO

日本原子力研究所
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Proceedings of The Workshop on Severe Accident Research held in Japan
SARJ-97

Organized by Severe Accident Research Laboratory
Department of Reactor Safety Research
Japan Atomic Energy Research Institute (JAERI)

and

Nuclear Power Engineering Corporation (NUPEC)

at
Pacifico Yokohama
Yokohama, Japan
October 6 - 8, 1997

Edited by Jun SUGIMOTO

Organization of the Workshop

The Workshop was organized by the following staffs of JAERI:

M. Maeda, The Chairman of the Workshop
(Severe Accident Research Laboratory)
J. Sugimoto, N. Yamano, A. Hidaka, Y. Harada, A. Maeda, Y. Maruyama,
K. Moriyama, H. Shibazaki, H. Park, H. Yang, M. Kataoka, M. Masui, T. Naraoka, E.
Kato and Y. Ishikawa

and the following staffs of NUPEC:

H. Ogasawara, The Cochairman of the Workshop
(Systems Safety Department)
H. Nagasaka, T. Hashimoto, T. Matsumoto, M. Iriyama, A. Watanabc, M. Kato,
Y. Haruguchi, and H. Noguchi

Severe Accident Research Laboratory wishes to express sincere appreciation to Messrs.
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Business Administration of Department of Reactor Safety Research, for their support and
assistance to hold the Workshop.
Proceedings of the Workshop on Severe Accident Research Held in Japan (SARJ-97)
October 6 - 8, 1997, Yokohama, Japan

(Ed.) Jun SUGIMOTO

Department of Reactor Safety Research
Nuclear Safety Research Center
Tokai Research Establishment
Japan Atomic Energy Research Institute
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(Received April 28, 1998)

The Workshop on Severe Accident Research held in Japan(SARJ-97) was taken place at Pacifico Yokohama on October 6 - 8, 1997, and attended by 180 participants from 15 countries and one international organizations. A total of 58 papers, which covers wide areas of severe accident research both in experiments and analysis, such as in-vessel melt retention, fuel-coolant interaction, fission products behavior, structural integrity, containment behavior, computer simulations, and accident management, were presented. The panel discussion titled "Severe Accident Research for Future Reactors" has been successfully conducted, and the wide variety of opinions and views are exchanged among panelists and experts.

Keywords: Severe Accident, Source Term, Accident Management, Fuel-Coolant Interaction, Fission Products, Containment, Structural Integrity, Advanced Analysis, PSA
シビアアクシデント研究ワークショップ(SARJ-97)論文集
1997年10月6－8日，横浜

日本原子力研究所東海研究所安全性試験研究センター原子炉安全工学部
（編）杉本 純
（1998年4月28日受理）

1997年10月6日から8日にかけて、横浜パシフィコにおいて、シビアアクシデント研究ワークショップ（SARJ-97）が開催された。このワークショップには、15ヶ国、1国際機関より180人の参加者があった。ワークショップでは、各国の研究概要、圧力容器内浸融炉心保持、水蒸気爆発、FP挙動、構造健全性、隔離容器挙動、シミュレーション、アクシデントマネジメントなど、シビアアクシデントに関する実験及び解析を含む幅広い領域を対象として、合計58件の発表があった。パネルディスカッションでは、「将来型炉のシビアアクシデント研究」をテーマに、パネリスト及び会場の専門家により活発な討論が行われ、様々な意見や見解が交換された。
Severe accidents have been one of the central reactor safety issues especially since the TMI-2 and the Chernobyl accidents. The objectives of severe accident research are first of all to understand physical phenomena associated with severe accidents, to develop analytical methods to predict such phenomena, to quantify the safety margin of nuclear reactors against severe accidents, to evaluate the effectiveness of accident management measures, and finally to increase the level of safety much further by combining these.

To meet these objectives, Japan Atomic Energy Research Institute (JAERI) initiated severe accident research in 1985. Wide range of severe accident research activities have been performed in accordance with National Five-year Safety Research Plan established by the Nuclear Safety Commission. On the contrary, Nuclear Power Engineering Corporation (NUPEC) has conducted Containment Integrity Tests to demonstrate the reactor containment vessel integrity and to respond to mitigative accident management concerns. The Workshop on Severe Accident Research in Japan (SARJ) has been held by JAERI and NUPEC since 1990 to provide results of severe accident research and to exchange information among international participants for more efficient and further progress of the research.

In the Workshop (SARJ-97) held in Yokohama on November 6 - 8, 1997, the topics included overview of severe accidents, in-vessel melt retention, fuel-coolant interaction, fission products behavior, structural integrity, containment behavior, computer simulations, and accident management. The panel discussion titled "Severe Accident Research for Future Reactors" has been successfully conducted, and the wide variety of opinions and views are exchanged among panelists and experts.

It is our great pleasure and wish that the information shared at the Workshop will be utilized for further reducing the phenomenological uncertainties and risk of the nuclear reactors associated with severe accidents, for the possible regulatory applications, and for highly enhancing the nuclear safety for future.

Lastly I would like to express my deepest sorrow for Mr. Yamano's sudden death on October 3, 1997, just three days before the Workshop. We have determined to dedicate these Proceedings to the memory of Mr. N. Yamano for his devoted efforts and contribution to the severe accident research in Japan.

Jun Sugimoto, Editor
Head, Severe Accident Research Laboratory
Japan Atomic Energy Research Institute
DEDICATION

These Proceedings are dedicated to the memory of Mr. Norihiro Yamano, who passed away on October 3, 1997.

Briefly recollecting his career, since Mr. Yamano joined JAERI in 1984 from University of Tokyo, he actively conducted research in the field of severe accidents. He developed the REMOVAL code to model fission product gas/aerosol behaviors in the containment. He established the degraded core coolability by experiment and analysis. From 1988, he initiated and led JAERI's large scale experiments, the ALPHA Program, aiming at FCI, molten core concrete interaction, aerosol re-entrainment, and electrical cable penetration leakage. His final efforts were directed towards FCI phenomenology investigations with ALPHA and basic experiments, and JASMINE code development.

Mr. Yamano was an exceptionally competent researcher with a deep knowledge and broad experience in both basics and applications. He enthusiastically participated in international activities, such as in OECD as a member of FPC, SAC and CAM in CSNI, willing to challenge a variety of roles such as lead author of the OECD technical report on ex-vessel FCI and debris coolability published in December, 1996. He played a crucial role for the OECD/CSNI Specialists Meeting on Fuel-Coolant Interactions on May 19 - 21, 1997, Tokai-mura, Japan. He also acted an important role in severe accident related committees for the Japanese regulators. Mr. Yamano stood out as a true technical expert.

The most lasting part of Mr. Yamano's legacy will be his human side, the example of cheerful character, open mind, kindness and personal integrity that he set for all who worked with him.

He is most sadly missed. May his soul rest in peace.
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   A. Omoto of Tokyo Electric Co., Japan
   C. S. Kang of Seoul National University, Korea
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Opening Remarks

M. Maeda
Director, Nuclear Safety Research Center
Japan Atomic Energy Research Institute

Good morning ladies and gentlemen. On behalf of Japan Atomic Energy Research Institute, JAERI, and coorganizer Nuclear Power Engineering Corporation, NUPEC, I would like to welcome all the participants to the Workshop on Severe Accident Research held in Japan, SARJ-97. The first Workshop was held in 1990 in Tokai JAERI with 78 participants, including 7 overseas participants from three countries. This year, as in these several years, we have about 200 participants, including over 40 overseas participants from 14 countries. It is my great pleasure that we have so many participants again to this eighth Workshop. However, first of all, it is my great regret that I must tell you a very sad news. Mr. Yamano, one of the most important contributor of this Workshop, has died on October 3 of a sudden heart attack. I would like to express my deepest sympathy for his families and I believe that all the participant here would share this grief.

The purpose of this Workshop is to provide an opportunity for exchanging information on severe accident research activities ongoing all over the world. From Japan side, achievements in the past year will be presented by JAERI, NUPEC, industries, and universities. Through the extensive world-wide severe accident research activities in these years, much knowledge has been accumulated. However some important issues still remain as challenges, and new issues have been created mostly in relation with the application to future reactor design. In the afternoon of Wednesday, we will have Panel Discussion entitled "Severe Accident Research for Future Reactors". It is my sincere hope that the outcome of all the presentations, discussions, and Panel Discussion will give us a sophisticated insight into the fruitful severe accident research for future.

This year the Workshop is held in Yokohama for the first time, and at the same place as The International Conference on Future Nuclear Systems, GLOBAL'97, which mostly aims at advanced fuel cycles. This is intended for the convenience of the participants to exchange information between SARJ and GLOBAL. By the way this area in Yokohama is called "minato mirai-ji", which means "Port for future twenty first century". So I think this is the right place to discuss the future reactors and systems in SARJ and GLOBAL.

Finally I would like to express my sincere gratitude to all the organizations and individuals concerned to this Workshop for your great help, assistance and cooperation. Also I do hope that all participants from abroad will enjoy your stay during this colorful and beautiful season of the year in Japan. Additionally, I would like to mention again that all the participants will long cherish the memory of Mr. Yamano and his contribution to this Workshop.

Thank you very much for your attention.
1. Plenary Session

Overview of Research Activities

Chairperson: H. Nariai (Tsukuba Univ.)
Co-chairperson: J. Sugimoto (JAERI)
1.1 OVERVIEW OF SEVERE ACCIDENT RESEARCH AT JAERI

SUGIMOTO, Jun

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ABSTRACT

Severe accident research at JAERI aims at the confirmation of the safety margin, the quantification of the associated risk, and the evaluation of the effectiveness of the accident management measures of the nuclear power reactors, in accordance with the governmental five-year nuclear safety research program from FY 1996 and FY 2000. JAERI has been conducting a wide range of severe accident research activities both in experiment and analysis, such as melt coolant interactions, fission product behaviors in coolant system and containment, containment integrity and assessment of accident management measures. Molten core and coolant interaction, molten core concrete interaction, the leakage through containment penetrations, and aerosol reentrainment behavior in containment have been investigated in ALPHA Program. Recently MUSE experiments in ALPHA Program has been initiated for the precise energy measurement due to steam explosion in melt jet and stratified geometries. In VEGA Program, which aims at FP release from irradiated fuels at high temperature and high pressure under various atmospheric conditions, the facility construction is in progress. In WIND Program the revaporization of aerosols due to decay heating and also the integrity of the piping from this heat source are being investigated. Code development activities are in progress for an integrated source term analysis with THALES, fission product behaviors with ART, containment aerosol behaviors with REMOVAL, steam explosion with JASMINE, and in-vessel debris behaviors with CAMP. Application of detailed and mechanistic codes, such as SCDAP/RELAP5, CONTAIN, and VICTORIA to experimental analyses has made progress by participating international standard problem and code comparison exercises. The outcome of the severe accident research will be utilized for the detailed implementation of the accident management measures and also for the future reactor development. JAERI participates in most international international activities and contributes to the progress of severe accident research. Consensus on severe accident issues should be reached among international society for future reactor development.
OVERVIEW OF SEVERE ACCIDENT RESEARCH AT JAERI

SUGIMOTO, J.
Japan Atomic Energy Research Institute
Presented at SARJ-97
October 6-8, 1997, Yokohama, Japan

1. Introduction
Objectives of Severe Accident Research
- Evaluation of Safety Margin
- Evaluation of Effectiveness of Accident Management
- Reduction of Risk

2. Plan and Schedule
- Safety research is conducted in accordance with national 5-year program authorized by Nuclear Safety Commission (NSC).
- Severe accident is selected as one of six important areas in new 5-year research plan for FY1996 to 2000.

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2. Plan and Schedule
3. Current Research Status
   3.1 Experiments
   3.2 Code Development and Assessment
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### 3. Current Research Status

#### 3.1 Experiments

**Containment Integrity**
- Steam Explosion - ALPHA
- In and Ex-Vessel Melt Coolability - ALPHA
- Penetration Leakage - ALPHA
- Structural Analysis with FEM

---

**ALPHA Program**

**Assessment of Loads and Performance of Containment in a Hypothetical Accident**

**Objectives**
- Evaluation of Mechanical Loads to Containment
- Characterization of Leakage Behavior Through Containment Penetrations
- Assessment of Possible Accident Management Measures

---

**Test Items in ALPHA Program**

- **Melt/Coolant Interactions**
  - Steam Explosion: Yamano 10/7
  - In-vessel Debris Coolability: Maruyama 10/6
  - MUSE/AMUSE: Park 10/6

- **Melt/Concrete Interactions**

- **Fission Products Aerosol Behavior**

- **Penetration Leakage**
Fission Product Behavior

Fission Product Release - VEGA Program
- Irradiated Fuel at High Temperature and High Pressure

Fission Product Transport
- Deposition in piping - WAVE Experiment
- Revaporization in piping - WIND Program

Fission Product Behavior in Containment
- Reentrainment - ALPHA Program

VEGA Program

Hidaka/Nakamura 10/7

Verification Experiments of FP Gas/Aerosol release

Objectives
To obtain fission product release data
(1) at high temperature above melting point (~2,800°C)
(2) at high ambient pressure (up to 1MPa)
(3) under various atmospheric conditions (steam, hydrogen, air, inert gas)
(4) with/without structural materials (control rod)
(5) from once molten core (TMI debris samples)
WIND Program
Wide Range Piping Integrity Demonstration

Objective
(1) Revaporization of FP aerosols in piping
(2) Integrity of piping due to FP decay heat

Facility
(1) FP aerosol simulant (CsI, CsOH)
(2) High temperature (~1,000°C)
(3) Full-scale piping (small and medium sizes)

Test Items in WIND

- Aerosol Behavior Test      Shibazaki 10/7
- Associated Analysis        Hidaka 10/7
- Piping Integrity Test      Maeda 10/7
- Metallurgical Examinations Harada 10/7
Studies on Accident Management

- **ROSA-V Program**
  - Prevention of Severe Accident

- **ALPHA Program**
  - Melt Coolability
  - Application to Future Reactor

- **Accident Management Analysis**
  - Containment Venting by THALES

Ishikawa 10/8

3.2 Code Development and Assessment

- **Code Development**
  - ART : FP in Primary Loop
  - REMOVAL : FP in Containment
  - JASMINE : Steam Explosion
    - Premixing
    - Propagation
  - CAMP : In-Vessel Debris Behavior

- **Code Assessment**
  - RELAP5/SCDAP, CONTAIN, ICARE2
  - CORCON, VICTORIA, MELCOR

- **International Standard problems/Benchmark**
  - VANAM (ISP-37), FARO (ISP-39), STORM (ISP-40), RTF (ISP-41)
  - PHEBUS/FP, Round Robin

Severe Accident Research Laboratory
3.3 Collaboration with Universities

- FP Vaporization and Chemical Interaction
  Prof. Yamawaki, Univ. of Tokyo
- Simulation of FP Aerosol Behavior in Piping
  Prof. Kataoka, Kyoto Univ.
- Visualization of Pre-mixing by Nutron Radiography
  Prof. Mishima, Kyoto Univ.
- Creep Behavior of Piping
  Prof. Miyazaki, Kyusyu Univ.

4. International Activities

- International Collaboration/Agreement
  CSARP (USNRC)
  ACE/MACE (EPRI)
  PHEBUS-FP (IPSN/EC)
  RASPLAV (OECD)
- International Meetings
  OECD Specialist Meeting on FCI
  May 19-21, JAERI Tokai, Japan
  SARJ Workshop
  Since 1990

5. Summary

- Extensive activities are being conducted in accordance with nuclear safety research plan, mainly to be reflected for the implementation of accident management.
- JAERI participates in most international activities and contributes to the progress of severe accident research.
- Consensus on severe accident issues should be reached for future reactor development.
1.2 Present Status of Containment Integrity Tests at NUPEC

Hideo Nagasaka
Systems Safety Department, NUPEC

ABSTRACT

Objectives of Containment Integrity Tests (CIT) conducted at NUPEC are to demonstrate reactor containment vessel integrity during severe accident and to respond to Phase II accident management (AM) concerns regarding uncertainty of phenomena, raised by Japan Atomic Energy Safety Commission. CIT consists of experiments and analysis of debris cooling phenomena, hydrogen combustion behavior, fission products transport behavior and containment structural behavior. The uncertain phenomena regarding Phase II AM (defined as AM after reactor pressure vessel failure) treated at NUPEC are fuel-coolant interaction (FCI) and molten core-concrete interaction phenomena during ex-vessel debris cooling, evaluation of peak pressure during hydrogen combustion, ultimate strength and leakage conditions in the containment vessel and effectiveness of fission products (FP) removal under Phase II AM conditions. Most of these studies have been conducted under the international cooperation including the participation of OECD project.

This paper summarizes the present status of CIT focusing the progress during fiscal year 1996. The progress is: a; UO$_2$ debris falling jet behavior test to support ex-vessel debris cooling tests completed and the evaluation of typical FCI tests of debris dropping into pool under the most dominant severe accident scenarios under way, b; 1/10 scale BWR II containment vessel structural test completed, c; evaluation of tendon tensile force distribution based on a newly proposed friction coefficient to better predict 1/4 scale prestressed concrete containment vessel structural test completed, d; test facility modification of GIRAFFE-FP aiming at demonstration of effective aerosol FP removal by containment spray under Phase II AM completed and the evaluation of preliminary test results under way, e; no leakage failure of any containment penetration below 500K and favorable FP trapping effect at leakage path confirmed, and f; large scale hydrogen combustion tests under AM conditions simulating transient hydrogen and steam generation completed.

This work is sponsored under the contract by the Ministry of International Trade and Industry,
Present Status of Containment Integrity Tests at NUPEC

SARJ-97
October 6, 1997 Pacifico YOKOHAMA

H. Nagasaka
Systems Safety Department

Features of Debris Cooling Experiment

- Ex-vessel debris cooling tests (COTELS project)
  - Observation of flow mode of falling debris jet and debris dispersion characteristics
  - Without FCI
  - Test parameters:
    - Debris composition and falling velocity

- FCI and MCCI experiments with conditions derived for the most plausible SA scenarios in LWR's simulating the most plausible SA scenarios in LWR's simulating
  - Real molten debris (UO2/ZrO2/Zr/Steel), temperature
  - Falling debris jet diameter
  - Ambient conditions (pressure, temperature, noncondensible gas partial pressure)
  - Flow rate of injecting water
  - Pool water temperature, depth (FCI only)
  - Ratio of accumulated debris diameter to thickness (MCCI only)
  - Decay heat (MCCI only)

Overall Scenario of Containment Integrity Tests

Demonstration of containment vessel integrity
Realistic evaluation of source term

COTELS Tests Program

[1] Test 01
- Objective: Observation of flow mode of falling debris jet and debris dispersion characteristics without FCI
- Test parameters:
  - Debris composition and falling velocity

[2] Test A
- Objective: Simulation of FCI for debris dropping into pool
- Test parameters:
  - Debris mass and composition, and falling velocity
  - Water pool depth, volume and temperature
  - Nitrogen partial pressure

[3] Test B
- Objective: Simulation of FCI for water injecting onto debris
- Tentative test parameters:
  - Debris composition and falling velocity
  - Geometry of concrete trap
  - Induction heater power
  - Flow rate and flow injection mode

- Objective: Simulation of MCCI with overlying water pool
- Tentative test parameters:
  - Same as Test B
Features of Test Facility

- Test A
  - Electrical Melting Furnace
  - Concrete Floor
  - Adjustable Melt Catcher
  - Concrete floor simulating wettability and surface boiling characteristics
  - Adjusted pool volume for a given pool depth

- Test B/C
  - Electrical Melting Furnace
  - Coolant Injection Nozzle
  - Induction Heater Coil
  - Concrete Trap

Dispersed Solidified Debris on Melt Catcher (Test A4)

- Almost all debris fragmented due to FCI different from Test 01
- Diameter of fragmented debris ranged ~0.5 mm to ~7 mm (Implication of no steam explosion)
- About 6.5 kg/m³ average density of debris
- Detailed analysis in progress

Pressure Response of Test Vessel (Test A4)

- Evidence of no steam explosion
  - No pressure difference between water and air space
  - No pressure spike observed
  - Large size fragmented debris ranged 0.5 to 7 mm

- Pressure transient characteristics
  - Initial rapid pressure increase due to heat transfer from fragmented falling debris in water pool
    (region A)
  - Pressure suppression due to heat-up of subcooled water pool
    (region B)
  - Gradual increase due to heat transfer from debris bed accumulated on melt catcher
    (region C)

Observation of Concrete Surface of Melt Catcher after Test A4

- Solidified debris very easily removed from concrete surface, suggesting the existence of water between the bottom of debris and concrete surface
- No damage observed on concrete surface and almost new (the concrete catcher can be used repeatedly)
Present Status of Large Scale Combustion Test

1. Objectives
   - Grasping the combustion behavior of uniform concentration hydrogen inside the test vessel simulating 11 containment compartments
   - Confirmation of the validity of the hydrogen combustion mitigation factor
   - Evaluation of combustion behavior under transient AM conditions

2. Typical test results simulating AM conditions
   - Test conditions
     - Continual transient supply of steam, H\textsubscript{2}, and spray
     - Continual ignition
   - Intermittent mild burning of H\textsubscript{2} without accumulation of high concentration H\textsubscript{2}S

Schematic diagram of test facility

Effect of Spray Operation on H\textsubscript{2} Combustion Behavior

1. Test Conditions
   - Initial gas concentration (steam:60%, H\textsubscript{2}:6%, air:70%)
   - 1 glow plug used

2. Test Results
   - Intermittent mild combustion associated with steam condensation observed
   - Almost uniform H\textsubscript{2} concentration observed

Summary of H\textsubscript{2} Combustion Tests under AM conditions

1. Test conditions
   - Initial gas concentration (steam:30%, air:70%)
   - Continual transient supply of steam and H\textsubscript{2}
   - 4 glow plugs used

2. Test Results
   - Mild Combustion in dome region after 40 min H\textsubscript{2} injection
   - Propagation of flame front toward H\textsubscript{2} injection region observed

Effect of Spray Operation on H\textsubscript{2} Combustion Behavior

1. Test Conditions
   - Initial gas concentration (steam:60%, H\textsubscript{2}:6%, air:70%)
   - 1 glow plug used

2. Test Results
   - Intermittent mild combustion associated with steam condensation observed
   - Almost uniform H\textsubscript{2} concentration observed

Present Status of Structural Behavior Tests

1. Objectives
   - Grasping the ultimate strength of reactor containment vessels by static pressure load
   - Refinement of structural behavior evaluation technique

2. Features
   - Joint test between NUPEC and NRC/SLN (except for the hatch test)
   - 1/10 Steel containment test facility simulating contact structure
   - Hatch test facility modeled in actual size
   - 1/4 Concrete containment test facility with liners

3. Current situation
   - Steel shell: Data evaluation under way
   - Steel hatch: Data evaluation under progress
   - PCCV: Analysis of pre-stressed tendon system test completed

Steel containment vessel (1/10)
Summary of 1/10 SCV Structural Integrity Test

**Objectives**
- To evaluate ultimate strength of BWR MARK-1 steel containment vessel
- To enhance predictive capability of structural behavior up to failure

**Test Results**
- Leakage at reinforcement plate of equipment hatch
- Maximum pressure up to 6 Pd
- Evaluation of test data under way

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Derivation of Empirical Correlation Tendon Friction

1. Tensile force distribution assuming constant friction coefficient
   \[ \ln \left( \frac{P}{P_0} \right) = m \]

2. Tensile force distribution considering \( q \) and \( P_0 \) dependency of friction coefficient assumed to be
   \[ \ln \left( \frac{P}{P_0} \right) = m_0 + e \exp \left( f \frac{P}{P_0} \right) \]

3. Good prediction of tendon elongation by the present correlation

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Tendon Tensile Force Distribution Test

**Objective**
- To obtain realistic tendon friction coefficient correlation as a function of loading end load and angle utilizing 1/4 PCCV ancillary test
- To evaluate the effect of the present correlation on ABAQUS structural response of 1/4 PCCV test during pressurization comparing conventional constant friction coefficient analysis

**Outline of Tendon Instrumentation Ancillary Test**
- Geometry of test specimens identical to that of 1/4 PCCV model
- Instrumentation
  - Load at both ends
  - Local strains of tendon
- Test procedure
  - 50 kN interval loading of tendon

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Tendon Tensile Force Distribution by the Present Friction Correlation Comparing with Constant Friction Case

- Prestressing - Convex curve in the vicinity of loading end due to functional characteristics of the present correlation
- Setting - Maximum value at 1.08 rad (Larger setting loss region due to smaller friction coefficient in the vicinity of loading load)
- 2.0 Pd - Flattening throughout the whole region
Summary of FP Trapping Tests

**Objectives**
- To investigate the containment penetrations (electrical penetration assembly/flange gasket) leakage criteria under severe accident conditions
- To demonstrate aerosol trapping effect at the leakage path of damaged actual penetration assembly

**Test Results**
- 12 Eps and 5 FG examined
- Leakage condition
  - No leakage failure below 230°C
  - Leakage failure above 280°C
- Aerosol Trapping Effect: DF>10 (8 Eps)
- Development of empirical DF correlation under way

**Temperature Profile of Degradation Test**

**CHARACTERISTIC FEATURES INSIDE CONTAINMENT UNDER AM CONDITIONS**
- **LOW FLOW SPRAY**
  - LOWER CONDENSATION RATE DUE TO LARGER SPRAY DROPLETS
  - EFFECT OF THE REDUCED CONDENSATION RATE ON DIFFUSION
- **CONTINUOUS FRESH WATER SPRAY SUPPLIED FROM WATER SOURCE OUTSIDE CONTAINMENT VESSEL**
  - LARGER FP CONCENTRATION DIFFERENCE BETWEEN DROPLET VOLUME AND CONTAINMENT AIRSPACE
- **HIGH HUMIDITY CONDITION DUE TO STEAM GENERATION AS A RESULT OF DEBRIS COOLING**
  - PROMOTION OF GRAVITATIONAL SETTLING AS A RESULT OF FP AEROSOL PARTICLE GROWTH
- **INTEGRAL FP REMOVAL EFFECT VIA VARIETY OF DEPOSITION MECHANISM AND COAGULATION MECHANISM**
- **TIME VARYING AMBIENT CONDITIONS**
  - TOTAL PRESSURE / STEAM PARTIAL PRESSURE / TEMPERATURE
  - FP AEROSOL GENERATION RATE

Objectives of GIRAFFE-FP Project

**To provide data demonstrating the effective aerosol FP removal by containment spray under the following PHASE II accident management (AM) conditions**
- Low flow spray
- High humidity condition due to debris cooling by water
- Long term fresh water supply

**To provide data for verification and modification of integral system codes such as MELCOR for analysis of FP transport behavior**

**CHARACTERISTIC FEATURES INSIDE CONTAINMENT UNDER AM CONDITIONS**
- **LOW FLOW SPRAY**
  - LOWER CONDENSATION RATE DUE TO LARGER SPRAY DROPLETS
  - EFFECT OF THE REDUCED CONDENSATION RATE ON DIFFUSION
- **CONTINUOUS FRESH WATER SPRAY SUPPLIED FROM WATER SOURCE OUTSIDE CONTAINMENT VESSEL**
  - LARGER FP CONCENTRATION DIFFERENCE BETWEEN DROPLET VOLUME AND CONTAINMENT AIRSPACE
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- **INTEGRAL FP REMOVAL EFFECT VIA VARIETY OF DEPOSITION MECHANISM AND COAGULATION MECHANISM**
- **TIME VARYING AMBIENT CONDITIONS**
  - TOTAL PRESSURE / STEAM PARTIAL PRESSURE / TEMPERATURE
  - FP AEROSOL GENERATION RATE
Overview of GIRAFFE-FP test facility

- Full height simulation of BWR
- 1/720 scaled volume of BWR
- Three separate vessel components simulating Drywell/Suppression Chamber/Reactor Pressure Vessel
- Actual plant spray nozzle

Heat loss control by heaters

Imposing transient boundary conditions of steam/non-condensable gas/CsI aerosol supplies

SPRAY FLOW PATTERN AND DROPLET DIAMETER DISTRIBUTION

- All droplets falling without attaching to the wall

Test condition
- Two nozzles
- Spray flow: 3 (litre/min)
- Pressure: 1 (bar)
- Atmosphere

TYPICAL AEROSOL CONDITION IN LOWER DRYWELL AFTER RPV FAILURE (TQUV)

- Aerosol condition to be imposed as boundary condition in GIRAFFE-FP determined by both MAAP and MELCOR analysis
  - Typical CsI flow rate as a result of re-evaporation: 1~5 mg/min
  - Typical CsI particle mass mean diameter: ~1 μm
- Both measured CsI concentration and diameter generated by atomizer nozzle confirmed close to prescribed values

TYPICAL TEST CONDITIONS (BWR TQUV CASE)

- Test conditions determined considering both MAAP and MELCOR analysis

Initial condition

- Drywell
  - Pressure: ~0.3 MPa
  - Temperature: ~420 K
  - Aerosol concentration: 10~100 mg/m³

Boundary condition

- Spray flow rate (transient)
- Steam supply (constant)
- Aerosol supply (nearly constant)
MELCOR ANALYTICAL RESULTS FOR BWR TQUV TEST

- Cs1 AEROSOL CONCENTRATION IN DRYWELL -

CALCULATED CONCENTRATION TRANSIENT AGREED WELL WITH THAT OF TEST DATA

Announcement of NUPEC Technical Facility Tour

- Date ... Afternoon, 9 October
- Place ... Toshiba Engineering Laboratory, Kawasaki city
- Test Facility ... GIRAFFE-FP and Fission products trapping test facility
- Meeting place and time (NUPEC will take care of participants.)
  -Place ... Registration at Washington Hotel
  -Time ... 13:00 AM

*Those who registered the participation, please fill up the participant list, prepared on the conference desk.

Summary of NUPEC Papers Presented at SARJ-97

1. Core-3LS Fuel Coolant Interaction Tests of UO2 Debris Dropping into Water Pool (Session IV)
2. Evaluation of Fission Products Release and Transport in the Circuit in PHEBUS FP Test by MACRES Code (Session V)
3. Failure Criteria and Fission Products Trapping Behavior at Containment Penetrations under Severe Accident Conditions (Session VII)
4. Fission Products Aerosol Removal Test by Containment Spray under Accident Management Conditions (Session VII)
5. Analysis of NUPEC's Large Scale Hydrogen Mixing Tests in a Reactor Containment vessel (Session III)
6. Summary of Hydrogen Combustion Tests at NUPEC (Session III)
7. Pressurization Test on a Full Scale Equipment Hatch Model (Session IV)
8. Pressurization Test of a 1/10 Steel Containment Vessel Model (Session IV)
9. Pre-test Analysis on the SCV Model Test (Session IV)
10. Preliminary Analysis and Instrumentation Planning of a Pre-stressed Concrete Containment Vessel Model (Session IV)
11. Analysis Study on Change of Tendon Tension Force Distribution during the Pressurization Process of Pre-stressed Concrete Containment Vessel (Session IV)

Conclusion

1. Ex-vessel FCI test of UO2-mixture debris dropping into pool, simulating most dominant severe accident scenario, completed with the results of no violent steam explosion
2. Large scale H2 combustion test demonstrated mild H2 combustion under phase II transient condition
3. 1/10 steel containment vessel test demonstrated containment integrity up to 6 times design pressure under room temperature
4. An empirical tendon friction correlation obtained as a function of loading end load and angle and tendon elongation well predicted by the correlation
5. One system integral test by GIRAFFE-FP, simulating BWR severe accident sequence, completed with the result of favorable FP removal even under low spray flow condition
6. No leakage failure of any actual containment penetration below 500 K and favorable FP trapping effect at leakage path confirmed
1.3 OVERVIEW OF SEVERE ACCIDENT RESEARCH AT THE USNRC

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Washington, D.C. 20555

ABSTRACT

The U.S. Nuclear Regulatory Commission is conducting severe accident research to provide the necessary technical data, analytical methods, and the expertise necessary to provide for an understanding of a range of severe accident phenomena. Considerable progress has been made in the understanding of the ways severe accidents can progress and challenge containments sufficient to support plant risk assessments, development of accident management strategies and resolve certain severe accident issues for operating and future plants, however, certain issues remain. Progress has been made in a number of areas. In addressing the issue of direct containment heating (DCH), considerable progress has been made toward the resolution of the DCH issue for PWRs. The major experimental program has now been completed, along with the resolution of DCH for a number of PWR plants. The resolution for the remaining PWR plants is nearing completion. The two remaining DCH experiments being planned will explore debris dispersal at lower pressures than those considered in the previous test series. In the area of lower head integrity, the experimental program to investigate boiling heat transfer on downward facing curved surfaces has been extended to expand the data base and model to take into account the effect of insulation around the outside surface of the RPV. Experiments on the reactor pressure vessel lower head failure are providing data on the strain behavior prior to creep rupture, rupture time, and the resulting rupture size from creep rupture of the lower head under the combined effects of thermal and pressure loads. As part of the ongoing research effort to address the issues of fuel-coolant interactions (FCI), experimental programs are addressing possible chemical augmentation of FCI energetics and effects of initial and boundary conditions on FCI energetics. In the area of temperature hydrogen combustion, the program to investigate high-temperature hydrogen combustion phenomena is continuing as well as the experiments to examine the performance of passive catalytic recombiners. THE USNRC is also continuing its active involvement in a number of international cooperative experimental programs such as RASPLAV, MACE, FARO, and PHEBUS. Finally, the USNRC is continuing its severe accident code development activities. Peer reviews of the VICTORIA and IFCI codes were recently completed.

Work is continuing on the SCDAP/RELAP5, CONTAIN and GASFLOW codes. Finally, a new version of MELCOR was recently released (MELCOR 1.8.4).
OVERVIEW OF SEVERE ACCIDENT RESEARCH AT THE USNRC

The Workshop on Severe Accident Research held in Japan SARJ-97

Presented by
Ali Behbahani
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

October 6, 1997

CURRENT DIRECTION OF SA RESEARCH PROGRAM

- Severe accident research continues to provide the technical bases for addressing a number of regulatory issues
- Experimental work is directed at areas of largest uncertainty or confirmatory work for code assessment
- Analytical work is directed toward completion of code development and assessment
- International cooperative efforts valuable in addressing severe accident issues
- Long term issue is maintenance of expertise

SEVERE ACCIDENT RESEARCH

Severe Accident Research has focused on phenomena and issues to understand and quantify phenomena and potential challenges to reactor vessel and containment integrity

- Direct Containment Heating
- Lower Head Integrity/Debris Coolability
- Fuel-Coolant Interactions
- Hydrogen Combustion
- Source Term
- Analytical Codes

DIRECT CONTAINMENT HEATING

OBJECTIVES
- To determine likelihood of early containment failure as a consequence of high pressure melt ejection and direct containment heating (DCH)

STATUS
- Completed large scale integral and separate effects tests simulating the Zion and Surry plants and completed evaluation of containment loads versus containment structural capability.
- Issued peer reviewed reports, NUREG/CR-6075, NUREG/CR-6109 and NUREG/CR-6338 for Zion, Surry and other Westinghouse plants with large dry or subatmospheric containments (41 plants). Issue resolved with finding of no significant failure probability for these plants
DIRECT CONTAINMENT HEATING
(Continued)

STATUS (CONT)

- Completed 7 integral effects tests for the CE configuration using "Calvert Cliffs - annular cavity design" for testing in SURTSEY (1/10 scale)
- Completing DCH issue resolution for CE and B&W plants and for W plants with ice condenser containments
- Initiating two additional tests, in conjunction with FZK and ISPN, with CE like geometry at lower reactor system pressure

LOWER HEAD INTEGRITY
(Continued)

EXTERNAL FLOODING (PENN STATE)

- Critical heat flux (CHF) experiments on downward facing curved surfaces (Penn State) performed to assess capability of removing decay heat via boiling on outside surface of vessel
- Completed final report on CHF phenomena and modeling NUREG/CR-6507 (June 1997)
- Extending experiments and analysis to study effect of insulation on CHF

CURRENT PROGRAM

- External Flooding (Penn State)
- Lower Head Failure Experiments (SNL)
- In-Vessel Debris Coolability Experiments (FAI)
- OECD RASPLAV Project (RRC-KI)

CURRENT PROGRAM

- Conducting experiments to measure strain behavior, time to rupture, and rupture size under combined effects of thermal and pressure loads, with and without penetrations
- Five experiments completed to date
- Three experiments being planned
- Complete experiments and assessment of analytical models of RPV lower head failure
LOWER HEAD INTEGRITY

(Continued)

IN-VESSEL DEBRIS COOLABILITY EXPERIMENTS (FAI)
- Participating in cooperative program initiated in mid FY96 (NRC, EPRI, 3 international organizations) to investigate inherent heat transfer mechanisms that promote in-vessel cooling of core debris and RPV lower head in the presence of water.

OECD RASPLAV PROJECT (RRC KL)
- Participating in OECD cooperative program to perform integral tests with prototypic materials to assess the possibility of retaining molten core material in the RPV lower head.

FUEL-COOLANT INTERACTIONS (FCI)

OBJECTIVES
- University of Wisconsin (UW) FCI propagation experiments with metallic and oxidic simulants to determine the effects of various fuel/coolant parameters on FCI energetics.

- Argonne National Laboratory (ANL) ZREX experiments to determine the chemical augmentation of FCI energetics in metal-water (Zr) systems, metal-oxide (Zr-ZrO₂)-water and metal-metal (Zr-SS)-water.

- Cooperative FARO/KROTOS program at JRC/hyspr to experimentally investigate non-explosive and explosive melt coolant interactions involving prototypic and simulant melts.

DEBRIS COOLABILITY

MACE Program (ANL)
- Participating in cooperative program to examine the mechanisms for ex-vessel quenching of core debris using prototypic materials.

FUEL-COOLANT INTERACTIONS (FCI)

(Continued)

UW FCI Program
- Series of one-dimensional propagation experiments using tin simulant have been conducted to investigate the effect of fuel coolant parameters on the steam explosion potential and yield.

- Initiated one-dimensional propagation experiments with oxidic simulants to investigate the effects of fuel/coolant parameters on steam explosion potential and energetics.

- Use results of the FCI experiments (at UW and elsewhere involving both oxidic and metallic simulants as well as prototypic melt materials) and an appropriate scaling rationale to extrapolate the experimental data to reactor situations.
FUEL-COOLANT INTERACTIONS (FCI) (Continued)

ZREX PROGRAM (ANL)

• Completed scoping tests with 200 gram Zr and ZrO₂ melt to investigate possible chemical augmentation of FCI energetics; all tests conducted at 0.1 Mpa in a one meter water pool at room temperature

• Completed first series of data tests involving one kilogram of melt material; preliminary results indicate little augmentation

• Completed second series of experiments to investigate the effects of melt composition, triggering and water subcooling on possible chemical augmentation

• Complete experiments and analyze experimental data to determine the extent of augmentation to FCI energetics

HYDROGEN COMBUSTION (Continued)

OBJECTIVE

• Assess potential challenges to containment integrity resulting from various modes of hydrogen combustion during severe accidents

CURRENT PROGRAM

• Low speed hydrogen combustion experiments (Cal Tech)

• Experiments on H₂ combustion to investigate deflagration to detonation transition (DDT) and criteria for placement of hydrogen igniters inside containment (RRC KI)

• Experiments on passive autocatalytic recombiner performance (SNL)

• High temperature, high speed hydrogen combustion experiments (BNL)

HYDROGEN COMBUSTION RESEARCH

(Continued)

LOW SPEED HYDROGEN COMBUSTION EXPERIMENTS (CAL TECH)

• Performing experiments to investigate diffusion flame stability and hot jets of hydrogen and steam to study ignition of lean hydrogen, air, nitrogen mixtures

LARGE-SCALE EXPERIMENTS AT RUSSIA RESEARCH CENTER (RRC) KURCHATOV INSTITUTE

• Performing large-scale experiments to investigate scaling phenomena for DDT and criteria for placement of hydrogen igniters

• Coordinating plans with RRC, FZK and IPSN on large-scale hydrogen-air-steam DDT experiments at elevated temperatures

HYDROGEN COMBUSTION RESEARCH

(Continued)

PASSIVE AUTOCATALYTIC RECOMBINERS (PAR) PERFORMANCE (SNL)

• Performing confirmatory tests to verify PAR performance to control combustible gases in containment

HIGH TEMPERATURE, HIGH SPEED HYDROGEN COMBUSTION EXPERIMENTS (JOINT NRC/NUPEC SUPPORT) (BNL)

• Cooperative program to perform experiments to predict conditions that may result in detonation and detonation transmission at high temperatures
SOURCE TERM RESEARCH

**PHEBUS**
- Participating in ongoing PHEBUS integral experiments being used to confirm our general understanding (if needed, upgrade computer codes) of fission product release and transport.

SEVERE ACCIDENT CODES

**MELCOR**: Integral systems level code to analyze severe accidents and consequences in nuclear power plants from initial core uncovering, through reactor vessel failure and containment response
- MELCOR 1.8.4 was released

**SCDAP/RELAP5**: Detailed mechanistic code to analyze in-vessel severe accident progression including thermal hydraulics, core melting, and reactor vessel failure
- SR5/CAP was initiated

**CONTAIN**: Detailed code for analysis of containment transient response (i.e., pressure and temperature conditions) during severe accidents and design basis accidents for a variety of containment types

**VICTORIA**: Detailed code to analyze fission product release and transport in the reactor coolant system during a severe accident including vapor deposition, resuspension and revaporization
- Completed VICTORIA peer review

**IFCI**: Integrated Fuel-Coolant Interactions code to model in-vessel and ex-vessel explosive and non-explosive phenomena
- Completed IFCI peer review

**GASFLOW**: Best estimate 3D finite difference code to predict transport, mixing, and combustion of hydrogen and other gases in the containment
PWR SAFETY RESEARCH AT IPSN

J. BARDELAY

INSTITUT de PROTECTION et de SÛRETE NUCLEAIRE

SARJ MEETING, YOKOHAMA October, 6-8 1997

IPSN has developed an important program on reactor safety in the frame of the European and international research. The goal of IPSN studies is to acquire necessary knowledge to analyze and to improve:

- accident prevention with as the first priority to ensure protection of installations, populations and environment,
- accident management with the identification and the assessment of effective accident management measures, and the identification of key events and measurements for accident management decisions,
- emergency preparedness.

The work performed at IPSN takes into account the inherent specificity of reactor safety: multiplicity of situations, limitation of available database, complexity of measurements, etc. The research effort includes the modeling of phenomena, the development of codes and the realization of experiments.

Codes:

In a first period of severe accident studies, some characteristic severe accident scenarios liable to occur in French pressurized water reactors have been identified and quantified, with the objective to determine typical releases for emergency planning and mitigation features. This work has implied the development of a whole set of scientific tools, the ESCADRE integrated codes system, designed to investigate the potential release in the case of an hypothetical severe accident. This system of codes has been also used for the elaboration and the validation of simplified computer tools which could be used in case of emergency situation so as to evaluate the situation and to forecast its development. In parallel, an experimental program for ESCADRE validation has been set up by IPSN, particularly in the area of fission product behavior. In the frame of the IPSN-GRS cooperation on severe accidents studies, it has been decided to develop a common integral code for the calculation of severe accidents.
The study of the hydrogen risk is the subject of an important program at IPSN. First of all, analytical tests are envisaged in a new facility, called TOSGAN, to ensure a proper validation of the models for wall condensation and aspiration, under representative PWR severe accident conditions. Two phases are foreseen, tests of condensation on the walls (φ 1), tests of condensation on the spray droplets and tests with a sump (φ 2). The first tests are foreseen in 1998-1999. Secondly, the flame acceleration mechanisms and the deflagration-to-detonation transitions (DDTs), which could possibly occur in the containment subcompartments, are currently studied. As a first step, research efforts will be focused on criteria for preventing any DDT. A joint research program based on relatively large-scale (500 m²) DDT experiments, including a steam partial pressure, is on going in the RUFI facility in Russia. Thirdly, the equipment of French PWRs with hydrogen recombiners is presently under discussion, and tests on the potential poisoning mechanisms of catalytic recombiners, H2PAR program, are performed in adverse environments representative of severe accident conditions (representative mixture of fission product simulants).

In the frame of the research on fuel-coolant interaction the BILLEAU program which consisted of the study of the premixing phase during a steam explosion, is now achieved. The study of the detonation phase is the new objective. It will be done with the DETHER facility. The experimental devices requirements are very strict because the conditions of the explosions must be well-known and each explosion must be significant (limitation of border effect) and effective. The experiment is at the stage of the conception. In parallel, IPSN takes part in FARO experiments (JRC ISPRA).

In case of a vessel meltthrough, one major parameter controlling corium cooling is its propensity for spreading. The objective of the CORINE experimental program is to deduce, from results of analytical tests using simulants with a low melting point, reliable laws describing corium spreading on various materials (concrete, steel), with or without water.

The last domain of investigation is the study of uncertainties of in/ex vessel corium properties in order to improve the corium data base. The experimental program should include the corium interaction and chemistry project in the frame of 4 FWP and a project of new separate effect tests on core-corium properties set with two configurations: true fresh or irradiated fuel and crucible with internal metals. Otherwise, IPSN takes part in RASPLAV OECD project performed by Kurchatov Institute.

Tests are planned in 1998: FPT1 (low volatile FP & actinides release from UO₂, ZrO₂ debris bed, up to melting) and FPT2 (as FPT-1 under steam poor conditions: evaporating sump, H₂ recombiner coupons, recirculating spray).

The study of the hydrogen risk is the subject of an important program at IPSN. First of all, analytical tests are envisaged in a new facility, called TOSGAN, to ensure a proper validation of the models for wall condensation and aspiration, under representative PWR severe accident conditions. Two phases are foreseen, tests of condensation on the walls (φ 1), tests of condensation on the spray droplets and tests with a sump (φ 2). The first tests are foreseen in 1998-1999. Secondly, the flame acceleration mechanisms and the deflagration-to-detonation transitions (DDTs), which could possibly occur in the containment subcompartments, are currently studied. As a first step, research efforts will be focused on criteria for preventing any DDT. A joint research program based on relatively large-scale (500 m²) DDT experiments, including a steam partial pressure, is on going in the RUFI facility in Russia. Thirdly, the equipment of French PWRs with hydrogen recombiners is presently under discussion, and tests on the potential poisoning mechanisms of catalytic recombiners, H2PAR program, are performed in adverse environments representative of severe accident conditions (representative mixture of fission product simulants).

In the frame of the research on fuel-coolant interaction the BILLEAU program which consisted of the study of the premixing phase during a steam explosion, is now achieved. The study of the detonation phase is the new objective. It will be done with the DETHER facility. The experimental devices requirements are very strict because the conditions of the explosions must be well-known and each explosion must be significant (limitation of border effect) and effective. The experiment is at the stage of the conception. In parallel, IPSN takes part in FARO experiments (JRC ISPRA).

In case of a vessel meltthrough, one major parameter controlling corium cooling is its propensity for spreading. The objective of the CORINE experimental program is to deduce, from results of analytical tests using simulants with a low melting point, reliable laws describing corium spreading on various materials (concrete, steel), with or without water.

The last domain of investigation is the study of uncertainties of in/ex vessel corium properties in order to improve the corium data base. The experimental program should include the corium interaction and chemistry project in the frame of 4 FWP and a project of new separate effect tests on core-corium properties set with two configurations: true fresh or irradiated fuel and crucible with internal metals. Otherwise, IPSN takes part in RASPLAV OECD project performed by Kurchatov Institute.
IPSN RESEARCH PRIORITIES

AIM: TO PREVENT AND MITIGATE THE ACCIDENTS IN COMPLIANCE WITH THE DEFENSE IN DEPTH STRATEGY.

1. CODE DEGRADATION AND MELT PROGRESSION:
   - Highly important for the assessment of core cooling possibilities.

2. FISSION PRODUCTS BEHAVIOR:
   - To assess the risk and the magnitude of F.P. released out of the containment.

3. CONTAINMENT BEHAVIOR:
   - To maintain as far as possible containment integrity.

SAFETY QUESTIONS: CORE COOLING POSSIBILITIES

1. WATER INJECTION INSIDE THE VESSEL:
   - Feasibility: Depends on the delay to restore a failed system.
   - Efficiency: Not proved.
   - Associated risks: Steam explosion.

2. FLOODING THE REACTOR PIT:
   - Feasibility: Plant dependent (Geometrical, Inerting...).
   - Efficiency: Questionable (Large fuel vessel, fast flooding).
   - Associated risks: Core coolant interaction.

3. EX-VESSEL CORIUM COOLING:
   - Feasibility: Not proved.
   - Efficiency: Corrosion of fuel cladding.
   - Associated risks: Corium spreading.

SAFETY QUESTIONS: FISSION PRODUCTS BEHAVIOR

1. F.P. DETECTION IN THE CONTAINMENT (CHEMICAL AND PHYSICAL PROPERTIES, DEPOSITION, Dose BEHAVIOR...)

2. EFFECT OF FISSION PRODUCTS ON VESSEL SYSTEMS... (OR PROCESSES (ULTIMATE PROCESSES)..., OR SOURCE TERM...)

3. INFORMATION NEEDED FOR AN ON LINE CRISIS MANAGEMENT (EXPECTED QUANTITIES,...).

4. EVACUATION OF RESIDENT HEAT WITHOUT VENTILATION EXTREME PHASES.

SAFETY QUESTIONS: CONTAINMENT BEHAVIOR

1. SLOW TRANSIENTS:
   - Overpressurisation and heat transfer in the containment atmosphere.

2. RAPID PRESSURE AND/OR TEMPERATURE TRANSIENTS:
   - Fast circuit depressurisation.
   - Core melt convection.
   - Steam explosion.
   - Role of corium spreading.

3. CONTAINMENT LEAKTIGHTNESS: Key parameter.

USE OF CODES FOR SAFETY STUDIES AT IPSN

1. PARAMETRIC CODES:
   - Ordering of magnitude.

2. INTEGRATED CODES:
   - Pressure/Temperature, Loadings...

3. MECHANISTIC CODES:
   - Core degradation and fission products emissions.

VALIDATION OF CODES: 2 STEPS:

1. QUALIFICATION:
   - To verify the ability of the code for describing a single event.
   - Calculation of analytical experiment (Verification).

2. VERIFICATION:
   - To verify the ability of the code for describing a set of events (Global experiment)
INTEGRATED CODE: ASTEC

• OBJECTIVE:
  - Development of an I&N integral code for the calculations of severe accidents & levels.
  - Accident management strategies.

• FRAMEWORK:
  - I&N: Core cooperation on severe accidents.

• CODE SPECIFICATIONS:
  - The best model from both institutes.
  - Reasonable calculation time.

CORE DEGRADATION AND CORE COOLING STUDIES: CODES

• CATHARE: Thermal-hydraulic calculations until beginning of the core degradation:
  - Version current: CATHARE V2.01 (6).

• ICARE: Calculations of severe fuel damage:
  - Operational cooperation.
  - Version current: ICARE V1.1.
  - Code is used up to lower head rupture.

• CATHARE-ICARE: Thermal-hydraulic/core degradation coupling:
  - First version: ICARE V2.0IC.

• CHXCO: Core behavior in a reactor pit & core catcher:
  - First version: CATHARE V1.32.
  - Development of the next version: ICARE V2.0.

CORIUM CHEMICAL INTERACTIONS AND PROPERTIES

• IMPROVEMENT OF THE CORIUM DATA BASE USED IN SEVERE ACCIDENTS CODE.

• EXPERIMENTAL EFFORT:
  - Combustion/interaction experiments.
  - Irradiated rods.

• ANALYTICAL EFFORT:
  - Improvement and validation of data.

SIMULATED CORIUM SPREADING EXPERIMENT: CORINE

• PREDICTION OF THE SPATIAL DISTRIBUTION OF THE CORIUM IN A SPREADING AREA.

• FACILITY, EXPERIMENTAL CONDITIONS:
  - Reactor conditions (P, T, flow).
  - Presence of sump.

• 1997: Achievement of tests with electric fluid.

• 1998: Tests with non-electric fluid.

HYDROGEN STUDIES: CODE

• TONUS: Hydrogen risk in severe accident:
  - Evaluation of operational means for hydrogen risk mitigation.

• FACILITY: Reactor conditions (T, turbulent flow):
  - Flow rate: 5-8 m/s, 6-8 m/s, presence of fume, presence of steam.

HYDROGEN STUDIES: TOSQAN

• TO VALIDATE SOME MODELS OF THE TONUS CODE:
  - Steam condensation at the walls with non-condensables.
  - Condensation of the walls.
  - Condensation of the droplets.
HYDROGEN STUDIES : H2PAR
- TO STUDY THE EFFICIENCY OF RECOMBINERS (POISONING TO)
- FACILITY: REPRESENTATIVE OF REACTOR CONDITIONS.
  - EXIT VOLUME: 9 m3, RATE 1.1 m3/s, RPT.
  - R.E. 3.7 AND STEAM GLAC., EROSIONS GLAC., NATURAL INCIDENT GAS.
  - PULVING GENERATES STEAM SPECIES: DISSOLVES FREE PRODUCTS.
    - OIL, VAPOR.
- 1994: INFLUENCE OF STEAM AND AEROSOLS.
- 1997: INFLUENCE OF PARTICULATE CONCENTRATIONS.
- 1997: INFLUENCE OF STEAM AND AEROSOLS.

STEAM EXPLOSION : CODES
- MC3D:
  - CALCULATIONS OF CORIUM-WATER INTERACTION.
    - GOOD DESCRIPTION OF PREMIXING PHASE.
    - PRODUCED BY EXPLORATORY PHASE.
    - CURRENT VERSION: MC3D-
- PLEXUS:
  - USED FOR THE EVALUATION OF CONSEQUENCES OF CORIUM-WATER INTERACTION.
    - IN VESSEL AND IN VESSEL.

STEAM EXPLOSION EXPERIMENTS (1)
- OBJECTIVES:
  - STUDY OF THE DETONATION PHASE: FRAGMENTATION AND PROPAGATION.
- CONDITIONS OF EXPERIMENT:
  - THE EXPLOSION MUST BE SIGNIFICANTLY DISTINGUISHABLE FROM A REAL EXPLOSION.
  - NON-THERMAL CONDITIONS.
  - INTERACTION OF BALLS (DIAQ) WATER (5-230', 0.1 M),
  - INITIAL PRESSURE 100-800 bar.
  - PROBLEMS: DURATION OF THE EXPLOSION.
- SCHEDULE:
  - PROBLEMS: MEASUREMENTS.

STEAM EXPLOSION EXPERIMENTS (2)
- MICRONIS:
  - OBJECTIVES:
    - DETERMINATION OF PREMIXING PHASE, DETERMINATION OF THE FRAGMENTATION.
    - PRESSURE WAVE.
  - EXPERIMENT:
    - EXPLOSION OF A TIN OR UO2 DROP IN WATER (80-230', 0.1 M),
    - MEASUREMENTS: DURATION, SIZE OF THE FRAGMENTS.
  - SCHEDULE:
    - 1996: BIBLIOGRAPHY.
    - 1997: FIRST TEST.

STEAM EXPLOSION EXPERIMENTS (3)
- TREPAM:
  - OBJECTIVES:
    - DETERMINATION OF THE TRANSFER CORIUM DEBRIS: WATER.
    - EXPERIMENT:
      - DETERMINATION OF THE TRANSFER CORIUM DEBRIS: WATER.
        - SIMILAR PRESSURE 100-800 bar.
        - SCHEDULE: SAME AS H2PAR.
  - RELEASE:
    - OBJECTIVE:
      - DESCRIPTION OF THE TRANSFER CORIUM DEBRIS: WATER.
        - THE PROGRAM IS COMPLETED.
        - 1000 TESTS: ANALYSIS AND INTERPRETATION OF THE TESTS.

F.P., ACTINIDES AND SIC RELEASE FROM CORE (VERCORS - EMAIC (1))
- VERCORS HT: VOLATILE AND LESS VOLATILE F.P. RELEASE RATES,
- VERCORS RT: TRANSURANIAN RELEASE.
- EMAIC: CONTROL RODS MATERIALS RELEASE RATES.
- VERCORS HT: HIGH TEMPERATURE: 400-400'C-4 TESTS:
  - 1997: TEST AT HIGH TEMPERATURE (400'; 4 TESTS),
  - HT: TEST AT 600'C-400'; 4 TESTS,
  - HT: TEST AT 800'C-400', 4 TESTS,
  - HT: TEST AT 1000'C-400', 4 TESTS,
  - HT: TEST AT 1200'C-400', 4 TESTS,
  - HT: TEST AT 1400'C-400', 4 TESTS.

- 1999: INTERACTION BETWEEN F.P. LCP, BORIC ACID AND CORIUM DEBRIS.

- 1999: INTERACTION BETWEEN F.P. LCP, BORIC ACID AND CORIUM DEBRIS.
F. P., ACTINIDES AND SIC RELEASE FROM CORE: VERCORS -EMAIC (2)

VERCORS RT: 6 TESTS -
- 1997;
- MWtTH STEAM, NO 5!C, Ti «0°C.
- RT1 TO BE COMPARED WITH HT7, RT1 WITH HT8.

FUTURE TESTS:
- STUDY OF THE RELEASE FROM A DEEP BED.
- USE WITH A HIGH CONVECTION RATE.
- TEST IN FREE STEAM.

EMAIC: 3 TESTS IN LW7.
- TEST: TEMPERATURE 17°C, ATMOSPHERE HI.
- 2nd TEST: TEMPERATURE 17°C, ATMOSPHERE HI.

F. P. RETENTION IN PIPES:
TRANSAT - TUBA

STUDY OF THE F. P. AEROSOL RETENTION IN PIPES (ACHIEVED).

TRANSAT (11 TESTS): SURGE (PRESSURIZER) OR SAFETY INJECTION LINE (TURBULENT REGIME).

- PRESSURE:
- VELOCITY (DEFINITION DECREASE WITH CARBON-GAS VELOCITY).
- TEMPERATURE.

- PARTICLE DISTRIBUTION MUST BE DETERMINED VERY ACCURATELY.
- GOOD AGREEMENT WITH LIFE AND ALAMSI'S DEPOSITION MODEL.

TUBA DIFFUSION PORES (15 TESTS): STEAM GENERATOR, D=3-34 mm.

- GOOD AGREEMENT BETWEEN CALCULATION AND EXPERIMENTAL DATA.
- THERMOPHORESIS (TALBOT MODEL); DIFFUSION PORES (WALDMAN MODEL).

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SEMI-GLOBAL IODINE EXPERIMENT: CAIMAN

MOLECULAR AND ORGANIC IODINE BEHAVIOR IN CONTAINMENT.
- MASS TRANSFER: LIQUID - GAS AND PAINTINGS: GAS.

EXPERIMENTAL DEVICE:
- MVR WITH PLATE (350 mm), PRESENCE OF PAINTING PLATES
- EXPERIMENTAL CONDITIONS:
- TEMPERATURE: 7-14°C, PRESSURE: 0.1-1.0 atm.
- WATER FLOW RATE: 3-5 m/s.

- EXPERIMENTAL PROGRAM:
- INSTALLATION OF SIMILAR EXPERIMENTAL PERSONAL.
- VALIDATION OF THE MODEL AND SIMULATION UNDER RADIATION:
- TEST TRANSFER LIQUID - GAS.
- IODINE REMOVAL FROM THE CONTAINMENT.

FRONTAL AGREEMENTS:
- OECD/RINF,
- IPR: ACE/MACE,
- CCE: 4th FRAMEWORK PROGRAM.

ACHIEVEMENT OF EXPERIMENTS:
- DCH: KAERI (EXPERIMENTS WITH SIMULANTS),
- DDT: KVI (RTH EXPERIMENTS).

TRANSFER OF KNOWLEDGE:
- PHARE, TACIS PROGRAMS.

INTERNATIONAL COOPERATION

- BILATERAL AGREEMENTS:
- NRC, CRF, FIU,
- JRC:

- OECD/RINF,
- IPR: ACE/MACE,
- CCE: 4th FRAMEWORK PROGRAM.

ACHIEVEMENT OF EXPERIMENTS:
- OECD: EXPERIMENTAL STUDIES WITH SIMULANTS.
- ENVIRONMENTAL EFFECTS: EXPERIMENTAL STUDIES.

- TRANSFER OF KNOWLEDGE:
- PHARE, TACIS PROGRAMS.
### Codes and Experiments

#### Codes and Phenomena

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<td>R.C.S. Thermalhydraulics, Core Blump, Vessel Failure</td>
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<td>Aerosols and Vapor F.P. Behavior in R.C.S</td>
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<td>CORIUM IN LOWER HEAD</td>
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<td>WECHSL</td>
<td>ESCADRE (JERICHO mod), TONUS</td>
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<td>SYSINT</td>
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#### Ex Vessel Phenomena

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<td>H2 Distribution/Stratification</td>
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<td>TONUS (currently PLEXUS, CASTEM)</td>
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<td>H2 Recombinbers</td>
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Configurations de l'enceinte TOSQAN

Phase 1

Phase 2

RECOMBINEUR CATALYTIQUE
CAIMAN

zone d'influence

Section d'essai DETHER (principe)
### Schéma de principe de l'installation

**FPT0**

- Facteur de réduction par rapport à un REP : -5000

- Générateur de vapeur
- Circuit de refroidissement du dispositif d'essai
- Dispositif d'essai
- Circuit de refroidissement du cœur

**Cœur du réacteur**

- Assemblage d'essai
  - 20 cr. UO2
  - 1 cr. AlC
  - Long. : 1m

**Enceinte de confinement**

- Condenseurs (74°C)
- Parois (110°C)
- Humidité 40 à 60%

**Puisard**

- (100L/90°C, pH 5)

**Ligne d'injection de vapeur**

---

**Tableau:**

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<th>Type of fuel</th>
<th>Fuel bundle</th>
<th>Primary circuit</th>
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1.5 Overview of Severe Accident Research at KAERI

Sang-Baik Kim and Hee-Dong Kim
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Korea Atomic Energy Research Institute
P.O. Box 105, Yoo-Sung, Taejon, 305-600, Korea

During the first phase (1993-1996) of the current severe accident research program under the Nuclear Reactor Safety Enhancement Project at KAERI, emphasis was placed on the experiment and model development of the separate effects for major severe accident phenomena occurring in the reactor vessel and the reactor cavity when it fails. The second phase of the severe accident research, which was launched in the middle of this year, focuses on development of the integral severe accident analysis code and large scale integral tests for accident management and design improvement for advanced reactor.

Separate effect tests in the first phase are spanning the high pressure melt ejection (HPME) resulting in the direct containment heating (DCH), crust formation during cooling of the high temperature melt, fuel coolant interaction (FCI) in the progress of injecting coolant onto the reactor cavity, and molten core concrete interaction (MCCI). Also, small-scale experiments were performed to visualize the fundamental phenomena of boiling in narrow spaces that may exist between the debris crust and the reactor vessel lower head.

A large-scale experimental program, SONATA-IV (Simulation of Naturally Arrested Thermal Attack In-Vessel), as a main experiment in the second phase of the project (1997-2001), has been developed to investigate the inherent nature of degraded core coolability inside the lower head. In the first phase of SONATA-IV, two separate experiments, so called, LAVA (Lower-plenum Arrested Vessel Attack) and CHFG (Critical Heat Flux in Gap) are being performed to validate the gap cooling mechanism of gap formation and critical heat flux, respectively. Once the relevant physics are interpreted and fully understood, Phase II and III will involve real-material tests with a larger experimental apparatus.

Along with the separate effect experiments, detailed analyses for unresolved issues such as \( \text{H}_2 \) burn and steam explosion using mechanistic models have been performed in Phase I. The LILAC (Lower-Head Integral Analysis Code) code is being developed to simulate phenomena in the reactor vessel lower head including the interaction between the molten corium and RPV wall and internal structures during severe accident. Development of ISAAC (Integral Severe Accident Analysis Code) is an integrated effort of severe accident analysis in Phase II, by combining the assessment of currently available integral analysis codes such as MELCOR and MAAP and the development of new models for unresolved issues. The ISAAC code is aimed to be used to assess severe accident sequences for the future reactor, SMART (Small and Medium Advanced Integral Reactor), being developed at KAERI.
Overview of Severe Accident Research at KAERI

Y.H. Jin, S.B. Kim and H.D. Kim

Korea Atomic Energy Research Institute

Contents

1. Introduction
2. Major Research Activities & Current Status
   • Code Development & Assessment
   • Experimental Studies
   • International Collaboration
3. Summary
**Code Development and Assessment**

**Phase II**
- Development of SA Analyst Codes
- Assessment of Advanced Reactor

**Phase I**
- Development of 3D Eulerian/Lagrangian Code
- Fuel Behavior Tracing with High Accuracy
- Application of Real Geometry

**H₂ Analysis**
- Hydrogen Mixing and Distribution
  - Analysis of the local hydrogen concentration in a compartment using FLUENT code
  - Visualization experiment for H₂ concentration
- Hydrogen Burn Behavior
  - Development of Flame Propagation Speed Correlation
  - One-Dimensional Flame Propagation Code: COMPAC
- Hydrogen Diffusion/Ignition in Two Compartments
  - Examination of the quenching mesh
- Analysis of Detonability
  - Evaluation of Ignition energy of direct initiation of detonation (DID)
  - Evaluation of DDT: to be planned

**Analysis of Fuel Coolant Interaction**
- Development of Expansion Phase Simulation Code
  - Including Void Fraction in Premixing Phase for Total Work Estimation
  - Improving Closure Laws in Expansion Phase
  - Considering Effects of Virtual Mass
- Assessment and Improvement of IFCI 6.0
  - Inclusion of Hydrodynamic Fragmentation Model
  - Analysis of ISP-39 (FARO L-14)
    - UO₂/ZrO₂ Melt Quenching and Mixing Behavior
    - Results: Good Agreement in Pressure and Temperature Rise Trend
    - Underscore the Importance of Jet Modelling and Lagrangian CFD
- Development of 3-D Eulerian/Lagrangian Code: Planning
  - Fuel Behavior Tracing with High Accuracy
  - Application of Real Geometry

**Analysis of Lower Head Integrity**
- Development of Structure Analysis Code: CALF
  - Finite Element Method
  - Heat Transfer in Corium and Lower Head Vessel
  - Elastic and Plastic Deformation of Lower Head Vessel
  - Simulation of the Gap Formation due to Deformation
- Simulation of TMI-2 Reactor Vessel Lower Head, depending on the Initial Corium Temperature and the Gap Formation.
- Supporting Analyses for the SONATA-IV Experiments
  - Thermal Response of the Lower Head Vessel
  - Deformation and Creep Analysis of Lower Head Vessel due to Pressure Load
**Analysis of In-Vessel Corium Cooling**

- Development of LILAC I Beta Version
  - Two Independent Modules to Model Mass and Energy Transfer
  - Debris Behavior Module
    - Heat and Mass Transfer in Oxidic Layer
    - Heat and Mass Transfer in Metallic Layer
    - Heat and Mass Transfer in Particulated Debris Layer
  - Lower Head Heat Structure Module
    - Thermal Behavior Analysis
    - RPV Integrity Analysis using Larson-Miller Method
    - Quenching of Corium Crust by Water Ingression Into the Gap
- Development of Uncertainty Analysis Methodology for LILAC

**Severe Accident Experimental Program**

- Development of LILAC II: Planning

**Assessment Program of MELCOR 1.8**

- Participation in USNRC's MCAP
  - TMLB' Sequence Analysis of Kori-1 NPP (WH PWR)
  - TMLB' Sequence Analysis of Ulchin 3&4 NPP (CE PWR)
- Analysis of PHEBUS Experiments
  - International Collaboration with IPSN, France
  - Code Validation in the Area of Core Melt Progression and Fission
    - Product Release to the Containment
    - Analysis of PHEBUS SFD B9+ and Comparison with ICARE 2 Results
    - Pre & Post Calculation of FPT0
    - Pre & Post Calculation of FPT1
      - Good Agreement in Shroud Temperature Trend
- Analysis of OECD/CSNI ISP-37 (VANAM M3 Experiment)
  - Thermal Hydraulic and Aerosol Transport in the Containment

**Molten Core Concrete Interaction Exp.**

- Objectives
  - Provide MCCI Data of YGN 3&4 Concrete for Model Validation
  - Test Facility
  - YGN 3&4 Concrete Crucible
  - Corium Simulant: SS 304 or Al2O3/Fe Thermite, 20 - 40 kg
  - Decay Heat Simulation using Induction Heater
  - Gas Chromatography
- Results
  - Thermophysical Properties of YGN 3&4 Concrete
  - Gas Composition of YGN 3&4 Concrete during MCCI
  - Ablation Characteristics of YGN 3&4 Concrete
- Applications
  - Validation of MCCI Model in CORCOR Codes
### MCCI Test Matrix

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<th>MHK1-C2</th>
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<td>Sealing Test of Thermal Furnace &amp; C-Type TC</td>
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<td>YGN-2</td>
<td>YGN-1</td>
<td>YGN-2</td>
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### Location of Thermocouples in MEK Tests

#### MEK-51 Test

- MEK-51 Test
- Location of Thermocouples in MEK Tests

#### MEK-52 Test

- MEK-52 Test
- Location of Thermocouples in MEK Tests

### CORCON Validation with MEK1 Test

- CORCON Validation with MEK1 Test

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**POWER-KAERI**

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Page 8
Fuel Coolant Interaction Experiment

- Objectives
  - Find Optimized Flooding Method
  - Providing Minimum Energetic FCI and Coolable State of Corium
- Test Facility
  - Corium Simulant: Tin, 10-20 kg
  - Coolant Injection Mode: Upper Injection, Spray, Side Injection
  - Reaction Chamber: 1/50 Linear Cavity

UO₂ Pouring FCI Experiment

- Objectives
  - Experiment for Premixing and Propagation
  - 10 kg of UO₂ or UO₂/ZrO₂
  - Normal and Elevated Pressure
  - Subcooled and Saturated Coolant

High Pressure Melt Ejection Experiment

- Objectives
  - Quantification of Debris Dispersal Phenomena in Reactor Cavity
  - Improve Reactor Cavity Design to Mitigate DCH
- Test Facility
  - 1/20 Linear Scale Down System
  - Simulant: Water/Wood Metal/Nitrogen
  - Pressure: 0.40 MPa
  - Rupture Dia.: 10-20 mm
  - Cavity: Korli, YGN3&4, Ulchin 1&2, etc
- Results
  - Debris Dispersal Correlation Including Geometric Effect
  - Quantification of the Effect of Reactor Capture Volumes
- Applications
  - DCH Evaluation of CONTAIN & TCE

High Pressure Melt Ejection Test Results For Zion & Ulchin 1&2 Cavity

- Quantification of Debris Dispersal Phenomena In Reactor Cavity
- Improve Reactor Cavity Design to Mitigate DCH
- 1/20 Linear Scale Down System
- Simulant: Water/Wood Metal/Nitrogen
- Pressure: 0.40 MPa
- Rupture Dia.: 10-20 mm
- Cavity: Korli, YGN3&4, Ulchin 1&2, etc
- Debris Dispersal Correlation Including Geometric Effect
- Quantification of the Effect of Reactor Capture Volumes
- Applications
  - DCH Evaluation of CONTAIN & TCE
Molten Pool Coolability Experiment

- Objectives
  - Analyze Crust Formation Process depending on B.C. and Coolant Injection Mode
- Test Facility
  - Phase 1 (Convection Exp.)
    - 45 cm x 15 cm x 15 cm Chamber
    - Wood's Metal as Corium Simulant
  - Phase 2 (Boiling Exp.)
    - 35 cm x 25 cm x 35 cm Chamber
    - Tin as Corium Simulant
  - Measurement of Corium Thickness and Temperatures
- Results
  - Relationship between Crust Thickness and B.C.
  - Modelling of Crust Heat Transfer
- Applications
  - Improvement of CORCON Model

Objectives of SONATA-IV Research

- to understand the mechanism that sustained the integrity of the RPV during the TMI-2 accident,
- to develop and validate models for debris behaviour in the lower plenum and for vessel response during a severe accident,
- to assess the potential for melt retention within the RPV in view of operating and future NPP.

Motivation of SONATA-IV

- A large amount of molten core material (~ 20 ton) relocated to the lower plenum in the TMI-2 Accident. The temperature of hot spot in reactor vessel reached ~ 1300 K.
- However, the reactor vessel did not fail, even though state-of-art analyses using SA analysis codes suggested it should fail.
- Was there an unrecognized "inherent" retention mechanism that might be more generally applicable?

Phased Approach of SONATA-IV Exp.

- Present Stage
  - PHASE I: SONATA-IV (Physical Principle Test)
  - PHASE II: SONATA-IV (Material Properties Test)
  - PHASE III: SONATA-IV (Mixing Process Test)
  - PHASE IV: SONATA-IV (Engineered Test)

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Integrated Effort of SONATA-IV Phase 1 Study

Separate Effect Exp.  Main Experiments  Analysis

- TETRIS: Preliminary Test of Thermal Attack
- VISUL&II: Visualisation in Gap
- CCFL: Counter-Current Flow Limit
- 2-D SLICE: Full Scale High Heat Flux Test
- LAVA: Gap Formation and Cooling
- T/Cs: Thermal Resistance and Heat Transfer
- CHFG: Heat Transfer in Hemispherical Gap
- Scaling
- Development of CHF Correlation

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LAVA Experiment

- Suppress steam explosion and any jet impingement attack
- Use thermite (Fe + Al₂O₃ and Al₂O₃), no sustained heating
- Use delivery conduit and diffusers for large mass
- Test peripheral & central paths for delivery of molten material
- Develop high temperature & gap measurement techniques

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CHFG Experiment

- No correlations are available in literature
- New correlations will account for periodic, turbulent, churn, wavy, and possibly liquid-deficient regimes for gap boiling
- Consider the effects of water depth, debris depth, edge configuration, surface roughness, azimuthal variation, etc.
- Cover both the maximum nucleate boiling & minimum film boiling heat fluxes

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International Collaboration

- USNRC Cooperative Severe Accident Research Program (CSARP)
- Cooperation with U. of Wisconsin
- France PHEBUS-FP Program
- OECD/NEA RASPLAV Program
- OECD/NEA CSNI PWG 2 & 4
- Participation of OECD/NEA ISP
  - ISP-37: VANAM M3
  - ISP-39: FARO L-14
- JRP (Joint Research Program) with IPSN
  - HPME Experiment for French 900 Mwe PWR
  - LILAC Code Development
- Joint Workshop on PSA between JAPAN & KOREA

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Summary

During the Phase I (1993-1997) of the SA Research Program at KAERI:

- The Code System to Analyze Severe Accident in NPPs is Established,
- Integrated Efforts of Separate Effect Tests and Model Development has been carried out.

The Phase II Severe Accident Research Program is planned to:

- Develop the Integral Severe Accident Analysis Code (ISAAC),
- Resolve Issues of In-Vessel Corium Retention and Coolability for AMP and Improvement of Design,
- Enhance Containment Integrity by Considering Ex-vessel Phenomena.
2. Session I

In-Vessel Retention 1

Chairperson: K.Y. Suh (Seoul National Univ.)
Co-chairperson: Y. Abe (Yamagata Univ.)
2.1 IN-VESSEL CORIUM RETENTION : PROPOSAL FOR A DUAL STRATEGY

I. Szabo and P. Richard

Commissariat à l'Energie Atomique - CEA (France)
Division of Nuclear Reactors - DRN
Department of Reactor Studies (DER - C.E. Cadarache)

ABSTRACT

The present paper describes a strategy investigated at CEA/DRN, aiming at retaining the corium in the reactor pressured vessel (RPV). This strategy, based upon the combined use of RPV external cooling and of an in-vessel core-catcher, is therefore referred to as a dual strategy.

During the last few years, promising results have been obtained with regard to the possibility of retaining the corium in vessels of reactor of mid-range operating power (e.g., up to about 600 MWe). For higher power reactors, the ability of in-vessel corium retention by reactor cavity flooding or by other similar RPV external cooling is still to be demonstrated since the higher residual heat to be removed possibly lead to boiling crisis and burn-out.

A core-catcher is proposed to be implemented in the vessel lower head of a future 1400 Me Nuclear Power Plant. The role of this device is to collect and to cool the corium down to a power level at which the external cooling system can efficiently operate, thus achieving definitely the retention of the molten core in reactor vessel.

The in-vessel core-catcher and associated external cooling system are described and discussed. Related R&D presently in progress at CEA/DRN and further R&D needs are also outlined.
In-vessel Corium Retention: Proposal of a "Dual" Strategy

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1 - BACKGROUND AND FRAMEWORK

Terminating LWR severe accidents and mitigating their consequences require the implementation of management strategies to arrest the core melt (corium) progression, preferably inside the reactor pressure vessel (RPV). If the reactor emergency core cooling systems (ECCS) are recovered and still available after the onset of core melting, water can be added to the degraded core to cool it down. In a later phase in-vessel cooling of the core debris relocated in the RPV lower head would be still feasible. The idea of cooling the core debris and confining the molten metal in vessel is very attractive, since in-vessel concerns about containment integrity, subsequent to the ejection of the corium into the containment can be avoided. These concerns include containment basement melt-through, slow over-pressurization by the additional amount of gases generated by an unmitigated molten core-concrete interaction (MCCI), direct containment heating (DCH) by corium dispersion subsequent to a high pressure melt ejection (HPME), and in the event of water present in the reactor cavity, by steam spikes or possible ex-vessel steam explosions.

However, under certain circumstances, adding water to a degraded core may have adverse effects [1], leading, for instance, to increased hydrogen production, changes in core geometry (cladding failure and fuel pellets dispersal through thermal shock), that would complicate the core cooling recovery, induce pressurization of the RCS from steam generation, in-vessel steam explosion, or the recalcitrance of the core if unboiled water is injected. Considerable efforts are currently devoted to overcome or exclude these undesirable effects. As a matter of fact, it is the opinion of some experts, that the difficulties associated with water injection into the RPV can be excluded in applying the so-called "dry" strategy to cool the corium from outside and to retain it in vessel.

Ex-vessel flooding as a strategy for in-vessel molten core retention has been analysed a few years ago by a group of OECD experts, who have issued in 1994 a situation report [2], which has highlighted the need for diversified alternative solutions, e.g. in-vessel molten core cooling strategies and ex-vessel molten core retention systems, the ultimate safety objective being the retention of corium in the containment with no (or minimal) radioactivity release in the environment.

Up to now, the trend in Europe is to take into account - at the very early design stage of future reactors - severe accident scenarios with RPV lower head penetration. Furthermore, from the technical standpoint, in-vessel corium retention is usually considered much more difficult to achieve for large operating power reactors.

As a result, a higher priority has been attributed to ex-vessel corium retention R&D and, at CEA/ORN, large efforts have actually been devoted to ex-vessel core-catcher studies. More particularly, in the frame of the "Innovations-Severe Accident Research Program" (ISARP), application of the so-called "dry" strategy has led to the elaboration of two concepts: the multi-crucible core catcher [3] and the flat corium catcher with promotion of radiative heat transfer [4]. An overview of R&D aimed at demonstrating the feasibility these two concepts was presented at SARJ'96 [5].

Recently, encouraging results have been obtained in in-vessel corium retention strategy provided for mid-range operating power NPP (AP 600 [6]) while implementation of design features for reactor cavity flooding is being done at Lovisa reactor (VVER-440 operating in Finland, ~ 450 MWe) [7]. An assessment of the ability of the FT Calhoun Station design features to external vessel cooling was made [8]. Results of this Probabilistic Risk Assessment showed significant benefits from this management strategy.

Assessment of cavity flooding, made for Sury NPP [9], showed however that the final results (benefits measured against adverse effects) are very sensitive to the values of uncertain parameters (such as possible occurrence of an ex-vessel steam explosion) and that before an evaluation of the flooding strategy can be carried out in a meaningful way, further research on the phenomena associated with the above parameters must be done.

From the brief discussion above, it is important to emphasize that assessment of in-vessel retention strategy must be done on a reactor-specific basis and in an integrated fashion (benefits measured against adverse effects: cost, impact, etc.) and that several remaining issues in core progression need still to be solved by further investigation.

Since the severe accident management objective is to ensure a "continuous and homogeneous" defense in depth against core melt accident consequences, in-vessel retention strategy must aim at arresting the core progression, at any time, from the very early core melt onset to the relocation of a large mass of corium in the RPV lower head. The dual strategy, outlined here, is to be considered as an attempt to meet this objective.

In the following, the basic principles of the dual strategy is presented first. The in-vessel water injection systems, in-vessel core-catcher and associated external cooling system are then described and discussed. Finally, related R&D presently in progress at CEA/ORN and further R&D needs are outlined.

2 - BASIC PRINCIPLES OF THE "DUAL" STRATEGY

The "dual strategy" proposed for application in high operating power range (1400 MWe) reactor is based upon the use of gravity-driven water injection systems:

1) to reflood either a decored core or a corium pool collected in a core-catcher installed in the lower head and;
2) to flood the reactor cavity for ex-vessel cooling purposes.
The basic principles of the dual strategy are derived from lessons learned from the TMI-2 post accident analysis and from recent results stemmed from the worldwide intensive investigation on the potentiality of ex-vessel cooling as a candidate strategy for in-vessel corium retention.

2.1 - Lessons learned from the Three Mile Island Unit-2 (TMI-2) Accident

The very first lesson, learned from TMI-2, is quite obvious and can be summarized as follows: after the core melt onset, water supply is absolutely necessary to retain the corium in a RPV not provided with ex-vessel cooling.

It was fortunate that the TMI-2 Unit 2 primary coolant pumps could be re-started and the primary coolant flow re-injected. However accident more severe than TMI-2 must be considered, in which active means are no more available (Station Black Out). Whenever the active injection systems are recovered, operators are usually required to flood the degraded reactor core, the overall advantages of water injection being considered to exceed possible adverse effects. It is worthwhile to notice that existing reactors are provided with water re-injection systems designed for design basis accidents (DBA). Different Emergency Core Cooling systems (ECCS) are effectively provided and progressively adapted according to the importance of the loss or the reactor inventory (continuous and homogeneous defense). No provision of water resources dedicated for degraded core re-reflooding was envisaged before, since the TMI-2 event, core melt accidents were not to be taken into account in reactor design. For future reactors however, provision of in-containment water resources more appropriate to severe accident management can be anticipated, since the core melt cannot be definitely ensured for all possible severe accident scenarios.

There is then a need for extending the in-containment water resources, making provision for water injection in vessel after the core melt onset.

Note: It is worthwhile to point out that we implicitly assume that the actual water resources are designed to meet the prevention safety objectives (i.e., the overall reactor CDF (core melt frequency) is less than 10^-7/year per reactor) and hence, the additional water resources can be listed as a «severe accident» management and consequence mitigation. Extensive passive injection systems, activated as core melt detection, are then preferred.

Water injection in the RPV during the course of any severe accident is to be achieved preferably by passive means, e.g., by gravity-driven injection systems. This solution, only valid for already designed RCS, will be applied in the dual strategy described hereafter, the «reference» PWR, considered being assumed to be provided with an Automatic Depressurization System (ADS).

A second and complementary «learned» case comes out of post-accident debris and RPV bottom wall samples analyses. Post-accident examinations, performed in the frame of the OECD-NEA-TMI-2-Vessel Investigation Project [10] indicated that a localized hot spot (1 m^2) has existed on the lower head of the RPV after 19 metric tons of molten core material has relocated onto it. The temperature reached 1100°C at this region and remained at that high temperature for approximately 30 minutes before cooling occurred (temperature decrease rate of 10 to 100 K/s). Given under the combined loads of high temperature and high Reactor Coolant System (RCS) pressure (> 8.5 MPa), the lower head did not fail. Post accident calculations, using best-estimate computer codes predicted that under such conditions, the TMI-2 RPV lower head should fail through local or creep rupture. As the vessel did actually not fail, assumption has been made about rapid cooling mechanisms by the water present in the RPV.

The actual mechanisms of how such cooling occurs are not yet known and established. One explanation has been proposed by R. E. Henry and Dube [11]: «the corium draining into the lower head does not wet or adhere to RPV wall and if the vessel wall becomes hot local material creep could create a small but significant cooling path between the debris and the RPV wall. This cooling path could have a characteristic dimension of 1 mm. While this is one explanation for the behavior in the TMI-2 accident, there may be other mechanisms which could aid and control the cooling process, for example the water ingestion through cracks or small gap between frozen core debris and internal structure.

The above explanations still need to be experimentally validated and several facilities were built for these purposes. Experiments [12] performed by R. E. Henry, Fauske & Associates, Inc. (FAI) focused on the potential for an in-vessel cooling mechanism, either as a result of the mechanism proposed by Henry and Dube or water ingress through cracks, or a combination thereof. Several tests were also performed in the ALPHA facility [13] of JAERI devoted to better understanding of the in-vessel cooling mechanisms. In Korea, the SONATA-IV Program [14] has been developed with similar objectives, i.e., to identify in-vessel debris cooling mechanisms.

The answer to the important question of whether the RPV would fail or not in severe accident conditions which are not TMI-2 like, need obviously further R&D than the ones summarized above. During the last few years, comprehensive in-vessel retention strategies have been developed. They are briefly reviewed hereafter.

2.2 - Overview of current in-vessel retention strategies

Although, the idea of retaining the corium in the reactor vessel is very attractive, very few strategies have been proposed up to now for application in future reactors. They can be classified into two main categories:

* External cooling of the RPV, e.g., by reactor cavity flooding
* Implementation of a core-catcher inside the RPV lower head.

As mentioned above, the first strategy is intensively investigated for new LWR designs (AP600) or for retrofitting purposes in existing LWRs (Lovisa) while the second type of solutions can be found mostly in patents, proposed mainly for Fast Breeder Reactor (FBR) application (The Super-Phoenix FBR is provided with an in-vessel core-catcher) and, more rarely, for LWRs.

External cooling of the RPV

In the concept of external passive cooling of the core melt relocated in the RPV lower head, cooling water is made available by flooding the reactor cavity during a severe accident. As the lower head is heated by the relocated corium melt, the decay heat generated in the melt is removed upward by convection and/or thermal radiation and downward from the external lower surface of the RPV by heating the water in the flooded cavity. Sustaining the flooding of the reactor cavity is usually achieved by providing the feed-back flow paths which direct the water resulting from vapor condensation in the reactor containment back to the reactor cavity.

The idea of in-vessel retention by external cooling has been investigated both theoretically and experimentally by several authors (T. G. Theofanous et al. [6], E. A. Hodge [17], H. Pohl and V. K. Shio [48], Hawkins and O'Brien [19], J. M. Solier et al. [20]). The related experimental facilities were designed to study local CHF on the external bottom (CYBL) [21], ULFUC [22], SULTAN [23], and CORIUM [24], etc.) and corium pool thermohydraulics (COPO [25], ACOPA and Mini-AEPT [26, BALL [24], etc.). Furthermore, new computer codes (COUPLAN [27], Cipion 2D [29], LOWHED [29], etc.), have been developed for corium pool behavior prediction while two phase flow codes (e.g., the CATHARE code [30]) are usually used for coolant thermohydraulic calculations. Due to this worldwide interest and related intensive activity, in-vessel retention issue can now be considered as to be ripe for decisive steps, such as those needed in the regulatory context.
Implementing a core-catcher in the RPV lower head

Well before TMI-2, regulatory authorities consider that typical (bounding) hypothetical core melt accidents must be taken into account in Fast Breeder Reactor (FBR) designs. To meet this safety objective, intensive R&D have been made since the early 70s on the coolability of core debris beds and several concepts of core-catcher to be implemented in the FBR vessels were proposed. After the TMI-2 events, a few concepts have been patented as possible devices to be implemented in PWRs for in-vessel corium retention. However none of these concepts was theoretically and experimentally investigated and their applicability to future reactors is questionable.

2.3 - Basic principles of the « dual » strategy

The « dual » strategy is an attempt to take benefits from both the lessons learned from the TMI-2 accident and the intensive R&D efforts which are being done worldwide on in-vessel corium retention concepts.

From TMI-2, we consider that the conditions identified by Henry and Dube [11] and also, by Wolf and Range [10], would actually favor the cooling of the corium relocated in the RPV lower head:

- presence of water in vessel
- no adherence of the relocated corium to the RPV wall (no wetting) and existence of a gap between the corium crust and the RPV wall,
- increase of heat exchange surface by cracks formation in debris (water ingression)

Such favorable conditions can happen - in a random manner however - during severe accidents (TMI-2-like) which may occur in existing reactors. For future reactors, the RPV of which can be modified (within the limit of a reasonable extra cost), an internal device, we called in-vessel core-catcher, can be envisaged, aiming at providing « by design » such favorable conditions.

From the intensive R&D on cavity flooding, we take due note that this promising in-vessel retention concept is expected to be validated for mid-range operating power reactors. For higher power reactor however, the specific internal power of the molten core may generate at the RPV lower head bottom local heat fluxes higher than critical heat flux (Boiling Crisis not totally excluded). Furthermore ablation of the RPV wall may induce a global creep rupture of the RPV lower head.

An example of practical application of the « dual » strategy is described and discussed in the following. It is to be considered as a very preliminary conceptual design, aimed mainly:

- at illustrating the main features of the strategy, the methodology pursued and the R&D scheme,
- at identifying both the theoretical and experimental means available at CEA/DRN for in-vessel retention study, and finally,
- at identifying the remaining issues and further R&D needs.

3 - CONCEPTUAL STUDIES OF A «DUAL» IN-VESSEL DETENTION (DIVER) SYSTEM

The methodology used to investigate the feasibility of the dual in-vessel retention concept is progressive and includes several steps as shown in Figure 1. In the following, presentation of analyses, R&D works and preliminary results will reflect the same stepwise approach.

![Figure 1 - Study scheme of the Dual In-Vessel Retention system](image-url)
cannot be cooled (worse case than TMI-2), the remaining part of the initial pool will relocate (e.g., reflector, etc.) (Figure 2). Since the total mass (19 tons in TMI-2) and mass flow-rate of the first relocation depend on the side break location and the volume of the initial corium pool, there is a large number of possible relocation scenarios. The corium relocated at the vessel periphery into the lower plenum. The initial corium pool is voided at the upper part at higher temperature, thinning and weakening the crust which will break at some location. The way the debris relocate from the core region into the lower plenum is important for the in-vessel core catcher design. A TMI-2 like melt release scenario is usually considered: in a first stage core debris accumulate above a bold shape lower crust resulting from resolidified core material of low melting point. A corium pool develop then, surrounded by crust. Convection inside the pool bring its constituents of the core is equivalent oxidation of 75% of the Zircaloy content of the cladding. The way the core melt in the reactor cavity can be arranged to accommodate the ex-vessel cooling system (inlets for water supply and outlets for generated vapor or two phase flow).

3.1.2 - Melt relocation scenarios

In some « happy-end case », the badly degraded core would be cooled by in vessel water injection without relocation into the lower plenum. Worse case must however be considered and relocation of core debris can not be totally excluded, even with water injection in the RPV after the core melt onset. As a result, the post accident reconstitution of the TMI-2 accident (19 tons in about 2 minutes).

The approach we actually use to define in-vessel relocation scenarios is quite similar to the ones used in ex-vessel corium release scenarios described in [3].

First relocation sequence

We consider that higher are the total mass relocated and the mass flow-rate during the first sequence, greater are the challenges on the in-vessel core catcher. As a result of this, the TMI-2 like first relocation sequences briefly described above can be bound by the following first sequence:

Mass and composition of the corium relocated

1) Before the first relocation, the molten material (in the pool) can reach 60% (100,000 kg) of the initial core inventory.

2) All the content of the pool relocate (e.g., through several quasi-simultaneous breaks, a "longitudinal (vertical) break" or through a more improbable but possible "bottom break"). Since the corium progression is essentially tridimensional, the location of the break, its propagation kinetics and initial size cannot be mechanistically predicted yet (considering the present state of the art) and more or less sophisticated assumptions have to be made (D effects did exist in TMI-2 leading to a single side jet at the upper part of the corium pool instead of an "axi-symmetric gross failure" of the upper crust). The assumptions we made above are quite simple and can obviously be improved further.

3) From the onset of the core melting to the pool dislocation, oxidation of the metallic constituents of the core is equivalent oxidation of 75% of the Zircaloy content of the cladding.

Duration of the relocation

It is assumed that the mass flow-rate during relocation is of the same order of the one deduced from the post accident reconstitution of the TMI-2 accident (19 tons in about 2 minutes). As a result, the duration of the first relocation is about 10 minutes for the reference reactor.

Single/multiple relocation flow paths:

Corium flowing from the core region into the lower plenum may have single or multiple flow path and can enter the plenum either near the RPV wall or near the RPV axis.

In summary, the loads and challenges that the in-vessel core catcher has to cope with during the first relocation sequence correspond to continuous single or multiple axisial corium flows welding a corium pool of about 100,000 tons in ten minutes. The main challenges during this first relocation phase are: in-vessel steam explosion and core catcher structure attacks.

Sequences following the first one are more difficult to predict. Upward convective and radiative energy transfers continue to heat up and melt the structures above the debris accumulated in the lower plenum. Heat up would result either in liquid corium mass flowing down or in dislocation of partly destroyed structures which are still solid. "Late" slumps of solid and/or liquid debris are essentially random.

However, if the in-vessel cooling remains still insufficient to arrest the in-vessel material melt-out, we can arrive at an extremely situation where most of the core material, the core support plate, portions of the core barrel and reflector, etc. will be relocated in the lower plenum. The resulting corium pool contains a higher fraction of non oxidised metals than the one developed previously in the core region. Solidification can occur, regrouping the pool in two parts: heavy oxide pool at the bottom overspun by a metallic layer. That is this extreme configuration which is generally examined in detail and quantified as specifications for design purposes. In such situation, it is then necessary to define some simplified relocation scenario with the aim to bound - in a best estimate but conservative manner - the various loads the in-vessel retention system has to cope with.

Figure 2 - Examples of « Generic » relocation sequences

duration of the relocation: 

Sequences following the first one are more difficult to predict. Upward convective and radiative energy transfers continue to heat up and melt the structures above the debris accumulated in the lower plenum. Heat up would result either in liquid corium mass flowing down or in dislocation of partly destroyed structures which are still solid. "Late" slumps of solid and/or liquid debris are essentially random.

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are how to cool the corium pool and confine it in vessel or in other words how the in-vessel retention concept fulfills its assigned mission.

For the reference reactor, the characteristics of the corium collected at the end phase are similar to those of scenario B described in reference [3 and 38]

| UO₂  | 75 000 kg | 37.7 % |
| Zr   | 4 500 kg  | 3.0 %  |
| SS   | 17 500 kg | 8.5 %  |
| Oxide power: 16 MW |
| Metal power: 2 MW |

Table 1: Corium characteristics for scenario B

3.2 - Basic concept: main features and operation principle

The Dual In-Vessel Retention system studied at CEA/DRN is schematically shown on figure 3

This system includes mainly:

- an In-vessel injection water storage tank (IVIWST), providing gravity-driven injection of water in the RPV;
- a second and much larger reactor cavity flooding water storage tank (RCFWST) supplying cooling water for reactor cavity flooding purpose and possibly water injection in the RPV;
- an In-vessel core-catcher

Cooling water supply systems

Water supply from the two water storage tanks is controlled by In-Vessel Injection valves (IVI valves) and Reactor Cavity Flooding valves (RCF valves). In vessel injection lines can end either in the cold/hot legs of the RCS or in the reactor pressure vessel (AP 600 like configuration: direct in-vessel injection nozzle).

At the core melt onset the primary system is depressurised down to a pressure in equilibrium with the containment atmosphere. Loss of water inventory by depressurisation and afterwards by vaporisation by the core debris can be compensated first by water injection from the smaller water storage tank (volume of IVIWST is about a hundred cubic meters). Meanwhile the cavity begins to be flooded. Once the IVIWST is empty, water injection in the vessel continue via the opening of the IVI valve of the RCFWST, located at sufficiently high elevation with regards to the injection inlets. The vapor produced in vessel flow out the RCS through the ADS opened valves or through the break in primary system, if any. Vaporisation may increase the primary pressure and stop from time to time the water injection in the vessel. Thus, the water injection in the vessel may have some oscillatory or intermittent behaviour. Nevertheless, it is expected that the two water storage tanks can ensure the presence of water inside the vessel for the short term (a few hours after core melt onset) and outside the reactor vessel for the long term (volume of RCFWST is about 1000 m³). In principle, water supply can be achieved by the larger tank RCFWST, since it can both inject water in-vessel and flood the reactor cavity. The use of an additional smaller IVIWST, aimed at:

- providing the possibility to dissolve boron (re-criticality issue) and also, some "additives", such as surfactants [31,32,34], which would modify specific water properties (e.g., surface tension) with a view to reduce the steam explosion risk at the very early phase of core degradation,
- providing the possibility to install this reservoir at a higher elevation than the RCFWST in order to increase the driving force for water injection in the reactor vessel.

Figure 3 - Dual In-Vessel Retention (DIVER) system

The volume of the IVIWST can be roughly estimated, using a similar approach than that used by S. Langer for TMI-2 accident analysis [34]. It would be of the order of the RPV coolant volume. More detailed further investigation aiming at optimising volumes of IVIWST and RCFWST should be done on a reactor-specific basis.

Since the IVIWST volume is small, water quality control (borated water with additives) is thought to be easier and seismic constraints less demanding (reservoir can be installed at high elevation).

In-vessel core-catcher

The In-vessel core-catcher (figure 4) is aimed to ensure by design the conditions identified in TMI-2 VIP as favourable to in-vessel corium cooling.
As described above, presence of water in the reactor vessel, after the core has melt, is ensured by injection at the early phase by the IVIWSST and afterwards by the RCFWST.

Non adherence of the corium to the RPV lower head bottom and existence of a gap is anticipated by design by collecting the cerium in a crucible, which can be either suspended or supported by radial "stiffeners" arranged in such a way to allow water ingress beneath the crucible for cooling purpose.

The necessary increase of the heat exchange surface between corium and cooling water is achieved in providing "water ingressions" from the gap into the corium debris bed/pool accumulated inside the crucible. For this purpose, vertical tubes, the lateral wall of which is perforated, develop in the crucible, having melted all its internal structures (figure 5). Since the diameter of the inlet of the vertical tubes is small (a few centimetres) by design, it is expected that liquid corium will freeze at this location and cannot enter the gap. At this late phase, the power density of the corium is lower and cooling of the corium can be achieved by external cooling even if the in-vessel core-catcher happens to be deficient.

In the case of a severe accident, liquid and solid core debris flows/falls down into the core-catcher. Corium flows at the periphery are directed into the core catcher by means of the collector. Internal structures such as the core support plate, the flow distributor plate and the bottom plate (a device devoted to direct the primary coolant flow from the down-comer to the core during normal operation of the reactors are partly destroyed as well as some vertical tubes. Relocation of the corium in the crucible will cause natural circulation of water within the gap and water ingestion into the corium debris bed/pool through the vertical tubes (figure 5). At the end phase, a large corium pool (possibly stratified) develop in the crucible, having melted all its internal structures (figure 6). Since the diameter of the inlet of the vertical tubes is small (a few centimetres) by design, it is expected that liquid corium will freeze at this location and cannot enter the gap. At this late phase, the power density of the corium is lower and cooling of the corium can be achieved by external cooling even if the in-vessel core-catcher happens to be deficient.

Figure 4 - In-vessel core-catcher concept

4 - FUNCTIONAL ANALYSIS AND ANALYTICAL STUDIES

4.1 Functional analysis and requirements for reactor application

Operability of the in-vessel injection systems depend on both the reliability and the efficiency of the ADS and on possible adverse effects (e.g., back-pressure in RPV due to vapour production, hydrogen generation and associated energy release, etc.). The main requirements on these system design are:

- effective and efficient injection into the vessel;
- passive operation or no large electric power supply need;
- no significant adverse effects;
- easily accommodated in containment building;
- no prohibitive extra cost.

Functional requirements of the in-vessel core-catcher are quite similar to those already identified in [3] for ex-vessel core-catcher: one, collect (1st requirement) all the molten core materials, two, confine (2nd requirement) and prevent them to penetrate the RPV lower head and sheet, to cool (3rd requirement) them down to solidification, in association with the ex-vessel cooling through reactor cavity flooding.

The core-catcher has also to survive (4th requirement) severe accident conditions or - at least - to operate efficiently during the course of the accident.

The whole in-vessel retention system must be designed to prevent excessive fission products release and dispel the present (5th requirement), to operate passively or without needing high electric power supply (6th requirement). It must not induce adverse effects (7th requirement) both in reactor normal operation and in accidental conditions. There are also some "recommendations" related to the demonstration of the feasibility of the concept and at a later stage, to the industrial design. It is desirable for the concept:

- to be based on proven and simple principles, i.e., one, not to induce novel, complex and costly R&D; two, to be of moderate cost, three, of easy inspection, maintenance; four, to be easily accepted by the safety authority (licensing) and last but not least, to be accepted by the public (public acceptance).
In the following, discussions are focused mainly on requirements related to the in-vessel core-catcher. Practical and operational recommendations as well as economic and socio-cultural aspects are not discussed further during this very preliminary stage of our studies. Requirements on the in-vessel water injection system have been mentioned above as well as a brief description of the operation of the IVIWSST and RCFWST, designed to meet the requirements.

Collecting the corium (1): As the core catcher is installed just beneath the core, only peripheral corium flows, near the RPV wall, may fill the gap. The collector is designed to derive peripheral flows in the crucible. Since the dilution of the structures by corium, during fast transient relocations under water is expected to be small (cf. TMI-2), the collector can be made of refractory material covered with external stainless steel (SS) liner, or more simply, made of stainless steel (SS).

Confining the corium (2): this function is assigned to the crucible, made of a refractory layer (e.g., ZrO₂ or MgO layer) sandwiched by two SS water tight layers (figure 7). The tightness of the crucible to primary water is needed during the life time of the reactor since interaction of the refractory layer with primary water is expected to be small (Cf. TMI-2), the collector can be made of refractory material covered with external stainless steel (SS) liner, or more simply, made of stainless steel (SS).

Cooling the corium (3): at the early phase of relocation this function is ensured by natural circulation of water through both the « gap » and the vertical tubes. The crucible is cooled externally and a major part of the gap flow is injected from below inside the relocated corium bed/pool. At that phase, one

**Figure 7 - In-vessel crucible: example of architecture (« artist's » view, not to scale)**

Cooling the corium (3): at the early phase of relocation this function is ensured by natural circulation of water through both the « gap » and the vertical tubes. The crucible is cooled externally and a major part of the gap flow is injected from below inside the relocated corium bed/pool. At that phase, one

important question is how the corium energy can be removed upwards via the primary system alone since downward energy flow (i.e., through the RPV bottom head) is blocked. In other words, the control and removal of the vapor and gas generated in-vessel are of crucial importance. If the openings (depressurization valves and break) are not sufficient, in vessel pressure increase may prevent injection water from entering. Due to « steam binding effects » water injection in vessel may be intermittent. At the late phase, the vertical tubes is « out of service » and cooling of the crucible continues via the gap flows. The RPV wall being externally cooled by cavity flooding, condensation is enhanced in inner surface generating a water film flowing downwards to the gap.

Survive to severe accident conditions (4): in stated above the main challenges during the first relocation are: (oxidic) corium jet impingement and possible steam explosions. These are generic problems still unresolved, even for existing reactors. Impingement of corium flows on the lower plenum as its periphery or at it centre continues to be intensively investigated, the parameters being the corium characteristics, the size of the « jets ». As shown on figure 5, the corium transient flow encountered several obstacles before reaching the crucible bottom. Direct impingement of jets on it seems unlikely.

Steam explosion risk : Besides the use of additives (whenever their efficiency can be assessed) in the early phase of in-vessel injection, the second means consists in using specific internal structure (honeycomb-like boxes, grids, etc.) with a view to divide the internal water volume inside the crucible in (communicating) cells containing a small amount of water each. When impinging the transient flow from above, this cellular structure would divide an inlet corium flow in several smaller ones which will relocate in different «cells». As a consequence, the interacting masses (corium and water) in each cell would be small and controllably, the energy release. It is thought that both producing conditions favorable to a steam explosion and its propagation to the entire volume of cellular structure of the crucible is less probable than for the configuration of a lower plenum void of structure and full of water. Effects of a grid on the likelihood of steam explosion have been investigated in ALPHEA facility but no definitive conclusions can be drawn from several tests performed with grids, since explosions were « randomly » observed in some of the tests and not in others. Further R&D are still needed in order to determine the effects of obstacles on jet fragmentation, on course mixing, propagation and expansion phases of the steam explosion mechanisms. It is worthwhile to point out that it is a real « generic » problem since the liquid corium mostly flows along or over rods and internal structures in the core region and impinges generally on solid obstacles before entering the lower plenum which is also provided with internal structures.

High Fission Products retention (5)

To meet this requirement, one needs to freeze rather quickly the core debris, to prevent corium and FP from being released and entrained by cooling flow turbulence, and subsequently, from being transported into the containment atmosphere through the ADS valves and RCS break. Evaluation of this source term can only be done at a later phase of design.

Pasive operation (6):

Dedicated power supply are provided in order to activate the valves, even in case of a station blackout; the water content of the RCFWST is maintained almost constant up to the long term by draining back water resulting from the vapour condensation in the containment compartments. The containment is provided with means (e.g., heat exchangers) dedicated for ultimate heat removal.

No adverse effects (7)

The in-vessel core-catcher must not disturb the normal operation of the reactor, most particularly the nominal flow rate and distribution at the core inlet. As suggested on figure 4, a « bottom plate » could be installed above the crucible with a view to provide similar paths to the RCS flow which merge
As shown in figure 3, the gravity-driven water injection in-vessel lines - from the IVIWST or from the RCFWST (flow pressure end ~ 0.1 MPa) to the cold legs or RPV down-comer (high pressure end ~ 15 MPa) - need to be controlled by appropriate valves, which are permanently submitted to a large differential pressure. Much attention has then to be put on the design of these valves and their actuation, and furthermore, consequences of a possible inadvertent actuation or failure during normal operation or accidents, must be considered.

4.2 Analytical studies

4.2.1 - Selection of crucible wall materials

The choice of appropriate material for in-vessel core catchers must take into account specific constraints due to the contact with the corium in presence of water:

* Good resistance against physical and chemical attacks by molten core materials (both oxides and metals);
* Good stability up to very high temperatures and good resistance to thermal shocks;
* Good overall mechanical behaviour,
* No significant modification of characteristics during normal operation of the reactor, more particularly under neutron and gamma irradiation and;
* Primary coolant chemical compatibility or Provision of a protective shell (e.g. sandwiched in water-tight SS layers).

From the material survey and analysis [37], performed previously in the frame of the CEA/DRN ex-vessel core-catcher studies, cerainics such as zirconia (ZrO₂) and magnesia (MgO) are the most promising materials.

4.2.2 - Preliminary calculations : thickness of the crucible wall

Rough estimations of the crucible wall thickness has been performed, considering the extreme situation of figure 8, i.e. the « end » phase of the corium melt relocation (internal structures of the core-catcher such as the vertical perforated tubes and the inner SS liner are assumed to be entirely melt and mixed in the stratified corium pool).

![Figure 8 - End phase considered in preliminary calculations](image)

Water Circulation

Crucible

Metallic Phase

Stiffeners

Cylindrical Part

Metamorphic Part

(a) Simplified geometry modelled

(b) Heat-exchange paths considered

Moreover, as shown in figure 10, the predicted thermal behaviour of the crucible is satisfactory. The corium temperature decreases regularly and complete solidification of the oxidic phase of the corium...
Ol (supposed here to be the minimum solidus temperature of UO2/ZrO2 mixture, i. e. 2500 °C) is obtained after 45 000 seconds (12.5 hours).

Figure 10 - Example of F.E. calculation results : Temperature transients in the crucible

4.3 Other analytical studies

Other analytical studies presently in progress are related to the mechanical behaviour of the crucible and to thermo-physical aspects of corium-refractory material interactions. The latter item concerns both in-vessel core catcher and ex-vessel core-catcher. Results obtained from the CEA/DRN ongoing R&D program on ex-vessel corium-materials interactions will be also useful in in-vessel core catcher studies.

5. OVERALL BEHAVIOUR STUDIES

5.1 - Identification of existing means and further needs

Since the in-vessel retention study is in its very early stage at CEA/DRN, only qualitative analysis have been achieved with regards to the overall behaviour of the Dual In-Vessel Retention (DIVER) system. The straightforward transposition of CEA/DRN ex-vessel core-catcher strategy allows the identification of existing CEA/DRN R&D which can be useful for in-vessel study purposes. This is schematically shown on figure 11.

5.2 - Ongoing works

The in-vessel retention studies under investigation at CEA/DRN are:

- Multi-D global behaviour of the in-vessel crucible investigated by means of integrated TOLBIAC/CATHARE codes
- Tentative integrated calculations of the whole DIVER system

The first item has been successively solved for ex-vessel crucible [40] by using the PVM (Parallel Virtual Machines) technique to couple the TOLBIAC code (corium pool thermalhydraulics) with the CATHARE code (external coolant thermalhydraulics). Adaptation to the in-vessel crucible geometry and specification of (approximate) initial and boundary conditions in vessel are already achieved. First calculations results are expected in early 1998.

Figure 11 - Theoretical and experimental means available at CEA/DRN in relation with the Dual In-vessel Retention system study

A preliminary study concerning the second item was recently presented at ICONE 5 [38]. The behaviour of the reference reactor assumed to be equipped with an in-vessel crucible has been investigated by means of the MAAP 4 computer code [41]. Although the structure of the code does not allow the user to easily model an in-vessel crucible, strong similarities do exist between the in-vessel crucible (figure 9, end phase) and the model implemented in MAAP which is based upon the same postulated explanation of the coolability of the corium relocated in the TMI-2 RPV bottom head. As a result, the in-vessel crucible can be singly modelled, by setting the characteristics of the lower crust to be specified in MAAP input data deck, identical to those of the zirconia or magnesia, which the in-vessel crucible is made of. Similarly the dimension of the gap below the crucible can also be introduced in MAAP.

In these preliminary calculations, the accident scenario considered is of TMLB type: Station Black-out (SBO) with a subsequent pump seal leak of 1.6 cm² after 2 700 seconds and a hot leg break at about 3 hours. At the last stage (> 3 hours) of this scenario, the RCS is entirely depressurised (without ADS actuation). The efficiency of the in-vessel core-catcher is evaluated by analysing the results of three calculations: the first one, performed without the core-catcher integrated in the RPV lower head is considered as the « reference case ». In the second one, the reactor cavity flooding is actuated alone (i. e. without in vessel crucible modelled) and, in the third one, both reactor cavity flooding and the in-vessel crucible are considered. As shown on figure 12, the results obtained reflected the trends expected : Implementation of an in-vessel crucible prevent the RPV lower head from failing.
5.3 - Further R&D needs

Most of the R&D needed for demonstration purposes of the feasibility of the DIVER system are "generic" in that they are of concerns for most in-vessel and ex-vessel corium retention strategies. They are mainly related to:

- Resolution of remaining issues in in-vessel corium progression; more particularly:
  - Coolability of in-vessel corium beds/pools and in-vessel steam explosion risk.
- Search for appropriate refractory materials, able to arrest the corium progression, including possible impingement of corium jets/ex-vessel core-catcher studies, the results of which can be applied to in-vessel retention systems.

The R&D needs specific to the DIVER system are related to:

- In-vessel passive injection systems dedicated to severe accident management;
- Protective or mitigative features devoted to ensuring core-catcher survival and satisfactory operation after a postulated in-vessel steam explosion. Preliminary candidate solutions, identified here (e.g., mitigation by additives and cellular structure core-catcher, shock absorbers, etc.) have obviously to be investigated further. More particularly, the likelihood of steam explosions, subsequent to a corium flow into a water pool containing immersed structures (obstacles) is to be experimentally investigated in complement to the more usual tests in which the corium jets fragment in a shallow or deep pool free of immersed structures.

6. CONCLUSIONS

The "dual strategy", based upon the combined use of:

- Water injection in-vessel, at the core-melt onset detection, to cool down either a badly degraded core or a large mass of debris collected in an internal core-catcher,
- Cavity flooding for ex-vessel cooling purpose,

is proposed here as a candidate strategy for in-vessel corium retention applicable for high operating power-range LWRs (~1400 MWe).

An example Dual In-vessel Retention (DIVER) system is described, aiming at illustrating the methodological approach pursued and the study themes envisaged. Identification of both theoretical and experimental means, available at CEA/DRN, which can be useful for the feasibility study of the DIVER concept, shows that most of the "generic" R&D works, originally developed at CEA/DRN for ex-vessel core-catcher studies, can also be used for predictive calculations and experimental validation of the DIVER system.

Further R&D needs include more comprehensive in-depth investigations on:

- In-containment water resources provision and management, and more particularly, provision for in-vessel injection after the core melt onset;
- Protective and mitigative features against in-vessel explosion;
- Reduction (or exclusion) of potential adverse effects on normal and incidental operation of the reactor;
- Reduction of cost impacts.

These more detailed studies are envisaged at CEA/DRN for the next few years.

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2.2 Thermomechanical Analysis for Advanced In-Vessel Retention Design

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Abstract

In the TMI-2 accident, approximately twenty (20) tons of molten core material drained into the lower plenum. Early advanced light water reactor (LWR) designs assumed a lower head failure and incorporated various measures for ex-vessel accident mitigation. However, one of the major findings from the TMI-2 Vessel Investigation Project was that one part of the reactor lower head wall estimated to have attained a temperature of 1100 °C for about 30 minutes has seemingly experienced a comparatively rapid cooldown with no major threat to the vessel integrity. In this regard, recent empirical and analytical studies have shifted interests to such in-vessel retention designs or strategies as reactor cavity flooding, in-vessel flooding and engineered gap cooling of the vessel. Accurate thermohydrodynamic and creep deformation modeling and rupture prediction are the key to the success in developing practically useful in-vessel accident management strategies. This paper presents some of the state-of-the-art thermal and mechanical results from improved creep and rupture modeling via validation of commercially-available computational schemes for heavy sections with thermal gradient and accumulation of the accurate creep strain and rupture database. As an advanced in-vessel design concept, this work also presents the COrium Attack Syndrome Immunization Structures (COASIS) that are being developed as prospective in-vessel retention devices for a next-generation LWR in concert with existing ex-vessel management measures. Both the engineered gap structures in-vessel (COASISI) and ex-vessel (COASISO) are demonstrated to maintain effective heat transfer geometry during molten core debris attack when applied to the TMI-2 and the Korean Standard Nuclear Power Plant (KSNPP) reactors. The likelihood of lower head creep rupture during a severe accident is found to be significantly suppressed by the COASIS options.

I. INTRODUCTION

Metallographic studies in the TMI-2 Vessel Investigation Project [1] determined that a significant part of the reactor vessel lower head was substantially overheated. Specifically, one part of the vessel wall is estimated to have reached a temperature of at least 1100 °C for about 30 minutes and then experienced a comparatively rapid cooldown. The cause and nature of this rapid cooling are of considerable importance since the TMI-2 vessel was at a pressure of 11 MPa during this time. With this internal pressure, the vessel wall would have undergone significant creep and perhaps eventual rupture, had it been sustained at 1100 °C for an extensive interval. Consequently, this rapid cooling of the vessel at some time after four hours into the accident may have been responsible for maintaining the vessel integrity. Major research programs [2,3,4] have recently been developed to investigate this inherent nature of degraded core coolability inside the lower head due to boiling in a narrow gap between the debris crust and the vessel wall [5,6,7] coupled with the primary system heatup and degradation models [8,9].

The TMI-2 findings [10,11] have led to a nuclear industry standard for advanced LWR development to add measures to mitigate the progression of severe accidents. The first kind of
severe accident management strategy was developed in mid 1980's assuming the lower head failure focusing on the capture and cooling of the escaped core debris which otherwise would react with concrete floor to generate additional heat and pressure [12]. The reactor cavity flooding, but not wetting the lower head, has been introduced as a mitigative measure to core debris-concrete reaction. This so-called ex-vessel management approach, however, has to cope with outstanding severe accident issues ranging from core debris induced steam explosion to containment direct heating.

In early 1990's, the lower head protection methods were sought aiming at the retention of core debris within the vessel. Varying methods of the proposed in-vessel retention design are portrayed in Figure 1. Several investigators have suggested that ex-vessel flooding combined with reactor depressurization is adequate to maintain the lower head integrity [13]. The in-vessel management approach is now seen as a key safety feature of recent advanced LWR designs, such as AP-600 [14].

![Figure 1. In-Vessel Retention Strategies for LWRs](image)

(a) in-vessel flooding  (b) engineered gap cooling  (c) ex-vessel flooding (d) external spraying

The Korean Next Generation Reactor (KNGR) development has been undertaken with the initiative of Korea Electric Power Corporation since early 1990's. The principal objective of the KNGR development is to significantly improve the safety and economy over those of current LWRs. Construction of the first KNGR is expected in early 2000s. In order to take advantage of existing foundation with the KSNPP technology, a reference design for the KNGR was designated to be System 80+ of ABB-CE [15] which had adopted the ex-vessel management approach. The reactor cavity flooding is deliberately limited to keep the outer wall of lower head dry for the sake of investment protection. Reactor cavity and containment structures are designed to withstand potential steam explosion in the flooded cavity in the event of core debris falling.

Although System 80+ design is the current baseline of severe accident management strategy for the KNGR, in-vessel management strategies are being pursued in parallel. Options for the latter avenue may include ex-vessel flooding to wetting and engineered gap cooling techniques, as will be described in this paper. Development of in-vessel management strategies presents nuclear materials challenges in that understanding of high temperature behavior of core debris, core structural materials and lower head has to be improved significantly, as illustrated in the Vessel Investigation Program (VIP) for TMI-2 accident by disagreement between creep analysis results [11]. By examining the source of disagreement in the lower head creep analysis results between investigators, this paper identifies important factors affecting the prediction accuracy. Evolving options for the in-vessel management for KNGR and their effectiveness in the lower head protection are also described from the thermomechanical standpoint.

II. CRITICAL REVIEW OF LOWER HEAD CREEP ANALYSIS IN OECD VIP

Under the auspices of the OECD-NEA, TMI-2 VIP was carried out from 1988 through 1993 [11]. The program was aimed at determining the likelihood of the creep rupture failure of the TMI-2 lower head by a multi-disciplinary effort. Detailed results of VIP have been extensively published [10,11]. Major achievements among others include 1) determination of approximate
vessel temperature distribution and history, 2) development of materials property database and constitutive models, and 3) margin to failure assessment. Margin to failure assessment effort was, however, met with limited success. All three investigators in the effort predicted significantly large creep strain whereas the actual TMI-2 lower head deformation was practically unnoticeable [16]. Furthermore there was no reasonable agreement in creep rupture prediction between investigators. Detailed benchmark on analysis accuracy was apparently absent until the end of the VIP.

The VIP creep analyses by three independent investigations are summarized in Table 1. Among others an axi-symmetric two dimensional model including the hot spot effect was chosen as the basis for the comparison since the particular case is covered in all the three investigations and because it is considered to be a good representation of the actual TMI-2 situation. Analysis method and data have been reviewed by repeating the analysis using the commercial finite element analysis programs: ABAQUS version 5.5 [17] and ANSYS version 5.2 [18] executed on workstations. Both programs are capable of handling a large strain non-linear visco-plastic problem typical of the lower head rupture case. In the later stage, however, only ABAQUS is used since ANASYS ran much slower.

A series of analyses described in Table 2 were made to quantify the impact of three principal variables in the analysis: 1) thermophysical history, 2) computational tool, and 3) creep data. ABAQUS analysis of the INEL procedure is not made yet due to the lack of complete description on the thermophysical history.

### Table 1. Comparison of VIP Creep Analyses

<table>
<thead>
<tr>
<th>Analyst</th>
<th>Thermophysical history</th>
<th>Computational tool</th>
<th>SA533B creep data</th>
<th>Predicted max. creep strain</th>
<th>Failure (rupture)</th>
</tr>
</thead>
</table>
| GRS [23] | $T_{\text{max}} = 1320$ K  
$\rho (t)$  
$h_g = 30 \text{ W/m}^2\text{K}$ | ADINA | Simplified EPRI | 0.06 at $t=10,800$ s | No failure |
| INEL [20] | $T_{\text{max}} = 1400$ K  
$p(t)$  
$h_g = 50 \text{ W/m}^2\text{K}$ | TSM | NRC ($T \geq 1000$ K)  
EPRI ($T < 1000$ K) | Large | Fail at $t=5,400$ s |
| JAERI [24] | $T_{\text{max}} = 1320$ K  
$p = 11.0 \text{ MPa}$  
$h_g = 30 \text{ W/m}^2\text{K}$ | ABAQUS | NRC ($T \geq 1000$ K)  
truncated ($T < 900$ K) | 0.05 at $t=2,200$ s | No failure |

Note: 1) $h_g$ is the heat transfer coefficient at the outer wall

### Table 2. ABAQUS Analyses for the TMI-2 Lower Head

<table>
<thead>
<tr>
<th>Thermophysical history</th>
<th>Creep data of GRS</th>
<th>Creep data of INEL</th>
<th>Creep data of JAERI</th>
</tr>
</thead>
<tbody>
<tr>
<td>GRS</td>
<td>x</td>
<td>x</td>
<td>x</td>
</tr>
<tr>
<td>INEL</td>
<td>x</td>
<td>x</td>
<td></td>
</tr>
<tr>
<td>JAERI</td>
<td></td>
<td></td>
<td>x</td>
</tr>
</tbody>
</table>

**Thermophysical History**

Although each investigator derived the thermophysical histories of the TMI-2 vessel from the same metallurgical and thermohydraulic study of VIP, there prevail significant differences. Thermophysical histories of GRS and INEL were tuned to fit the metallographic examination results
on the maximum temperature at the inner wall, the duration at the maximum temperature (about 30 minutes) and the final cooling rate of the hot spot [19]. The maximum temperature values at the inner wall in the hot spot agreed well with the range of metallographic conclusion, i.e.; 1320 ~ 1370 K. However, the initial heatup rate upon core debris relocation was different significantly between GRS and JAERI. JAERI terminated thermal history at about one hour from the relocation. JAERI’s creep analysis on ABAQUS was concluded at 2,200 s to show a maximum creep strain of about 5% [20]. When the analysis was extended in this study for up to 3,300 s a gross deformation with sagging of hot spot was observed. This highlights the impact of a large variation in thermal models on core debris relocation amongst the investigators. The variation is not narrowed since TMI-2 metallographic examination revealed vague information on the core debris relocation behavior. Limited information on INEL thermophysical history also indicates a difference from that of GRS or JAERI. Given the large variation in thermophysical input and its importance in final creep analysis results, the refinement of core debris relocation behavior modeling is in demand.

Computational Tool

ADINA analysis results obtained earlier by GRS are reproduced closely by ABAQUS analysis using the same thermophysical and creep data. This is an important confirmation that both commercial FEM models are capable of properly handling the non-linear deformation problem. While ANSYS also produced comparable results for elastic and non-linear plastic problems, it required significantly longer run time than ABAQUS when high strain rate creep was involved. ADINA analysis by GRS was made using strain-hardening rule for creep modeling. However time-hardening rule used by ABAQUS did not make any difference. It may be attributed to the fact that the convergence criterion for ABAQUS analysis under time hardening rule is tight enough to finely divide the time steps. When time step is adequately divided two modeling should produce the same results. A disadvantage of ADINA lies in the fact that creep data have to be in certain form of constitutive equation whereas those of ABAQUS have a range of flexibility from raw data input to fitted equations.

The Shell Model (TSM) used by INEL was developed for VIP in order to handle a large number of cases in the given amount of time. TSM is formulated based on the finite deformation shell theory. While it can determine elastic and inelastic stresses from the calculated deformation, its inability to account for the radial stress can introduce considerable amount of error depending on the nature of lower head thermophysical history. A series of benchmark calculations of TSM as part of VIP showed up to 50% deviation in the stress in the high temperature region from the accurate ABAQUS results [21]. Overall it is considered that ABAQUS is the most appropriate computational tool for the lower head analysis.

Although ABAQUS is one of the most accurate and versatile tools for creep analysis, it has some shortcomings. First, the multi-axial creep modeling based on the von Mises effective stress strain formalism needs to be validated with experimental data for heavy section steels with thermal gradient. The current creep model used in ABAQUS does not take account of the primary creep strain memory effect which becomes significant as stress fluctuation amplifies with time. This deficiency is of great importance for ALWR severe accident analysis where feed and bleed mode of the emergency procedure will alternate thermal stresses between local tension and compression. Least significant disadvantage of ABAQUS is associated with long program run time, which may easily be overcome by using on supercomputers available at the Seoul National University.

Creep Data

In general, creep strain rate varies as a steep function of temperature and stress. The importance of the accuracy in thermophysical history and stress analysis is evident as discussed previously. In addition, the creep data for SA533B are found to be generated from a variety of materials and test conditions. Selection of proper set of creep data is subject to the investigator’s discretion. In order to examine if the difference between the investigators is significant, the creep strain was calculated as a function of time at a fixed stress that is the yield strength of SA533B at a
given temperature. A least square fitted correlation for the yield strength from a single database was used in this comparison.

Resultant creep strains were compared amongst the investigators. INEL and JAERI used the same creep data for temperatures of 1,000 K and above. JAERI creep data gave zero creep strain for temperatures below 900 K. This combined with the large creep strain in the high temperature region caused excessive buildup of stress in the outer region of the lower head leading to high damage when stress-based failure criterion, i.e. the Larson-Miller Parameter (LMP), was used [20]. The GRS creep model predicted significantly lower strain in the high temperature region. This explains why GRS predicted the maximum creep strain of TMI-2 to be only 5% at the end of 10,800 s. The INEL data set was selected as the most accurate creep data among the three. The ABAQUS analysis of the TMI-2 case was made with GRS thermophysical history and this creep data set for a period of up to 10,800 s from the core debris relocation since it was considered to be the most reasonable combination. When the INEL creep data were applied to the GRS case, considerable deformation in the hot spot was found.

Maximum creep strain in the hoop direction was calculated to be about 8% occurring at the outer wall. The effective creep strain was much higher due to the compressive radial strain. The LMP was calculated to be very low indicating the low probability of vessel rupture. The result is consistent with the actual TMI-2 end state far better than the results reported from the TMI-2 VIP.

III. CREEP SUPPRESSION OPTIONS FOR ADVANCED IN-VESSEL RETENTION

It becomes clear that the predominant mechanism of core debris escape in LWRs is the lower head failure due to creep rupture, as the results of TMI-2 VIP indicate. The creep strain rate increases more or less exponentially with temperature. The likelihood of creep rupture, predicted by either stress- or strain-based criterion, increases with creep strain. Therefore the lower head can be protected most effectively by suppressing the creep against the core debris attack. We have developed a range of creep suppression options for advanced in-vessel retention design. The bottomline idea centers about the structural designs that help maintain coolable geometry even under massive core debris relocation to the lower plenum. The structures designated as COrium Attack Syndrome Immunization Structures (COASIS) are made of metallic hemispherical shells with joints that form gaps toward either the inner wall (COASISI) or the outer wall (COASISO) of the lower head or extended up to the beltlne (COASISX), as sketched in Figure 2. Cooling water to the gap can be supplied from safety injection lines or the water tanks in the containment vessel for the COASISI, COASISO and COASISX, respectively. The COASISO nozzle protection design is presented in Figure 3 against the possible thermal shock with ensuing quench crack that may be caused by the emergency coolant being injected into the gap during a severe accident as mitigative measures. This is perhaps the most salient feature of the COASISO design that distances itself from the ex-vessel flooding approach that may in cases run into severe vessel outer surface and nozzle weld damage by the quench crack during the submergence of the reactor.

The effectiveness of the COASISI/COASISO options for the lower head protection was studied for the case of TMI-2 accident using ABAQUS. The reference TMI-2 case for this purpose used 1) thermophysical history of GRS, 2) ABAQUS with strain hardening creep rule, and 3) creep data set of INEL for SA533B. COASISI is simply remodeled by applying a higher heat transfer coefficient at the outer wall of the lower head from the reference value of 30 W/m²K. COASISO on the other hand is modeled as a 5 cm thick hemispherical shell made of SA533B and simply supported at its rim with a hot spot on the inner wall consistent with the GRS thermophysical history while water cooled in the gap surfaces. The heat transfer coefficient at the water cooled surfaces in the gap has to be determined experimentally for the curved annulus cooling geometry. A 30 cm diameter hemisphere with internal heating and about 2 cm gap on the outside was used to determine the heat transfer coefficient for wetted surfaces in the gap over a wide range of superheat including nucleate boiling and post critical heat flux (CHF) regimes [22]. Results are compared in Figure 4 with those of full scale ex-vessel flooding experiment and scaled quench experiment of the Sandia.
National Laboratories designated as CYBL [13]. A lower bound of the experimental data, 630 W/m²K, is taken as the uniform gap heat transfer coefficient applicable to all wetted surfaces for COASIS analyses. In line with thermohydrodynamic considerations we have also devised a means to determine the creep rupture probability as a function of creep usage factor defined in Figure 5 for SA533B used in the TMI-2 VIP.

![Engineered Gap Structures for High Temperature Creep Suppression of the Vessel](image)

**Figure 2.** Engineered Gap Structures for High Temperature Creep Suppression of the Vessel

![Nozzle Protection Against Thermal Shock by Incoming Emergency Coolant](image)

**Figure 3.** Nozzle Protection Against Thermal Shock by Incoming Emergency Coolant
Figure 4. Comparison of Heat Transfer Coefficients for Vessel Cooling

SA533B (VIP) Date fitted with R=0.97

\[ P_f = 1 - \exp\left[-\left(\text{Creep Usage Factor}\right)^{2.36}\right] \]

Figure 5. Creep Rupture Probability for SA533B
Results of the COASIS study are shown in Figures 6 to 8 for the KSNPP during a station blackout scenario in which the power was recovered when the debris mass reached 20,000 kg in the lower plenum. Figure 6 compares global deformation of the lower head (a) in the reference case, (b) with the COASISO shell, and (c) with the COASISI shell. Only the COASISI shell was presented for the last case since the lower head is assumed to remain at the initial temperature being protected from the core debris attack. The creep strain distribution in the lower head with COASIS is illustrated in Figure 7. Creep strain is shown to be localized near to inner walls where core debris attack is the most severe. Virtually no creep damage is seen near to the outer wall with COASIS. The LMP, the stress-based creep rupture indicator defining a failure when its value reaches unity, is calculated as shown in Figure 8. It is evident that COASIS cooling capacity is high enough to effectively suppress the creep damage of the lower head.

IV. REACTOR APPLICATION FOR SEVERE ACCIDENT MANAGEMENT

As described earlier the debris thermal and mechanical attack of the reactor pressure vessel is an urgent issue from the severe accident management point of view. The gap cooling strategies are being pioneered in Korea as part of the SONATA-IV program [4]. Currently the national in-vessel retention research and development efforts follow along the line of the natural gap cooling by the Korea Atomic Energy Research Institute (KAERI), the engineered gap cooling by the Seoul National University (SNU) supported by the Korea Electric Power Research Institute (KEPRI), and the basic critical heat flux databank accumulation by the Korea Advanced Institute of Science and Technology (KAIST). The engineered gap cooling structure study is being conducted at SNU for potential application to the KNGR. The advisory review meetings are being held to review fundamental concept and rationale, direction of future development, and necessary licensing requirements. It was also presented at the OECD Special Meeting on In-Vessel Debris Coolability and Lower Head Coolability on November 18-19, 1996 as part of the SONATA-IV Program.

The accident management may now be pursued along the line of in-vessel melt retention in concert with existing ex-vessel management measures for the KNGR. The dual defense-in-depth strategy may be summarized as in Table 3.

The benefit of having COASISI and/or COASISO is to be able to deal with the late phase of the melt progression in much more an efficient manner than with flooding the whole reactor cavity. Be aware that the ex-vessel flooding ought to be executed at the cost of time and water resources, and that with potential for jeopardizing the vessel integrity itself via thermal shock to the lower head penetrations with ensuing quench crack at the time of submergence. The influence of quenching may differ by medium, specimen size and geometry. In general, the more rapid the quench, the more severe the impact. Relatively large pieces that are rapidly quenched may crack as a result of internal stresses. This becomes a problem especially when the carbon content is greater than 0.5 % by weight. For higher carbon steels, a water quench may be too severe because of resultant cracking and warping.

On the other hand, the COASISO design can ensure prompt delivery and highly effective utilization of the emergency cooling water for only the part of the vessel that needs to be cooled at the time of accident. Furthermore, should it turn out to be an inadvertent action to have initiated the COASISO injection system, the operator should easily be able to immediately drain the water from the gap thus protecting the vessel investment. In contrast, it'll be impractical to drain the cavity once it has been flooded, and that by mistake not to mention the potential disaster of under-water steam explosion induced by the draining molten debris from the vessel rupture. Also, when considering ex-vessel flooding, an account must be taken of newly surfacing technical issues such as the reactor vessel insulator design and impacts of inadvertent flooding during otherwise normal operation on the reactor vessel material properties. The cavity flooding system (CFS) should also be designed with additional water source. If the CFS is placed external to the containment, its reliability should be ensured, else if incorporated into the containment, structural loads on the CFS should be evaluated.
Figure 6. Deformed Shape of KSNPP Lower Head after Core Relocation

Bare Design (Top), COASISO (Middle), COASISI (Bottom)
Figure 7. Creep Strain Distribution at the Center of KSNPP Lower Head

Figure 8. Fractional Cumulative Creep Strain Damage of KSNPP Lower Head

Bare Design (Top), COASISO (Middle), COASISI (Bottom)
Table 3. Dual Defense-in-Depth Accident Management Strategies for the KNGR

<table>
<thead>
<tr>
<th>In-Vessel Management</th>
<th>Ex-Vessel Management</th>
</tr>
</thead>
<tbody>
<tr>
<td>1) Hardware: COASISI</td>
<td>Hardware: IRWST</td>
</tr>
<tr>
<td>Procedure: Partial Depressurization if Necessary</td>
<td>Ex-Vessel Debris Catcher</td>
</tr>
<tr>
<td>DVI (HPSIT and SIT)</td>
<td>Capture Volume</td>
</tr>
<tr>
<td>2) Hardware: COASISO</td>
<td>Hydrogen Igniters/Recombiners</td>
</tr>
<tr>
<td>Dedicated Water Supply Valve and Controller</td>
<td>Containment Spray</td>
</tr>
<tr>
<td>Procedure: Complementary and Prior to COASISI (minimize thermal shock) Water Injection to Bottom Sparging from Top Annulus Flow Rate Control</td>
<td>Containment Cooling</td>
</tr>
<tr>
<td></td>
<td>Procedure: Similar to System 80+ SAMG</td>
</tr>
</tbody>
</table>

The roadmap to design certification and licensing of the COASIS is shown in the tree of Figure 9. The process may be segmented into three principal routes involving a number of nuclear organizations with varying tasks. The first one is of course the design verification and validation through extensive and thorough testing and analysis in combination of probabilistic and deterministic approaches. The safety analyses and tests must also be accompanied by the normal operation, thermohydraulic design and mechanical component design analyses to secure proper margins during the plant lifetime. The manufacturing sector shall concentrate on maintenance and inspection plan and on materials and process specification. The specification shall not only consider the expected cost-benefit but also any undue risk to plant safety. To summarize, both design basis and severe accident evaluations ought to be performed in tandem when considering actual implementation of the COASIS structures for retaining the molten core debris within the reactor vessel during and after the accident.

V. CONCLUSIONS AND FUTURE WORK

Severe accident management strategies for advanced LWRs are being shifted from ex-vessel to in-vessel retention of molten core debris. Possible in-vessel management options have been explored for the KNGR with its baseline design of ex-vessel management strategy.

Capability for accurately determining temperature distribution, hydrodynamic boundary conditions, creep deformation and rupture time and its mode is the key to success in the development...
of in-vessel management strategies. However, current creep and rupture modeling needs improvements in the following areas:

- accurate modeling of core debris relocation behavior in the lower head region
- validation of computational scheme of creep analysis for heavy sections with multi-axial stress and thermal gradients
- accurate creep strain and rupture database

The COASIS structures have been developed as a potential in-vessel management option for the KNGR. Both the gap structures for in-vessel (COASIS1) and ex-vessel (COASIS0) are shown to maintain effective heat transfer geometry during core debris attack when applied to TMI-2 and KSNPP case studies. The likelihood of lower head creep rupture during a severe accident is found to be effectively suppressed. Further analysis for COASIS effectiveness in KNGR is in progress.

COASIS can thus protect the lower head without thermal shock and direct water resource to the hot spot with efficiency. Synergism of COASIS with existing ex-vessel management can be utilized to free KNGR from emergency evacuation requirement.

ACKNOWLEDGMENT

The authors would like to express their sincerest gratitude to Drs. Peter Gruner of GRS, K. Hashimoto of JAERI and Prof. Robert Witt of University of Wisconsin for providing with valuable information during the course of this study. They would also thank KEPR and KAERI for their support during the course of this work.

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Thermomechanical Analysis for Advanced In-Vessel Retention Design

Kune Y. Suh, II Soon Hwang
Seoul National University

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- In-Vessel Retention Strategies
  - In-vessel flooding
  - Engineered gap cooling
  - Ex-vessel flooding
  - External spraying
- Lower Head Creep Analysis for TMI-2 VIP
  - Thermophysical history
  - Computational tool
  - Creep data
- Creep Suppression Options for Advanced IVR
  - COASIS, COASISO, COASISX
- Reactor Application for Severe Accident Management
- Roadmap to COASIS Design Certification

In-Vessel Retention Strategies for LWRs

- Under what conditions will the mechanism take effect?
- Will require large-scale real material TH & material tests
- How to secure the structure to RPV
- How to get the water to fill the RPV in time?
- How to design a spraying mechanism?
- Cooling might not be enough to remove decay heat from the RPV
- Will require examination of possible quench cracks
- How to do it?
External Flooding Issues

- Thermal Insulator
- Penetration Tubes

COASISO Nozzle Protection Design

- Lower Head
- RPV support

COASIS - COrium Attack Syndrome Immunization Structure

- Internal Gap Structure
- External Gap Structure
- Double Wall Structure

COASIS - Advanced IVR Design for LWR

- Debris Attack of Reactor Pressure Vessel is an Urgent Issue.
- Gap Cooling Strategies Are Being Pioneered in Korea as SONATA-IV:
  - Natural Gap Cooling: KAERI
  - Engineered Gap Cooling: SNU/KEPRRI
  - Critical Heat Flux Databases: KAIST
- Engineered Gap Cooling Structure Study Is Being Conducted at SNU for Potential Application to KNPR.
- The Advisory Review Meetings Are Being Held:
  - Review of Concept and Rationale
  - Direction of Future Development
  - Necessary Licensing Requirements
- The Engineered Gap Cooling Concept Was Presented at The OECD Special Meeting on In-Vessel Debris Coolability and Lower Head Coolability on November 18-19, 1996 as part of the SONATA-IV Program.
**D2R2 - Defense-in-Depth Reactor**

- May incorporate
  - the internal structure COASISI (would require detailed normal operation analysis and maintenance)
  - the external structure COASISO (would require dedicated water supply)
  - the double-wall structure COASIX (would require thermal shock and irradiation-induced embrittlement analysis)

---

**Lessons Learned from OECD TMI-2 VIP**

- Determination of RPV Temperature
- Development of Material Property Database & Constitutive Models
- Assessment of Margin to RPV Failure
  - Limited success only
  - All 3 (GRS, INEL, JAERI) investigators predicted significantly large creep strain
  - TMI-2 lower head deformation was practically unnoticeable
  - There was no reasonable agreement in creep rupture prediction

---

**Comparison of TMI-2 VIP Analysis**

<table>
<thead>
<tr>
<th>Analyst</th>
<th>Thermophysical history</th>
<th>Computational tool</th>
<th>Creep data</th>
<th>Predicted max. creep strain</th>
<th>Failure (rupture)</th>
</tr>
</thead>
<tbody>
<tr>
<td>GRS</td>
<td>$T_{max}=1220 , K$</td>
<td>ADINA</td>
<td>Simplified EPRI</td>
<td>0.004 at $t=18,400 , s$</td>
<td>No failure</td>
</tr>
<tr>
<td>INEL</td>
<td>$T_{max}=1400 , K$</td>
<td>TSM</td>
<td>NRC (75,1000 K)</td>
<td>Large</td>
<td>Failure at $t=3,400 , s$</td>
</tr>
<tr>
<td>JAERI</td>
<td>$T_{max}=1320 , K$</td>
<td>ABAQUS</td>
<td>NRC (78,600 K)</td>
<td>0.06 at $t=4,200 , s$</td>
<td>No failure</td>
</tr>
</tbody>
</table>

---

**ABAQUS Analysis for TMI-2 Lower Head**

- A Series of Analyses Done to Quantify the Impact of 3 Principal Variables:
  - Thermophysical history
  - Computational tool
  - Creep data

<table>
<thead>
<tr>
<th>Thermophysical history</th>
<th>Creep data of GRS</th>
<th>Creep data of INEL</th>
<th>Creep data of JAERI</th>
</tr>
</thead>
<tbody>
<tr>
<td>GRS</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>INEL</td>
<td>X</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>JAERI</td>
<td></td>
<td></td>
<td>X</td>
</tr>
</tbody>
</table>
**Margin-to-Failure Analysis for RPV**

- Analytical Tool Development
  - ABAQUS - a non-linear FEM for stress & strain calculation
  - SA533B creep database from INEL (U.S. NRC & EPRI)
- TMI-2 Accident Case
  - Thermophysical history from VIP - GRS
    - bare (w/o COASIS)
    - COASISO (5 cm outer gap)
    - COASISI (5 cm thick, 5 cm inner gap)
    - Lower bound gap heat transfer coefficient (630W/m²K)
- KSNPP Station Blackout Case
  - Thermophysical history from MAAP4.2
    - bare (w/o COASIS)
    - COASISO (3 cm outer gap)
    - COASISI (5 cm thick, 5 cm inner gap)
    - Monde's narrow gap heat transfer coefficient plus others

**Heat Transfer Coefficients for RPV Cooling**

---

**Severe Accident Mitigation w/ COASISO**

- Core melt detection
- COASISO injection (1 min after CM)
- Core relocation (10 min after CM)
- Long-term recirculation cooling

---

**Severe Accident Mitigation w/ COASISO/O**

- Core melt detection
- COASISO injection (1 min after CM)
- Core relocation (10 min after CM)
- Long-term recirculation cooling

---

*Department of Nuclear Engineering
Seoul National University*
Severe Accident Progression w/o COASIS

- Core melt detection
- Core relocation (10 min after CM)
- RPV failure (45 min after CM): melt ejection & possible SX (60 min after CM)

Potential Major SX Underneath Water

- Core melt detection
- Core relocation (10 min after CM)
- RPV failure (45 min after CM): melt ejection & possible SX (60 min after CM)

Proposed SAM for the KNGR

**In-Vessel Melt Retention:**

1. **Hardware:** COASIS
   - Dedicated Water Supply Valve and Controller
   - Procedure: Complementary & Prior to COASIS (to minimize thermal shock)
   - Water Injection to Bottom
   - Sparging from Top Annuli
   - Flow Rate Control

2. **Hardware:** COASSO
   - Dedicated Water Supply Valve and Controller
   - Procedure: Similar to System 80+

**Ex-Vessel Management:**

- **Hardware:** IR5/V3T
  - Ex-Vessel Debris Catcher
  - Capture Volume
  - Hydrogen Recombination
  - Containment Spray
  - Containment Cooling

Roadmap to COASIS DC

- Severe Accident Management Strategy is Shifting Focus from Ex-Vessel to In-Vessel.
- COASIS Can Protect Lower Head without Thermal Shock and Direct Water Resource to the Hot Spot with Efficiency.
- Synergism of COASIS with Existing In-Vessel and Ex-Vessel Management Can be Utilized to Free KNGR from Emergency Evacuation Requirement.
### Conclusion

- SAM strategies for ALWR are shifting from ex-vessel to in-vessel retention of molten core debris.
- Possible IVR options (COASISO/COASISI) were explored in parallel with existing ex-vessel measures.
- The key is to accurately determine thermal, hydrodynamic boundary conditions, creep deformation, and rupture time and mode of failure.
- Current creep and rupture model needs improvements in:
  - core material relocation behavior in the lower plenum
  - validation of computational scheme of creep analysis of lower head
  - accurate creep strain and rupture database.
- The likelihood of RPV failure significantly reduced with COASISO/COASISI.
2.3 Modelling Lower Plenum Core Debris

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Constructing heat transfer models for the lower plenum core debris is important for assessing accident management strategies. Involving natural convection, crust formation, molten metal layers and interaction with pressure vessel wall, etc., the models can be quite complex. On the other hand, for exploring the many possible scenarios and for incorporation into integrated thermal hydraulics codes such as SCDAP/RELAP, they should be relatively simple, capturing the essential physics with enough accuracy. In this paper, we describe our progress in developing such a model based on a lumped-parameter approach similar to the one used by Theofanous et al (In-vessel Coolability and Retention of a Core Melt, DOE/DD-10460, Vol.1, July 1995) in connection with AP600 and the one used in MAAP4.0 for LWR's in general.

As depicted in the figure, the model, which is applicable to transients, assigns temperatures nodes to the ceramic pool, the molten metal layer, the bottom and top crusts (which sustain parabolic temperature profiles), and the vessel wall. Empirical correlations are invoked for the various heat transfer paths. The following work will be reported:

1. The use of correlations developed for simpler geometries to the molten metal layer is validated against CFD calculations.
2. Correlations for the ceramic pool established for steady states are converted into forms applicable to transients.
3. The description of the top crust is validated using a finite-difference method which solves the phase change (Stefan) problem.
4. The role played by the top crust is explored.
Modelling Lower Plenum Core Debris

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Presented at the Workshop on Severe Accident Research held in Japan
Yokohama, Japan
6th-8th October 1997

In-Vessel Melt Retention

- Safety mitigation method for LWR severe accidents, e.g. TMI.
- If core meltdown occurs, molten steel and UO₂ relocate to lower head of pressure vessel.
- Radioactive decay still produces a significant amount of heat within this molten UO₂.
- Proposed to flood outside of pressure vessel with water to remove this decay heat.
- Pressure vessel integrity is maintained and the radioactive material contained.

Lumped Parameter Model

- Temperature nodes: Metallic layer, Oxidic pool, Crust.
- Parabolic profile assumed across crust.
- Empirical equations for heat fluxes.
Need for Transient Simulation

- Timing for accident management.
- Determine most challenging conditions for vessel integrity.
- Steady state calculations show thin upper crust supports large temperature gradient. "What happens if crust breaks?"

Approach to Transient Simulation

- Existing correlations for the molten pool are of the form:
  \[ Nu \propto Ra^{n} \]
  where \( Ra' \) is the internal Rayleigh number:
  \[ Ra' = \frac{g \beta L^3 Q_v}{k \alpha v} \]
- Cannot be applied to transients.
- Re-express in terms of \( \Delta T \) instead of \( Q_v \) (first suggested by Turland & Morgan) - can then be used for transients.

Transient Correlations

Require correlations in terms of:

\[ External \ Rayleigh \ Number = Ra = \frac{g \beta L^3 \Delta T}{k \alpha v} \]

Internal and external Rayleigh numbers related by:

\[ Ra' = Ra \left( \frac{Q_v L^3}{k \Delta T} \right) \]

From conservation of energy:

\[ Q_v \sim \sum_{\text{surfaces}} \]

\[ \Rightarrow \left( \frac{Q_v L^3}{k \Delta T} \right) - \sum N_h \]

Example - Hemispherical Geometry

Mini-ACOPO correlations:

\[ Nu_{up} = 0.345 \, Ra'^{0.233} \]
\[ Nu_{dn} = 0.045 \, Ra'^{0.27} \]
\[ 10^{12} \leq Ra' \leq 3 \times 10^{13} \]

Conservation of energy gives:

\[ Ra' = \frac{3}{2} Ra \left( 0.345 \, Ra'^{0.233} + 2 \times 0.045 \, Ra'^{0.27} \right) \]

which may be simplified using a power law fit:

\[ Ra' = 0.477 \, Ra^{5/4} \]
Power Law Fit

Resulting in the transient correlations:

\[ \begin{align*} 
N_{up} &= 0.290 Ra^{0.311} \\
N_{ide} &= 0.0393 Ra^{0.366} \\
1.74 \times 10^6 &\leq Ra \leq 2.23 \times 10^{10} 
\end{align*} \]

Application of Transient Correlation

- Simulation of ACOPO

- Experiment establishes correlation in terms of \( Ra' \) by transient cooldown.

- New transient correlation can be used to predict temperature history (or \( Ra' \) history).

"Dam Breaking" Effect

- Thin top crust (6mm for AP600).

- Large stored heat in oxidic pool.

- Metallic layer heats up when crust breaks.

- Transient correlations can be used to analyse this phenomenon.

"Dam Breaking" Effect - Results

Remove crust suddenly, not allowed to reform. Metallic layer/oxidic pool interface assumes single temperature.

- Oxidic pool quickly drops below solidification point.

- Metallic layer heats up.
“Dam Breaking” Effect - Heat Fluxes

- Heat flux to side wall rises rapidly and exceeds critical heat flux of 1500 kW/m² for ex-vessel cooling.

Temperatures and Heat Fluxes

- Impact on oxidic and metallic layers negligible.

Crust Healing

In reality, crust will reform. Assume crust suddenly thins out to 1 mm. What is response of system?

- Crust quickly rebuilds to original thickness.

Validation of Integral Method

- Analysis uses heat balance integral method assuming parabolic temperature profile across crust.
- Parabolic profile accurate for steady state but not necessarily valid for transients.
- Validate using finite difference treatment for crust.
Finite Difference Method - Details

- Stefan problem with prescribed heat flux boundary condition at the melt front.
- Used fixed grid enthalpy formulation.
- Used virtual domain with low heat capacity.

Results from Thermal Resistance Model

- Calculated steady state conditions for range of resistance values for AP600 design.
- Found narrow range of resistance consistent with liquid phase in both oxidic and metallic layers.

Thermal Resistance Model for Top Crust

- Crust is thin and susceptible to breakage.
- If it breaks, it quickly heals itself.
- Process of breakage/healing is likely to be continuous.

Conclusions

- Re-expressing correlations in terms of $\Delta T$ instead of $Q_v$ enables transients to be simulated.
- Used finite difference method to validate treatment of top crust in lumped parameter model.
- If crust damaged, quickly heals itself with negligible impact on oxidic or metallic layers.
- Can model crust as thermal resistance - find only narrow range of resistance consistent with liquid metallic and oxidic layers.

Can model crust using an effective thermal resistance.
2.4 Study on the One-Dimensional Flow Characteristics of the Counter-current Flow in Debris Beds

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4-3-16, Jonan, Yonezawa-shi,
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ABSTRACT

In the course of a severe accident, a debris bed is formed from once molten and fragmented fuel elements as observed in the Three-Mile Island unit-2 accident. The debris bed must be cooled to avoid further degradation of the core since the degraded core still releases decay heat. Even if the degraded core is in water, it cannot be assumed that the coolability of the degraded core would be maintained, since the degraded core may be melted again if dryout occurs. It is thus necessary to evaluate the dryout heat flux for the judgment of the coolability of the debris bed during the severe accident. Dryout phenomena in the debris bed are dominated by two-phase flow behavior in the debris bed. Especially, it is indicated that dryout phenomena in debris bed is strongly affected by counter-current flow limitation (CCFL) in the debris bed. Therefore, it is important to know the CCFL characteristics in the debris bed.

If one hopes to analyze counter-current flow in the debris bed by using the same method as usual two-fluid model, it is necessary to know the interfacial and wall friction factors. However, it is not clear that the correlations for interfacial and wall friction factor for a pipe can be used for counter-current flow in the debris bed. In order to determine the interfacial and wall friction factors, it is necessary to obtain wall and interfacial shear stresses. The wall and interfacial shear stresses depend on the void fraction and differential pressure. However, the wall and interfacial shear stresses should be changed in the flow direction since counter-current flow limitation in the debris bed occurs in some height of the debris bed. At present, there is no available data for local void fraction and local pressure distribution in the debris bed at CCFL since it is very difficult to measure the flow characteristics in the complex geometry such as the debris bed.

In the present study, local void fraction and local pressure distributions are measured simultaneously as well as the flow rates for gas and liquid. From the measurement data, the wall and interfacial shear stress are estimated. Finally the wall and interfacial shear stresses are determined from the experimental data.
1. Introduction
Background - 1

- If one hopes to analyze counter-current flow in the debris bed by using the same method as usual two-fluid model, it is necessary to know the interfacial and the wall friction factors.
- However, it is not clear that the correlations for interfacial and wall friction factor for a pipe can be used for counter-current flow in the debris beds.
- In order to determine the interfacial and wall friction factors, it is necessary to obtain wall and interfacial shear stresses.

Objectives

- To measure local void fraction and local pressure distributions simultaneously as well as the flow rates for gas and liquid.
- To estimate the wall and the interfacial shear stresses from the measurement data.
- To determine the wall and interfacial shear stresses from the experimental data.

Background - 2

- The wall and the interfacial shear stresses are depend on the void fraction and differential pressure.
- At present, there is no available data for local void fraction and local pressure distribution in the debris bed at CCFL, since it is very difficult to measure the flow characteristics in the complex geometry such as the debris bed.

2. Experimental apparatus and procedures
Schematic diagram of the test apparatus

Main parameters

- Particle diameter: \( d = 4.5 \sim 14.5 \) (mm)
- Bed height: \( H_b = 200 \) (mm)
- Bed inner diameter: \( D_b = 35, 50, 70 \) (mm)
- Upper water level: \( H_{\text{up}} = 200 \) (mm)
- Input current: \( I_m = 0.2 \sim 0.4 \) (mA)
- Air flow velocity: \( V_g = 0.0 \sim 2.1 \) (m/s)

Schematic diagram of the test rig

Measurement method of void fraction in debris bed under CCFL:

1. After CCFL establishment, apply a constant current \( I_m \) to the upper and lower part of the test section.
2. Measure the voltage of \( V_m \) at the measurement location.
3. After CCFL experiment, measure the current \( I_m \) and the voltage \( V_m \) at the single phase water condition.
4. Estimate the void fraction by
   \[
   \alpha = 1 - \frac{V_m}{V_m^0} - \frac{V_m}{V_m^0}
   \]
3. Experimental results

- Measurement results of the flow rates

Volumetric water flow rate $M_w$ ($m^3/s$)

- Measured pressure and void fraction distributions

- Measured CCFL line
The black symbol shows the measurement data at the height of 30 mm from the top of the debris bed. The white symbol shows the measurement data at the height of 140 mm from the top of the debris bed. The line shows the calculated results by the drift flux model for bubbly flow in a pipe.

\[ \alpha = \frac{J_g}{C_0 \cdot (J_g + J_l) + V_g} \]

Where \( C_0 = 1.2 \) and \( V_g = \sqrt{2 \cdot \left( \frac{g \cdot (\rho_l - \rho_g)}{\rho_g} \right)^{1/2}} \).

This result indicates that the debris bed can accumulate much water than in a pipe.

4. Discussion

- Non-dimensional water film thickness is estimated by

\[ \delta^* = \frac{\delta}{\sqrt{\frac{g \cdot (\rho_l - \rho_g)}}} \]

Where

\[ \delta = \frac{D_d}{2} \left( 1 + \sqrt{\alpha} \right) \]

- Water film thickness decreases with increasing the non-dimensional superficial gas velocity.

- Water film thickness for larger particle diameter is larger than that for smaller particle when the gas velocity is the same.

- The effect of the debris bed diameter is not significant.
Estimated shear stresses

- The wall and the interfacial shear stress are evaluated with following equations by using the experimental data of the void fraction and differential pressure.
  \[ \tau_w = \frac{\rho g D w}{(1-c)^2} \left( \frac{D}{D_w} \right)^2 \]
  \[ \tau_i = \frac{D_i}{D_w} \left( \frac{D}{D_i} \right)^2 \tau_w \]
  \[ \tau_w = \frac{\rho g D w}{(1-c)^2} \left( \frac{D}{D_w} \right)^2 \]
  \[ \tau_i = \frac{D_i}{D_w} \left( \frac{D}{D_i} \right)^2 \tau_w \]

- The evaluated wall shear stress is of the same order of the interfacial shear stress.
- The wall shear stress decreases with increasing the interfacial shear stress and decreasing the particle diameter.

![Graph showing shear stress vs. non-dimensional velocity](image1)

![Graph showing shear stress vs. non-dimensional velocity](image2)
Estimation of wall friction factor

$$f_w = \frac{Z \cdot \tau_w}{\rho_l \cdot V_l^2}$$

- Estimated wall friction factors are larger than the following Ergun's equation.

$$f = \frac{33.33}{Re} + 0.5833 \frac{1}{\varepsilon^2}$$

where

$$Re = \frac{D_p \rho V}{\mu}$$

- The wall friction factors for larger particles are larger than that for smaller particles of the Reynolds number is the same.

- The wall friction factors depend strongly on the Reynolds number.

5. Conclusions

- Local void fraction, local pressure distribution were simultaneously measured for debris bed under CCFL condition as well as flow rates for gas and liquid.
- The wall and interfacial shear stresses were estimated from the measurement data of the local void fraction, local pressure distribution and flow rates.
- The estimated wall shear stress was almost the same order as the interfacial stress. It was clarified that the relationship between the interfacial and wall shear stress are theoretically well correlated.

- The wall and the interfacial friction factors were experimentally estimated for the debris bed under CCFL.
- It was clarified that wall friction factors are depend on the liquid phase Reynolds number and particle diameter.
- It was also clarified that the interfacial friction factors are depend on the water film thickness and the particle diameter.
2.5 OECD RASPLAV Project: Phase I Results.

V.G. Asmolov, A.V. Merzliakov
RRC “Kurchatov Institute”, Moscow, Russia.

Abstracts

OECD RASPLAV Project dealing with experimental study of the behavior of the molten prototypic material of the core (corium) in the lower heard of nuclear reactor. During fist phase of the Project two lager scale experiments with 200 kg of corium were conducted. A lot of tests with corium mass up to 40 kg have been done.

The main results of Phase I may be formulated as
- technical problems to conduct large scale experiment were resolved
- methods of the heating up of core material beyond liquidus temperatures
- compatible materials
- measurements techniques
- new data on corium properties were obtained
- analytical tools for corium test analysis were developed
- separation of corium melts was observed
RASPLAV Phase I Major Activities

- Large scale corium tests
  - RASPLAV-AW-200-1 9 of October 1996
  - RASPLAV-AW-200-2 24 of May 1997
- Separate effect tests (Tulpan, Korpus, Tigel)
- Material properties measurements
- Salt tests
- Code development and usage

Major Goals of OECD RASPLAV Project

- Study behavior of prototypic molten core materials in the lower head of the RPV
  - Heat transfer
  - Chemical interactions of core materials
  - Crust formation
  - Core material properties data at high temperatures
- Develop codes for consistent analysis of experimental data
  To create a knowledge base to support analysis of core melt retention in the lower head of the RPV

Matrix of Corium Property Measurements

<table>
<thead>
<tr>
<th>Corium</th>
<th>Composition, w%</th>
<th>Measured properties vs. temperature</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>UO2</td>
<td>ZrO2</td>
</tr>
<tr>
<td>C-22</td>
<td>81.5</td>
<td>5</td>
</tr>
<tr>
<td>C-50</td>
<td>80</td>
<td>11.5</td>
</tr>
<tr>
<td>C-100</td>
<td>77.8</td>
<td>22.2</td>
</tr>
</tbody>
</table>
Corium Tests Conducted before Large Scale Experiments

<table>
<thead>
<tr>
<th>Tests</th>
<th>Objective</th>
<th>Composition</th>
<th>Corium mass, kg</th>
<th>Maximum temperature</th>
<th>Number of tests</th>
</tr>
</thead>
<tbody>
<tr>
<td>Laboratory scale tests (Tigsl, Korpus, Toksv, Tulpan)</td>
<td>Material properties, Material interactions</td>
<td>C — 22, C — 50, C — 100</td>
<td>up to 5</td>
<td>3000°C</td>
<td>&gt;106</td>
</tr>
<tr>
<td>RASPLAV-A-1.1c</td>
<td>Feasibility of pouring</td>
<td>C — 22</td>
<td>12</td>
<td>2900°C</td>
<td>13</td>
</tr>
<tr>
<td>RASPLAV-AW-2.5</td>
<td>Feasibility of SWH</td>
<td>C — 22</td>
<td>12</td>
<td>2450°C</td>
<td>1</td>
</tr>
<tr>
<td>RASPLAV-AD-2.5</td>
<td>Feasibility of DEH</td>
<td>C — 100</td>
<td>40</td>
<td>&gt;2600°C</td>
<td>1</td>
</tr>
</tbody>
</table>

Large Scale Test: Experimental Approach

- Slice geometry
- Side wall inductive heating method to simulate Q
- Use of compatible materials to prevent interactions of corium with structural materials

RASPLAV-AW-200-1 Test: Thermal Hydraulic Results

Large Scale Test: Experimental Approach

- Slice geometry
- Side wall inductive heating method to simulate Q
- Use of compatible materials to prevent interactions of corium with structural materials

L=12.6 cm
R=40 cm
RASPLAV-AW-200-1 Test (C-22)

Main Results

- Heat transfer due to convective and conduction modes
- Measured peak temperature of molten corium pool - 2680°C
- Peak heat flux ~ 130 kW/m²
- Molten fraction ~ 70%
- Almost complete stratification of initial melt compatible with data from phase diagram of the state
  - upper layer U/Zr ratio ~ 0.37
  - lower layer U/Zr ratio ~ 1.9-2.15
  - light liquid volume ~ 2 dm³
  - heavy liquid volume ~ 11 dm³
  - \( V_L/V_H = 0.18 \)
- Living spreading temperature of corium during melt process

RASPLAV-AW-200-1 Test (C-22)

\( \frac{U}{Zr} \) distribution along height of section #9

RASPLAV-AW-200-2 Test (C-22)

Main Results

- Heat transfer due to convective and conductive modes
- Peak heat flux ~ 250 kW/m²
- Molten fraction higher than 70%
- "Transition" stratification of initial melt with different U/Zr ratio
  - upper layer ~ 0.5-0.8
  - lower layer ~ 2.0
  - light liquid volume ~ 5 dm³
  - heavy liquid volume ~ 9 dm³
  - \( V_L/V_H = 0.55 \)
- Corium-vessel steel interaction due to molten corium attack

RASPLAV-AW-200-2 Test:

Left side of corium ingot during facility disassembling
**TULPAN Corium Stratification Test (T3 test C-22)**

**Main Results**
- Complete corium melting
- Stratification of corium melt
  - upper layer U/Zr ratio - 0.3-0.7
  - lower layer - 1.4-1.8
  - light liquid volume ~0.17 dm$^3$
  - heavy liquid volume ~0.87 dm$^3$
  - $V_L/V_H$ ~ 0.2
- Influence of convection role on stratification process:
  - negligible during heat up phase of test
  - could take place during cool down phase
- No stratification inside slim tube (without convection)

### Comparision of Stratification Obtained in Corium Experiments

<table>
<thead>
<tr>
<th>RASPLAV-AW-200-1</th>
<th>RASPLAV-AW-200-2</th>
<th>TULPAN T3</th>
</tr>
</thead>
<tbody>
<tr>
<td>U/Zr</td>
<td>Volume ()</td>
<td>$V_L/V_H$</td>
</tr>
<tr>
<td>Light</td>
<td>-0.37</td>
<td>-2.0</td>
</tr>
<tr>
<td>Heavy</td>
<td>1.9-2.15</td>
<td>-11.0</td>
</tr>
</tbody>
</table>

$ZrO_2 + UO_2$ phase diagram:
- light - 0.3
- heavy - 2.3

### Salt Test Matrix

<table>
<thead>
<tr>
<th>No series</th>
<th>Salt composition</th>
<th>Heating method</th>
<th>P [kW]</th>
<th>Ra*</th>
<th>Pr</th>
</tr>
</thead>
<tbody>
<tr>
<td>S1</td>
<td>NaF-NaBF$_4$ 8:92</td>
<td>SWH</td>
<td>0.5</td>
<td>7.1·10$^1$</td>
<td>5.2</td>
</tr>
<tr>
<td>S2</td>
<td>NaF-NaBF$_4$ 8:92</td>
<td>SWH</td>
<td>0.5-8.3</td>
<td>(0.6-5.0)·10$^3$</td>
<td>3.6-7.5</td>
</tr>
<tr>
<td>S3</td>
<td>NaF-NaBF$_4$ 8:92</td>
<td>SWH</td>
<td>0.8-5.7</td>
<td>(0.2-2.2)·10$^3$</td>
<td>4.2-7.5</td>
</tr>
<tr>
<td>S4</td>
<td>NaF-NaBF$_4$ 8:92</td>
<td>DEN</td>
<td>0.97-3.6</td>
<td>10$^1$,1.5·10$^3$</td>
<td>4.4-7.5</td>
</tr>
<tr>
<td>S5</td>
<td>NaF-NaBF$_4$ 25:75</td>
<td>SWH</td>
<td>1.4-5.8</td>
<td>$-1$·10$^4$</td>
<td>$-6$·10$^4$</td>
</tr>
<tr>
<td>S6</td>
<td>Li-NaF-KF 46.5:11.5:42</td>
<td>SWH</td>
<td>1-10</td>
<td>$10^{15}$-$10^{18}$</td>
<td>0.8-4.0</td>
</tr>
</tbody>
</table>

$^*$ to be refined after salt properties measurement.

### Code Development and Distribution

**Recent features of developed and distributed CONV-2D and 3D codes**
- Convection due to heat generation in the liquid melt or in the side heated wall
- Turbulence model for high Ra numbers
- Model of a "mushy" region
- Heat conductivity in solid phases (crust, vessel, structures)
- Melting and re-freezing of a crust
- Melting of steel structures
Project Findings from Phase 1

- The experimental program reached the following major findings:
  - the C-22 corium melt employed stratified into two liquids having different densities. An equilibrium was reached in the AW-200-1 test. In AW-200-2 and TULPAN stratification tests the equilibrium was not attained.
  - The axial stratification in two liquids produced axial regions having different U/Zr ratios along the height.
  - As a consequence the corium melt physical properties varied in different zones with own:
    - density
    - heat conductivity
    - liquidus temperature
    - \(T_c - T_{\text{melting}}\)
- The Project believes that such behaviour of corium melts will also be valid for other initial compositions with metallic Zr
- The Project submits that stratification effects the melt-pool natural convection

Summary

- Technical issues to conduct large scale experiments were resolved
  - Methods of the heating up of core materials beyond liquidus temperatures (T>2400 °C)
  - Compatible materials
  - Measurement techniques
- New data on corium properties were obtained
- Analytical tools for pre-test and post test analysis were developed
- Results of corium tests showed significant difference in corium behaviour in comparison to simulant materials
3. Session I

In-Vessel Retention 2

Chairperson: A. V. Jones (JRC Ispra)
Co-chairperson: A. Serizawa (Kyoto Univ.)
3.1 EXPERIMENT AND ANALYSIS ON IN-VESSEL DEBRIS COOLABILITY IN ALPHA PROGRAM

Yu Maruyama, Norihiro Yamano, Kiyofumi Moriyama, Hyun Sun Park, Tamotsu Kudo, Yanhua Yang and Jun Sugimoto

Severe Accident Research Laboratory
Department of Reactor Safety Research
Japan Atomic Energy Research Institute

ABSTRACT

In-vessel debris coolability experiments have been performed in ALPHA program at JAERI, applying aluminum oxide (Al$_2$O$_3$) produced by a thermite reaction as a debris simulant. Approximately 30 kg or 50 kg of Al$_2$O$_3$ was poured into a pool of nearly saturated water (450 K at 1.3 MPa) formed in a lower head experimental vessel (LHEV). Post-test examination with an ultrasonic technique and thermal responses of the LHEV wall implied that the interfacial gap was formed between the solidified Al$_2$O$_3$ and the LHEV wall. The LHEV temperature was sharply increased at the beginning of the experiments. The observed temperature increase rate was much smaller than a calculated value based on heat conduction through the LHEV wall. Later, steep temperature decrease was found on the LHEV outer surface while Al$_2$O$_3$ was kept at a high temperature. It is supposed that the gap acted as a thermal resistance during the heat-up stage and water subsequently penetrated into the gap to realize the effective cooling of the LHEV wall. The maximum heat flux at the inner surface of the LHEV facing to Al$_2$O$_3$ was roughly evaluated to be 320 kW/m$^2$ through 600 kW/m$^2$. In parallel with the experiments, the development of a computer code, CAMP (Coolability Assessment of Melt Pool), is in progress. The framework of the code was completed, which is capable of analyzing laminar natural convection of an incompressible viscous fluid, solid-liquid phase change and thermal responses of a lower head wall. The code was applied to the post-test analysis of the in-vessel debris coolability experiments, and the influence of the interfacial gap on the thermal behavior of the LHEV and Al$_2$O$_3$ was examined. A new experimental apparatus is in design with major aims to realize the simulation of a decay heat generation within debris and a pressure load imposed on the lower head. Improvement of CAMP code will be made in this fiscal year for internal heat generation, turbulent flow, boiling heat transfer in the gap and water penetration into the gap. A concept of a packed bed type in-vessel debris retention design is proposed. A correlation for critical heat flux was derived, based on the results from the experiments for the flooding phenomena in a packed bed. A condition was found, in which the predicted critical heat flux sufficiently exceeded the heat flux expected in a realistic situation.
2. IN-VESSLE DEBRIS COOLABILITY EXPERIMENTS

Conceptual Diagram of Experiments

- Model Containment Vessel
- Water Supply System
- Thermite Melt Generator
- Water
- Al2O3
- Fe
- Stainless Steel Liner
- Carbon Steel
- Lower Head Experimental Vessel (LHEV)
- Nitrogen Supply System (Pressurization)

1. OBJECTIVES

- To investigate interactions between debris and water filled lower head
- To identify inherent cooling mechanisms of debris and lower head structure, and to clarify the conditions to achieve the cooling
- To develop and validate analytical models, and to propose in-vessel debris retention design based on accumulated experimental insights
Findings from Post-Test Observation and Examination

- Accumulation of a small amount of Al₂O₃ particles on the solidified Al₂O₃ layer
- Rough surface on the top of the solidified Al₂O₃ layer
- Smooth surface on the bottom of the solidified Al₂O₃ layer with several hollows, channels and crevices
- Formation of a thin porous layer between the core and the outer layer of the solidified Al₂O₃
- No thermal damage on the LHEV wall
- Detection of the interfacial gap between the solidified Al₂O₃ layer and the LHEV wall by the ultrasonic measurement

Major Conditions of In-Vessel Debris Coolability Experiments

<table>
<thead>
<tr>
<th></th>
<th>Al₂O₃ Mass (kg)</th>
<th>Initial Water Depth (m)</th>
<th>Initial Water Temperature (K)</th>
<th>Ambient Pressure (MPa)</th>
</tr>
</thead>
<tbody>
<tr>
<td>IDC001</td>
<td>-30</td>
<td>0.3</td>
<td>445</td>
<td>1.3</td>
</tr>
<tr>
<td>IDC002</td>
<td>-50</td>
<td>0.3</td>
<td>450</td>
<td>1.3</td>
</tr>
</tbody>
</table>
3. ANALYSIS WITH CAMP CODE

Development of CAMP (Coolability Assessment of Melt Pool) Code

Features of the first version
- Finite volume scheme
- Semi-implicit time integration
- Hybrid numerical grid (triangle and rectangle cross-sectional cells)
- Laminar natural convection of incompressible viscous fluid
- Solid-liquid phase change of debris
- Thermal interaction of lower head with debris (heat conduction of lower head)

Cross-Sectional View of Noding in Post-Test Analysis (IDC002)

Initial and Boundary Conditions in Preliminary Post-Test Analysis for IDC002 with CAMP Code

Initial Temperature
- Aluminum Oxide: 2500K
- LHEV: 440K
- Water: 468K

Observed Temperature History of LHEV Outer Surface and Comparison with Heat Conduction Calculation for IDC002
Temperature History of LHEV Wall Calculated with CAMP Code (Gap Width 1mm) and Comparison with Experimental Results in IDC002

Temperature History of LHEV Wall Calculated with CAMP Code (Gap Width 2mm) and Comparison with Experimental Results in IDC002

Solid Fraction in Debris Simulant Calculated with CAMP Code for IDC002

Temperature History of Debris Simulant Calculated with CAMP Code and Comparison with Experimental Result in IDC002

4. FUTURE PLAN

Schematic of Apparatus for Planned In-Vessel Debris Coolability Experiments
5. CONCEPT OF IVR DESIGN

Concept of IVR Design and Geometry in Evaluation of Critical Heat Flux

\[ q_{\text{lim}} = \frac{\Delta h \cdot A_{\text{rev}} \cdot C^2 \cdot \sqrt{g \cdot D_e \cdot (\rho_f - \rho_l)}}{\rho_f^{1/4} + \rho_l^{1/4}} \]

\[ j^* = \sqrt{\frac{\rho_f}{g \cdot D_e \cdot (\rho_f - \rho_l)}} \quad j^* = \frac{v_{\text{inl}}}{\sqrt{g \cdot D_e \cdot (\rho_f - \rho_l)}} \]

\( D_e \): Hydraulic Equivalent Diameter of Packed Bed

Schematic of Apparatus for Air-Water Flooding Experiments
6. SUMMARY

- Post-test examination and thermal responses of the LHEV wall during the initial phase of the experiments implied that the interfacial gap was formed between the solidified Al₂O₃ layer and LHEV wall, and water penetrated into the gap.

- The first version of CAMP code was applied to the analysis of the in-vessel debris coolability experiments, and the influences of the interfacial gap on thermal transient behavior were examined.

6. SUMMARY (CONT'D)

- A new experimental apparatus is under design with major aims to realize the simulation of decay heat generation within debris and pressure load onto lower head.

- A concept of packed bed type IVR design was proposed. A condition was found, in which predicted critical heat flux sufficiently exceeded the heat flux expected in a realistic situation.
3.2 SONATA-IV Experiments on In-vessel Debris Coolability and Retention

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Fax) +82-42-861-2574

Summary

The results obtained from the metallurgical examinations in the TMI-2 Vessel Investigation Project showed that the lower head experienced a comparatively rapid cooldown. The evaluation of this rapid cooling mechanism is very important in the prediction of severe accident progression and the establishment of an accident management strategy. A cooling mechanism due to boiling in gaps between the debris crust and the RPV wall was proposed for this rapid cooling. KAERI launched a research project named SONATA-IV(Simulation Of Naturally Arrested Thermal Attack In Vessel) to investigate the possibility of in-vessel debris cooling through this narrow gap. Two major experiments of LAVA(Lower-plenum Arrested Vessel Attack) and CHFG(Critical Heat Flux in Gap) are in progress.

The LAVA experimental facility was constructed and the functional tests have been completed. The experiments were performed inside the pressure vessel, which has an inner diameter of 2.4 m and a height of 4.8 m. The experimental apparatus is composed of a thermite melt generator, a melt holder, a test section of a lower head vessel and a gas supply system. The lower head vessel is made of carbon steel, is composed of hemispherical and cylindrical parts having inner diameter of 0.5 m and thickness of 0.025 m. In the present study, the validation of gap formation and the measurement of gap thickness are of prime importance. Therefore, the experiments have been performed under varying the initial conditions of melt and water. Depending on the test conditions, the melt holder is to be designed to separate iron and alumina, and only Al₂O₃ will be used as a corium simulant. The first experiment, LAVA-1, was performed, in which 40 kg of Al₂O₃/Fe thermite melt was poured into a 60 K subcooled water pool under the elevated pressure of 17 bar. During the initial period of the debris cooling phase, the temperature reduction rate of the debris and lower head vessel were 6 K/s and 0.24 K/s, respectively. These temperature reduction rates may imply that when the melt was poured into the lower head vessel, the iron was welded to the inner surface of the lower head vessel. For a precise examination of the interface configurations between the debris crust and lower head vessel, the test section was cut along a center line. Also, metallurgical inspections of the debris and lower head vessel samples will be executed to observe the structural deformations.

The VISU-II experiments have been done to visualize the flow behaviour inside a hemispherical gap and to understand the CHF-triggering mechanism. According to visual observation, the counter-current flow limit(CCFL) phenomenon prevents water from wetting the heater surface and induces CHF. The lessons obtained from the VISU-II experiments were reflected on the design of the CHFG test facility. The CHFG test is currently in progress. The purpose of the experiment is to assess the heat removal capacity through a hemispherical narrow gap. The experimental facility consists of an electric heater, a copper shell, a pressure vessel, a heat exchanger and coolant storage tank. The maximum heat flux at the shell surface is 90 kW/m². The outer diameter of the inner copper shell is 500 mm and various gap sizes of 0.5, 1 and 2 mm will be used. The experiments are being performed using distilled water; experiments with Freon-113 will be performed later. The CHF measurements will be made in the range of 1 to 10 atm. The CHF in a hemispherical narrow gap is detected using 66 K-type thermocouples embedded in a heated copper vessel. A basic form of a CHF correlation has been developed to correlate the measurements that will be made in this experiment. This correlation is based on the fact that the CHF in hemispherical gaps is enhanced by CCFL, and the Kutateladze number correlates CCFL data well in narrow gaps.
SOVATA-IV Experiments on In-vessel Debris Coolability and Retention

J.H. Jeong
Severe Accident Research Lab
Korea Atomic Energy Research Institute

Contents

- Motivation and Objectives
- LAVA Experiment
- CHFG Experiment
- Summary

Objectives of SONATA-IV Research

- to understand the mechanism that sustained the integrity of the RPV during the TMI-2 accident,
- to develop and validate models for debris behaviour in the lower plenum and for vessel response during a severe accident,
- to assess the potential for melt retention within the RPV in view of operating and future NPP.

Integrated Efforts of SONATA-IV Phase 1 Study

Separate Effect Exp. | Main Experiment | Analysis
--- | --- | ---
TETRIS: Prompt key Test of Thermal Shock | LAVA: Gap Formation and Cooling | TEXAS: Melt Resuspension and Steam Explosion
VOSI & H.: Valuation in Gap | | CONV: Melt Pool Behaviour and Heat Transfer
COPL: Closure-Outlet Film Test | CHFG: Heat Transfer In Hemispherical Gap | CALF: Thermal Response and Failure of RPV
2-D SLICE: Full Scale Heat Flux Test | | Scaling
| Development of CHF Correlation

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LAVA Experiment
Lower-plenum Arrested Vessel Attack

Objectives

- Investigate Relocation of Corium Into Lower Head Vessel
- Investigate Interactions between Debris and Lower Head Vessel and Gap Formation
- Investigate Cooling of Melt and Thermal Response of Lower Head Vessel

Design Features

- Suppression of steam explosion by pressurization of the system
- Design Pressure of Pressure Vessel: 3 Mpa
- Pressurization: Nitrogen Supply System
- Use iron thermite (Fe + Al₂O₃ or Al₂O₃ only), no sustained heating.
- Test peripheral & central paths for delivery of molten material
- On-line gap measurement between RPV wall and corium crust
- Pressure difference between the inside RPV and the cavity
  - Pressurized RPV: 2 Mpa
  - Pressure in Cavity: 0.1 / 2 Mpa
- Linearly scaled-down lower plenum of RPV
  - Lower Head Vessel Inner Diameter/Thickness: 500 / 25 mm
  - Vessel Material: Carbon Steel

Instrumentation & Measurements

- LHV Outer Surface Temperature: K-type Thermocouple; 9 Channel
- Melt Initial Temperature: Pyrometer, W/Re Thermocouple
- Dynamic Pressure: Melt / Water Interaction in Lower Head Vessel
- Coolant Temperature: K-type Thermocouple; 4 Channel
- System Pressure: Pressure Gauge; Full Range - 4 Mpa
- Gap Thickness: Ultrasonic Technique
- Melt & LHV Post Test Specimen Inspection
LAVA Experiment (Cond.)

Out-Door Preliminary Test

Objectives:
- Integrity of Low Head by thermal attack by corium
- Thermal response of Lower Head in case of the dry condition as reference data for LAVA experiments
- Performance test of thermite ignition, melting and delivery

Results:
- Lower head failure occurred by jet impingement
- Need a caution for a possibility of vessel failure in LAVA experiment when the superheated iron melt contacts with the vessel wall directly.

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Test Matrix

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>LAVA-PRE</td>
<td>20 kg</td>
<td>0.4 m</td>
<td>1.7/1.7*</td>
<td>Al₂O₃/Fe</td>
<td>480.0 K</td>
</tr>
<tr>
<td>LAVA-1</td>
<td>40 kg</td>
<td>0.5 m</td>
<td>1.7/1.7</td>
<td>Al₂O₃/Fe</td>
<td>480.0 K</td>
</tr>
<tr>
<td>LAVA-2</td>
<td>40 kg</td>
<td>0.5 m</td>
<td>1.7/0.1</td>
<td>Al₂O₃/Fe</td>
<td>480.0 K</td>
</tr>
<tr>
<td>LAVA-3</td>
<td>30 kg</td>
<td>0.5 m</td>
<td>1.7/1.7</td>
<td>Al₂O₃</td>
<td>480.0 K</td>
</tr>
<tr>
<td>LAVA-4</td>
<td>30 kg</td>
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<td>1.7/0.1</td>
<td>Al₂O₃</td>
<td>480.0 K</td>
</tr>
</tbody>
</table>

* / / = In-Vessel / Ex-Vessel Pressure

LHV Temperature History : LAVA_PRE Test

POWER-KAERI
Cross-section of LHV: LAVA_PRE Test

Cross-section of LHV: LAVA-1 Test

LHV Temperature History: LAVA-1 Test

Continuous Ball Indentation Test Result
**Experimental Method**

- VISU-I: To visualize boiling heat transfer in a small scale and small heat flux conditions (Completed)
- VISU-II: To visualize boiling heat transfer in a medium scale and medium heat flux conditions (Completed)
- CHFG (Critical Heat Flux in Gap): To measure heat transfer characteristics and to develop CHF correlation in a large scale condition (In Progress)
- CCFL & 2-D SLICE: To investigate counter-current flow limitation in annular gap and 2-D full scale effect, respectively (Planning)

**VISU-II Experiments**

- Test Geometry and Conditions
  - Heater Diameter: 250 mm
  - Gap Size: Approximately 1.0 mm
  - Coolant: Water
  - Pressure: 1 atm
- Heater
  - Electric heater with Wood's metal filled
  - Max. Heat Flux: 70 kW/m²

**Observations**

- Wide bubbles are getting more energetic with power
- Steam and water make multiple separated flow-paths around the heater periphery at random
- Irregular gap size due to non-symmetric Pyrex glass vessel
- CCFL occurs at the top-end of the gap where its size is smaller
  => water supply is limited
  => heater surface dries out (CCFL controlled CHF)
- Dryout region expands with a heater power increase
**Objectives**

- To Identify Heat Removal Capacity through the Gap in a Hemispherical Narrow Gap
- To Study Boiling Heat Transfer and to Develop CHF Correlation in a Hemispherical Narrow Gap

- No Correlations are available in literature
- New Correlations will account for Periodic, Turbulent, Churn, Wavy, and possibly Liquid-deficient Regimes for Gap Boiling

**Test Facility Layout**

- Diameter: 500 mm
- Gap Size: 0.5, 1.0, 2.0 mm
- Coolant: Water, Freon 113
- Pressure: 1 - 10 atm
- Max. Heat Flux: 0.1 MW/m²
T/C locations

Tests Matrix

<table>
<thead>
<tr>
<th>Gap (mm)</th>
<th>0.5</th>
<th>1.0</th>
<th>2.0</th>
<th>5.0</th>
</tr>
</thead>
<tbody>
<tr>
<td>Heater 1.2</td>
<td>W, F</td>
<td>W, F</td>
<td>W, F</td>
<td>W, F</td>
</tr>
<tr>
<td>Heater 1</td>
<td>W</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Heater 2</td>
<td>W</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Heater 1: Upper Heater, Heater 2: Lower Heater,
W: water, F: freon
Pressure: 5 points in the range of 1 ~ 10 atm

Temperature history (2mm gap, 1atm)

Temperature variation with q''

Gap size: 2mm
Press: 1atm
Avg. q'' (kW/m²)
32, 42
52, 60
Self-propagation of dryout region

Gap size: 2mm  
Press: 1atm

At time = 0,  
Avg. $q'' = 68 \text{ kW/m}^2$

$0s, + 500s$  
$+ 1400s, + 1600s$

Summary

SONATA-IV Phase 1 studies are in progress very actively at KAERI, with an integrated manner in the areas of experiments and analyses. The construction of major experimental facilities of LAVA and CHFG are finished. The experiments are being carried out.

- LAVA-0&1 tests were performed using Al2O3/Fe thermite melt. Temperature reduction rate of the lower head vessel is relatively low and the iron is welded to the inner surface of the vessel.

- According to the VISU-II experiment, CCFL phenomenon prevent water from wetting the heater surface and induce CHF. CHFG experiment will confirm the gap cooling mechanism using detailed measurement of the temperature histories of the inner wall.

- The SONATA-IV Phase II studies is under planning stage, in which the test facilities will be scaled up to use a large amount of real material and simulant (~ 200 kg).
3.3 Molten Material Heat Transport Tests with Coolant Boiling

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Abstract

This paper presents results of experimental studies on the heat transfer and solidification of the molten metal pool with overlying coolant with boiling. The metal pool is heated from the bottom surface and coolant is injected onto the molten metal pool. As a result, the crust, which is a solidified layer, may form at the top of the molten metal pool. Heat transfer is accomplished by a conjugate mechanism, which consists of the natural convection of the molten metal pool, the conduction in the crust layer and the convective boiling heat transfer in the coolant. This work examines the crust formation and the heat transfer rate on the molten metal pool with boiling coolant. The simulant molten pool material is tin (Sn) with the melting temperature of 232°C. Demineralized water is used as the working coolant. The crust layer thickness was ostensibly varied by the heated bottom surface temperature of the test section, but not much affected by the coolant injection rate. The correlation between the Nusselt number and the Rayleigh number in the molten metal pool region of this study is compared against the crust formation experiment without coolant boiling and the literature correlations. The present experimental results are higher than those from the experiment without coolant boiling, but show general agreement with the Eckert correlation, with some deviations in the high and low ends of the Rayleigh number. This discrepancy is currently attributed to concurrent rapid boiling of the coolant on top of the metal layer.

I. INTRODUCTION

During a hypothetical severe accident in nuclear power plants, it is possible to form stratified fluid layers. These layers may be composed of high temperature molten debris pool and water coolant in the lower plenum of the reactor vessel or in the reactor cavity.[1-5] Also, molten debris pool may be stratified into a metal layer and an oxide layer on account of their density difference.[6-8] Molten metal layer is located in the upper region and cooled by overlying coolant which may undergo boiling. As a result, the crust, which is a solidified layer of the molten pool, may form at the top. Heat transfer is accomplished by a conjugate mechanism consisting of the natural convection of the molten metal pool, the conduction through the crust layer and the convective boiling heat transfer to the coolant.

The heat transfer and solidification processes in the molten metal pool are of fundamental importance in the severe accident progression. A number of experimental and theoretical investigations were performed to understand the solidification and the change of heat transfer rate of the debris pool which greatly affects the accident progression.

An experimental study on the crust formation and heat transfer characteristics of the molten metal pool with overlying coolant with boiling was performed to determine the heat transfer rate of the molten metal pool. Tests were performed under the condition of the bottom surface
heating in the test section and the forced convection of the coolant, which is injected onto the molten metal pool. The test parameters spanned the heated bottom surface temperature of the molten metal pool, the coolant injection rate and the coolant injection temperature.

II. EXPERIMENTAL SETUP

Figure 1 shows the schematic diagram of the test section. The inner dimension of the rectangular test section is 25cm in length, 35cm in height, and 25cm in depth. The test section is made of 10mm thick STS304 stainless steel. The heights of the molten metal and the coolant layer are 20cm and 15cm, respectively. A 20kW heater is installed in the bottom horizontal plate of the test section. The viewports are installed using a quartz glass at the front and the back of the test section. Four sides of the test section are insulated with a 4cm thick Fiberfrax material to reduce heat loss. A digital pump is installed to deliver a uniform mass flow of the coolant onto the molten metal pool. The temperature distribution inside the test section is measured using 51 thermocouples, which are placed in three arrays of thermocouple bundles located at the one-fourth, one-half and three-fourth positions of the length of the test section. The simulant molten pool material is tin (Sn) with the melting temperature of 232°C, and demineralized water is used as the working coolant. The test parameters are the bottom surface temperature ranging from 253°C to 266°C, the injection coolant mass flow rate in the range of 0.5kg/min to 2.5kg/min, and the injection coolant temperature ranging from 82°C to 95°C.

In this experiment, the molten metal is injected to the test section, and the bottom surface temperature is set at a prescribed value. Next, the coolant is injected onto the molten metal in the test section at the preset mass flow rate. The coolant is recirculated in a closed loop until the steady state condition is achieved.

III. RESULTS AND DISCUSSION

Figure 2 shows temperature distribution of the test section as a function of the coolant injection rate at a bottom heating temperature of 255°C. The portion below the horizontal dotted line is the metal layer, and the above is the coolant layer. The vertical dotted line is the melting temperature of tin. The temperature varies linearly in the solidified region, and is almost uniform in the molten pool and in the coolant. The crust thickness and temperature distribution are barely affected by the coolant injection rate. Figure 3 displays the temperature profile in the metal and coolant layers for the bottom heating surface temperatures of 253°C, 255°C, 258°C, 262°C and 266°C with the coolant injection rate of 1.5kg/min. The results illustrate that crust thickness and temperature of the metal layer are affected by the bottom surface temperature. As can be seen from Figures 2 and 3, the crust layer thickness may be greatly varied by the heated bottom surface temperature of the test section, but not much affected by the coolant injection rate.

Table 1 presents the natural convection heat transfer rates and crust thickness in the molten metal layer. The values were obtained after the steady state had been accomplished. In actuality, the quasi-steady state apparently produced fluctuations in temperature in the metal layer and the coolant. The data in Table 1 were obtained from the average values for the duration of measurement after the quasi-steady state had been reached. The crust thickness is determined by the linear interpolation method from the thermocouple reading data and the melting temperature (232°C) for tin. The heat flux can be derived from the temperature difference between the top surface and the bottom surface of the crust layer using the heat conduction equation.
In this study, the actual heat flux was calculated from the temperature measurements by the thermocouples located right underneath the metal layer and coolant interface and just above the melting point, and the distance between the two points in-between.

This was necessary because the interfacial temperature $T_i$ and the melting location of the metal layer normally fell between the two thermocouple locations.

The heat transfer coefficient of the molten metal pool is derived from this heat flux as follows.

This calculation is based on the assumption that there is no heat loss to the environment. Natural convection heat transfer in the molten pool is generated by the buoyancy force arising from the density difference. The Nu number and the Ra number are defined as follows.
Where \( h \) : heat transfer coefficient in the molten pool
\( L \) : height of molten pool layer
\( \Delta T \) : temperature difference \((T_b - T_m)\)
\( g \) : gravitational acceleration
\( \alpha \) : thermal diffusivity
\( \beta \) : thermal expansion coefficient
\( \nu \) : kinematic viscosity

The relationship between the Nusselt number and the Rayleigh number in the molten metal pool region was determined and compared against the experiment without coolant boiling and the literature correlations. The experiment without coolant boiling was performed using the low temperature melting alloy which has a composition by weight percentage of Bi(49.92%), Pb(26.93%), Sn(13.28%) and Cd(9.85%) with the melting temperature of 70°C. The bottom heating method was the same as this work but the cooling mechanism was subcooled coolant natural convection using the heat exchanger at the top of the test section.[7] Figure 4 shows a comparison of the present experimental results with the experiment without coolant boiling and other correlations in the molten metal pool region. Many experimental studies were performed on the Rayleigh-Bernard problem which deals with the natural convection heat transfer. Their results are generally presented in the following fit correlation.

\[
Nu = aRa^b
\] (6)

Available correlations are the Eckert correlation[8] for tin, the Globe and Dropkin correlation[9] for mercury, and the Chu and Goldstein correlation[10] for water, all of which were developed in an enclosure without phase changes. The Globe and Dropkin correlation and the Chu and Goldstein correlation are developed empirically. The Eckert correlation is a theoretical relation for natural convection of low Prandtl number materials for vertical plates. The equation is generally presented in the following correlation.

\[
Nu = 0.68[Pr/(0.952 + Pr)]^{0.25}Ra^{0.25}
\] (7)

Empirical correlation:

- Globe and Dropkin: \( Nu = 0.051Ra^{0.333} \) (8)
- Chu and Goldstein: \( Nu = 0.183Ra^{0.278} \) (9)

Theoretical correlation:

- Eckert: \( Nu = 0.24Ra^{0.25} \) (10)

Equation (10) is developed from equation (7) by substituting for Pr the value of 0.015 for Tin.

The present experimental results for the heat transfer from the molten metal pool are apparently higher than those without coolant boiling, but show better agreement with the Eckert correlation than with the other correlations. However, the experimental results of the heat transfer are lower than the Eckert correlation in the low Rayleigh number region and higher in the high Rayleigh number region.
IV. CONCLUSION

An experimental study was performed to investigate the heat transfer characteristics and crust formation of the molten metal pool natural convection concurrent with forced convective boiling of the overlying coolant. The temperature distribution and crust layer thickness in the metal layer were appreciably affected by the heated bottom surface temperature of the test section, but not much by the coolant injection rate.

In this experiment, the heat transfer is achieved with accompanying solidification in the molten metal pool by coolant with boiling. The present experimental results of the heat transfer on the molten metal pool are apparently higher than those without coolant boiling, but show general agreement with the Eckert correlation. However, the present experimental results of the heat transfer show deviations in the low and high Rayleigh number regions. This is probably because this experiment was performed in concurrence of solidification in the molten metal pool and the rapid boiling of the coolant. On the other hand, the comparison experimental tests were performed without coolant boiling and the correlation was developed for the pure molten metal without coolant phase change. Note however that the test results may not directly be applied to the actual reactor accident condition because this study was performed in the lower Ra number region than for the actual molten debris. During a severe accident, the molten debris reaches the $10^{15}$–$10^{16}$ range of Ra number for the oxide pool and the $10^{9}$–$10^{10}$ range for the metallic layer. Further study is planned to investigate the effect of the boiling coolant in the high temperature and high Rayleigh number region.

REFERENCES

Table 1. Heat Transfer Rate and Crust Thickness of the Molten Metal Pool

<table>
<thead>
<tr>
<th>Bottom Surface Temp. (°C)</th>
<th>Coolant Injection Rate</th>
<th>Heat Flux (W/m²)</th>
<th>Crust Thickness (cm)</th>
<th>Nu Number</th>
<th>Ra Number</th>
</tr>
</thead>
<tbody>
<tr>
<td>253</td>
<td>0.5</td>
<td>5.97E+4</td>
<td>12.79</td>
<td>7.14 (±8.3%)</td>
<td>1.62 E+6 (±7.5%)</td>
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<td>1.0</td>
<td>6.02E+4</td>
<td>11.83</td>
<td>7.71 (±6.6%)</td>
<td>2.50 E+6 (±6.5%)</td>
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<td>1.5</td>
<td>6.05E+4</td>
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<td>2.68 E+6 (±6.9%)</td>
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<td>2.0</td>
<td>6.14E+4</td>
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<td>6.21E+4</td>
<td>11.09</td>
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<td>3.50 E+6 (±6.1%)</td>
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<tr>
<td>255</td>
<td>0.5</td>
<td>7.74E+4</td>
<td>10.46</td>
<td>11.09 (±8.2%)</td>
<td>4.14 E+6 (±10.0%)</td>
</tr>
<tr>
<td>255</td>
<td>1.0</td>
<td>7.68E+4</td>
<td>10.68</td>
<td>10.85 (±7.8%)</td>
<td>3.83 E+6 (±8.9%)</td>
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<td>1.5</td>
<td>7.89E+4</td>
<td>10.02</td>
<td>11.21 (±8.9%)</td>
<td>5.01 E+6 (±9.9%)</td>
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<td>1.01 E+7 (±10.6%)</td>
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<td>1.10 E+7 (±11.7%)</td>
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<td>1.06 E+7 (±9.3%)</td>
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<td>1.11 E+7 (±11.2%)</td>
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<td>1.04 E+7 (±8.9%)</td>
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<td>3.55 E+7 (±6.8%)</td>
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<td>24.09 (±7.8%)</td>
<td>3.52 E+7 (±5.8%)</td>
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<tr>
<td>266</td>
<td>2.0</td>
<td>1.74E+5</td>
<td>3.96</td>
<td>20.23 (±8.4%)</td>
<td>2.24 E+7 (±9.5%)</td>
</tr>
<tr>
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<td>2.5</td>
<td>1.80E+5</td>
<td>4.26</td>
<td>23.04 (±9.4%)</td>
<td>3.49 E+7 (±6.6%)</td>
</tr>
</tbody>
</table>

Fig. 1 Schematic Diagram of the Experimental Setup
Fig. 2 Temperature Distribution in Metal Layer and Coolant (Bottom Temperature: \(255\, ^\circ\text{C}\))

Fig. 3 Temperature Distribution in Metal Layer and Coolant (Coolant Injection Rate: \(1.5\, \text{kg/min}\))

Fig. 4 Comparison of the Experimental Results with Literature Correlations
Molten Material Heat Transport Tests with Coolant Boiling

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Contents

- Introduction
  - Needs and Background
  - Objectives
- Experimental Setup
- Experimental Method
- Results and Conclusion

Needs and Background

- Possibility of the Stratification of Two Fluids of the Corium Pool and the Coolant under Severe Accidents
- Importance of the Heat Transfer and Solidification in the Molten Debris Pool in the Severe Accident Progression
- No Investigations on the Heat Transfer Characteristics of the Molten Debris Pool with Solidification by Overlying Coolant with Boiling

Objectives

- Experimental Study on the Heat Transfer and Solidification Process in the Molten Metal Pool
- Examination of the Conjugate Heat Transfer Mechanisms in the Molten Metal Pool with Solidification by Overlying Coolant with Boiling
  - Natural Convection in Molten Metal Pool
  - Conduction Heat Transfer in the Crust Layer
  - Boiling Heat Transfer in the Coolant
- Development of the Natural Convection Heat Transfer Coefficient in the Molten Metal Pool
Experimental Setup

- **Test Section**
  - 25 cm x 25 cm x 35 cm Rectangular Shape
  - Bottom Heating with 19.8 kW Heater
  - Molten Metal 20 cm, Coolant 15 cm in Height
  - Installing View Ports in Front and Back
  - 4 cm Thick Fiberfrax Insulation to Reduce Heat Loss

- **Bottom Plate Heater**
  - 35 cm x 35 cm Area
  - 2.2 kW Heating Rod 9EA (19.8 kW)

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Experimental Setup

- **Schematic Diagram of the Experimental Setup**

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Experimental Method

- **Closed Loop (Coolant Recirculation)**
- **Pressure in Test Section**: 1 atm
- **Experimental Procedure**
  1. Molten Metal Injection in the Test Section and Heating the Coolant Supply Tank
  2. Heating the Test Section Bottom Surfaces to a Preset Temperature
  3. Coolant Injection to the Upper Surface of Molten Metal
  4. Accomplishment of the Steady State Condition
Experimental Method

- Experimental Parameters
  - Temperature of the Test Section Bottom Surface: 253 ~ 266 °C
  - Coolant Injection Flow Rate: 0.5 ~ 2.5 kg/min

- Measurement Parameters
  - Temperature Distribution of the Metal Layer and Coolant
  - Crust Thickness

Experimental Results

- Heat Flux Calculation
  - Using the Heat Conduction Equation in Crust Layer
    \[ q^* = k \frac{T_{coolant} - T_{crust}}{x} \]

- Crust Thickness Measurement
  - Interpolation of the Melting Temperature from the Thermocouple Reading Data

- Temperature Profile in Tests

- Heat Transfer Rate and Crust Thickness

<table>
<thead>
<tr>
<th>Control Injection Rate</th>
<th>Temperature of the Test Section Bottom Surface</th>
<th>Coolant Injection Rate</th>
<th>Heat Flux (W/cm²)</th>
<th>Crust Thickness</th>
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</thead>
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<td>12.5 kg/min</td>
<td>255 °C</td>
<td>10 kg/min</td>
<td>1.05</td>
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<td>255 °C</td>
<td>25 kg/min</td>
<td>1.67</td>
<td>7.56</td>
</tr>
</tbody>
</table>
Experimental Results

• Nu Number and Ra Number Calculation in the Molten Metal Pool

\[ h = \frac{T_s - T_w}{T_s - T_m} \]

- Nu Number

\[ Nu = \frac{hL}{\frac{1}{\alpha} + \frac{1}{\beta}} \]

- Ra Number

\[ Ra = \frac{g \beta \Delta T L^3}{\alpha} \]

YOKOHAMA, JAPAN, October 6-8, 1997

Experimental Results

• Comparison with the Experiment without Coolant Boiling

- Low Temperature Melting Alloy (70°C)
  : Bi(49.92%), Pb(26.93%), Sn(13.28%), Cd(9.85%)

- Bottom Heating Method

- Cooling Mechanism
  : Subcooled Coolant Natural Convection

Experimental Results

• Literature Correlations for Heat Transfer

\[ Nu = \alpha Ra^\frac{3}{4} \]

- Eckert(1954) \[ Nu = 0.23Ra^{0.67} \] for Tin

- Globe and Dropkin(1959) \[ Nu = 0.051Ra^{0.36} \] for Mercury

- Chu and Goldstein(1973) \[ Nu = 0.183Ra^{0.47} \] for Water

YOKOHAMA, JAPAN, October 6-8, 1997
**Experimental Results**

- Comparison with the Literature Correlations

![Graph showing comparison with literature correlations](image)

**Conclusions**

- Temperature Distribution and Crust Thickness in the Metal Layer are more affected by the Bottom Heating Temperature than by the Coolant Injection Rate
- General Agreement with the Eckert Correlation
- Some Deviation of the Heat Transfer in the Low and High Rayleigh Number Regions
- Literature Correlations were Developed for the Pure Liquid without Phase Change
- Nucleate Boiling Region of the Coolant vs. Film Boiling

**Further Study**

- High Temperature and High Ra Number Region
- Bottom Heating Condition
  - Constant Temperature $\rightarrow$ Constant Heat Flux
- Coolant Boiling Mechanism
  - Nucleate Boiling $\rightarrow$ Film Boiling
- Coolant Simulant
  - Water $\rightarrow$ R113

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**Department of Nuclear Engineering**
**Seoul National University**
3.4 Lattice Gas Automata Simulations of flow through Porous Media

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ABSTRACT

In the course of a severe accident, a debris bed may be formed from once-molten and fragmented fuel elements. In order to avoid further degradation of the reactor core, it is necessary to remove the heat from the debris bed since the debris bed still release the decay heat. So as to predict the coolability of the debris bed, it is important to precisely estimate flow patterns through complex geometry of debris bed in microscopic level. Lattice gas automata could be powerful tool to simulate such a complex geometry. As a first step of the study, fundamental numerical simulation were conducted in two dimensional systems by using the lattice gas automata method to clarify single phase flow patterns through porous media in mesoscopic level.

Immiscible lattice gas model is one of the lattice gas automata method and utilized for spinodal decomposition simulation of binary fluids. This model was applied to generate the complex flow geometry simulating porous media. It was approved that the complex flow geometries were successfully generated by the present method.

Flow concentration was observed in specified flow channels for lower Reynolds number. Two dimensional flow concentration was caused by the irregular flow geometry generated by the present method, since the flow selects the channels of lower friction. Two dimensional pressure distribution was observed relating to the concentrations of flow in specified channels.

The simulating results of the flow through the porous media by the present method qualitatively agree with the Ergun's equation. Quantitatively, the present results approach to Ergun's equation in higher Reynolds number than 10, although concentration of the flow in a specified flow channels were observed in lower Reynolds number than 10. It can be concluded that this technique is useful to simulate flow through complex geometry like porous media.
Lattice Gas Automata Simulations of Flow through Porous Media

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Presented at the Workshop on Severe Accident Research in Japan (SARI-97)
October 6-8, 1997, Yokohama, Japan

Contents

1. Introduction
2. Lattice gas automata models
   - ILG model
   - Technique to generate complex flow geometry
3. Simulation results (flows through porous media )
   - Computational domain and parameters
   - Flow pattern observations
   - Dependence of grain size distribution
   - Comparison with Ergun's equation
4. Conclusions

1. Introduction

Schematic diagram of debris bed formation during severe accident

ECC water

pressure vessel

Break

steam

core

debries bed

Large break of pipe → Debris bed formation → Core cooling CCFL

Background

- It is important for reactor safety analysis to precisely estimate flow patterns in the debris bed.

- Flow patterns in the debris bed are mainly complicated due to the phase change of the coolant activity and the complex geometry of the debris bed.

- Numerous experimental and numerical studies have clarified the flow through complex geometry in macroscopic level.

- Microscopic studies are, however, required to predict complicated flow in complex geometry of the debris bed.
Objectives

- To propose the technique to generate complex flow geometry simulating the debris bed using lattice gas automata method.

- To confirm applicability of the present technique after observation of flow patterns and comparison of friction factors with Ergun's equation.

- Single phase laminar flow of no phase change
  → Preliminary study of flow through the debris bed

2. Lattice gas automata models

Immiscible Lattice Gas automata (ILG) model

Evolution of the system

is specified by assumed collision and translation rules.

- collision
  binary or triple particle collide to change a microscopic lattice states

- translation
  particles move along lattice links

- coarse graining
  obtain macroscopic variables by averaging lattice states over a region of space

Collision rules of ILG model

Red particles move to the most red-riched neighboring site
to generate macroscopic behavior of surface tension

Lattices and lattice sites of ILG model
Separation of two components mixture by ILG model

Complex geometry of the surface is spontaneously formed between the two components.

Technique to generate complex flow geometry

Schematic of lattice states near the surface at a certain time step $t_p$

Red rectangle illustrates that the lattice site contains more red particles than blue

Lattice sites located at the surface are surrounded by both red- and blue-riched lattice sites.

3. Simulation results

Two 2-D simulations

Case (A): $t_p=200$
Case (B): $t_p=10000$.  

Time step $t_p$ will determine the grain size and shapes of the obstacles formed by the blue components.
- Computational domain

The average number of particles per lattice: $\rho = \rho_\gamma + \rho_\alpha$

for $t < t_p$  
$\rho_\gamma = 3.2$  
$\rho_\alpha = 1.6$

for $t > t_p$

Simulation results in case (A): $t_p = 200$

Simulation results in case (B): $t_p = 10000$

Simulation results
• Grain size distribution of solid obstacles as a function of the parameter $t_0$

![Graph showing grain size distribution]

• Comparison with Ergun's equation

Ergun's equation

$$f = \frac{33.3}{Re} + \frac{0.5833}{\epsilon}$$

Reynolds number: $Re = \frac{D_\text{w} \cdot \rho \cdot u_c}{\mu}$
Porosity: $\epsilon$

Hydraulic diameter: $D_\Delta = \frac{4 \cdot \text{(Fluid Volume)}}{\text{(Wetted Surface Area)}}$

Fanning's equation

$$f = \left( \frac{dp}{dx} \right) \frac{D_\Delta}{4} \frac{2}{\rho u_c^2}$$

• Comparison of the wall friction factors between Ergun's equation and estimated values by the present method

![Graph comparing wall friction factors]

4. Conclusions

- The complex flow geometry was successfully generated by the present method.

- Flow concentration was observed in specified channels for lower Reynolds number due to the irregular flow geometry.

- The simulating results by the present method qualitatively agree with Ergun's equation higher Reynolds number than 10.

- This technique is useful to simulate flow through complex geometry.
4. Session II

Computer Code Development

Chairperson: C. Allison (Innovative Systems Software)
Co-chairperson: K. Muramatsu (JAERI)
4.1 Status of ICARE2 and ICARE/CATHARE Development

Florian Fichot, Fabrice Babik, Magali Zabiego, Marc Barrachin, Patrick Chatelard, Valia Souillard, Stéphane Melis and Francine Camous

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CEA-Cadarache - 13108 ST PAUL LEZ DURANCE (France)

Workshop on Severe Accident Research held in Japan - SARJ-97
Yokohama (Japan), October 6-8, 1997.

Abstract

The ICARE2 code is developed by IPSN as a tool for calculating severe core degradation in PWRs. The aim is to provide a mechanistic analysis of fuel rods heat-up, cladding oxidation, debris bed formation, molten corium relocation, chemical interactions, hydrogen release, FP release and lower head integrity.

It is also used for the preparation and analysis of the experimental tests of the PHEBUS-FP program. A new version, named V3modO, will be released in 1998. That version will be coupled with the thermalhydraulics code CATHARE2 and will include three major improvements with respect to the previous version (V2mod2):

- The possibility to use a two-phase thermalhydraulics model in the core, instead of a single gas phase model.
- The possibility to calculate a whole accident sequence, including the primary circuit.
- The possibility to calculate the accident evolution up to the vessel failure thanks to a more complete modeling of the late degradation phenomena and a thermal and mechanical modeling of the vessel lower head.

The late degradation modeling is based on a porous medium approach. Oxidation of Zircaloy particles and dissolution of UO₂ particles by liquid ZrO₂ are modeled. The relocation of liquid corium is calculated on a 2D cylindrical meshing. The collapse of solid debris under specific conditions may also be taken into account.

After substantial melting of the debris, one or several empty cavities may appear, and a large molten pool may form. To deal with that configuration, models for the convective heat transfers inside a molten pool and for the 2D radiative heat transfers in cavities have been developed. The prediction of a thin crust on top of the molten pool is also possible.

Moreover, a 2D, gas phase thermalhydraulics model has been implemented, to be able to deal with a complex geometry, where intact rods, debris and void areas exist simultaneously in the core. This model is also based on a generalized porous medium approach.

The transport and release of fission products in the early and late degradation phases are calculated, and the resulting local power is deduced.

Finally, the mechanical and thermal modeling of the vessel lower head allows to calculate the evolution of the debris after their collapse into the lower plenum and to predict a possible vessel failure.

The coupling of ICARE2 with the advanced system code CATHARE2, developed by CEA, IPSN, EDF and FRAMATOME has been done, and several tests have been performed to check the operability of the coupled version in all the phases of the accident. So far, comparisons have been made between the coupled version and CATHARE2 during the LOCA phase of a LBLOCA accident, simple tests with rod degradation have been performed, as well as a TMI-2 transient up to the end of the oxidation phase.
Late degradation features - ICARE2 modeling strategy

- Several phases: 1 gas, 1 liquid (melt), 1 solid as particles and several solids as rods.
- Each phase is characterized by a volume fraction, composition, temperature and geometry.
- The gas and liquid flows are calculated in 2 dimensions (or 1D with parallel channels for the gas, according to user's choice).
- Solid particles may move (vertical collapse).
- Heat and mass balances are solved on a 2D meshing.
- Discretization of the structures is made automatically to simplify input data creation.
- Most of the models are not restricted to the core region but may also be applied in the lower plenum.
- Need for relevant material properties and interactions: Phase diagram U-Zr-O et U-Zr-O-Fe.

ICARE2 V3 mod2 version

- **Modeling (additional features with respect to V2):**
  - Thermohydraulics:
    - 2-D cylindrical, one-phase gas flow through rods or porous medium
    - Two-phase 1-D parallel channels in rod geometry
    - Variable geometry, unrestricted number of walls with possible oxidation
    - Modified properties of steam at high temperature (dissociation)
  - Late phase:
    - Transition from early to late phase with debris formation (according to user's criteria).
    - Implicit coupling of the momentum and energy balance for the melt.
    - IMURA-YAGI effective conductivity (valid, ACRR DC-1 and MP-1) + dispersion effects
    - 1-D model for temperature in the corium pool (boundary layer approach)
    - 2-D cylindrical radiative heat transfers in cavities (non-participative gas)
    - Coupling with PHARD - U-Zr-O phase diagram (H, Co, Ti, Ts, phases and composition)
    - Collapse of the debris (according to user's criteria)
    - UO$_2$ dissolution by ZrO$_2$ in debris configuration (solubility limit may be chosen)
Validation and applications:

**Validation**

- V2mod2.2: PHEBUS (B9+, FPT0 et FPT1), CORA W2
- V3mod0: ACRR MP-1, DC-1, BAFOND, SCARABEE BF1

**Applications**

- with V2mod2.2: Preparation of PHEBUS FPT2
  - Interpretation of PHEBUS FPT0 et FPT1: late phase
  - ICARE2/SET application to PHEBUS FPT1 et B9+ for statistical analysis of the results according to input data uncertainties
- with V3mod0: Preparation of PHEBUS FPT4S (with private V3mod0 version)
  - TMI-2 Phase II calculation

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1. ICARE2: 1.2 Future developments

**Version ICARE2 V3 mod0**

- Developments from mid-97 to mid-98 (new model; improved model):
  - Modeling of core plate and lower head (2D heat conduction)
  - Coupling with IBRAE model for lower head creeping/failure: LOHEY
  - Fission products transport with relocated materials (calculation of residual power)
  - Generalisation of the corium relocation model to treat in the same way:
    - early degradation: candling, crust formation,
    - flow of corium in the porous medium (Barracuda),
    - flow of corium in the by-pass or through the core plate.
  - Simple model for Zry oxidation by a mixture air/steam (coupled with the 2D gas flow)
  - Crust formation on the upper surface of the molten pool
  - Coupling with ELSA model for fission products release
  - Improvement of IBRAE models: UZRO and PHADI2

- Coupling with PHADI2 module: U-Zr-O-Fe diagram
- Extension of the 2D gas flow model for reactor applications:
  - algebraic model for turbulence;
  - friction and heat transfer coefficients for flow through rods;
  - model to take into account the presence of liquid water in the lower part of the vessel;
  - preliminary study of the dissociation of steam and the consequences on oxidation.
- Application of the 2D solver to a model for natural convection in a corium pool with solidification (Ph. D. thesis)
Calculation of phase 2 of TMI-2 Accident

Computational options

Boundary conditions:
- top: exchange with steam at 600K
- bottom: saturation temperature under the water level (~ 500K)
- external vessel: exchange with air at 320K

The pressure evolution inside the vessel is specified.
The steam mass flow rate evolution is specified.
The water level evolution is specified.

Degradation criterions:
- Cladding failure is declared if $T > 2250K$ and zirconia thickness is lower than 300 microns.
- Fuel debris are formed as soon as the cladding is molten or dislocated or absent.
2. ICARE/CATHARE 2.1 Status of the coupling

ICARE/CATHARE V1.3U (1D thermal hydraulics)

- The current version (V1) is based on the coupling of ICARE2 V3 mod 0 with the V1.3U version of the thermal hydraulics code CATHARE2.

- The aim of this coupling is to allow a pertinent description of the whole sequence of severe LWR accidents up to the vessel failure, using the same code.

- To achieve this, a two phase thermal hydraulics model based on the CATHARE2 two fluid six equations model - adapted to severe accidents conditions - has been implemented into ICARE2. The resulting two phase ICARE2 version has been introduced into CATHARE2 as a new core module.

ICARE/CATHARE - Validation and Applications

- As a first step, debugging with simple test cases and comparison to CATHARE2 or ICARE2 depending on the situation studied (1 or 2 phase flow).

- Comparison to CATHARE2 and analysis of the discrepancies on PWR transients (e.g. LB LOCAs).

- Beginning of validation on LOFT FP-2 and comparison with SCDAP/RELAP5.

- Preliminary analysis of TMI2 (phases 1 and 2). Some results are presented in the following.

- VVER 440 reactor: a small break LOCA calculation has been performed up to the core degradation. Results have been compared successfully to the CATHARE2 ones during the LOCA phase.
ICARE/CATHARE: Future work up to mid 98

- Improvement of oxidation calculation (semi-implicit treatment).
- Adaptation of the two phase flow model to porous media (debris bed).
- Improvement of numerical performances.
- Validation on TMI2 (phases 1 and 2) and LOFT-FP2.
- Calculation of LWR accidental sequences.
- Improvement of post processing.

ICARE/CATHARE V1

Till the release of this version (mid 98) some work remains to be done, in particular:

ICARE/CATHARE V2 (1D or 3D thermalhydraulics)

- In 1998, beginning of a coupling of ICARE2 V3 mod 0 to CATHARE2 V1.4E and its 3D two phase thermalhydraulics module. Besides 1D calculations, this version will allow a 3D description of the core and of the upper and lower plena.
- Work implied by this coupling will consist mainly in introducing ICARE2 V3 mod 0 into CATHARE2 V1.4E as an optional thermics module and coupling this module to the 1D and 3D CATHARE2 thermalhydraulics modules, after adaptation of these modules to severe accidents conditions.
- A first mock-up of ICARE/CATHARE V2 is foreseen to be ready at the end of 1998.
- Then, starting from this mock-up, the first ICARE/CATHARE V2 version will be developed. This will imply physical developments concerning reflooding, corium water interaction, extension of physical laws to severe accidents conditions, ..., improvement of numerical performance, validation and reactor calculations.

This first version is foreseen for 2000.
COMPARISON BETWEEN CATHARE AND ICARE/CATHARE CODES IN THE TMI-2 CALCULATION

TMI-2 CALCULATION
PRESSURE IN THE PRIMARY CIRCUIT

TMI-2 CALCULATION
TOTAL FLOW OF THE PRIMARY CIRCUIT

PHASES 1 AND 2
Abstract

The development and validation of the Containment Code System COCOSYS is needed for a comprehensive simulation of severe accidents in containments of light water reactors. COCOSYS is to enable the simulation of all relevant processes and conditions during the course of severe accidents, covering design basis accidents.

Models mechanistic as far as possible are used for analyzing the physical-chemical processes in containments. Essential interactions between the individual processes, like e.g. between temperature and flow distribution, hydrogen distribution, hydrogen combustion as well as fission product and aerosol behaviour will be treated in a comprehensive form. New models, like e.g. models for describing melt distribution and melt cooling will be integrated.

An early version of COCOSYS naturally not complete in modelling all of the relevant phenomena is ready so that validation is just taken up.

With its detailed approach to analyze containment behaviour during severe accidents COCOSYS will not only be able to describe important individual phenomena close to reality, but will also make it possible to demonstrate the interaction between these phenomena as well as the overall behaviour of the containment.

The ongoing development and validation of COCOSYS by GRS is supported by the Federal Ministry for Education, Science, Research and Technology.
COCOSYS

(Containment Code System) —
a detailed approach to analyze containment behaviour during severe accidents

H.-J. Allelein, J. Rohde
GRS-Cologne

Workshop on Severe Accident Research in Japan (SARJ - 97)

meaning of the acronym COCOSYS:
Containment Code System
general structure of COCOSYS:

Three main modules are linked to the global interface module.

These main modules are:

- **module THY** for thermal-hydraulics
  (based on RALOC MOD 4)
- **module AFP** for aerosol-fission products
  (based on FIPLOC 3)
- **module CCI** for corium-concrete interaction
  (based on a re-engineered version of WECHSL)

An early version of COCOSYS naturally not complete in modelling all of the relevant phenomena is ready so that validation is just taken up.

experiments calculated with COCOSYS:

- F 2 thermal-hydraulics
- HYJET 4 jet and stratification
- G x 4 recombination
- E 11-4 H₂ distribution
- E 11-8.1 catalytic foil
- M 7-1 thermal-hydraulics and spray system
- VANAM-M 3 aerosols and thermal-hydraulics
- KAEVER aerosols
- JAERI hydrodynamics

Essential interactions between the individual processes, like e.g. between temperature and flow distribution, hydrogen combustion as well as fission product and aerosol behaviour will be treated in a comprehensive form. New models, like e.g. models for describing melt distribution and melt cooling will be integrated.
planned activities for the medium term:

- review, modifications and completion of the COCOSYS concept
- extension of modelling
  - simulation of the interactions of internal spraying with thermal-hydraulics, aerosols, iodine
  - integration of a computational fluid dynamic (CFD) model
  - complete simulation of H₂-combustion (deflagration, DDT, detonation)
- continuation of validation and calculations for plant sequences
Summary

- The general structure of COCOSYS is realized.
- The main modules for thermal-hydraulics, aerosols and fission products and concrete - concrete interaction are coupled, but not all of the interactions are modelled up to now.
- The ongoing development and validation of COCOSYS by GRS is supported by the Federal Ministry for Education, Science, Research and Technology.
4.3 Development of Super Simulator "IMPACT"

Part(1) IMPACT System Configuration

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Advanced Simulation Systems Department, Nuclear Power Engineering Corporation

ABSTRACT

A large-scale project is underway at the Nuclear Power Engineering Corporation to develop the software for a "Super Simulation" system, capable of analyzing scenarios ranging from normal operation to hypothesized accident conditions. Designed as a system of interconnected, hierarchical modules, IMPACT's distinguishing features include mechanistic models and high speed simulation on parallel processing computers. The present plan is a ten-year program starting from 1993. The IMPACT software will be completed within the year 2000 followed by two years for refinement through extensive verification and validation against test results and available real plant data.

The first technical phase, which ends in fiscal 1997, consists of development of basic physics modules, boiling transition code, flow induced vibration code and fast-running modules for severe accident analysis. Initial versions of the modules are currently being verified and upgraded to be able to simulate various aspects of severe accidents.

These fast-running modules will be integrated into a prototype system to analyze severe accident scenarios within Fiscal 1997. The Control System to manage and coordinate the various modules is also being developed within NUPEC. This system will enable complete simulation of a severe accident without need for user intervention.

This work is sponsored under the contract by the Ministry of International Trade and Industry, Japan.
Development of Super Simulator "IMPACT" Part(1)

IMPACT System Configuration

SARJ-97 Oct.6, Yokohama, Japan

Nobuhide Sato, Hiroshi Ujita, Hiroichi Nagumo, Masanori Naitoh and Masayoshi Shiba

Nuclear Power Engineering Corporation

IMPACT : Integrated Modular Plant Analysis and Computing Technology

Objectives

Demonstrate the Available Safety Margin of Nuclear Power Plants

Means

Physics-based Mechanistic Simulation (Phenomenological Models)
Parallel Processing & GUI
Modular Structure

Goals

High-accuracy, detailed calculations
Fast Running
User assistance from input generation to comprehension
Code Maintainability & Flexibility

Contents

IMPACT Overview
Objectives, Goals
Implementation of Severe Accident Code
Control System / Simulation Supervisory System
Modeling of Severe Accident Modules (in Part-2)
Module Verification Tests (in Part-2)

Ultimate Software Structure of IMPACT

- Analysis System
  - Basic Physics Modules
    - Phenomenon-Specific Modules
  - Detailed Analysis
- Human Interface System
  - Mesh Generation
  - Documentation Support
  - Visualization, etc
- Knowledge Base
  - for Analysis Support
    - Artificial Intelligence
      - Expert Systems
- Information Management System
- Data-Base Management System
- Data Base
  - Plant Data
  - Physical Constants
  - Experimental Data
  - Model Correlations
  - Material Properties
- Microscopic Model Libraries
- Developed in Phase 1

IMPACT Overview

Objectives, Goals
Implementation of Severe Accident Code
Control System / Simulation Supervisory System
Modeling of Severe Accident Modules (in Part-2)
Module Verification Tests (in Part-2)
### Development Schedule

<table>
<thead>
<tr>
<th>Year</th>
<th>Phase</th>
<th>Description</th>
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<tbody>
<tr>
<td>1993</td>
<td>1993-1997</td>
<td>Conceptual Design</td>
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### Phase-1
- Basic Physics Modules
- High Speed, Large Scale Calculation Using Parallel Processors

### Phase-2
- Detailed Analysis Modules
- Total Control System

### Phase-3
- Detailed Analysis Modules
- Communication between Modules

### Execution as a Severe Accident Analysis Code

- **Steady-State/Transient**
  - Core Heat-Up
  - FP Release from Fuel
  - Thermal Hydraulics in RCS
  - FP Behavior in PCV

- **After Fuel Failure**
  - Core Heat-Up
  - FP Release from Fuel
  - Thermal Hydraulics in RCS
  - FP Behavior in PCV

### Analysis Modules in Severe Accident Code

- Thermal Hydraulics in RCS
- Thermal Hydraulics in PCV (including Hydrogen Combustion)
- FP Behavior in PCV
- Core Heat-Up
- FP Release from Fuel
- Molten Core Relocation
- FP Behavior in RCS
- Debris Coolability in Lower Plenum
- Debris Spreading

### Control System / Simulation Supervisory System

#### Subjects
- Module Selection
- Allocation of Analysis Modules to Processor Elements
- Time Step Control (Synchronization)
- Visualization

#### Parallel Computer System

- **IBM SP-2 (72PEs)**
- Distributed Memory, MIMD (Multiple Instruction Stream Multiple Data Stream)
- Performance 19.2 GFLOPS (total)

- IBM's Parallel Environment
  - MPI (Message Passing Interface) for linkage to other modules
Time Step Control and Dynamic Allocation of Processor Elements by Simulation Supervisory System

<table>
<thead>
<tr>
<th>PE Module</th>
<th>Time Step Δt</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 Supervisory System</td>
<td>Δt for PE Control</td>
</tr>
<tr>
<td>2 Analysis Module</td>
<td>Δt for Communication Among PEs</td>
</tr>
<tr>
<td>3 Analysis Module</td>
<td>Δt for Calculation</td>
</tr>
<tr>
<td>4 Analysis Module</td>
<td></td>
</tr>
<tr>
<td>5 Analysis Module</td>
<td></td>
</tr>
<tr>
<td>...</td>
<td></td>
</tr>
<tr>
<td>j-th Analysis Module</td>
<td>Δt</td>
</tr>
<tr>
<td>n-1 Analysis Module</td>
<td></td>
</tr>
<tr>
<td>n Analysis Module</td>
<td>Δt for Communication Among PEs</td>
</tr>
</tbody>
</table>

Interprocess Communication

3 Layers
Each Analysis Modules : SPMD (Single-Program Multiple-Data)
Severe Accident Analysis Code : MPMD (Multiple-Program Multiple-Data)

Visualization

Online Monitoring
Running Modules, PEs & Log Messages
Typical Process Parameters (Time History)
AVS (Offline)
Profile, Vector & Contour Plot
Generalized Output Format Common to Modules
Conclusions

System Configuration of IMPACT is Reviewed

Supervisory System Design has been Completed, and All Modules for IMPACT (Phase-1) is under Coding and/or Testing

Prototype System of the Integrated Severe Accident Analysis Code will be Completed by the end of this fiscal year
4.4 Development of Super Simulator "IMPACT"
Part(2) Modeling of Severe Accident Phenomena and Initial Verification Tests

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Masataka Hidaka, Akira Susuki and Naoyuki Ishida
Power & Industrial System R & D Division, Hitachi, Ltd.

Makoto Yamagishi
Reactor Control & Safety Engineering Department, Mitsubishi Heavy Industries, Ltd.

Nobuaki Abe
Nuclear Engineering Laboratory, Toshiba Corporation

ABSTRACT

IMPACT employs advanced methods of physical modeling and numerical computation and can simulate a wide spectrum of scenarios ranging from normal operation to hypothetical, severe accidents. The simulator models major phenomena in the accident such as thermal hydraulics in the reactor coolingsystem (RCS), heat up of fuel rods, core melt, molten core relocation (freezing, slumping etc.), debris cooling in lower plenum, fission product (FP)’s release and transport in RCS, steam explosion, debris/concrete interaction, thermal hydraulics in the containment vessel (CV), FPs transport in the CV. The modeling and combination of these phenomena could supply the ability to simulate the severe accident progress at Light Water Reactor.

Initiated in fiscal year 1993, the project's conceptual and detailed design phases have been completed and coding and verification phases are in progress.

In the several analysis modules, verification studies for some modules are under way, and the steam explosion analysis module and the debris coolability analysis module are examined against typical experimental results to confirm the ability of each model.

The premixing submodule for analysis of steam explosion phenomena under severe accident conditions has been completed, and was shown to simulate the MIXA tests well. Calculation results of debris spreading model in debris cooling process are compared with the experimental results and calculated average location of the spearhead at each time shows good agreement with the experimental observation, though spearhead shapes are different.

This work is sponsored under the contract by the Ministry of International Trade and Industry, Japan.
1. Introduction

The IMPACT Simulation System is being developed to assess the available safety margin of nuclear facilities. Emphasis is on development of software for deterministic analysis of severe accidents.

IMPACT consists of:
- Severe accident code
- Control system

This presentation covers:
- Several of the mechanistic modules in the severe accident code
- Initial verification

2. Description of Analysis Modules

- Thermal Hydraulics in RCS Analysis Module (THA)
- Fuel Rod Heat-Up Analysis Module (FRHA)
- Molten Core Relocation Analysis Module (MCRA)
- Debris Coolability in Lower Plenum Analysis Module (DCA)
- FP Release from Fuel Analysis Module (FPRA)
- FP Behavior Analysis Module (MACRES)
- Thermal Hydraulics in CV Analysis Module (CVPA)
- Debris Spreading Analysis Module (DSA)
- Debris-Concrete Reaction Analysis Module (DCRA)
- Ex-vessel Steam Explosion Analysis Module (VESUVIUS)

3. Code Verification

- Verification of Steam Explosion Analysis Module (VESUVIUS)
- Verification of Debris Coolability Analysis Module (DCA)

4. Conclusions
**Thermal Hydraulics in RCS Analysis Module (THA)**

- Evaluate thermal hydraulics in Reactor Cooling System (RCS), using the existing code (RELAPS) developed in USA.
- Calculate two-phase flow behavior in RCS by node-junction model.

**Fuel Rod Heat-Up Analysis Module (FRHA)**

- Model fuel rod heat-up, hydrogen production due to cladding oxidation, and cladding deformation and failure in core region.
- Model UO2-Zr eutectic formation, and heat structure and control rod.

**Molten Core Relocation Analysis Module (MCRA)**

- Model core melting, molten core relocation, molten core/structure interaction using mechanistic model.
- Based on Multi Phase, multi Component and multi Velocity field (MPCV) method.
- Couple with core heat-up and debris cooling in lower plenum.

**Debris Coolability in Lower Plenum Analysis Module (DCA)**

- Model debris spreading, and cooling in lower plenum.
- Determine vessel failure occurrence.
**FP Release from Fuel Analysis Module (FPRA)**

- Model FP transport within pellet, release from fuel rod after failure, and from molten pool
- Model chemical change of FP and decay heat

**Thermal Hydraulics in CV Analysis Module (CVPA)**

- Model thermal hydraulics in CV under severe accident conditions
- Perform fast-running calculations by node-junction model

**FP Behavior Analysis Module (MACRES)**

- Calculate aerosol behavior and FP transport in RCS and Containment Vessel (CV) using the existing code (MACRES) developed by NUPEC

**Debris Spreading Analysis Module (DSA) / Debris-Concrete Reaction Analysis Module (DCRA)**

- Model falling debris behavior in CV
- Use DSA for short-term and DCRA for subsequent long-term behavior
**Ex-vessel Steam Explosion Analysis Module (VESUVIUS):**

- Model entire steam explosion process in the ex-vessel, from initial premixing phase to final expansion phase.
- Implement molten particle breakup model and jet breakup model to a -FLOW code which was modified to account for mixing phenomena.
- Complete first version of models of each steam explosion phase within fiscal 1997.

**Verification of Steam Explosion Analysis Module (VESUVIUS):**

- Particle breakup calculation.
- Calculation for MIXA-06 Experiment (conducted at AEA Winfrith).
  - Experimental conditions:
    - Melt material: UO2-Mo sphere
    - Melt mass: 2.75 kg
    - Melt temperature: 3600K
    - Particle inlet diameter: 0.003m
    - Melt inlet volume fraction: 0.03
    - Water temperature: Saturation temperature
    - Test tank: 0.57 m square, 1.6m in height.

**Progress Scenario in Severe Accident:**

- Multiple Failure
- Fuel Rods Dryout
- Fall of Core Melts (Debris) into Lower Plenum
- In-vessel
- Ex-vessel
- RPV Failure
- Debris Fall-Down onto Containment Vessel and FP Release
- Containment Vessel Failure

**Phenomena in Lower Plenum:**

- RPV Wall
- Baffle Plate under Core
- In-core Instrument Guide
- Heat Transfer of Debris-bed
- RPV Failure by Debris Heat
- Debris Spreading and Cooling
Mechanistic Model Requirement

1. Detailed Debris Coolability
   - Natural Convection

2. RPV Failure
   - Wall Temperature Distribution

3. Debris Offset Falling Effect
   - Debris Spreading

Objective

- Development of Debris Behavior Analysis Module, DCA* Based on Mechanistic Models

*: Debris Coolability Analysis Module

Analysis Methods of DCA Models

<table>
<thead>
<tr>
<th>Model</th>
<th>Analysis Method</th>
</tr>
</thead>
<tbody>
<tr>
<td>Debris Spreading</td>
<td>- Mass, Momentum and Energy Eqs.</td>
</tr>
<tr>
<td></td>
<td>- Quasi-3-Dimensional Scheme</td>
</tr>
<tr>
<td></td>
<td>- Volume of Fraction (VOF)</td>
</tr>
<tr>
<td>Detailed Coolability</td>
<td>- Navier-Stokes + SMAC</td>
</tr>
<tr>
<td></td>
<td>- 3-Dimensional Scheme</td>
</tr>
<tr>
<td></td>
<td>- Local Melting and Solidification</td>
</tr>
<tr>
<td></td>
<td>(Now under Development)</td>
</tr>
<tr>
<td>Simplified Coolability</td>
<td>- Two-Phase Energy Balance</td>
</tr>
<tr>
<td>RPV Failure</td>
<td>- 3-Dimensional Heat Conduction</td>
</tr>
<tr>
<td></td>
<td>- Creep Rupture and In-Core</td>
</tr>
<tr>
<td></td>
<td>- Instrumentation Penetration Mode</td>
</tr>
</tbody>
</table>

Analysis Options of DCA Module

<table>
<thead>
<tr>
<th>Opt</th>
<th>Analysis Method</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Simplified Coolability Model</td>
</tr>
<tr>
<td></td>
<td>RPV Failure Model</td>
</tr>
<tr>
<td>2</td>
<td>Debris Spreading</td>
</tr>
<tr>
<td></td>
<td>Spreading Stop</td>
</tr>
<tr>
<td></td>
<td>Coolability Model</td>
</tr>
<tr>
<td></td>
<td>RPV Failure Model</td>
</tr>
<tr>
<td>3</td>
<td>Debris Spreading</td>
</tr>
<tr>
<td></td>
<td>Spreading Stop</td>
</tr>
<tr>
<td></td>
<td>Coolability Model</td>
</tr>
<tr>
<td></td>
<td>RPV Failure Model</td>
</tr>
</tbody>
</table>

Quasi-3-Dimensional Analysis in Debris Spreading

- Analysis Method
  1. X-Y Coordinates with Height
  2. Conservation of $\rho H$ ($\rho$ : Density)
  3. Replacement of Pressure Term
  4. Energy Balance and Heat Transfer in Crust

(a) Analysis Coordinate

Debris Falling Point

(b) Longitudinal-Section

- Analysis Objectives
  - Debris Height, H (Unknown)
  - Crust
  - Melt
  - Debris Height, H (Unknown)
Results of Spreading Analysis

Concentric Wall

Flow Spearhead

Falling Point

Flow Exit

Water Flow Rate: 0.17 kg/s

(a) Experiment
(from NUREG/CR-5423)

(b) Calculation

Analysis Coordinates in Simplified
Debris Coolability Model

- Upper Crust (Lumped, Disc)
- Melt (Lumped, Semi-Sphere)
- Lower Crust (Meshed, Thickness)
- RPV Wall (3-Dimensional Shape)

4. Conclusions

- Modeling of severe accident phenomena
  - Conceptual design and detailed design have been completed
  - Coding and verification tests to confirm capabilities of several modules are in progress
- Initial Verification Tests
  - Ex-vessel steam explosion analysis module
    - The comparison of the pre-mixing calculation with the MIXA test results showed good agreement
  - Debris coolability in lower plenum analysis module
    - Calculation results of debris spreading model were well compared with the experiment results
- The RPV failure condition was preliminarily clarified by simplified debris coolability model
5. Session III

Hydrogen Behavior

Chairperson: V. Sidorov (RRCKI)
Co-chairperson: T. Hashimoto (NUPEC)
5.1 LARGE-SCALE EXPERIMENTS AND SCALING OF DDT CONDITIONS IN HYDROGEN-AIR-STEAM MIXTURES - AN OVERVIEW

V. P. Sidorov, and S. B. Dorofeev
Russian Research Centre “Kurchatov Institute”, Moscow, 123182, Russia

ABSTRACT

Results of large scale experiments on deflagration to detonation transition (DDT) in hydrogen-air [1], and hydrogen-air-steam mixtures [2, 3] are presented. The objectives of the tests were (1) study of the conditions for DDT at reactor relevant scale, (2) verification of DDT scaling criteria for steam diluted hydrogen-air mixtures at elevated initial temperatures, and (3) study of turbulent flame propagation and resulted loads in hydrogen-air-steam mixtures at large scale. These tests were aimed to cover the whole range of compositions (lean and stoichiometric hydrogen-air mixtures with variable steam dilution) which are above and below the critical compositions for DDT.

Experiments were carried out in the RUT facility. This facility consists of three parts: first channel (34.4 x 2.5 x 2.2 meters), canyon (10.5 x 2.5 x 2.2 m) and second channel (20.1 x 2.5 x 2.2 m) Total volume of mixture was about 480 m$^3$. Inner volume of the facility was preheated by specially designed heating system in tests with steam. Initial gas temperature inside the facility was 285 K, or 370 - 380 K. Ignition was made by weak electric spark near the closed end of the first channel. The combustion modes observed include shock-less (slow) deflagration, fast turbulent deflagration, and DDT and stable detonation propagation. Onset of detonations was observed in the obstructed channel, or in the largest compartment of the enclosure (canyon).

Critical conditions for DDT were analyzed in terms of the detonation cell size $\lambda$ as a measure of mixture sensitivity to detonation initiation. It was shown that the compositions with $\lambda \approx 1$ m are the critical compositions for DDT in the RUT facility. The critical compositions expressed in terms of $\lambda$ appeared to be very similar in hydrogen-air DDT experiments at $\approx 285$ K, and in hydrogen-air-steam tests at $\approx 375$ K in the same facility. It was shown that experimental results are in good agreement with the $7\lambda$ DDT criterion within the range of initial conditions used in the tests. These data are strong arguments in favor of the detonation cell size scaling of DDT conditions for severe accident analyses.

REFERENCES

2. S. B. Dorofeev, V. P. Sidorov, W. Breitung, J. Vendel, and A. Malliakos. Recent results of joint FZK-IPSN-NRC-RRCKI research program on large scale $\text{H}_2$ DDT experiments in the RUT facility. CSARP meeting, Bethesda, MD, USA May 5 - 8, 1997
3. S. B. Dorofeev, V. P. Sidorov, W. Breitung, and A. S. Kotchourko. Large scale combustion tests in the RUT facility: Experimental study, numerical simulations, and analysis on turbulent deflagrations and DDT. Proc. of 14th SMIRT, p. 815-1, 1997

*) The work was sponsored by FZK, Germany; IPSN, France; and US NRC.
LARGE-SCALE EXPERIMENTS AND SCALING OF DDT CONDITIONS IN HYDROGEN-AIR-STEAM MIXTURES - AN OVERVIEW*

S. Dorofeev, V. Sidorov,
RRC - Kurchatov Institute, Russia

SARJ Workshop
Yokohama, Japan
October 6 - 8, 1997

* joint project RRC Kl (Russia) - FZK (Germany) - IPSN (France) - NRC (USA)

MOTIVATION

- Hydrogen that might be generated during loss of coolant accidents can be mixed with air and ignited deliberately or accidentally in a containment building
- Under certain initial and boundary conditions accelerated flames, fast turbulent deflagrations and transition to detonation are principally possible
- DDT events and detonations are able to produce the most severe loads to confining structures
- Processes of flame acceleration, propagation of turbulent flames, and DDT are essentially scale dependent
- Small scale laboratory experiments must be supplemented by large scale tests
- Numerical and analytical models for flame acceleration, turbulent deflagration propagation and DDT must be validated against data of large scale experiments

OBJECTIVES

1. Investigation of the conditions for DDT in lean hydrogen-air mixtures at reactor relevant scale at normal initial temperature
2. Investigation of the conditions for DDT in hydrogen-air-steam mixtures at reactor relevant scale for widest possible range of hydrogen and steam concentrations at elevated temperatures (about 100°C)
3. Verification of DDT scaling criteria for lean hydrogen-air mixtures at normal initial temperature and steam diluted hydrogen-air mixtures at elevated initial temperatures
4. Investigation of turbulent flame propagation and resulted loads at large scale

RANGE OF COMPOSITIONS

Normal initial temperature:
- Hydrogen: 10 - 14 % vol.
- Steam: less than 1.5 % vol.

Elevated initial temperature:
- Hydrogen: 10 - 32 % vol. in dry mixture
- Steam: 6 - 45 % vol.
## TEST TABLE (NORMAL INITIAL TEMPERATURE)

<table>
<thead>
<tr>
<th>Test</th>
<th>H2, % vol.</th>
<th>BR, %</th>
<th>Explosion regime</th>
<th>Dcj, m/s</th>
<th>Dexp, m/s</th>
<th>Pcj, Bar</th>
</tr>
</thead>
<tbody>
<tr>
<td>#1</td>
<td>12.5</td>
<td>30</td>
<td>DDT (slow)</td>
<td>1405</td>
<td>-</td>
<td>9.1</td>
</tr>
<tr>
<td>#2</td>
<td>12.5</td>
<td>30</td>
<td>DDT (slow)</td>
<td>1334</td>
<td>1240-</td>
<td>8.25</td>
</tr>
<tr>
<td>#3</td>
<td>12.5</td>
<td>30</td>
<td>DDT</td>
<td>1405</td>
<td>1690-</td>
<td>7.53</td>
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<tr>
<td>#4</td>
<td>12.5</td>
<td>30</td>
<td>DDT</td>
<td>1345</td>
<td>1360-</td>
<td>9.9</td>
</tr>
<tr>
<td>#5</td>
<td>12.5</td>
<td>30</td>
<td>DDT</td>
<td>1405</td>
<td>1480-</td>
<td>8.37</td>
</tr>
</tbody>
</table>

Investigation of DDT conditions: tests 11-17 (BR=30%), 21-23 (BR=60%)

Notes:
- H2: Hydrogen
- BR: Burner ratio
- DDT: Delayed Detonation
- Dcj: Critical jet velocity
- Dexp: Critical explosion velocity
- Pcj: Critical pressure

Diagram: Top view
RESULTS (FLAME PROPAGATION IN FIRST CHANNEL)

X-t diagram of flame and detonation propagation in first channel

RESULTS (DDT IN CANYON)

Test 15, 12.5% H₂, dt = 0.57 ms

Flame shapes (dotted lines) and secondary explosion waves (solid lines). Local velocities (m/s) of explosion waves are shown in the plots. (o) - locations of transducers.

SUMMARY OF OBSERVATIONS IN DDT TESTS (NORMAL INITIAL TEMPERATURE)

- It was found that the critical hydrogen concentration for DDT is 12.5% vol. for the given geometrical size of facility.

- The largest volume part of the facility (canyon) was responsible for the DDT at critical concentration of hydrogen 12.5% vol.

- DDT in the first channel was observed for 14% vol. of hydrogen.

- The results are reproducible in the main features (critical conditions, DDT location, characteristic velocities and pressures).
### TEST TABLE (ELEVATED INITIAL TEMPERATURE)

<table>
<thead>
<tr>
<th>Test #</th>
<th>Average $H_2$ (dry)</th>
<th>Average steam content</th>
<th>Minimum steam content (canyon)</th>
<th>Explosion regime (comments)</th>
</tr>
</thead>
<tbody>
<tr>
<td>tm1</td>
<td>15.0</td>
<td>15.0</td>
<td>14.0</td>
<td>Deflagration (fast)</td>
</tr>
<tr>
<td>tm2</td>
<td>14.7</td>
<td>14.7</td>
<td>12.7</td>
<td>Deflagration (fast)</td>
</tr>
<tr>
<td>tm3</td>
<td>17.5</td>
<td>25.7</td>
<td>19.6</td>
<td>Deflagration (fast)</td>
</tr>
<tr>
<td>tm4</td>
<td>30.1</td>
<td>44.1</td>
<td>41.2</td>
<td>Deflagration (fast)</td>
</tr>
<tr>
<td>tm5</td>
<td>25.6</td>
<td>45.2</td>
<td>43.3</td>
<td>Deflagration (fast)</td>
</tr>
<tr>
<td>tm6</td>
<td>28.8</td>
<td>40.6</td>
<td>38.1</td>
<td>Deflagration (fast)</td>
</tr>
<tr>
<td>tm7</td>
<td>10.1</td>
<td>6.6</td>
<td>4.2</td>
<td>Deflagration (slow)</td>
</tr>
</tbody>
</table>

stm1 - stm7 - experimental campaign 95  
sth1 - sth9 - experimental campaign 96

---

### INITIAL CONDITIONS

- Channel floor
- Cooling (hot)
- Cooling (cold)

---

### RESULTS (DDT IN THE CANYON)

**General X-t diagram of flame and detonation propagation**

- Solid line - direct wave, $dt=0.5$ ms
- Dash line - reflected wave, $dt=0.5$ ms

**Detonation propagation in the canyon**

- Stm3, solid line - direct wave, $dt=0.5$ ms
- Dash line - reflected wave, $dt=0.5$ ms
RESULTS

DDT IN THE OBLSTRUCTED CHANNEL

General X-t diagram of flame and detonation propagation

ANALYSIS OF DDT CONDITIONS

- While a predictive computer tool describing all combustion regimes including DDT are a distant future, DDT criteria are capable to give immediate outcome for accident analysis.
- One of these criteria (RRC "Kurchatov Institute") states that characteristic size of compartment filled with combustible mixture should be >7 X for DDT.
- Application of this criterion for severe accident analysis requires:
  > detonation cell size data (mixtures and conditions typical for severe accidents)
  > clear definition of "characteristic size" for different geometrical configurations
  > experimental validation for mixtures and scales typical for severe accidents
- Analytical functions λ(H2, H2O, T, p) were constructed to make interpolation of cell size data. Mean deviation (calculated/measured) is about a factor of 1.5.
- Analysis of geometry has shown 4 typical cases. For each one characteristic geometrical size D was defined.
- DDT conditions from RUT experiments were analyzed in the frame of D and λ.
GEOMETRY
- Compartment sizes L, H, W, connection sizes d, and obstacle configuration (BR, S) are important.
- Typical cases:
  1. Room geometry: L ≈ H ≈ W, BR < 0.5, d < 0.5H
  2. Channel geometry: L ≫ H ≈ W, BR < 0.5, d < 0.5H
  3. Flat room geometry: L = H ≫ W, BR < 0.5, d < 0.5H
  4. Flat channel geometry: L ≫ H ≫ W, BR < 0.5, d > 0.5H

CHARACTERISTIC SIZES
- Maximum possible size of preconditioned mixture, which can be formed during mixture combustion
- Limited conservatism

ROOM GEOMETRY
Room L = H = W, BR < 0.5, d < 0.5H.
D = (L+H+W)/3
Flat room L = H ≫ W, BR < 0.5, d < 0.5H.
D = (L+H)/2

CHANNEL GEOMETRY
D = 2.5 S, for S close to H;
D = 2.5 H, for S ≫ H;
D = 2.5 d, for S ≪ H.

COMPARISON WITH EXPERIMENTAL RESULTS.
DDT CONDITIONS (GENERAL)

COMPARISON WITH EXPERIMENTAL RESULTS.
DDT CONDITIONS (CANYON)

Characteristic geometrical sizes of the enclosure are 6.25 m (obstructed channel), 8.25 m (canyon).
λ = 1 m - critical cell size for the RUT scale
λ = 1.2 m (8.25/7) - critical cell size for the canyon
COMPARISON WITH EXPERIMENTAL RESULTS.
DDT CONDITIONS (CHANNEL)

SUMMARY OF CRITICAL CONDITIONS

\[ \lambda = 0.9 \text{ m (6.25/7)} \] - critical cell size for the obstructed channel

SUMMARY AND CONCLUSIONS

- Results of \( \text{H}_2\)-air and \( \text{H}_2\)-air-steam DDT experiments form a detailed data base on propagation of turbulent deflagration, DDT limits, and resulted loads at normal and elevated initial temperatures and large scale.
- Critical conditions for DDT were analyzed in terms of \( \lambda \). Compositions with \( \lambda \approx 1 \text{ m} \) were shown to be critical for DDT in the RUT facility.
- Critical compositions of \( \text{H}_2\)-air-steam mixtures at 375°C (in terms of \( \lambda \)) appeared to be very similar to that found in hydrogen-air DDT experiments at \( \approx 285 \text{ K} \) in the same facility.
- Critical conditions for DDT found experimentally show good agreement with 73 DDT criterion including compositions, scales and initial conditions typical for severe accidents.
- This criterion can be used as a conservative estimate for analysis of hydrogen combustion behavior in severe accidents.
Laboratory of Induced Chemical Reactions, RRC "Kurchatov Institute"

5.2 NUMERICAL INVESTIGATION OF MISSILES ACCELERATION BY HYDROGEN EXPLOSION

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Institut für Neutronenphysik und Reaktortechnik Forschungszentrum Karlsruhe, Germany

ABSTRACT

In case of severe accidents in nuclear power plants large amount of hydrogen could be released into containment. Combustion, deflagration, and detonation of the hydrogen-air mixture are highly possible there. This leads to generation of intense gas flows. Equipment inside the containment and some parts of the inner structure can be accelerated by these flows and form hazardous missiles. Interaction of missile with the gas flow was studied numerically by means of three-dimensional gasdynamic computer code. Aerodynamic tube was modelled in which missiles of different shape were inserted in supersonic gas flow. Mach number and specific heat ratio of the flow as well as missile's orientation with respect to the flow were varied. Dependencies of drag coefficient of missiles on Mach number and specific heat ratio under conditions, that are typical to an accident in nuclear power plant, were evaluated. They are presented together with distributions of parameters of the gas flow around missile. It was shown that dependence of drag coefficient on specific heat ratio of the gas is not steep and may be omitted in evaluation of missile hazard. The method, which is used in evaluation of drag coefficients, is based on direct numerical simulation of hydrodynamics of the gaseous flow. It can be applied to obtain the exact values of drag coefficients under various conditions.

The data on drag coefficient values give way to application of the computer code, in which drag coefficient model of missile-flow interaction was implemented. This code is capable to model gas flows, subsequent pressure loads on the containment structure, and missiles' motion in case of detonation or explosion of hydrogen-steam-air mixture inside the containment.

Drag coefficient model used in the code assumes, that missiles are much smaller, than characteristic size of the flow. Hence, it limits applicability of this code to the case of large scale detonations. However, hazardous missiles can be expected in cases of smaller scale, i.e. local detonations. Direct simulations of missile acceleration were performed under the following conditions. Compartment of 8 by 6 by 2.2 m size was filled with stoichiometric hydrogen-air mixture. Detonation was ignited at the centre of the shorter wall. Part of the opposite wall (2.2 by 1.2 m) could move freely under the pressure stress from detonation wave and form the missile. In five tests wall thickness, missile thickness and its mass were varied. Resulting missiles' velocities, distributions of gas flow parameters, and pressure loads are presented. Results of these numerical tests give the data on the missile velocities and momenta in some typical cases. Scaling relationships are proposed. The results of numerical tests and scaling relationships provide the estimation of velocities and momenta of missiles that are possible under accidental conditions.

* Work was sponsored by Project of Nuclear Safety Research, FZK, Germany
Background

- Missile hazard estimation via computer modelling of possible velocities and momenta of the missiles dragged by the gas flow in case of an accident requires data on drag coefficients.
- Application of computer code, which uses drag coefficient model, is limited to the case of large scale, i.e. global detonations. Meanwhile, generation of hazardous missiles is possible at destruction of the inner containment structure due to local explosion.

Objectives

- Numerical evaluation of dependence of drag coefficient on Mach number and on specific heat ratio under conditions, that are typical to an accident in nuclear power plant.
- Direct numerical simulation of missile acceleration by local detonation wave.
Laboratory of Induced Chemical Reactions, RRC "Kurchatov Institute"

\[ M = 3.0 \quad \gamma = 1.40 \quad P_0 = 1.0 \text{ MPa} \]

Pressure: 11.89 ± 0.30 MPa

Pressure: 12.03 ± 0.16 MPa

Sound speed: 1448 ± 826 \[ c_{s0} = 837 \text{ m/s} \]

Flow speed: 2527 ± 0 \[ v_0 = 2510 \text{ m/s} \]
Laboratory of Induced Chemical Reactions, RRC "Kurchatov Institute"

$M = 3.0$  \ $\gamma = 1.40$

Transvers velocity

$+1897$  \ $+1079 \text{ m/s}$

Longitudinal velocity

$+2024$  \ $-2510 \text{ m/s}$

$\nu_{20} = -2510 \text{ m/s}$

Pressure

$46.66 \pm 0.32 \text{ MPa}$

$P_0 = 1.0 \text{ MPa}$

Sound speed

$2507 \pm 837 \text{ m/s}$

$c_{s0} = 837 \text{ m/s}$
Flow speed $v_0 = 5020$ m/s.

Sound speed

$M=2.0 \quad \gamma = 1.40$

$2395.8 \div 836.7$ m/s

$M=6.0 \quad \gamma = 1.40$

$1122.6 \div 776.9$ m/s

Pressure

$5.56 \div 0.27$ MPa

$43.55 \div 0.29$ MPa
**Laboratory of Induced Chemical Reactions, RRC "Kurchatov Institute"**

\[ M = 3.0, \quad \gamma = 1.40, \quad P_0 = 1.0 \text{ MPa} \]

**Square bar**

11.79 ± 0.41 MPa

**Cylinder**

11.89 ± 0.12 MPa

**Sphere**

11.04 ± 0.13 MPa

**Dependence of drag coefficient on Mach number**

\[ \gamma = 1.4 \]
Dependence of drag coefficient on Mach number and specific ratio of the gas is not steep and may be omitted in evaluation of missile hazard under typical accidental conditions. Corresponding error in this case is less than 2%.
Laboratory of Induced Chemical Reactions, RRC "Kurchatov Institute"

$P_0 = 1.0$ atm, $T_0 = 275$ K

Tests
- No 2 (200 kg)
- No 5 (500 kg)

Test No 1 ($M = 80$ kg)

Test No 3 ($M = 200$ kg)

Test No 5 Transducer #4

Pressure ($5.28 \pm 0.10$ MPa) at $t = 5.05$ ms

Test No 1 ($M = 80$ kg)

Test No 3 ($M = 200$ kg)

Test No 4 ($M = 200$ kg)
Laboratory of Induced Chemical Reactions, RRC "Kurchatov Institute"

Test No 3.

Test No 4.

Test No 5. Transducer #1 records of pressure and impulse.
Test № 5. Transducer #2 records of pressure and impulse.

Test № 5. Transducer #3 records of pressure and impulse.

Test № 5. Transducer #4 and transducer #5 records of pressure and impulse.

Test № 5. Transducer #6 and transducer #7 records of pressure and impulse.

Test № 5. Transducer #8 and transducer #9 records of pressure and impulse.

Test № 5. Transducer #10 and transducer #11 records of pressure and impulse.
Laboratory of Induced Chemical Reactions, RRC "Kurchatov Institute"

Test № 1 14.70 ms

Pressure
2.035
0.051 MPa

Transvers velocity
+1065
-1388 m/s.

Longitudinal velocity
+1371
-866 m/s.

Flow speed
1408 m/s

Density
6.39
0.09 kg/m³

Temperature
3100
281 K

Laboratory of Induced Chemical Reactions, RRC "Kurchatov Institute"

Test No 4.

9.10 ms
Pressure
2.31
0.06 MPa.

9.10 ms
Sound speed
1181
339 m/s.

25.09 ms
Sound speed
1145
375 m/s.

24.96 ms
Mach number
2.10 \(\pm\) 0.

37.70 ms
Sound speed
1073
357 m/s
Laboratory of Induced Chemical Reactions, RRC "Kurchatov Institute"

**Numerical results**

<table>
<thead>
<tr>
<th>Test</th>
<th>№ 1</th>
<th>№ 2</th>
<th>№ 3</th>
<th>№ 4</th>
<th>№ 5</th>
</tr>
</thead>
<tbody>
<tr>
<td>Missile mass, kg</td>
<td>80</td>
<td>200</td>
<td>200</td>
<td>200</td>
<td>500</td>
</tr>
<tr>
<td>Wall thickness, mm</td>
<td>300</td>
<td>600</td>
<td>600</td>
<td>600</td>
<td>600</td>
</tr>
<tr>
<td>Missile thickness, mm</td>
<td>300</td>
<td>600</td>
<td>100</td>
<td>100</td>
<td>600</td>
</tr>
<tr>
<td>Distance, mm</td>
<td>3000</td>
<td>2700</td>
<td>3200</td>
<td>2700</td>
<td>2700</td>
</tr>
<tr>
<td>Drift time, ms</td>
<td>11.19</td>
<td>18.70</td>
<td>21.30</td>
<td>19.51</td>
<td>32.40</td>
</tr>
<tr>
<td>Average speed, m/s</td>
<td>268.09</td>
<td>144.36</td>
<td>150.21</td>
<td>138.38</td>
<td>83.33</td>
</tr>
<tr>
<td>Final speed, m/s</td>
<td>313.14</td>
<td>204.95</td>
<td>194.86</td>
<td>171.94</td>
<td>111.87</td>
</tr>
<tr>
<td>Maximal speed, m/s</td>
<td>313.14</td>
<td>205.50</td>
<td>203.25</td>
<td>179.94</td>
<td>115.61</td>
</tr>
</tbody>
</table>

**Scaling relationships**

1) \( H_m \rightarrow \alpha \cdot H_m \Rightarrow m \rightarrow \alpha \cdot m \), \( U_m, a_m, t, P, \rho, V, \text{ etc.} = \text{const} \).
2) \( X \rightarrow \beta \cdot X' \Rightarrow t \rightarrow \beta \cdot t' \).

**Scaling of Test № 3**

\( \alpha = 0.5, \ \beta = 2.0 \)
- First compartment 16 m x 12 m
- Second compartment 5.4 m x 12 m
- Wall thickness 120 cm
- Missile sizes 2.2 m x 2.4 m x 0.2 m
- Missile mass 800 kg
- Maximal speed 203.25 m/s

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**Summary**

- Drag coefficients of missiles under accidental conditions are evaluated. This makes possible computer modelling of velocities and momenta of missiles in case of global detonation inside containment.

- Motion of missiles generated at destruction of the inner containment structure is investigated. The results of numerical tests and scaling relationships provide the estimation of velocities and momenta of missiles that are possible in case of local detonations inside containment.
5.3 Analyses of NUPEC’s Large Scale Hydrogen Mixing Tests in a Reactor Containment Vessel

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F-91191 Gif-sur-Yvette cedex
France

ABSTRACT

NUPEC has carried out « The Hydrogen Mixing and Distribution Tests » as a part of a project entitled «Proving Test on the Reliability for Reactor Containment Vessel » on the trust contract with MITI (Ministry of International Trade and Industry). This project aims to investigate hydrogen distribution behavior in a model containment vessel and provide a set of experimental data for validation of severe accident analysis codes. The Test was completed in March 1994.

IPSN (Institut de Protection et de Surete Nucleaire) sponsors the development of the TONUS hydrogen risk analysis code incorporating both lumped-parameter and multi-dimensional formulations. For the purpose of qualification and validation of the TONUS code using NUPEC hydrogen tests, IPSN has detached an engineer to NUPEC/Systems Safety Department, Tokyo. In a first step, the german RALOC code, developed by GRS, is used for test analyses, and to prepare the qualification and validation of TONUS code.

The Containment Vessel of Hydrogen Mixing and Distribution Tests has a free volume of 1,300 m3 in which there are 25 compartments with steel walls, which simulate the inside of a four-loop type PWR containment. Helium gas was used instead of hydrogen for safety considerations. Helium gas concentration and temperature were measured in the center of each compartment as a function of time during the test period.

The three tests that were assessed included M2-2, M4-3, and M8-1 tests. Test M2-2 had a low-point injection of helium, no steam, and the test was performed at room temperature (isothermal test). Test M4-3 had a low-point injection of helium and steam, and the containment was initially at ambient temperature. Test M8-1 is similar to Test M4-3, except helium and steam were released in the pressurizer compartment, in the middle of the containment.

Calculations predicted the helium concentrations well in most of the compartments. The most difficult compartments to predict were the in-core chase, which is an irregularly shaped room at the bottom of the facility; the pressurizer rooms, which include the dead-end compartment; and the source rooms. Calculations also predicted well the gas pressure, and the gas temperature in most rooms.

This work is sponsored under the contract by the Ministry of International Trade, and Industry, Japan.
INTRODUCTION

Since 1987, NUPEC (Nuclear Power Engineering Corporation) has been working on the project entitled "Proving Test on the Reliability for Reactor Containment Vessel". The aim of this project is to evaluate the integrity of a containment under various accident scenarios. The Test, performed at Tadotsu Engineering Laboratory, was completed in March 1994. The Test was carried out using a 1/2 scale model of PWR large dry containment, under simplified test conditions.

The purpose of NUPEC’s Hydrogen Mixing and Distribution Tests were to determine the mixing behaviour of helium throughout the containment, to determine the thermal hydraulics, and to produce data to assess analysis codes. As a safety consideration, helium gas was used for simulating hydrogen in order to avoid unexpected explosions during the tests.

BEL (Bureau d' Études et de Laboratoire) sponsors the development of the RALOC code, and the German RALOC code, developed by GRS, is used for test analysis and to prepare the qualification data to assess analysis codes. As a safety consideration, helium gas was used for simulating simultaneous physical effects in a complicated multi-compartment geometries. In particular, the ability of this code to predict helium mixing throughout the containment, gas temperatures, gas velocities, and gas pressure are assessed.

PRESENTATION OF THE NUPEC TESTS [2],[3]

2.1 Description of the NUPEC test facility

Figure 1 shows experimental apparatus flow chart. The facility consists of a model containment vessel, gas and steam supply systems, and spray water supply system. Steam and helium gas are introduced into the mixing chamber and mixed uniformly, and then discharged into a compartment of the model containment.

Table 1 gives details about the NUPEC facility which is a domed cylinder, approximately 10.8 m in diameter, 17.4 m high, and 1.310 m3 in volume. The model containment has three main floors. The gap between the first floor and the surrounding containment wall is sealed. There is a 50 mm cylindrical gap between the second floor and surrounding containment wall. The third floor has also a similar cylindrical gap. The dome compartment is located above the third floor.

The facility contains 20 compartments which are interconnected. The dome compartment constitutes approximately 71% of the total containment volume. The containment is constructed entirely of carbon steel. The containment shell and floors are 12 mm thick, except for the first floor, which is 16 mm thick. The compartment walls are 4.5 mm thick. The outside of the containment is covered with a layer of insulation, which is covered by a thin metal sheet to protect the insulation from weather damage. The insulators around the cylinder and hemisphere are 125 mm and 150 mm thick, respectively. A water storage tank is located below the first floor of the containment so that condensate and spray water can drain. The tank is separated from the rest of the containment by 100 mm of insulation. Water is pumped from the tank to 21 spray nozzles in the dome.

2.2 Experimental conditions

Nine series of tests were performed to determine the effect of mixing by natural convection as a result of density differences (differences in molecular weights and temperature), by forced convection due to steam release and water sprays, and by different release locations. Tests were performed injecting helium and steam individually as well as together, injecting helium with water sprays, and injecting helium and steam with water sprays. The gases were injected in either a steam generator room or pressurizer relief tank at the bottom of the containment or in the pressurizer compartment in the middle of the containment.

Selected tests from series M-2, M-4, and M-6 were used in the current assessment.

- Series M-2: only helium was injected into a room in the bottom of the containment.
- Series M-4: helium and steam were injected into a room in the bottom of the containment.
- Series M-6: repetition of selected tests in Series M-4 and M-7 (in which helium and steam were injected into a room in the bottom of the containment) at the same time the water sprays were operating except the gases were released in the pressurizer compartment in the middle of the containment.

The helium concentrations were sampled from the center of 24 rooms and at 9 locations in the dome. Samples were drawn through tubes, cooled, dried by passing through a dessicant, and analyzed by thermal conductivity gas chromatography. The gas temperatures (x°C) and wall temperatures (x146) were measured with Co-Al thermocouples. The gas pressure was measured by a pressure gauge in the top of the dome. The gas velocities through some junctions (x10) were measured using wind gauges with hot wires to evaluate gas circulation transient through the compartment.

Table 1 shows the major characteristics of the selected M2-2, M4-3, and M8-2 tests:

<table>
<thead>
<tr>
<th>PARAMETER</th>
<th>TEST</th>
<th>M2-2</th>
<th>M4-3</th>
<th>M8-1</th>
</tr>
</thead>
<tbody>
<tr>
<td>Initial system pressure (kPa)</td>
<td></td>
<td>101</td>
<td>101</td>
<td>101</td>
</tr>
<tr>
<td>Initial gas/structure temperature (°C)</td>
<td></td>
<td>25</td>
<td>30</td>
<td>10</td>
</tr>
<tr>
<td>Environment temperature (°C)</td>
<td></td>
<td>25</td>
<td>30</td>
<td>6</td>
</tr>
<tr>
<td>SUPPLY GAS</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Helium temperature (°C)</td>
<td></td>
<td>15</td>
<td>15</td>
<td>20</td>
</tr>
<tr>
<td>Helium mass flowrate (kg/h)</td>
<td></td>
<td>0.027</td>
<td>0.027</td>
<td>0.03</td>
</tr>
<tr>
<td>Steam temperature (°C)</td>
<td></td>
<td>160</td>
<td>140</td>
<td></td>
</tr>
<tr>
<td>Steam mass flowrate (kg/h)</td>
<td></td>
<td>0.33</td>
<td>0.33</td>
<td></td>
</tr>
<tr>
<td>Mixing gas temperature (°C)</td>
<td></td>
<td>115</td>
<td>115</td>
<td></td>
</tr>
<tr>
<td>GAS DISCHARGE</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Location</td>
<td></td>
<td>Cell 8</td>
<td>Cell 8</td>
<td>Cell 22</td>
</tr>
<tr>
<td>Duration (min)</td>
<td></td>
<td>30</td>
<td>30</td>
<td>30</td>
</tr>
</tbody>
</table>

TABLE I. NUPEC Tests, Experimental Conditions

Concerning the modelling of hydrogen distribution in a containment vessel, two approaches can be followed:
- the lumped-parameter approach based on mass and energy balance equations for completion volumes;
- the multi-dimensional approach based on a spatial discretization of the flow equations.

The first approach has the advantage that a complete geometry can be represented by a series of compartments and junctions, and that the simulation can cover several hours of physical time at a relatively low CPU cost. But, due to the limitations of the approach, multi-dimensional effects such as local heat transfer and flow gradients cannot be modelled.

The second approach allows a fully multi-dimensional (also called field) representation of the flow, including gradients of concentration due to possible stratification of accumulation of hydrogen. This information is absolutely essential before attempting to simulate combustion phenomena. But, the resolution of the flow equations on meshed volumes requires important CPU resources, as well as memory resources when explicit time-staging is used, making the multi-dimensional simulation of the complete containment geometry unfeasible.

Consequently, in the TONUS code, a coupled lumped-parameter/multi-dimensional approach has been followed. Steam and hydrogen release is simulated in certain designated volumes such as the dome by a multi-dimensional approach whilst the other volumes are dealt with using the lumped-parameter approach. At present, which compartment is meshed or not is a matter of choice, although a certain level of automatism will be implemented in a future version of the code.

The TONUS code is developing within the frame of the CASTEM-2000 code of CEA, which is a general computational tool for structural mechanics and fluid dynamics applications. A single numerical scheme to treat both the distribution and combustion aspects of a containment hydrogen release scenario is not efficient. So, the development of the TONUS code follows a modular approach which allows different physical models and numerical methods to be used when appropriate. These include lumped-parameter or multi-dimensional (field) formulations, explicit, implicit or partially implicit time discretizations, as well as finite elements or unstructured finite volume methods.

The global validation of the TONUS code will be made by using experimental results from tests performed in the following containment vessels: HDR (11,300 m^3), BATTELLE (640 m^3), and NUPEC (1,310 m^3). As far as the NUPEC experimental tests are concerned, in a first step the RALOC code is used for test analyses (present work), and to prepare the qualification and validation of the TONUS code, indeed.

RALOC input data will be used for setting input data for TONUS-ID calculations,
- RALOC output results will be compared to TONUS-ID output results,
- RALOC output results will serve as input data for 3-D TONUS calculations in the dome of the containment vessel.

Concerning the analytical qualification of TONUS code, IPSN has launched an experimental program named TSOGAN (TONUS Qualification Analytique) devoted to the qualification of models related to the steam condensation on walls in presence of non-condensable gas and of modes dealing with dynamic and thermodynamic effects of the spray [5]. The dimensions of TSOGAN containment vessel are the following: height equal to 5 m and diameter equal to 1,5 m.

Furthermore, CEA/DRN has also started an experimental program called MISTRA (Mitigation & STRategic, which is medium scale (100 m^3), especially designed to establish a link between separate effect and global experiments, introduce step by step physical phenomena and their coupling [6].

4. PRESENTATION OF THE RALOC CODE [7],[8]

RALOC is a lumped parameter code. The lumped-parameter approach is based on mass and energy balance equations for compartment volumes; examples of similar codes are CONTAIN[9], JERICHO[10], and MAX[11] in which the complete geometry can be represented by a series of compartments (zones) and junctions. Note that, in such codes, multi-dimensional effects such as local heat transfer and flow gradients cannot be modelled.

The computer code RALOC is able to evaluate each compartment, in particular, gas composition (steam and different noncondensable gas), gas/wall temperatures, gas pressure, hydrogen combustion and catalytic recombination.

4.1 Zone models
- For the determination of the conditions in the nodes, water is considered as vapour and/or liquid (water) according to the conditions in the node.
- The state of the zones can be calculated under equilibrium conditions (in which gas and water temperatures are similar) or non-equilibrium conditions (in which gas and water are their own equilibrium temperature).

The gaseous components are assumed to be homogeneously mixed. Actually, seven different gases (air, oxygen, nitrogen, hydrogen, carbon monoxide, carbon dioxide, and helium) are defined including their material properties.

4.2 Junction models
- Junctions are divided into two main types: atmospheric junctions and drain junctions (drainage of water at the bottom of the containment vessel). The mass transfer between nodes is described separately for gas and liquid flow by different momentum equations (unsteady, incompressible) taking into account height differences of the node centres. Furthermore, mass transfer by diffusion is considered.

4.3 Structure models
- The one-dimensional finite element representation of the conduction heat transfer equation (Fourier's equation) within the structures is used to account for the transient response of the containment vessel and internal structures.

4.4 Heat and mass transfer models between atmosphere and structures
- For the simulation of heat and mass transfer between the zone atmosphere and the structures, next transfer is described for the different physical phenomena of free and forced convection, radiation (wall to gas, gas to wall, wall-gas-wall), and condensation depending on the thermal status of the zone and structures.

4.5 Combustion
- In order to model the combustion of gases (hydrogen) which propagates, in general, very fast with large local pressure and temperature gradients, an one-dimensional flame front is assumed with the turbulence combustion rate and zone model for the combustion front.

4.6 Engineered systems
- For a realistic description of accident sequences, the simulation of engineering systems is possible.
5. INPUT DATA USED FOR RALOC

5.1 Meshing

5.1.1 Zones (x33)

The following nodalization scheme was used to analyze the NUPEC tests. This scheme is based on a 33-node representation of the facility, each room was represented by a computational node except for the dome and source rooms, which were subdivided. A schematic of the 33-cell representation of the facility is shown in Figure 3. The cell volumes and flow paths were represented by the boxes and the interconnected lines, respectively. The dome was divided into four cells. Furthermore, the source room was also subdivided into two cells (Cells 8 and 30) in order to reduce the impact of non-homogeneous helium concentration distribution in injected compartment (high helium concentration in the center of the source room, according to 3-D calculation results). Concerning the M2-1 test, the source room (Cell 22) was not subdivided.

5.1.2 Junctions (x150)

From the previous nodalization scheme, the following junctions were established: 73 atmospheric junctions, 47 drain_bal junctions (drainage of water at the bottom of the cell), and 18 drain_bal junctions (drainage of water for sump balance).

5.1.3 Structures (x206)

Among the 206 structures of the containment vessel, 11 were insulated structures in contact with the environment.

5.2 Equilibrium/non-equilibrium zone models

The non-equilibrium zone model was used for all the zones.

5.3 Initial and boundary conditions

Initial and boundary conditions were deduced from experimental conditions of NUPEC tests (Table I).

6. RALOC ASSESSMENT OF THE NUPEC TESTS

The results of the RALOC assessment of the M2-2, M4-3, and M8-1 tests are presented in this section.

6.1 M2-2 test (helium). Natural convection effect

a) Helium concentrations

The results for the predictions of the helium concentrations and their comparison with the data, for the dome, which constitutes 70% of the containment volume, and the source room, are shown in Figures 4 and 7. The final predicted concentration in the dome (at the end of injection period) was nearly equal to the measured value. Concerning the source cell, 3-D calculation results show that helium concentration is near-homogeneous [12], which can explain the discrepancies between experimental values (measurement at the center of the source room) and calculation results which provide average value inside the cells. Furthermore, the calculation poorly predicted the helium concentration in three of the rooms in the center of the containment (Cells 1, 16, and 22).

b) Gas temperatures

Because the test was nearly isothermal, differences between the measured and predicted gas temperatures are emphasized for this test. This is especially true on a relative basis, which has less meaning since the changes in the temperatures were small.

c) Gas pressure

The predicted pressure was in nearly perfect agreement with the measured values. Because the test was nearly isothermal, the pressure increase was due to the addition of the noncondensible gas. Because of the good agreement for this test, deviations between the predicted and measured pressures in other tests are most likely due to the inability to model the heat transfer processes correctly.

d) Gas velocities through junctions

The wind gauge sensors were not reliable in a wet environment, such as in experiments with steam and/or spray water injections. Consequently, only for M2-2 test performed under dry atmosphere, comparison between experiment and calculation related to gas velocities through junctions is reported.

Figure 10 shows the results of gas velocity measurements through some junctions (x10) conducted in the containment vessel. Table II below gives, at the end of injection period, the values of experimental and calculated values of the gas velocity at the center of junctions equipped with wind gauges.

<table>
<thead>
<tr>
<th>JUNCTION</th>
<th>FROM ZONE</th>
<th>TO ZONE</th>
<th>EXPERIMENT (MAXIMUM VALUE)</th>
<th>RALOC (AVERAGE VALUE)</th>
</tr>
</thead>
<tbody>
<tr>
<td>10</td>
<td>2—11</td>
<td>0.20</td>
<td>0.30</td>
<td></td>
</tr>
<tr>
<td>18</td>
<td>6—13</td>
<td>0.50</td>
<td>0.30</td>
<td></td>
</tr>
<tr>
<td>22</td>
<td>7—30</td>
<td>1.76</td>
<td>1.00</td>
<td></td>
</tr>
<tr>
<td>24</td>
<td>4—30</td>
<td>1.96</td>
<td>1.25</td>
<td></td>
</tr>
<tr>
<td>32</td>
<td>6—11</td>
<td>1.45</td>
<td>0.50</td>
<td></td>
</tr>
<tr>
<td>38</td>
<td>13—29</td>
<td>0.50</td>
<td>0.40</td>
<td></td>
</tr>
<tr>
<td>50</td>
<td>20—26</td>
<td>0.20</td>
<td>0.00</td>
<td></td>
</tr>
<tr>
<td>51</td>
<td>21—26</td>
<td>2.25</td>
<td>0.75</td>
<td></td>
</tr>
<tr>
<td>52</td>
<td>20—27</td>
<td>0.30</td>
<td>0.10</td>
<td></td>
</tr>
<tr>
<td>53</td>
<td>24—27</td>
<td>0.20</td>
<td>0.10</td>
<td></td>
</tr>
</tbody>
</table>

TABLE II M2-2 Test. Gas velocities through junctions expressed in m/s.

RALOC-Experiment Comparison

Figure 10 and Table II show that the agreement between experiment and calculation is good for junctions V10, V16, V38, V50, and V53 at low velocities (less than about 0.5 m/s), but at high velocities (more than about 1.5 m/s) the calculation underpredicts the experimental value for junctions V22, V23, V32, and V51. This disagreement can be explained partially by the fact that, on one hand, RALOC provides the average value of gas velocity through the junctions, on the other hand, the calculated gauges located at the center of the junction measure the maximum value of the gas velocity. Instead, gas velocities are distributed according to a parabolic profile, as shown by 3-D calculation results [12]. Furthermore, calibration of wind gauges has not been performed under NUPEC test conditions. This may also explain the experiment-calculation disagreement.

Figure 10 shows that the gas-velocity measurement through some junctions increases with time and it apparently is caused by the pressure build-up inside the containment vessel that creates differences among the pressure in the compartments. However, the velocity increase is not shown by the calculation. This point should be clarified later.

N.B.: Concerning gas velocity through junctions, a comparison of RALOC (0-D calculation) and TRIO-EF (3-D calculation) results is presented in reference [12].

6.2 M3-1 test (helium and steam). Steam release effect

a) Helium concentrations

Typically, the difference between final predicted and measured concentrations was less than 15% and the discrepancies were larger in the source room (Cell 30) and the steam generator rooms above it during the injection phase. The results for the prediction of the helium concentrations and their comparison with the data are shown in Figures 5 and 6 for the dome (Cell 29) and for the steam generator foundation room (Cell 30), which is the source compartment. In general, the trends of the predicted results agree with the data. The final predicted concentration in the dome was approximately 1% lower than the measured value. Concerning the In-Cow Chute room (Cell 1) and the pressurizer room (Cell 22), calculation overestimates experimental values, at the end of injection period, by a factor of 4 and 6, respectively.

b) Gas temperatures

In general, the calculation predicted the correct trends for the temperature histories as well as the vertical temperature gradients. The difference between predicted and measured final gas temperatures was typically less than 10 oC.
Laboratory by MHI (Mitsubishi Heavy Industries) in a steel containment vessel of 270 m³.

8.3 MB-1 test (item M4-3, except source location). Source location effect.

a) Helium concentrations

In general, the trends of the predicted results agreed with the data. The results for the predictions of the helium concentrations and their comparison with the data are shown in Figures 6 and 9 for the dome (Cell 20) and for the source compartment (Cell 22). The helium concentration in the dome, at the end of the injection phase, was approximately 6% higher than the measured value. For the source room (Cell 22), investigations are underway in order to understand the existence of the calculated concentration peaks at the beginning of the test (concentration peaks not observed during the test). Because the helium was quite stratified, concentrations below the dome were low. Predicted helium concentrations were up to a factor of ten higher in this region than measured values. However, because the absolute values of helium concentrations were so low, the magnitude of such discrepancy can be put into better perspective when actual deviations in helium concentrations are reported. There were also significant deviations in the In-Core Chase room (Cell 1) and the dead-end room (Cell 16).

b) Gas temperatures

Except for the pressurizer room (Cell 10), the calculation predicted the temperature histories fairly well. The predicted gas temperature in the dome deviated less than 10 oC from the measured value.

c) Gas pressure

The pressure was underpredicted although the general trends were correct. On an absolute basis, the final pressure was approximately 4 kPa lower than the data noted, during the test, the pressure increase was equal to 52 kPa.

7. CONCLUSION

Within the scope of the preparation of the qualification of TONUS hydrogen risk analysis code which the development is sponsored by IPSN, in a first step, the german RALOC developed by GHS is used for analyses of NUPEC hydrogen distribution tests. Calculations typically predicted the helium concentrations relatively well in most of the compartments in the facility, in all of the tests, the helium concentration in the dome (at the end of injection period) was predicted to be within 8% of the experimental value. In some special cases, such as dead-end rooms, rooms with complicated geometries, and source rooms, larger discrepancies were sometimes found. When predicting stratified conditions, calculations predicted helium concentrations relatively well, except in the lowest compartments although the helium concentration were low in this region.

Concerning gas velocities through functions, calculation-experiment agreement is good for low velocities (less than 0.5 m/s). For high velocities (more than 1.5 m/s), the calculations underpredicts about a factor of 2) the experimental value. A partial explanation can be that experiment and calculation provide maximal and average gas velocities, respectively. Calculations typically predicted the gas temperature well in most of the compartments in the facility, except in the dead-end rooms, rooms with complicated geometries, and source rooms. This was similar to the difficulties observed in the predictions for the helium concentrations. Concerning the gas pressure, calculations reproduced the general trends, but slightly underestimated the experimental values.

Comparison between the experimental and the RALOC results show an overall good agreement. RALOC is not only able to predict helium concentration, but also gas/wall temperatures, and gas pressure in complex situations. One has to underline anyhow that only average values can be deduced from this code. A fine detailed distribution of helium concentration would require a 3-D code.

Future work will be devoted to validate the combustion model included in the code from the experimental data of NUPEC hydrogen distribution tests performed at Takasago Engineering Laboratory by MHI (Mitsubishi Heavy Industries) in a steel containment vessel of 270 m³.

8. REFERENCES


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NUPEC's Hydrogen Mixing and Combustion Tests, 3-D Calculation of the M2-7 Test with TRIO-EF Code. Report to be published.
FIGURE 10. M2-2 Test. Gas velocities through the junctions
Calculation-Experiment Comparison
5.4 Summary of Hydrogen Combustion Tests Results at NUPEC

Takashi Hashimoto and Koki Inagaki
Systems Safety Department, NUPEC

ABSTRACT

Experimental and analytical investigations for \( \text{H}_2 \) combustion behavior during severe accident have been conducted in the world and a lot of data-base have been obtained. NUPEC has been carrying out three types of combustion test (small-scale basic test, large-scale demonstrative test and high-temperature condition test cooperated with USNRC-BNL) and finished these tests until fiscal year 1996. The latest test results are summarized below.

In the small-scale test, the turbulent flame acceleration was realized in the two compartment. The flame propagation was observed by detecting OH radicals and hot \( \text{O}_2 \) using LIPF (Laser Induced Pre Fluorescence). From the results of the small-scale test, the empirical equation for turbulent flame accelerations was obtained.

In the large-scale test, two system integral transient tests simulating severe accident conditions were conducted. Ignition occurred at the upper dome, then the combustion flame shifted to the gas injection region and continued until gas injection was stopped. The initial to peak pressure ratio was the same order with that of the premixed tests. From these results, it became clear that the integrity of large-scale dry reactor containment vessels are confirmed even if hydrogen combustion occurred.

This work is sponsored under the contract by the Ministry of International Trade and Industry, Japan.
Summary of Hydrogen Combustion Tests
Results at NUPEC

SARJ-97
October 6, 1997 Pacifico YOKOHAMA

Takashi Hashimoto / Koki Inagaki
Systems Safety Department

Outline
Background
Objectives
Large-scale test
   Apparatus
   Test Results
   - Influence of igniter position
   - Influence of initial pressure
   - Steam condensation by spray
   - System integral transient
Small-scale test
   Apparatus
   Test Result
   - Turbulent flame acceleration
Conclusions

Background of this study
- Experimental and analytical investigations for H2 combustion behavior during severe accident have been conducted in the world and a lot of data-base have been obtained.
- In order to evaluate containment integrity against rapid pressure and temperature increase associated with H2 combustion, H2 controls are recognized as one of the key issues.
- NUPEC has been carrying out three types of combustion test (small-scale basic test, large-scale demonstrative test and high-temperature condition test cooperated with USNRC-BNL) and finished these tests during fiscal year 1996.

Objectives of this study
- Large-scale test:
  A study of hydrogen combustion phenomena in the test vessel simulating PWR large-scale dry reactor containment vessels.
- Small-scale test phase II:
  A study of hydrogen combustion phenomena concerning flame acceleration between two compartments in the test vessel.
**Apparatus** - Large-scale Hydrogen Combustion Test

Spherical model containment vessel
- Inner diameter: 8m
- Volume: 270m³

Eleven compartments simulating PWR subcompartment room
- Steam Generator: 2
- General: 8
- Dome: 1

**Parametric tests**

**Test results**
- Influence of igniter positions: the initial to peak pressure ratios were irrespective igniter position.
- Influence of initial pressures (B-9 series): they were almost the same as atmospheric tests: the initial to peak pressure ratio, the elapsed time from the turning on the igniter to the temperature rising and the relationship between the combustion gas pressure and the elapsed time.
- Steam condensation by spray (B-10 series): initially unflammable H₂-Air-Steam mixture was turned flammable by steam condensing effect due to spray actuation and was ignited by a glow type igniter installed other place. The initial to peak pressure ratio was the same or low as the previous tests.

**System integral transient tests simulating severe accident**

Two tests (B-11 series) were conducted.

**Test results**
- In these tests the pressure increased gradually, suggesting mild combustion.
- In B-11-3 test (simulated SiH), ignition occurred at the upper dome then the combustion flame shifted to the gas injection region and continued until gas injection was stopped.
- In B-11-4 test (simulated-TML), ignition occurred at the upper dome and continued until H₂ gas injection was decreased.

<table>
<thead>
<tr>
<th>RUN No.</th>
<th>Initial Temp. (°C)</th>
<th>Initial Press. (kPa)</th>
<th>Hydrogen flow (g/s)</th>
<th>Steam flow (g/s)</th>
<th>Spray flow (m³/h)</th>
<th>Simulated Accident Scenario</th>
</tr>
</thead>
<tbody>
<tr>
<td>B-11-3</td>
<td>85</td>
<td>140</td>
<td>0~0.94</td>
<td>29~0</td>
<td>4.3</td>
<td>SiH</td>
</tr>
<tr>
<td>B-11-4</td>
<td>85</td>
<td>128</td>
<td>0~1.2</td>
<td>107~25</td>
<td>4.3</td>
<td>TML</td>
</tr>
</tbody>
</table>

**Conditions of system integral transient tests**

![Graphs showing pressure and flow rates over time]
System integral transient test result (1)

Ignition occurred at the upper dome (B) then the combustion flame shifted to the gas injection region (C) and continued until gas injection was stopped.

System integral transient test result (2)

Ignition occurred at the upper dome (B) and continued until $H_2$ gas injection was decreased.
**Apparatus** - Small-scale Hydrogen Combustion Test  Phase III

Test vessel: cylindrical (inner diameter: 150mm, height: 300mm) two compartments divided by an orifice.

Visualization of the flame near the orifice using Laser (LIPF)

Measurement of OH radical and hot O$_2$ band

Test Results in Small-scale test

The typical results in the latest small-scale test are as follows.

- Turbulent flame acceleration was realized in the two compartments.
- The flame propagation was observed by detecting OH radicals and hot O$_2$ using LIPF (Laser Induced Pre Fluorescence).
Conclusions

(1) In any large-scale test no detonation type hydrogen combustion was observed, because maximum hydrogen concentration and maximum gas temperature might be lower than 15vol% (dry) and 200 °C, respectively.

(2) In the small-scale tests turbulent flame acceleration was realized in the two compartments and the flame propagation was observed by detecting OH radicals and hot O₂ using LIPF (Laser Induced Pre Fluorescence).
6. Session IV

Structural Integrity

Chairperson: Vincent Luk (SNL)
Co-chairperson: Y. Maruyama (JAERI)
6.1 Pressurization Test on the Full Scale Equipment Hatch Model

Shoji ARAI*, Tomoyuki MATSUMOTO*, Masashi GOTO**, Tsutomu MIEDA***

**Systems Safety Department, NUPEC
***Nuclear Energy Division, Toshiba
***Nuclear Power Division, IHI

ABSTRACT

In order to evaluate the behavior for structural discontinuous flange portion in containment vessels during severe accident condition, the full-scale test model, simulating the equipment hatch in BWR containment vessel, has been manufactured and the pressurization test has been conducted.

The test model has been pressurized up to leakage initiation with hydrostatic pressure to understand the behavior and pressure-proof limit in the flange portion. The measuring points were about 300 channels, including strain gages, displacement transducers, temperature and pressure sensors. The leakage onset in the flange portion occurred at 19.5kg/cm².

This paper described the test results including the test procedure and comparison between the measured data and analysis results focusing on the flange opening displacement, bolt strain and flange ring strain.

In addition, preliminary analyses were performed to predict the structural behavior of the model using finite element models of ANSYS code. With regard to the analysis results, the flange opening displacement and flange ring strain were well predicted below 15kg/cm², and the average bolt strain was well predicted.
Pressurization Test on the Full Scale Equipment Hatch Model

SARJ-97
October 7, 1997 at Pacifico YOKOHAMA

S. ARAI/T. Matsumoto
Systems Safety Department

M. Goto
Toshiba Corporation

T. Mieda
Ishikawajima-Harima Heavy Industries Co., Ltd.

Contents of Presentation

1. Background and Objectives
2. Test Model Configuration
3. Instrumentation
4. Test Procedure
5. Test Results
6. Pretest Analysis Condition
7. Comparison of Test Data with Pretest Analysis
8. Conclusion

1. Background and Objectives

Structural Behavior Test Program for Steel Containment Vessel (SCV)

Cooperative Containment Research Program

Scale Shell Model Test Configuration: 1/10
Plate Thickness: 1/4

Objectives:
- to understand the behavior in the flange portion of the steel containment vessel using the full-scale hatch model
- to confirm a margin of the flange portion against a loss of sealing function
- to demonstrate the integrity of the flange portion against the leakage

2. Test Model Configuration

BWR Mark-II Type Containment Configuration
2. Test Model Configuration (continued)

3. Instrumentation

Explanatory Notes
- : Displacement Transducer
o : Strain Gage (inside)
O : Strain Gage (outside)
• : Clip Gage

4. Test Procedure

5. Test Results
6. Pretest Analysis Condition

- X : Fixed
- Y : Free

**Bolt Preload**
- Same as the specified value in the actual plant

**Analysis Code**
- FEA Code "ANSYS"
- Data obtained from the material tensile test

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7. Comparison of Bolt Strain between Test and Pretest Analysis

- Bolt Strain decreased up to Internal Pressure of 3kg/cm² due to Release of the Bolt Tension in both Test and Analysis
- Average Bolt Strain Increased Linearly with Internal Pressure due to the Lack of Contact between Flange Seal Surfaces and well Predicted by Analysis
- Analysis Predicted Larger Inside and Outside Strains of Bolts because of the Difference of Strain Evaluation Point between the Test and Analysis

---

7. Comparison of Flange Displacement between Test and Pretest Analysis

- ~3kg/cm²: Inner Flange Opened, Outer Flange not Opened
- Seal Surface Contact Maintained
- 3~15kg/cm²: Both the Inner and Outer Flange Opening Increased Linearly
- Test and Analysis Results in Good Agreement below 15kg/cm²
- 15kg/cm²+: Flange Opening Nonlinearly Increased Caused by Increasing Rate of Displacement due to Plasticity

---

7. Comparison of Flange Ring Strain between Test and Pretest Analysis

- Both Test and Analysis data of Hoop Strain Negative due to the inward Deformation Characteristic Shown in the Figure
- Both Test and Analysis data of Axial Strain Positive
- Larger Difference between Test and Analysis because of the Plasticity in the Flange Ring Portion Initiated above 15kg/cm²
8. Conclusion

(1) The leakage in flange portion occurred at the pressure level of about 19.5 kg/cm² and the average flange opening displacement of about 6.3 mm.

(2) Judging from the flange opening displacement, the flange seal surfaces remained contacting each other up to the pressure level of 3 kg/cm².

(3) Judging from the change in bolt strain, the seal surface separation was initiated from the pressure level of 2~3 kg/cm².

(4) Judging from the deformation of the flange ring, the bottom of flange ring deformed inward.

(5) With regard to the analysis results,
   - Flange opening displacement well predicted below 15 kg/cm²
   - Average bolt strain well predicted
   - Flange ring strain well predicted below 15 kg/cm²

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Mepo
6.2 Pre-test Analysis on the SCV Model Test

Kuniaki KOMINE and Tomoyuki MATSUMOTO

Systems Safety Department, NUPEC

ABSTRACT

Nuclear Power Engineering Corporation (NUPEC) of Japan and the U.S. Nuclear Regulatory Commission (NRC) have been co-sponsoring and jointly funding a Cooperative Containment Research Program at Sandia National Laboratories (SNL). The purposes of the program are to investigate the response of representative models of nuclear containment structure to pressure loading beyond the design basis accident and to compare analytical predictions with measured behavior. The Steel Containment Vessel (SCV) model uses a mixed scale design; 1:10 for the geometry scale, 1:4 for the thickness scale represent certain features of an improved boiling water reactor (BWR) Mark-II containment vessel in Japan.

NUPEC has conducted pre-test analyses using two types of modeling by finite element code ABAQUS Ver.5.5. The analyses consist 2-D analysis to grasp the global behavior of SCV test model and 3-D analysis (global model and submodels) to grasp the local behavior of SCV test model. The 3-D analysis showed that the strain at the boundary of equipment hatch reinforcement plate was maximum until the internal pressure reaches to about from 4.0 MPa to 7.3 MPa. Under the pressure design from 7.3 MPa to 11.8 MPa, the strain at the knuckle joint and at the top head flange joint was maximum. Beyond 11.8 MPa, the strain at the top head apex of the model exhibited the maximum value.

This work is sponsored under the contract by the Ministry of International Trade and Industry, Japan.
Pre-test Analysis on the SCV Model Test

SARJ-97
October 7, 1997 at Pacifico YOKOHAMA

K. Komine / T. Matsumoto
Systems Safety Department

Nupec

Contents of Presentation

1. Objective of SCV Test
2. Summary of Pre-test Analysis
3. Global Shell Model Analyses
   • Global 2-D Shell Model Analysis
   • Global 3-D Shell Model Analysis
4. Local 3-D Submodel Analyses
   • Submodel Analysis of Local 3-D Top Head section
   • Submodel Analysis of Local 3-D Equipment Hatch area
5. Investigation of Critical Area
6. Conclusions

Objectives of SCV Test

The objective of the steel containment vessel (SCV) test: To evaluate the structure response up to the failure of a scaled containment vessel model by pressurization.

Summary of Pretest Analysis

- Evaluation of Maximum Strain Portion
- Confirmation of structure behavior and pressure-proof limit
- Establishment of an adequate analytical method

2-D axisymmetric shell elements
- Analysis: elasto-plastic / large displacement / contact analysis
- Model: Global

3-D shell elements
- Analysis: elasto-plastic / large displacement / contact analysis
- Model: Global and Submodel

analysis code: ABAQUS Version 5.5
**Global 2-D Axisymmetric Shell Model**

- **Analysis Model**
  - Half symmetry model
  - 606 2-node, axisymmetric shell elements (shell, reinforcement ring, support ring)
  - 24 4-node axisymmetric solid elements (top head flange)
  - 227 4-node plane stress elements (support ring rib)
- **Boundary condition**
  - Axisymmetric boundary conditions at top head apex and bottom head
  - No vertical displacement on lower surface of ring support
- **Load conditions**
  - Gravity and internal pressure loading

**Global 3-D Shell Model**

- **Analysis Model**
  - Half symmetry model (180 deg.)
  - 7812 4-node shell elements
  - Total node = 7812, Total elements = 7640
- **Boundary condition**
  - Symmetric boundary conditions for from top head apex to bottom head
  - No vertical displacement on lower surface of ring support
- **Load conditions**
  - Gravity and internal pressure loading

**Global 2-D Shell Model Analysis (Equivalent Plastic Strain)**

The equivalent plastic strain of several position as follows.

**Global 2-D Shell Model Analysis (Displacement and Plastic Strain)**

**Deformed Shape**

**Global Distribution of Plastic Strain**

(First contact at Knuckle region)
Global 2-D Shell Model Analysis Results Summary

- Analysis Results Summary of 2-D Analysis

<table>
<thead>
<tr>
<th>Internal Pressure</th>
<th>Event</th>
</tr>
</thead>
<tbody>
<tr>
<td>~2.1 MPa</td>
<td>Occurrence of first yielding at knuckle region</td>
</tr>
<tr>
<td>~3.5 MPa</td>
<td>Initiation of first contact with contact structure (CS) at knuckle region</td>
</tr>
<tr>
<td>~7.6 MPa</td>
<td>Initiation of contact with CS at upper spherical shell section</td>
</tr>
<tr>
<td>~12.5 MPa (Final increment)</td>
<td>Extension of contact region from the knuckle region throughout lower cylindrical section</td>
</tr>
<tr>
<td></td>
<td>The largest plastic membrane strain at the SCV test model was about 40%, causing at top head apex</td>
</tr>
</tbody>
</table>

- Detailed investigation in top head section is needed

Global 3-D Shell Model Analysis (Equivalent Plastic Strain)

The equivalent plastic strain of several position are as follows.

- Top Head Apex
- Knuckle region
- Upper Spherical Shell
- Upper Conical Shell
- Middle Conical Shell
- Lower Conical Shell
- Lower Cylindrical Shell

Position

Equivalent Plastic Strain of SCV Shell Wall

Global 3-D Shell Model Analysis (Equivalent Plastic Strain at near Equipment Hatch)

The equivalent plastic strain of several position are as follows.

- Top Head Apex
- Knuckle region
- Upper Spherical Shell
- Upper Conical Shell
- Middle Conical Shell
- Lower Conical Shell
- Lower Cylindrical Shell

Position

Equivalent Plastic Strain at near Equipment Hatch

Global 3-D Shell Model Analysis (Displacement and Plastic Strain)

- Deformed Shape
- Global Distribution of Plastic Strain (Contact at Equipment Hatch region)

Position

4.4 MPa
Global 3-D Shell Model Analysis (Global Distribution of Plastic Strain)

- Maximum equivalent plastic surface strain occurred at near equipment hatch reinforcement plate

<table>
<thead>
<tr>
<th>Internal Pressure</th>
<th>Event</th>
</tr>
</thead>
<tbody>
<tr>
<td>~4.4 MPa</td>
<td>Initiation of contact with CS at near equipment hatch</td>
</tr>
<tr>
<td>~5.2 MPa</td>
<td>Initiation of contact with CS at lower conical shell section</td>
</tr>
<tr>
<td>4 ~ 7.3 MPa</td>
<td>Occurrence of maximum equivalent plastic surface strain at near equipment hatch reinforcement plate</td>
</tr>
<tr>
<td>7.3 ~ 11.8 MPa</td>
<td>Occurrence of maximum equivalent plastic surface strain at below knuckle joint and top head flange joint</td>
</tr>
<tr>
<td>11.8 MPa ~</td>
<td>Occurrence of maximum equivalent plastic surface strain at top head apex</td>
</tr>
</tbody>
</table>

Detailed investigation in Equipment hatch area is needed

Submodel Analysis of Local 3-D Top Head Shell section

- Analysis Model
  - circumference of 30 deg.
  - 4545 4-node shell elements
  - Total nodes = 4545, Total elements = 4290

- Boundary conditions
  - displacement and rotation given by global analysis results
  - SCV test model and CS nodes on symmetric surface

- Load conditions
  - internal pressure same as global analysis

Local 3-D Top Head submodel (Equivalent Plastic Surface Strain)
Submodel Analysis of 3-D Local Equipment Hatch area

- **Analysis Model**
  - 4816 4-node shell elements
  - Total node=4816, Total element=4572

- **Boundary conditions**
  - displacement and rotation given by global analysis results
  - SCV test model and CS nodes on symmetric surface

- **Load conditions**
  - internal pressure same as global analysis

Investigation of Critical Area

- **Top Head Apex**
- **Above Top Flange**
- **Below Top Flange**
- **Above Knuckle**
- **Below Knuckle**

Equivalent Plastic Surface Strain
Investigation of Critical Area (cont'd)

<table>
<thead>
<tr>
<th>Pressure Level and Critical Area</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pressure Level</td>
</tr>
<tr>
<td>----------------</td>
</tr>
<tr>
<td>4~7.3 MPa</td>
</tr>
<tr>
<td></td>
</tr>
<tr>
<td>7.3~11.8 MPa</td>
</tr>
<tr>
<td></td>
</tr>
<tr>
<td></td>
</tr>
<tr>
<td>11.8 MPa~</td>
</tr>
</tbody>
</table>

- Attention should be paid to these areas in the test.
- And the strains at several points in these areas have to be measured.

Conclusions

Analysis results, using global model and submodel, to understand the structure behavior up to failure, summarized below.

<table>
<thead>
<tr>
<th>Internal Pressure</th>
<th>Analysis results</th>
</tr>
</thead>
<tbody>
<tr>
<td>~2.1 MPa</td>
<td>Occurrence of first yielding at knuckle region</td>
</tr>
<tr>
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<td>Initiation of first contact at knuckle region</td>
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<tr>
<td>4~7.3 MPa</td>
<td>Occurrence of maximum equivalent plastic surface strain at near equipment hatch reinforcement plate</td>
</tr>
<tr>
<td>7.3~11.8 MPa</td>
<td>Occurrence of maximum equivalent plastic surface strain at below knuckle region</td>
</tr>
<tr>
<td>11.8 MPa~</td>
<td>Occurrence of maximum equivalent plastic surface strain at top head apex</td>
</tr>
</tbody>
</table>

Analysis results need a detailed comparison with the test.
6.3 Pressurization Test of a 1/10 Steel Containment Vessel Model

Tomoyuki MATSUMOTO*, Kuniaki KOMINE*
Vincent K. LUK**, Michael F. HESSHEIMER**
James F. COSTELLO***

*Nuclear Power Engineering Corporation, Tokyo, Japan
**Sandia National Laboratories, Albuquerque, New Mexico, USA
***United States Nuclear Regulatory Commission, Washington, D.C., USA

ABSTRACT

Nuclear Power Engineering Corporation (NUPEC) of Japan and the U.S. Nuclear Regulatory Commission (NRC) have been co-sponsoring and jointly funding a Cooperative Containment Research Program at Sandia National Laboratories (SNL). A series of static overpressurization tests of scale models of nuclear containment structures is being conducted under the program. One of the static overpressurization tests is a test of a steel containment vessel (SCV). The SCV model is a mixed scale model with 1:10 scaling of containment geometry and 1:4 scaling of shell thicknesses of a nuclear containment vessel that is representative of an improved, boiling water reactor (BWR) Mark II containment. A contact structure, which is a thick steel bell-shaped shell structure installed over the SCV model and separated at a nominally uniform distance from it, provides a simplified representation of some features of a reactor concrete shield building in the actual plant. There are more than 800 channels of data to be monitored and recorded during the test. A high pressure pneumatic test of the SCV model was conducted on December 11-12, 1996 at Sandia National Laboratories. This paper summarizes the planning and conduct of the high pressure test of the SCV model, and describes the preliminary results of the test.
Pressurization Test of a 1/10 Steel Containment Vessel Model

by

T. Matsumoto, K. Komine, (Nuclear Power Engineering Corp., Japan)  
J. F. Costello, (United States Nuclear Regulatory Commission)  
V. K. Luk, M. F. Hessheimer (Sandia National Laboratories)

at  
SARJ-97, Pacifico YOKOHAMA  
October 7, 1997

Outline of Presentation

• Brief description of Cooperative Containment Research Program  
• Objectives of SCV Model Test  
• Description of SCV model  
• Pretest preparations  
  • Instrumentation  
  • Pressurization system  
  • Data acquisition and display systems  
• Conduct of pressure tests  
  • Leak and instrumentation test  
  • Low pressure test  
  • High pressure test  
• Preliminary test results  
• Preliminary post-test observations  
• Summary

Brief Description of Cooperative Containment Research Program

• Program organization  
  • Nuclear Power Engineering Corporation (NUPEC) of Japan and United States Nuclear Regulatory Commission (NRC) cosponsor and jointly fund Cooperative Containment Research Program at Sandia National Laboratories.

• Purposes of the program  
  • To investigate the responses of representative models of nuclear containment structures to pressure loading beyond design basis accident, and  
  • To compare analytical predictions with measured behavior.

• Planned tests  
  • Steel containment vessel (SCV)  
  • Prestressed concrete containment vessel (PCCV)

Brief Description of Cooperative Containment Research Program (cont'd)

• Program activities  
  • Design and fabrication of scale models  
  • Instrumentation of scale models  
  • Pneumatic overpressurization tests of scale models at ambient temperature  
  • Pre- and post-test model analyses with ABAQUS code  
  • Round Robin analysis with international participants
Objectives of SCV Model Test

1. To provide experimental data for checking the predictive capabilities of analytical methods to represent some aspects of the static internal pressure response of a steel containment,
   a) beyond the elastic range, without consideration of contact with a surrounding shield structure
   b) after contact with a surrounding shield structure,
2. To investigate the failure mode of the model, and
3. To provide experimental data useful for the evaluation of actual steel containments.

Description of SCV Model

SCV Model Configuration

- Mixed-scale model
  - 1:10 in geometry and 1:4 in shell thickness
  - representing an improved boiling water reactor (BWR) Mark II containment
  - Approximately, model is 2900 mm in diameter (largest) and 6000 mm tall
- Containment details included in model:
  - Equipment hatch opening and reinforcement plate (hatch is not to scale and hatch cover is welded shut)
  - Drywell head (welded shut)
  - SGV 480 / SPV 490 material transition location
- Containment details omitted from model:
  - Other hatches, airlocks, and penetrations
  - Internal structures
  - Lower wetwell and wall-basemat junction (replaced by thick bottom head)

SCV Model Installation, March 22, 1995
Description of SCV Model (cont'd)

- Contact structure (CS)
  - A bell-shaped steel structure installed over the model, separated by a nominally uniform gap of 18 mm
  - 38 mm thick, ASTM SA516 Grade 70 steel
  - CS was welded to ring support girder and its top end was open at elevation of knuckle region of SCV model.
  - Purpose of CS
    - Intended to investigate SCV model responses against an almost rigid surrounding structure when the model expands under internal pressurization
    - Not intended to simulate concrete surrounding shield building in physical plants

Instrumentation

- More than 800 instruments were installed on SCV model and CS to obtain model responses under overpressurization conditions
  - to investigate deformation behavior of model, and
e  - to compare test data with analysis results.
- These instruments were strategically positioned and carefully placed to obtain the desirable test data.

1. Strain measurements
   - Single element and multi-element (strip and rosette) strain gages

2. Displacement measurements and contact detection
   - Resistance potentiometers on inside and outside of model
   - Interior potentiometers mounted on an interior central support column
   - Linear variable differential transformers (LVDTs) through holes in CS
   - Contact detectors through holes in CS

Pretest Preparations

- Instrumentation (cont'd)
  3. Pressure and temperature measurements
     - Pressure transducers to measure internal nitrogen gas pressure
     - Temperature detectors to monitor nitrogen gas temperature
     - Thermocouples to measure local surface temperatures of model
  4. Video coverage
     - Three high pressure rated interior video cameras and seven standard exterior video cameras
  5. Acoustic emission system
     - Twenty-four sensors to detect acoustic emission to signal the onset of a potential model failure
Conduct of Pressure Tests

- Three separate pressure tests:
  - Leak and instrumentation test @ 0.2 $P_d$ (October 3, 1996)
  - Low pressure test @ 1.5 $P_d$ (November 7, 1996)
  - High pressure test to model failure (December 11-12, 1996)
    - where $P_d = 0.78$ MPa
- Leak and instrumentation test
  - To check functionality of pressurization system, installed instruments and data acquisition system
- Lower pressure test
  - To provide a performance check on all operating units while model behaved elastically
  - To serve as a "dry-run" for high pressure test

Pressurization Sequence of High Pressure Test

- First stage
  - Model behaved as a stand-alone structure in the elastic domain
- Second stage
  - When average displacement at an elevation reached 9 mm, pressure condition was held constant for 30 minutes
- Third stage
  - Model behaved plastically and a longer time was required to achieve a state of steady structural response for each pressure rise
Conduct of Pressure Tests (cont’d)

High pressure test
- SCV model underwent a monotonic pressure rise to its failure without cycles of unloading and reloading.
- Test was planned to terminate when either a model failure occurred or the internal pressure reached 15.9 Pd, the design capacity of pressurization system.
- Model failure could mean
  - A structural failure - a catastrophic failure of model or development of a major tear in model
  - A functional failure - pressurization system could not maintain pressure at a given level inside model

Preliminary Test Results

Post-test inspection of SCV model revealed two tears:
- A large tear of approximately 190 mm long
  - was found along the weld seam at the outside edge of equipment hatch reinforcement plate
  - was responsible for the leakage of model leading to the termination of the high pressure test
- A small meridional tear of approximately 55 mm long
  - was found next to a semi-circular hole in the stiffening ring above equipment hatch
Preliminary Test Results (cont'd)

Strain Data Around Equipment Hatch Reinforcement Plate

- STG1-EDH-37b: 8.8% strain @ 4.66 MPa
- STG1-EDH-16b: 5.3% strain @ 4.66 MPa

- Note: All dimensions in mm

Free-field response of model was represented by the hoop strain data recorded at upper conical shell section above equipment hatch.

Narrow range of strain variations at this section suggests that the model experienced axisymmetric expansion there.

Preliminary Test Results (cont'd)

Layout of strain gages around equipment hatch reinforcement plate

A Tear Inside a Semi-Circular Opening at Middle Stiffening Ring

Middle Stiffening Ring

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Preliminary Test Results (cont’d)  
Horizontal Displacement Data at Middle Conical Shell Section

- The highest horizontal displacement data, ranging from 19 to 27 mm, were recorded at the middle conical shell section immediately above the material change interface.

Preliminary Post-Test Observations

- High pressure test was terminated when the pressurization system at its maximum flow capacity could not maintain pressure inside the model.
- Maximum internal pressure achieved during the high pressure test was 4.66 MPa or 5.97 $P_d$.
- Post-test inspection revealed a large tear, approximately 190 mm long, at the outside edge of equipment hatch reinforcement plate and a smaller tear, about 55 mm long, inside a semi-circular opening at the middle stiffening ring.

Preliminary Test Results (cont’d)

- Vertical displacement data (at the end of high pressure test)
  - Top head region: 17.3 mm
  - Center of equipment hatch: -2.8 mm
- Acoustic emission data
  - Acoustic emission data indicate that the two tears were initiated and started to propagate at an internal pressure around 4.25 MPa.
- Measurement of gap sizes between contact structure and model
  - Four arrays of holes were drilled through the contact structure.
  - Gap sizes between the two structures were measured before and after the high pressure test.
  - Post-test measurements indicate very small gap sizes at some hole locations, suggesting that local contact between the two structures might occur there.

Summary

- The high pressure test of SCV model was conducted on December 11-12, 1996.
- The model failed during the high pressure test when a tear developed along the equipment hatch reinforcement plate resulting in leakage.
- The maximum pressure achieved in the high pressure test was 4.66 MPa which is 5.97 times the design pressure.
- On-going efforts are focusing on post-test failure analysis and inspection, and metallurgical evaluation.
Preliminary Post-Test Observations

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- Maximum internal pressure achieved during the high pressure test was 4.66 MPa or 5.97 P_d.
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Summary

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- On-going efforts are focusing on post-test failure analysis and inspection, and metallurgical evaluation.
6.4 Preliminary Analysis and Instrumentation Planning of a Prestressed Concrete Containment Vessel Model

David W. PACE*, Michael F. HESSHEIMER*, Robert A. DAMERON **
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ABSTRACT

A series of static overpressurization tests of scale models of nuclear containment structures is being conducted by Sandia National Laboratories for the Nuclear Power Engineering Corporation of Japan and the U.S. Nuclear Regulatory Commission. As part of this program, a 1:4-scale model of a prestressed concrete containment vessel (PCCV) will be pressurized up to its ultimate capacity.

One of the key program objectives is to develop validated methods to predict the structural performance of containment vessels when subjected to beyond design bases loadings. An analysis effort including two and three-dimensional nonlinear finite element analysis of the PCCV test model is being conducted to evaluate the models structural performance under very high internal pressurization. The associated tasks include the preliminary prediction of failure pressure and probable failure locations and the development of models to be used in the detailed failure analysis.

This paper describes the preliminary analysis effort and plans for the instrumentation of the PCCV model. The preliminary analysis effort serves to provide guidance for placement of instrumentation and identify candidate failure modes. The instrumentation, consisting of approximately 2000 channels of data, is designed to monitor the response of the model during prestressing operations, during structural integrity and integrated leak rate testing, and during the test to failure of the model.

This work is jointly sponsored by the Nuclear Power Engineering Corporation and the U.S. Nuclear Regulatory Commission. The work of the Nuclear Power Engineering Corporation is performed under the auspices of the Ministry of International Trade and Industry, Japan. Sandia National Laboratories is operated for the U.S. Department of Energy under Contract Number DE-AC04-94AL85000.
Preliminary Analysis and Instrumentation of a Prestressed Concrete Containment Vessel Model

D. W. Pace, M. F. Hessheimer, V. K. Luk
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PCCV Model Description

- The model is representative of a large, dry PWR prestressed containment vessel.
- The model scale is uniformly 1:4.
- Design Pressure is 0.4 MPa (57 psig).
- Model was designed by NUPEC (Mitsubishi/Obayashi).
- Liner segments fabricated in Japan and shipped to Sandia.
PCCV Model Analysis

- Preliminary Analysis
  - investigate and rank order potential failure modes
  - support instrumentation planning
  - demonstrate analytical methods for predicting failure of PCCV
- Pretest Analysis
  - Refine and finalize pretest predictions using actual material properties, as-built conditions, etc.
- Posttest Analysis
  - validate pretest predictions and methods by comparing with actual behavior
  - demonstrate analytical methods for simulating failure of PCCV

Global Axisymmetric Analysis

- Global Axisymmetric and 3D Analysis
- Local Liner Analysis
  - Wall-Base Junction
  - Equipment Hatch
- Local Shear Failure
  - Wall-Base Junction
  - Tendon Gallery
- Tendon Analysis
- 3D Equipment Hatch Analysis
Local Liner Analysis - Equipment Hatch

3D Equipment Hatch Model

Local Shear Failure

Potential Failure Modes

- Free-field Liner tearing
- Liner tearing at discontinuity
- Liner anchor failure
- Shear failure at wall-basemat junction
- Shear failure in basemat (@ tendon gallery)
- Equipment Hatch leakage
- Personnel Air Lock leakage
- Small Penetration leakage
- Tendon wire or strand failure
- Tendon anchor or bearing failure
PCCV Instrumentation

- Displacement: 140 Channels
- Reinforcement Strains: 364 Channels
- Concrete Strains: 36 Channels
- Liner Strains: 675 Channels
- Tendon Strains: 336 Channels
- Tendon Forces: 132 Channels
- Total: 1774 Channels
- Video and acoustic monitoring will also be provided inside and outside the model.
- Two suites of instrumentation will be used:
  - Low range/high Resolution to monitor elastic behavior during prestressing and low pressure testing
  - High range/Lower Resolution to monitor inelastic behavior during high pressure testing

Displacement Measurements

- Instrumentation Layout

- Reinforcing Strains
Liner Strains

Test Sequence
6.5 Analytical Study on Change of Tendon Tension Force Distribution during the Pressurization Process of Pre-Stressed Containment Vessel

Takako KASHIWASE and Hideo NAGASAKA
Systems Safety Department, NUPEC

ABSTRACT

NUPEC has been planning the ultimate strength test for structural integrity of 1/4 scale Japan Pre-stressed Concrete Containment Vessel (PCCV), simulating severe accident condition, in cooperation with USNRC. The ancillary test for the PCCV test, in which friction coefficient of hoop tendon was evaluated using the same tendon as that of 1/4 PCCV model, was also conducted. An empirical correlation as a function of loading end load and circumferential angle within the elastic limit of tendon was obtained, utilizing measured data of the ancillary test. The validity of the correlation was confirmed by comparing tendon elongation estimated by the present correlation with the measured one.

The obtained correlation was incorporated into ABAQUS Ver.5.5 and the tendon tensile force distribution change during pressurization process of PCCV was analyzed in order to increase the predictive capability of analysis for the 1/4 PCCV test. The analysis results using the present friction coefficient correlation was compared with the results of analysis using constant friction coefficient, and the effect of the difference on tendon tensile force distribution and concrete strain were evaluated. The present friction correlation provided higher internal pressure to cause concrete cracking compared with constant friction case. This was due to the smaller friction coefficient in the present correlation and the resultant larger compression force of tendon.

This work is sponsored under the contract by the Ministry of International Trade and Industry, Japan.
Analysis Study on Change of Tendon Tension Force Distribution during the Pressurization Process of Pre-stressed Concrete Containment Vessel

**SARJ-97**
October 7, 1997 Pacifico YOKOHAMA

T. Kashiwase / H. Nagasaka
Systems Safety Department

**Outline of NOPEC 1/4 PCCV Test**

- **Objectives**
  - To evaluate ultimate structural behavior under severe accident conditions
  - To increase the predictive capability of analysis
- **Features of test facility**
  - 1/4 uniform scale
  - Major parts correctly simulated in the model
    - Dome, Cylinder, and Base mat
    - Tendon ballary and Buitress
    - Liner anchor system
- **Penetrations also simulated**
  - Equipment hatch
  - Air rock
  - Main steam and Feed water
- **Scaled number of hoop tendons and vertical tendons**

**Outline of Tendon Tensile Force Distribution Measurement Test**

- **Geometry of test specimen**
  - 559mm radius identical to that of 1/4 PCCV model
  - Tendon consisting of 3 strands within a 35 mm diameter sheath
  - 13.7mm strand diameter (corresponding to (1/4)² of actual plant total tendon area), close to actual plant strand diameter of 12.7mm
- **Instrumentation**
  - Load at both ends --- Load cell
  - Local strains of strand
    - Tensmeg gauge (4 locations)
    - Wire strain gauge (4 locations × 3)
- **Test procedure**
  - 50 kN interval loading of tendon
  - Maximum tensile force of up to 490 kN (allowable tensile stress of tendon)
  - 3 times experiment

**Objectives**

- To obtain realistic tendon friction coefficient correlation as a function of loading end load and angle, utilizing 1/4 PCCV ancillary test
  - Incorporation of the present correlation into ABAQUS Ver.5.5
- To evaluate the effect of the present correlation on ABAQUS structural response of 1/4 PCCV test during pressurization comparing conventional constant friction coefficient analysis
  - Tendon tensile force distribution
  - Concrete cracking behavior
  - Strain distribution of liner, rebar, and concrete

Increasing predictive capability of 1/4 PCCV analysis
Current Status of Tendon Tensile Force Distribution Evaluation Method

- Equation based on local force balance of tendon and integrated from $P=P_0$ at $\theta=0$, assuming constant friction coefficient
  $$P=P_0 \exp(-\mu_\lambda \theta) \quad (1)$$

- Modified design equation, considering friction loss along tendon length ($\lambda=0.001$)
  $$P=P_0 \exp(-\mu_\lambda (\theta+\lambda_0)) = P_0 \exp(-\mu_\lambda \lambda R \theta) \quad (2)$$

- Experience during tendon setting process of actual containment vessel
  - Friction coefficient increasing as tensile force increasing
  - Disagreement of measured elongation data with that using Eq.(2)

Existence of loading load and angle dependency on friction coefficient

Evidence of Angle and Loading End Load Dependency on Friction Coefficient

- Angle dependency ($\mu_\lambda = \ln(P/P_0)/(\mu+\lambda_0)$)
  - Lower value in the vicinity of loading end (like straight part behavior)

- Loading end load dependency
  - $\mu_\lambda = 1/\theta \ln(P_0/P_0)$
  - Friction coefficient increased as load increased

Previous sector size models test also consistent with the above results

Data Evaluation Method of Tendon Tensile Force Distribution

- 1/3 of the measured force at anti-loading end by load cell = Force estimated by averaging 3 wire strain gauges (WSG) measured at $\pi$
  - All tensmrg gauges failed
  - Neglect of friction force at straight part (between $\pi$ and load cell location)
  - All 3 WSGs available data only adopted considering strain variation at one cross section

- Status of all 3 remaining WSG
  1/4 $\pi$ up to ~200 kN
  1/2 $\pi$ up to ~150 kN
  3/4 $\pi$ up to ~300 kN
  $\pi$ up to ~500 kN

Derivation of Empirical Correlation Considering Dependency of Angle and Loading End Load

- Tensile force distribution assuming constant friction coefficient
  $$\ln(P/P_0) = \mu_\lambda \theta$$

- Tensile force distribution considering $0$ and $P_0$ dependency of friction coefficient assumed to be
  $$\ln(P/P_0) = \mu_\lambda (a_0+b_0 t+c_0)$$
  $$\mu_\lambda = d + e \exp(f P/P_{\text{fre}})$$

Empirical constants $a_0, b_0, c_0, e$, and $f$ evaluated by least square fitting
Tensile Force Distribution and Loading
End Load by Present Correlation

Verification of the Present Correlation by
Measured Data of Tendon Elongation

- Tendon elongation by the present correlation
  \[ \delta = J P R \exp\left(\frac{a+60+b+c}{e}\right) \]
  Data Evaluation method
  - Straight part elongation in data eliminated to compare with the correlation
  - Data less than 100 kN excluded due to inclusion of initial looseness of tendon
    and extrapolated from loading data of 150 kN and 200 kN
  - Prediction of elongation by the present correlation
    - Good agreement except for the 3rd time test above 450 kN due to the plastic
      deformation of tendon
  - Prediction of elongation by constant friction coefficient used in 1/4 PCCV design
    - Smaller elongation due to larger friction coefficient

Outline of ABAQUS Analytical Modeling

- One hoop tendon, the corresponding sheath, concrete, liner, rebars and vertical tendons
- Buttress portion
  - Concrete geometry correctly simulated
  - Straight part of tendon in the vicinity of loading end not simulated for simplicity

Analysis Method

- Analysis procedure
  - Step 1 Initial stressing of vertical tendon 1.4 kN/mm²
    (corresponding to setting loss value)
  - Step 2 Prestressing of hoop tendon 453 kN
  - Step 3 Setting of hoop tendon 350 kN
  - Step 4 Imposing internal pressure up to 2.5 Pd
    (corresponding to liner yielding)
- Major modeling adopted
  - Smooth option to correctly simulate contact length between nodes
  - Elastic slip value (ESV) in penalty method to stabilize friction force optimized
  through sensitivity study
Tendon Tensile Force Distribution Using Constant Friction Coefficient

- Prestressing: Decreasing continuously
- Setting: Maximum force at 0.63 rad close to 0.65 estimated by tensile force distribution equation
- From 1.5 Pd: Flattening from anti-loading end region
- From 2.0 Pd: Flattening also from loading end region

Relation between Tensile Force Distribution and Cracking Behavior Using Constant Friction Coefficient (2.0 Pd)

- Smaller tensile force in the vicinity of anti-loading end region
  → Smaller compression force
  → Initial concrete cracking
- Concrete cracking in the anti-loading end region
  → Larger radial deformation
  → Flattening of tensile force in the anti-loading end region

Hoop Strain Distribution Using Constant Friction Coefficient

- 2.0 Pd: Region
  Concrete cracking → Large deformation → Large strain of liner and rebar
  Bending behavior associated with larger strain in outer element
  Smaller strain due to existence of buttress
- 2.5 Pd: Whole concrete cracking → Larger strain in whole region
  Minimum strain at 0.63 rad corresponding to the largest tendon tensile force

Relation between Tensile Force Distribution and Cracking Behavior Using Constant Friction Coefficient (2.0 Pd)

- Smaller tensile force in the vicinity of anti-loading end region
  → Smaller compression force
  → Initial concrete cracking
- Concrete cracking in the anti-loading end region
  → Larger radial deformation
  → Flattening of tensile force in the anti-loading end region

Tendon Tensile Force Distribution Using the Present Friction Coefficient Correlation

- Prestressing: Decreasing continuously
  Convex curve in the vicinity of loading end due to functional characteristics of the present correlation
- Setting: Maximum value at 1.08 rad (Larger setting loss region due to smaller friction coefficient in the vicinity of loading load compared with constant friction case)
- 2.5 Pd: Flattening throughout the whole region
Relation between Tensile Force Distribution and Cracking Behavior Using the Present Correlation Comparing Constant Friction Coefficient Analysis (2.0 Pd)

- Smaller friction coefficient $\rightarrow$ Larger tendon tensile force $\rightarrow$ Larger compression force $\rightarrow$ Delay of concrete cracking
- Initial cracking pressure -Using constant friction coefficient 1.6 Pd -Using the present correlation 2.0 Pd
- Smaller force difference along tendon length $\rightarrow$ Smaller difference in strain $\rightarrow$ Nearly simultaneous concrete cracking

Friction Force Distribution and Radial Deformation Using the Present Correlation Comparing Constant Friction Analysis

- Smaller friction coefficient $\rightarrow$ Larger tensile force $\rightarrow$ Smaller radial deformation
- Reduction in friction force due to concrete cracking
- Smaller change in tensile force distribution $\rightarrow$ Larger setting loss region

Strain Distribution Using the Present Correlation Comparing Constant Friction Coefficient Analysis

- 2.0 Pd: Smaller region of concrete cracking $\rightarrow$ Smaller and nearly uniform strain
- 2.5 Pd: Simultaneous concrete cracking $\rightarrow$ Smaller strain difference along $\theta$

Conclusion

- An empirical friction correlation was obtained as a function of loading end load and angle within the elastic limit of tendon and was incorporated into ABAQUS Ver.5.5, utilizing measured data of the 1/4 PCCV ancillary test
  - Validity of the present correlation coefficient was confirmed by tendon elongation data
- The present friction correlation provided smaller friction force and the resultant higher internal pressure to cause concrete cracking compared with constant friction analysis
  - Conventional friction coefficient provided conservative analysis results
- Structural response of 1/4 PCCV during pressurization process using the present correlation became different from that using constant friction coefficient
  - The present correlation will provide realistic prediction
- The present analysis was reflected in the instrumentation planning of 1/4 PCCV
6.6 Studies on Reactor Piping Integrity during Severe Accident in WIND Project

Akio Maeda, Yu Maruyama, Yubei Harada, Hiroaki Shibazaki,
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Japan Atomic Energy Research Institute
*System Twenty One Inc.

In a severe accident of a light water reactor, the reactor coolant piping might be subjected to thermal loads resulted from decay heat release of the deposited fission products, and heat transfer from high temperature gases generated in the reactor core region in addition to an internal pressure load. Thermal and structural responses of the reactor coolant piping under elevated temperature and pressure conditions are being investigated in piping integrity tests in WIND (Wide Range Piping Integrity Demonstration) project at Japan Atomic Energy Research Institute.

Four piping failure tests have been conducted using straight stainless steel pipes with a diameter of 114.3mm or 355.6mm which simulate a part of the reactor coolant piping. In a recent failure test in which a pipe with an outer diameter of 355.6mm and a wall thickness of 35.7mm was used and an internal pressure was kept at 10MPa, a failure of the pipe was observed when the maximum temperature of the pipe reached 1040°C after the temperature was sustained at 1000°C for about one hour. Several small openings were formed at the pipe failure in an area showing the maximum temperature. A history of the change in a vertical diameter was obtained from the measurement of displacement at the top and the bottom of the pipe at the center of heated area. An acceleration of piping ballooning was observed when the maximum temperature of the pipe became approximately 1000°C. It was confirmed in post-test observation that the pipe diameter in vertical and horizontal direction was remarkably enlarged and the pipe wall thickness was largely reduced.

In parallel with the tests, a post-test analysis was performed using ABAQUS code. The results from 2D elasto-plastic creep analysis for a piping failure test with an outer diameter of 114.3mm and a wall thickness of 13.5mm qualitatively reproduced the final deformation including the diameter enlargement and the wall thinning.
1. Introduction

WIND (Wide Range Piping Integrity Demonstration) Project
- FP aerosol behavior in reactor piping
- Integrity of reactor piping

Objectives of Present Study
- To obtain experimental insights on thermal and mechanical behavior of a reactor coolant pipe under various thermal and internal pressure loads
- To investigate the applicability of analytical models through post-test analyses
Structure of Medium Diameter Test Pipe

Thermocouple Locations over Test Pipe Outer Surface at Center of Heating Area (WPH10 and WPH11)

Test Conditions

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Piping Material</th>
<th>Internal Pressure (MPa)</th>
<th>Maximum Temperature (°C)</th>
<th>Outer Diameter / Wall Thickness (mm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>WPH3(^1)</td>
<td>SUS316 (\text{nuclear grade})</td>
<td>9.8</td>
<td>1070(^2)</td>
<td>114.3 / 13.5</td>
</tr>
<tr>
<td>WPH4(^1)</td>
<td>CF8M</td>
<td>9.8</td>
<td>1000(^2)</td>
<td>114.3 / 13.5</td>
</tr>
<tr>
<td>WPH6(^1)</td>
<td>SUS316 (\text{nuclear grade})</td>
<td>4.9</td>
<td>1130(^2,3)</td>
<td>114.3 / 13.5</td>
</tr>
<tr>
<td>WPH10</td>
<td>SUS316</td>
<td>9.8</td>
<td>1000</td>
<td>355.6 / 35.7</td>
</tr>
<tr>
<td>WPH11(^4)</td>
<td>SUS316</td>
<td>9.8</td>
<td>1040(^3)</td>
<td>355.6 / 35.7</td>
</tr>
</tbody>
</table>

1) Presented at SARJ-96
2) Temperatures at piping failure
3) Estimation (failure of thermocouple at the highest temperature area)
4) Deformed pipe in WPH10 was used

Method for Radial Displacement Measurement
Test Results

History of Temperature Observed in WPH11

Observed Radial Deformation in WPH11

Variation of Post-Test Wall Thickness for WPH11
Summary of Conditions at Piping Failure and Diameter Enlargement

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Piping Material</th>
<th>Internal Pressure (MPa)</th>
<th>Difference of Temp. (°C)</th>
<th>Ratio of Diameter (Post-Test/Initial)</th>
</tr>
</thead>
<tbody>
<tr>
<td>WPH11</td>
<td>SUS316</td>
<td>9.8</td>
<td>76</td>
<td>Vertical: 1.28, Horizontal: 1.19</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>One hour hold at 1000°C in WPH10.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Failed at 1040°C during heating phase with 130°C/h.</td>
</tr>
<tr>
<td>WPH3</td>
<td>SUS316 (Nuclear Grade)</td>
<td>9.8</td>
<td>96</td>
<td>Vertical: 1.31, Horizontal: 1.24</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>After 2 hours hold at 1000°C, reheated with 130°C/h.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Failed at 1070°C.</td>
</tr>
</tbody>
</table>

*Circumferential stress on outer surface caused by internal pressure
WPH11: 34.7 MPa, WPH3: 27.5 MPa

3. Post-Test Analysis
- 2D Elasto-plastic creep analysis for WPH3 (Failure test with small diameter pipe) with ABAQUS V5.5
- Creep equation was approximated using data obtained from specimen creep tests
- Boundary conditions:
  - Constant internal pressure: 9.8 MPa
  - Constant temperature at top of pipe: 1000°C
  - Circumferential temperature distribution: Based on test result
  - Temperature difference between outer and inner surfaces: 1.4°C

Approximated Creep Equation

\[
\log(\sigma) = 5.36024 - 0.00016T(19 + \log(t_{10\%}))
\]

\(\sigma\): Stress (MPa)  
\(T\): Temperature (K)  
\(t_{10\%}\): Time required to generate 10\% strain in secondary creep strain

Specimen test  
Fitting
Circumferential Temperature Distribution
Used in Analysis (at Outer Surface)

Measured temperature in WPH3
Interpolated temperature distribution

Degree from top of pipe

Analytical Result on Deformation
Sustained at 1000°C
for 5 hours

Evaluation of Creep Damage
During Temperature Increase Period
Definition of creep damage: \[ \sum \left( \Delta t / tr(t) \right) \times \text{time(h)} \]

Stress vs Larson-Miller Parameter
\[ \log e = 47352 - 6000170(17+\log tr) \]
\[ \sigma : \text{Stress(MPa)} \]
\[ T : \text{Temperature(K)} \]
\[ tr : \text{Time to rupture(h)} \]

Time to give identical creep damage when constantly heated at 1000°C
\[ Tc = 4.7 \text{ h} \]
\[ Tc = 2.35 \text{ h} \]
\[ Tc = 0.25 \text{ h} \]

Comparison for Deformation
between Test and Analysis
Unit: mm

<table>
<thead>
<tr>
<th></th>
<th>Post-Test Configuration</th>
<th>Analytical Result*</th>
</tr>
</thead>
<tbody>
<tr>
<td>Vertical diameter</td>
<td>150.0</td>
<td>152.4</td>
</tr>
<tr>
<td>Horizontal diameter</td>
<td>141.5</td>
<td>140.0</td>
</tr>
<tr>
<td>Wall thickness at upper part</td>
<td>6.0</td>
<td>6.8</td>
</tr>
<tr>
<td>Wall thickness at lower part</td>
<td>12.4</td>
<td>11.4</td>
</tr>
</tbody>
</table>

* Deformation due to thermal expansion was excluded.
4. Conclusions

- The following insights were obtained.
  - Acceleration of piping ballooning
  - Remarkable reduction of piping wall thickness
  - Formation of small opening at piping failure
  - Identification of piping failure conditions

- The post-test analysis with ABAQUS reproduced quantitatively the final deformation of the failed piping.
6.7 Metallurgical Examination of Piping Failed at High Pressure and High Temperature in WIND Project

Yuhei Harada, Akio Maeda, Yu Maruyama, Hiroaki Shibasaki, Akihide Hidaka and Jun Sugimoto

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Abstract

The WIND (Wide range piping INtegrity Demonstration) project is being performed at JAERI in order to prove the integrity of LWR piping under severe accident conditions. The failure behavior of pipings under beyond-design-basis conditions, i.e., if a considerably elevated temperature, has been investigated by means of heating tests with small diameter straight pipings. The test pipe had a length of 2000 mm, an outer diameter of 114.3 mm and a wall thickness of 13.5 mm. The cold-drawn type 316 (SUS316) and cast (CF8M) stainless steel pipings were used as the test materials. Since the high-temperature strength data of piping materials are not enough, the specimen tests are performed above 800 °C. The metallurgical examination was conducted to observe microstructures of failed pipings.

As a result of the tensile test and short term creep test, 0.2% proof stress, tensile strength and creep rupture strength of SUS316 were superior to those of CF8M at high temperature. Ductility of SUS316 was higher than that of CF8M. The SUS316 and CF8M pipings failed under a constant internal pressure of 10 MPa using N₂ at 1070 and about 1000°C, respectively during heat-up at a rate of 130°C/h. The deformation (increase of diameter and reduction of wall thickness) of SUS316 piping was greater than that of CF8M. The piping failure behavior could be qualitatively explained by the results of the specimen tests.

As a result of the metallurgical examination, many cracks occurred from the outer surface of the pipings at heating area probably due to circumferential tensile stress. The cracks were observed mostly on the grain boundaries in SUS316 and within the dendrite phases in CF8M. The distribution of main elements was uniform in SUS316, as received. On the other hand, the microstructure of CF8M was two phase, which consisted of dendrite and matrix. The Ni content, which influences high temperature strength, was lower in dendrite than in matrix. The difference in the piping failure behavior can be attributed to the chemical composition and the microstructure due to the production method.
1. Introduction

- The failure behavior of pipings under beyond-design-basis conditions, i.e. a considerably elevated temperature, has been investigated by means of heating tests with small diameter straight pipings in WIND.
- The specimen tests of piping materials are performed above 800°C to obtain high temperature strength data.
- Metallurgical examination was conducted to observe microstructures of failed pipings.

2. Specimen Test

- Materials
  - cast stainless steel (CF8M) pipe
  - cold drawn type 316 stainless steel (SUS316) - nuclear grade pipe
- Specimen taken from the piping
  - Dimension (mm) of the piping: 2000 L, 114.3 OD, 13.5 WT
- Chemical composition of piping (wt.%)
• Tensile test
  - standard specimen (axial direction)
  - subsize specimen (circumferential direction)
  - atmosphere: air
  - temperature: from RT to 1000°C
  - strain rate:
    standard specimen; 0.3%/min up to 0.2% proof stress level
    7.5%/min beyond it
    subsize specimen; 0.9%/min up to 0.2% proof stress level
    7.5%/min beyond it

• Creep test
  - specimen: standard (axial direction)
  - atmosphere: air
  - temperature: from 600 to 1000°C
  - target of time to rupture: 1, 10, 100h

Geometry of test specimen

![Geometry of test specimen](image)

![Tensile strength of piping materials at high temperature](image)

![0.2% proof stress of piping materials at high temperature](image)
3. Failure Conditions of Piping

Pipe Structure Used in Piping Integrity Test
4. Metallurgical Examination

- Crack pattern by optical microscopy
  - polished surface
  - chemically etched surface in aqua regia
- Element distribution
  - by XMA (X-ray microanalyzer)
- Main chemical composition (Ni, Cr, Mo)
  - by ZAF (Z: atomic number effect, A: absorption effect, F: fluorescence effect) method
Area analysis by XMA of WPH3 (SUS316) at failure location

Microstructure of WPH3 (SUS316) at failure location (0 degree)

- Circumferential Cross-section
- Axial Cross-section

Microstructure of WPH4 (CF8M) at failure location (0 degree)

- Circumferential cross-section
- Axial cross-section
5. Summary (1/2)

1. Proof stress, tensile strength and creep rupture strength at high temperature of SUS316 (cold drawn pipe) were superior to those of CF8M (cast pipe). Ductility of SUS316 was higher than that of CF8M. The piping failure condition could be qualitatively explained by the results of the specimen tests.

2. The cracks occurred from the outer surface of the pipings at heating area probably due to circumferential tensile stress.

5. Summary (2/2)

3. The cracks were observed mostly on the grain boundaries in SUS316 and within the dendrite phases in CF8M.

4. The distribution of main elements was uniform in SUS316, as received.

5. The microstructure of as received CF8M was two phase which consisted of dendrite and matrix. The Ni content, which influences high temperature material strength, was lower in dendrite than in matrix.
7. Session V

PHEBUS/FP Program

Chairperson: J. Fermandjian (NUPEC/IPSN)
Co-chairperson: A. Hidaka (JAERI)
7.1 Fission Product Release, Transport and Chemistry
Indications from the First Two Phebus-FP Tests

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The paper begins by presenting the status of the experimental information on fission product release
and transport coming from test FPT0 using trace-irradiated fuel, and the first data on the same topic
from FPT1, which used irradiated fuel with a substantial inventory of fission products, and then goes on
to look in more detail at the chemistry of iodine in the Phebus containment vessel.

Release calculations made simultaneously with the bundle degradation analyses presented in a
companion paper show that two kinds of difficulty have been encountered in FPT0 in predicting the
releases of volatile elements. Analysts using CORSOR or other models adopted to irradiated fuel
predicted releases of the right order, but beginning considerably too early. Others used the Petti
correlation, which had been adjusted to fit experiments using fresh fuel. The outcome was a predicted
release which began too late. This suggests that the advanced degradation observed in FPT0 was
more effective at liberating fission products from the fuel matrix than expected on the basis of previous
experiments with fresh fuel. The picture for less-volatile elements is more complex. The link between
release and fuel degradation is clear in the first test, and it has been possible to quantify significant
deposition of certain elements just above the bundle in the second. Explanations have also been
provided for the behaviour of Ru but more effort is still needed.

Fission product transport information from the circuit and deposition information from the containment
vessel strongly suggest that, with the notable exception of iodine, all elements behaved in the same
way, very probably riding on the much more copious emissions of structural and fuel materials such as
silver, rhenium (from bundle instruments), and uranium. Neither simple models of the TRAP type nor
the full equilibrium treatment of SOPHAEROS or VICTORIA appear able to reproduce the measured
deposition in the steam generator. Mechanical resuspension may have played a role (it was observed
in post-test operations). For the elements present in vapour form (iodine and cadmium), an influence of
non-equilibrium chemical reactions is suspected. Both hypotheses need further investigation.

Iodine behaviour in the containment of FPT0 and FPT1 has been the object of detailed investigation
because of its safety importance. The iodine behaviour in the sump was strongly influenced by the
presence of large excess of Ag. The main stable iodine species in the sump was AgI. On-line sump
activity measurements and post-test analysis of sump samples suggest that both Ag and I behaved as
colloids. Large silver and iodine deposits were found on the sump vertical walls at the end of the test.
Post test chemical analysis of aerosol material collected in the containment atmosphere indicated that
the colloidal behaviour of Ag may be attributed to oxidation processes in the containment atmosphere.

The formation of stable AgI in the sump prevented the radiolytic formation of volatile molecular iodine in
the water phase. There was however a peak in the gaseous iodine concentration in the containment
atmosphere during the fission product release phase. This peak could only be interpreted as a direct
injection of a small but significant amount of gaseous iodine to the containment. The gaseous iodine
concentration then fell to a plateau after 5 hours, and by 10 hours had reached a very small fraction of
its initial value and remained constant for several days. Gas phase iodine speciation measurements
indicated that this small fraction could be essentially composed of organic iodides, probably formed by
reaction of iodine with the painted steel surfaces provided in the containment.

Submitted for SARJ-97
FISSION PRODUCT RELEASE, TRANSPORT AND CHEMISTRY INDICATIONS FROM THE FIRST TWO PHEBUS FP TESTS

Didier Jacquemain and Bernard Clément
Institut de Protection et de Sécurité Nucléaire

Yannis Drossinos, Elizabeth Kraussmann and Alan Jones
European Commission Joint Research Centre

FISSION PRODUCT RELEASE

FOR VOLATILE FPs, CALCULATED INTEGRAL RELEASE FOR FPT-0 IS GENERALLY OF THE GOOD ORDER BUT IT STARTS TOO EARLY WHEN USING MODELS ADAPTED TO IRRADIATED FUEL. IT STARTS TOO LATE WHEN USING CORRELATION ADAPTED TO FRESH FUEL.

FP RELEASE IS NOT ONLY TEMPERATURE DEPENDENT; IMPORTANT EFFECT OF THE DEGRADED FUEL STATE.

Schematic presentation of the circuits

PWR => Phebus scaling-down factor : -5000

containment vessel

condensers
(FPT0 : 347 K
FPT1 : 363 K)

423 K

973 K

sump
(363 K, pH 5)

steam injection line

20 UO2 rods (230GWd/ft
fuel rods in FPT1)
1 Ag-In-Cd rod
fissile length : 1m

test assembly:

steam generator

driver core cooling system

test device cooling circuit (438 K, 25 bar)

driver core

wall (383 K)

steam generator
TRANSPORT IN THE CIRCUIT

MOST OF RELEASED MATERIAL TRANSPORTED AS AEROSOLS AT 970K EXCEPT IODINE, CADMIUM, NOBLE GASES
CHEMICAL FORM OF CAESIUM?

AEROSOL DEPOSITION IN THE SG LOWER THAN CALCULATED
MECHANICAL RESUSPENSION?

PART OF IODINE TRANSPORTED AS A GAS AT 420K
REACTION OF I WITH Cs AND Rb AT 970K POSSIBLY LIMITED BY KINETICS
REACTION OF I WITH Cd IN THE SG POSSIBLY LIMITED TOO

AEROSOLS ARE MULTI-COMPONENT MAINLY COMPOSED OF
Ag, Re, Sn, In, Cd, U, Fe AND Ni

IODINE BEHAVIOUR IN THE SUMP WATER

IODINE BEHAVES AS
A NON SOLUBLE SPECIES

FORMATION OF Ag/Agl COLLOIDS

Ag COLLOIDAL BEHAVIOUR
LINKED TO OXIDATION PROCESSES
IN THE CONTAINMENT ATMOSPHERE

WATER SOLUBLE OR COLLOIDAL
FRACTION OF SILVER

Water solubility of Ag at different locations in FPT0

IODINE ACTIVITY IN THE CONTAINMENT GAS PHASE

IODINE ACTIVITY MEASUREMENT

DEGRADATION AND RELEASE PHASE

DEPOSITION ON WALLS

AEROSOL SETTLING

AEROSOL PHASE

10000 25000

Time (s)
VOLATILE IODINE BEHAVIOUR IN FPT-0

1. AgI formation in the sump prevents radiolytic production of volatile molecular iodine.
2. Gaseous iodine concentration in the short term can only be explained by injection from the circuit.
3. In the long term, organic iodine amount attributed to reaction with the painted surfaces.

IODINE BEHAVIOUR IN THE CONTAINMENT ATMOSPHERE

CONCLUSIONS

Fission product release is not only temperature dependent; the physical state of the degraded bundle plays a key role.

Caesium is transported as aerosols at 970K.
Deposition in the steam generator lower than calculated.
Part of the iodine is still gaseous at 420K.
Possible kinetic effects.
Silver traps iodine in the sump water by colloids formation; this reduces drastically radiolytic volatile iodine formation.
Gaseous iodine concentration in the containment can be explained in the short term by iodine source from the circuit.
In the long term, by organic iodides formation from paints.
7.2 THE PHEBUS EXPERIMENT FPT1

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The Fission Product Test FPT1 was conducted in the PHEBUS facility of the French <Institut de Protection et de Surete Nucleaire> (IPSN) at the Research Centre of CADARACHE, on July 26th, 1996.

The test was the second in a series of six in pile tests in an internationally sponsored program involving IPSN, Electricite de France (EDF), European Commission, Japanese Nuclear Power Engineering Corporation (NUPEC) and Japan Atomic Energy Research Institute (JAERI), US Nuclear Regulatory Commission (NRC), Canadian Candu Owners Group (COG), Korean Atomic Research Institute (KAERI), the Swiss Bundesamt fur Energiewirtschaft, Hauptabteilung fur die Sicherheit der Kernanlagen (HISK), and Paul Scherrer Institut (PSI).

The objective of the FPT1 test was to study the phenomenology of low pressure severe accident sequences for which the Fission Products (FPs) flow path involves the primary side of the steam generator and the reactor containment building. FPT1 was performed using irradiated fuel (mean burn up 23.4 GWd/MtU).

The objectives of the test may be stated separately for the bundle, circuit and containment.

For the bundle the objective was to obtain a large degradation by reaching the liquefaction temperature of the fuel, producing 2kg of liquefied fuel and the release of 70 to 80% of volatile fission products. For the circuit the objective was to investigate FPs deposition without steam condensation and to provide data on FPs chemistry including interactions of these FPs with the pipe walls and structural aerosols at high temperature. For the containment the objectives were to study FPs chemistry and specially iodine radio chemistry in the sump water and in the atmosphere, and the effects of paints in the realistic chemical conditions of a reactor accident.

The test fuel was firstly submitted to a short re-irradiation in PHEBUS in order to generate short-lived fission products in the fuel. Then the experiment itself was performed including several main stages:

The degradation phase of the bundle, caracterised by FP and structural materials releases into the experimental circuits.

An aerosol phase devoted to the study of aerosol settling in the containment.

A washing phase for the collection of the settled aerosol on the containment bottom.

A final chemistry phase whose objective was to study the sump and atmosphere chemistry, especially iodine behaviour one under radiolysis conditions.

POWER TRANSIENT

The degradation phase led to events briefly described below:
- an oxidation phase beginning quite early and slowly when the maximum temperature was 1370K and coming to an escalation (2500K maximum temperature). The cladding degradation leading to its destruction and to the interaction with the fuel, is supposed to begin during this phase.
- a heat up phase consisting in increasing the bundle power up to 29 kW with a 2.2 g.s\(^{-1}\) constant steam flowrate, then increasing the power up to 41.7 kW (end of the experiment) with a decreasing steam flowrate. Two fuel material movements were detected during this phase inducing increased material releases into the circuits and defining the moment of final test shut-down.

A description of the bundle degradation, illustrated by non-destructive examinations results, will be given in the presentation.

FP AND AEROSOL MEASUREMENTS

The first main results coming from the FP and aerosols measurements will be presented to the workshop.

They will mainly concern the iodine and aerosols behaviour in the containment about which it can be briefly said that:
- the total amount of aerosols entering in the containment was about 200g, which settled down rather quickly,
- the maximum aerosol flowrate was reached during the oxidation escalation phase,
- more than 50% of iodine was released from the bundle and a small fraction was in gaseous form in the containment at the end of the transient,
- iodine had a non soluble behaviour in the sump.

- 250 -
FPT1 - OBJECTIVES

BUNDLES (irradiated: 30 GWd/t)

- Fuel degradation
  - 2 kg liquefied fuel
  - FP release

CIRCUITS:

- FP deposition, chemistry and interactions

CONTAINMENT:

- FP chemistry
- Iodine radiochemistry in the sump water
  and in the atmosphere
- Effect of paints

FPT1 TEST SECTION

18 irradiated fuel rods
Ag-In-Cd Control rod
2 instrumented fresh fuel rods
3 thermocouples
2 Ultrasonic thermometers
Outside shroud thermocouples
Inside shroud thermocouples
Zirconia shroud

SCHEMATIC PRESENTATION OF THE CIRCUITS

PWR = Pressurized water reactor
FPT1 - EXPERIMENTAL PHASES

RE-IRRADIATION phase (6 days)
To generate short-lived FP

DEGRADATION phase (5 hours)
FP and structural material release

AEROSOL phase (12 hours)
Aerosol settling in the containment
- WASHING phase (15 minutes)
To collect settled aerosols on the bottom
- CHEMISTRY phase (4 days)
Sump and atmosphere chemistry
Aerosols inlet flowrate in the containment
(CAROLE and G point weighing)

End of validity of the CAROLE signal

- detail point G
- detail norm.
- detail flagged

FP before and after scram in the containment

CONCLUSION - FPT1

- The test met its objectives
  - ~ 2Kg instantaneous melted fuel — high release
  - Big quantity of information collected concerning FP and aerosols
    - 60,000 Gamma spectrometry spectra
    - Almost all the sampling devices operated correctly
    - Even those which didn't operate properly
    (mainly because unexpected phenomenology)
      - give informations about the physical phenomena
        (ex : overloading of impactors)

- The analysis is still in progress
  - Signals analyses
  - Numerous physical and chemical analyses
    in 7 hot laboratories in Europe

- Preparation of the next two tests FPT4 and FPT2
7.3 Analysis of Bundle Degradation Behaviour in the First Two PHEBUS-FP Tests

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Fridric Serre and Bernard Climent Institut de Protection et de Sûreté Nucléaire IPSN/DRS - CE Cadarache F - 13108 St. Paul lez Durance Cedex, France

The first experiment of the PHEBUS FP programme was performed using trace irradiated fuel in December 1993. Post-test analyses revealed that FPT-0 had reached a more advanced state of bundle degradation than any previous LWR bundle in-pile experiment, well comparable to what was observed in TMI2. The second experiment, FPT-1, was performed in July 1996 using irradiated fuel (23 GWd/tU). A similar degradation pattern has been observed, though the two fresh fuel rods in the bundle were less degraded than the irradiated ones.

None of the codes used to pre-calculate FPT-0 had predicted the early and rather large bundle degradation observed under steam rich conditions. Post Irradiation Examinations have revealed important features and helped the test interpretation.

The codes were able to overcome the associated difficulties and to calculate the temperature evolution. They have also shown, by a number of sensitivity calculations, that even extreme variations of the thermal shroud conductivity should not account for the observed phenomena. FPT-0 and FPT-1 transients can be described by calculation codes only if assuming some hypotheses and parameter changes.

The maximum temperature measured during the oxidation runaway can be re-calculated assuming that the cladding dislocation occurs at a temperature higher than observed in tests like PBF-SFD, CORA and PHEBUS CSD. Some of the codes, like ICARE2, are able to reproduce the measured hydrogen production during that phase. Some have needed modelling improvements, authorising for instance the oxidation of molten cladding inside a zirconia shell. This also results in a significant fuel dissolution in the steam rich conditions of the PHEBUS experiments.

During the subsequent heat-up phase, the best re-calculation of temperature field and final degradation state is obtained by assuming a progressive relocation of material liquefied at a temperature far below the melting point of urania. Relocated materials are retained, for a while, by the lower grid that probably failed near the end of the experiment.

Post Irradiation Examinations give some insights about the role of the silver-indium-cadmium control rod and its cladding on the degradation, mainly due to silver and iron oxides. Fuel hyper-stoechiometry effect on liquefaction temperature is also considered, as well as the irradiation effect in the case of FPT-1. Further investigations, maybe including analytical experiments are needed to quantify these phenomena.
Analysis of Bundle Degradation Behaviour in the First Two Phebus-FP Tests

I. Shepherd, A. Jones, F. Serre, B. Clément

The Phebus Bundle has 20 rods, each 1 metre long. The first test, FPT-0 produced more core damage than had been seen in any other bundle experiment.

Comparison with other tests

<table>
<thead>
<tr>
<th>Test</th>
<th>Conditions</th>
<th>Results</th>
</tr>
</thead>
<tbody>
<tr>
<td>CORA</td>
<td>electrical heating, maximum</td>
<td>maximum temperature 2673K</td>
</tr>
<tr>
<td>SFD</td>
<td>mostly in reducing atmosphere,</td>
<td>maximum temperature ≈2700K</td>
</tr>
<tr>
<td>Phebus</td>
<td>maximum ≈2800K, 18% of fuel</td>
<td>melted or liquefied</td>
</tr>
<tr>
<td>SFD 1.4</td>
<td>mostly in reducing atmosphere,</td>
<td></td>
</tr>
<tr>
<td>LOFT-FP</td>
<td>maximum ≈2900K for short period,</td>
<td></td>
</tr>
<tr>
<td>Phebus-FPT0</td>
<td>approximately 50% melted or liquefied,</td>
<td>Molten pool maintained for almost an hour</td>
</tr>
</tbody>
</table>

The Bundle Interpretation Circle coordinates activities of the bundle degradation behaviour in Phebus-FP. The aim is to:
- Build a consensus view of what happened in the first test, FPT-0
- Distribute experimental measurements
- Discuss calculational results
- Advise the experimental team as to the probable outcome of future tests.

As part of this work a code comparison exercise for FPT-0 has been conducted.

Tomograms of FPT-0

922mm 772mm 622mm 422mm

Tomograms of FPT-0

300mm 255mm 163mm 63mm
Participants to Post-Test Exercise

<table>
<thead>
<tr>
<th>Participants</th>
<th>Code</th>
</tr>
</thead>
<tbody>
<tr>
<td>IPSN, Cadarache</td>
<td>ICARE-2</td>
</tr>
<tr>
<td>JAERI, Japan</td>
<td>ICARE-2</td>
</tr>
<tr>
<td>JRC, Ispra</td>
<td>ICARE-2</td>
</tr>
<tr>
<td>UPM, Madrid</td>
<td>MELCOR</td>
</tr>
<tr>
<td>NUPEC, Japan</td>
<td>MELCOR</td>
</tr>
<tr>
<td>KEMA, Netherlands</td>
<td>MELCOR</td>
</tr>
<tr>
<td>KAERI, Korea</td>
<td>MELCOR</td>
</tr>
<tr>
<td>GRS, Garching</td>
<td>ATHLET-CD</td>
</tr>
<tr>
<td>FZK, Karlsruhe</td>
<td>SCDAP/BELAPS</td>
</tr>
<tr>
<td>ENEA, Bologna</td>
<td>SCDAP/BELAPS</td>
</tr>
<tr>
<td>CSN, Madrid</td>
<td>SCDAP/BELAPS</td>
</tr>
<tr>
<td>AEA, Winfrith</td>
<td>SCDAP/BELAPS</td>
</tr>
</tbody>
</table>

Calculations have also been performed with MAAP-4 (by EDF), KESS-III (University of Stuttgart) and MERIS (by Sandia) and presented to the group.

SARJ-97, October 1997

Calibration Phase

- These were not blind calculations. Everybody had a chance to calibrate their codes against experimental measurements.

SARJ-97, October 1997

Calibration Phase (2)

- The early phase of the experiment was particularly useful. This series of plateaux was designed so that the experimental team could check the instruments. They were also useful for checking that the heat loss boundary conditions (shroud conductivity etc) were specified correctly in the codes.

SARJ-97, October 1997

Oxidation Phase

\[ \text{Zr} + 2\text{H}_2\text{O} \rightarrow \text{ZrO}_2 + 2\text{H}_2 \]

The hydrogen mass was
- deduced from measurements in the containment (instrument saturated),
- and confirmed by heat transfer considerations in the steam generator.

SARJ-97, October 1997
Hydrogen Generation (2)

In general the codes predicted too little hydrogen production from the Zircaloy oxidation.

- The fundamental reason for the codes' low hydrogen production was the melting and slumping of cladding material. Oxidation of this slumped material is not modelled in any code except ICARE-2. Other codes assume that molten, dissolved or relocated Zircaloy cannot oxidise.

Liquefaction

It is well known that:

- Zircaloy dissolves fuel and liquifies it at a considerably lower temperature than melting.
- Silver and iron liquify Zircaloy.

Modelling these effects is not easy especially as they compete with the oxidation of cladding. Interaction between $\text{ZrO}_2$ and $\text{UO}_2$ is theoretically only at temperatures (>2800K).

Post Irradiation Examination.

Visual Observation of level 787.3mm

Post Irradiation Examination.

Visual Observation of level 787.3mm

Liquefaction (2)

The degradation seen here in the upper parts of the bundle probably happened during the oxidation phase because this is when the highest temperatures occurred.

There are some indications that there was some steam starvation during this period.

Clad Failure

Thermocouples within the bundle provided good measurements through the oxidation phase. In FPT-1 measurements from ultrasonic thermometers are also available.

The peak temperature marks either:

- complete oxidation of the available Zircaloy
- a slumping of the cladding ("clad failure")

Deciding which of these was responsible is difficult and requires careful analysis.

Clad failure criteria depend on thickness of oxide layer and temperature. A typical example could be:

if oxide thickness $\tau < 300\mu$ at 2300K, clad fails at 2300K
else clad fails at 2500K

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Core Degradation in Phebus

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Core Degradation in Phebus

SARJ-97, October 1997 11/34

Core Degradation in Phebus

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Core Degradation in Phebus
The exact criteria can only be determined by fitting to experiments (such as FPT-0).

Clad Failure (2)

<table>
<thead>
<tr>
<th>User</th>
<th>Code</th>
<th>Criteria</th>
</tr>
</thead>
<tbody>
<tr>
<td>CEA, JRC, CARE2V2</td>
<td>$T &gt; 2300$ and $\delta &lt; 300$</td>
<td></td>
</tr>
<tr>
<td>JAERI</td>
<td></td>
<td>$T &gt; 2500$</td>
</tr>
<tr>
<td>ENEA, AEA</td>
<td>SCDAP/RELAPS</td>
<td>$T &gt; 2300$ and $f_{ox} &lt; 0.6$</td>
</tr>
<tr>
<td>AEA</td>
<td>SCDAP/RELAPS</td>
<td>$T &gt; 2500$ and $f_{ox} &lt; 0.6$</td>
</tr>
<tr>
<td>FZK</td>
<td>SCDAP/RELAPS</td>
<td>$T &gt; 2350$ and $f_{ox} &lt; 0.6$</td>
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<tr>
<td></td>
<td></td>
<td>$\varepsilon &gt; 0.18$</td>
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<tr>
<td>ENEA, AEA</td>
<td>SCDAP/RELAPS</td>
<td>$T &gt; 2400$ and $\delta &lt; 100$</td>
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<td>KAERI</td>
<td>MELCOR</td>
<td>$T &gt; 2500$ and $\delta &lt; 0.6$</td>
</tr>
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<td>KEMA</td>
<td>MELCOR</td>
<td>$T &gt; 2500$ and $\delta &lt; 0.6$</td>
</tr>
<tr>
<td>NUPEC</td>
<td>MELCOR</td>
<td>$T &gt; 2088$ and $\delta &lt; 0.1$</td>
</tr>
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<td>$T &gt; 2980$ and $\delta &lt; 100$</td>
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<td>UPM</td>
<td>MELCOR</td>
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<tr>
<td>CEA</td>
<td>MAAP4</td>
<td>$d_f \geq 1$ and $f_{ox} &lt; 0.675$</td>
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<td></td>
<td></td>
<td>mesh molten, blocked or collapsed</td>
</tr>
</tbody>
</table>

where $T$ (K) is the temperature of the cladding, $T_f$ (K) is the temperature of the fuel, $\delta$ (um) is the oxide layer thickness, $f_{ox}$ is the fraction of the Zircaloy in the oxide layer, $\varepsilon$ is the rupture strain, $d_f$ is the damage fraction.

It is not clear at this stage:
- why clad failure criteria need to be changed for different tests with apparently similar conditions
- why different codes have different default parameters
- whether they are a convenient tuning factor that mask the effect of other, more subtle, parameters.

Clad Failure (3)

In general higher limits for clad failure allow the cladding to:

1) oxidize more and produce more hydrogen. Otherwise it falls to colder places that cannot sustain the oxidation.

2) heat up and produce more fuel liquefaction.

Some codes even have models that allow part of the cladding to remain in place and part to relocate. Again the models are justified a-posteriori by their agreement with experiment.

Previous calculations were worse than these but better agreement with experiment was obtained by using clad failure criteria that kept the cladding in place longer. This meant using a clad failure temperature at least 200K higher than criteria developed for CORA or Phebus-SFD.

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Beginning of Slumping

The next events were at about 14500 seconds when a number of jumps in the thermocouple readings indicated the movement of material.

Events seen by thermocouple in bundle - 14730 seconds. tcw3 (700), tcw10 (400), tcw12, tcw13 (300)

Events seen by thermocouple in bundle - 14730 seconds. tcw3 (700), tcw10 (400), tcw12, tcw13 (300)

Temperature inside shroud at 200mm elevation

Events seen in coolant (inlet and outlet) 15180 seconds

Blockage of the bundle at 15180 seconds.

Formation of Blockage

The standard version of ICARE-2 suggested that all the Zircaloy was oxidised in the central and upper regions. The eutectic temperature of UO2-ZrO2 mixtures (≈2850K) is well above the temperatures predicted so no liquefying was possible. Therefore the rods did not collapse. Subsequently IPSN made a number of calculations that achieved some success.

- manual movement of material
- reduction of exchange areas following degradation
- assuming a more insulating shroud

ATHLET-CD currently only models early phase behaviour.

End-state of Bundle after FPT-9

From the time of the blockage (15180 seconds) till the end of the experiment an hour later the bundle was again in a fairly stable state. The codes submitted to this exercise had difficulty with this part of the exercise.
Molten Pool Behaviour

- Both SCDAP/RELAP5 and MELCOR had problems modelling the molten pool. In SCDAP/RELAP5's case this was because radial heat transfer from the pool is not taken into account.

- MELCOR made such gross errors in the molten pool phase that it was found better to turn off all the degradation models and assume that the rods were intact. Even though this was not the case.

- For these reasons the modelling of molten pools was not really tackled by the calculations reported here. Nevertheless the observed uniformity in the material content of the pool means that modelling should not pose severe problems.

The post-irradiation analysis of the frozen melt shows a rather uniform ceramic melt of around 60 wt%U, 20 wt%Zr and 20 wt%O.

Unreactorlike features

The Phebus-FP experiment requires the analyst to model a number of features that are not present in a reactor. In a reactor the radial heat losses are small important but in Phebus-FP they dominate. It is therefore necessary to model a porous zirconia shroud containing gaps that open and close as the material expands.

One should not condemn a code because it provides inadequate models of these unreactorlike features but, equally, it is difficult to assess the core degradation models of a code if the temperatures are wrong because the heat losses are not modelled correctly.

Unreactorlike features (2)

Obviously ICARE-2 has the most sophisticated models for shroud behaviour because it was a design requirement that ICARE should be able to model Phebus-FP. Nevertheless the temperature comparisons show that most participants were able to reproduce the heat heat losses with sufficient accuracy. There were one or two outlying plots but each of the four codes used was able to provide at least one calculation that was sufficiently accurate.

† Although even ICARE-2 has difficulty modelling all the details of the shroud behaviour.
Unreactorlike features (3)

The stiffeners

Some analysts blamed the extra Zircaloy present in the Phebus bundle for their poor performance in calculating the hydrogen release. This mainly included the Zircaloy springs, upper and lower plugs, absorber guide tube, spacer grids and stiffeners. However some of these materials are present in a reactor (guide tube, spacers) and most analysis managed to model in some way the stiffeners.

Conclusion about Unreactorlike Features

The unreactorlike features cannot be blamed for the codes' failure to reproduce the observed core degradation because
- the temperatures were generally well predicted
- a number of sensitivity calculations showed that even extreme variation in shroud conductivity did not, on its own, produce the desired results.
- Oxidising the extra Zircaloy would not have produced enough hydrogen.

Effect of Control Rod

The codes' assumption that control rod material does not interact with anything else is plainly false. Whether it made a lot of difference is harder to decide. On balance it probably did.

By interacting with the cladding of the inner fuel rods it led to their early collapse.

Effect of Control Rod (2)

This early collapse could have happened either through structural weakening through de-cladding or by lowering the temperature of fuel liquefaction. Unfortunately introducing a model will not be easy.

Map of silver at 57mm

Cut at 57mm showing line L2

Mass fraction of silver along line L2
Fission Product Release

It can be immediately seen that the calculations divide into two camps:

- Those that took no account of the fact that fresh fuel was used. In this case the release was predicted too early.
- Those that used the Petti correlation (Petti, 1991) that had been tuned to match release data from the fresh fuel. In this case the release was predicted too late.

Highly volatile isotopes

<table>
<thead>
<tr>
<th>radionuclide</th>
<th>lower</th>
<th>upper</th>
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</thead>
<tbody>
<tr>
<td>$^{131}$I</td>
<td>85%</td>
<td></td>
</tr>
<tr>
<td>$^{137}$I</td>
<td>5%</td>
<td></td>
</tr>
<tr>
<td>$^{137}$Te</td>
<td>82%</td>
<td></td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>65%</td>
<td></td>
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<tr>
<td>$^{137}$Ba</td>
<td>59%</td>
<td></td>
</tr>
<tr>
<td>$^{137}$Cs</td>
<td>57%</td>
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Volatile isotopes

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<tr>
<th>radionuclide</th>
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<th>upper</th>
</tr>
</thead>
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<td>$^{54}$Mn</td>
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<tr>
<td>$^{88}$Co</td>
<td>30%</td>
<td></td>
</tr>
<tr>
<td>$^{89}$Sr</td>
<td>18%</td>
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</tr>
</tbody>
</table>

Low volatility isotopes

<table>
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<tr>
<th>radionuclide</th>
<th>lower</th>
<th>upper</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{106}$Ru</td>
<td>2.5%</td>
<td></td>
</tr>
<tr>
<td>$^{103}$Ru</td>
<td>1.8%</td>
<td></td>
</tr>
</tbody>
</table>

Fission products deposited before point C not included.

Post-Test Analysis of Bundle

Release of high volatiles from molten pool looks almost complete as expected. But .......

The ability of codes to model FPT-0 (2)

- We were over-confident in our codes' ability to predict this test. Codes had been calibrated under different conditions
  - the electrical heaters in CORA may have provided artificial stability.
  - Most Phebus-SFD tests were run under reducing conditions
  - PBF was not so "oxidising" as Phebus FPT-0.

- The amount of liquefaction obtained under highly oxidizing conditions was unexpected.
- Fission product release from fresh fuel was higher than expected.
- The unreactor-like features in FPT-0 (high heat losses, presence of stiffeners) were not a significant factor in the codes ability to calculate the scenario.
- The highly uniform nature of the molten pool helps our analysis and is a positive sign for reactor calculations.
The ability of codes to model FPT-0 (3)

It is thought that the control rod and its cladding played a significant role in the early degradation process but the evidence was mostly destroyed in the later degradation. Analysis of FPT-1, which stopped earlier, may help us to understand the process better.

And lastly

Although water and natural convection make reactor calculations harder than Phebus calculations the lessons learnt from Phebus will improve our ability to assess the consequences of reactor sequences and the efficacy of accident management strategies.

FPT-1

There is not enough time to discuss all the details of FPT-1 but it was rather similar to FPT-0 except that:
- Pre-irradiated fuel was used
- The experiment was stopped rather earlier.

Even better data is available from this test because the ultrasonic thermometers worked throughout the test - in FPT-0 we only had temperatures in the shroud for most of the high temperature part of the transient.

Initial indications are that the bundle became severely degraded even earlier than FPT-0 - codes such as ICARE-2, RELAP5/SCDAP and MELCOR had to assume a liquefaction temperature of 24500-26000K. This is well below the minimum melting temperature of UO2-SrO2 mixtures.
7.4 Evaluation of Fission Products Release and Transport in the Circuit of PHEBUS FP Test by MACRES code

Yoshishige KAWADA* and Itaru KANEKO**
*Systems Safety Department, NUPEC,
**Nuclear Energy division, Toshiba

ABSTRACT

NUPEC has joined “PHEBUS-FP PROGRAM” in which the first test FPT-0 was performed in 1993 and the second test FPT-1 was also done successfully in 1996. While, almost all of the data in FPT-0 was just before finalization, the data in FPT-1 has been just reported as preliminary data.

In the first test FPT-0, 20 fresh fuels and one CR (control rod of AIC:Ag-In-Cd) were melted within a shroud located in the center of the test core under steam rich atmosphere condition. FP (fission product) generated during 9 days irradiation, prior to the power transient, was released in a test circuit simulating steam generator and containment vessel for PWR reactor. FP and aerosol distributions were measured partly on time and partly in post test analysis. Their chemical forms as well as characteristics were also investigated.

In the second test FPT-1, 20 fuels, irradiated in BR3 reactor up to 23 GWd/t, were melted in a modified shroud (thermal resistance improvement) in the center of the test core. FP already generated in the fuels was released and distributed to the same circuit in FPT-0. The FP and aerosol measurements were done with almost same manner as that in FPT-0.

FP behavior in the FPT-0 circuit was analyzed by MACRES code and was compared with the test results. The region of the primary circuit from upper plenum of the shroud for the bundle to the end of cold leg was divided into 23 sections in MACRES analysis.

The code predicted the chemical provable forms for each fission product of Cs, Iodine, Te, Ag and In respectively. The code predicted well concentration of Iodine, Cs, Ag and In also in the cold leg region, but it did underestimate for Te concentration.
Evaluation of Fission Products Release and Transport in the Circuit of PHEBUS-FP Test by MACRES

SARJ-97
Pacifico Yokohama
Oct. 7th, 1997

Yoshishige KAWADA
Itaru KANEKO

Nupec

Objectives

- Verify of MACRES code through comparison with PHEBUS-FP test FPT0
- Prepare to construct data base of S.A.
- Get hints of modification of the code

Outline

- Objectives
- PHEBUS-FP test
- Measurement in PHEBUS test circuit
- Model of circuit used in MACRES
- Analysis & test results
  - Deposition of fission products
  - Airborne element mass concentration
- Summary

Fig. 1 Test circuit of PHEBUS-FP
Measurements in PHEBUS test circuit

- On-line monitor
  - On-line aerosol monitor
  - On-line maypack
  - Photo-meter
- Post test analysis
  - Sampling of filter, impactor and piece of test circuit
  - Chemical analysis
  - Tomography of test fuel

MACRES

[Mechanistic computer code of Aerosol and gaseous radioactive material behavior in LWR Cooling system for Realistic Estimation of the Source term]

- Aerosol & Gaseous radioactive material behavior
- Core, Pipe, Containment: up to 25 compartments
- Composed by around 200 modules
- Developed by NUPEC under the contract of MITI [Ministry of Institutional Trade and Industry]

Materials treated in MACRES

- Noble gas: Kr, Xe
- Halogen: I, Br
- Alkali: Cs, Ba, Sr
- Carkogen: Te
- Noble: Ru, Mo, Rn, Pn, Tc
- Rare metal: Ce, Sm, Pm, Pr, Nd, La, Y, Zr, Nb
- Fuel materials: U, Zr, Sn
- Control rod mat.: Ag, Cd, In, B
- Structure mat: Fe, Cr
- Atmospheric mat: H2, O2, He, air
Materials from fuels and control rod assumed in the analysis

- Fuels (UO\textsubscript{2}): 6 days irr. \(\rightarrow\) Iodine, Cs, Te, Ba
- C.R.(AIC) \(\rightarrow\) Iodine, Ag, Cd
- Iodine, Cs, Te, Ag, In, UO\textsubscript{2}, Ba
- UO\textsubscript{2} & Ba are not released to the circuit

<table>
<thead>
<tr>
<th>Table 1:</th>
<th>Element</th>
<th>Fuel bundle inventory [kg]</th>
<th>Released masses [kg]</th>
<th>Ratio of released masses to initial inventory [%]</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Iodine</td>
<td>3.6E-05</td>
<td>3.51E-05</td>
<td>97.5</td>
</tr>
<tr>
<td></td>
<td>Cs</td>
<td>1.44E-04</td>
<td>9.74E-05</td>
<td>67.6</td>
</tr>
<tr>
<td></td>
<td>Te</td>
<td>4.7E-05</td>
<td>3.89E-06</td>
<td>8.3</td>
</tr>
<tr>
<td></td>
<td>Ag</td>
<td>4.768E-01</td>
<td>1.94E-01</td>
<td>40.7</td>
</tr>
<tr>
<td></td>
<td>In</td>
<td>8.94E-02</td>
<td>5.97E-02</td>
<td>66.8</td>
</tr>
<tr>
<td></td>
<td>UO\textsubscript{2}</td>
<td>1.05332E+01</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td></td>
<td>Ba</td>
<td>1.49E-04</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

Chemical form of fission product

- Iodine & Cs
  - CsI, I\textsubscript{2}, HI, HOI, CsOH
- Te
  - Te\textsubscript{2}, H\textsubscript{2}Te
- Ag
  - AgI
- In
  - InI

Table 2: Chemical species ratio
Table 2 Chemical species ratio by MACRES for PHEBUS test

<table>
<thead>
<tr>
<th>Element Species</th>
<th>Cs</th>
<th>Iodine</th>
<th>Te</th>
<th>Ag</th>
<th>In</th>
</tr>
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<tbody>
<tr>
<td>Cs</td>
<td>0.000</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>I</td>
<td>-</td>
<td>0.000</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Te</td>
<td>-</td>
<td>-</td>
<td>0.000</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>CsOH</td>
<td>99.765</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>I₂</td>
<td>-</td>
<td>0.498</td>
<td>-</td>
<td>-</td>
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</tr>
<tr>
<td>HI</td>
<td>-</td>
<td>4.845</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>HOI</td>
<td>-</td>
<td>0.000</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>Te₂</td>
<td>-</td>
<td>-</td>
<td>99.336</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>H₂Te</td>
<td>-</td>
<td>-</td>
<td>0.664</td>
<td>-</td>
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</tr>
<tr>
<td>Ag</td>
<td>-</td>
<td>-</td>
<td>100.000</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>In</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>-</td>
<td>99.964</td>
</tr>
<tr>
<td>CsI</td>
<td>9.233</td>
<td>24.494</td>
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<td>-</td>
</tr>
<tr>
<td>AgI</td>
<td>-</td>
<td>1.980</td>
<td>-</td>
<td>0.000</td>
<td>-</td>
</tr>
<tr>
<td>InI</td>
<td>-</td>
<td>68.182</td>
<td>-</td>
<td>-</td>
<td>0.036</td>
</tr>
<tr>
<td>CH₃I</td>
<td>-</td>
<td>0.000</td>
<td>-</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>

Analysis and Test Results

- Deposition of fission product
- Airborne element mass concentration

Comparison of analysis with test results

Iodine Deposition Profile

- Iodine
  - major of species: InI
  - InI aerosol from section No10 [SG1]
  - 68% of released Iodine deposit by InI

Fig. (1/5) Deposition of Iodine predicted by MACRES
Cs
- major of species: CsOH
- Cs aerosol from section No 10 [SG1]
- Vapor of Cs: absorption to wall
- 50% of released Cs deposit in the form of aerosol

Fig. (2/5) Deposition of Cs predicted by MACRES

Te
- major of species: Te₂
- Te₂ aerosol from section No 11 [SG2]
- vapor of Te₂: absorption to wall
- absorption in the form of vapor: 35%
deposition in the form of aerosol: 7%

Fig. (3/5) Deposition of Te predicted by MACRES
Ag, In
- Major of species: Ag, In
- Huge amount of release
- Same behavior: both of Ag & In
- 55% of released materials deposit in the form of aerosol

Silver Deposition Profile

Indium Deposition Profile

Airborne element mass concentration
Comparison of analysis with test results

- FPT-0 data (TG302)
  Airborne FP elements mass concentration at point C & G
- Predict well Cs, Iodine, Ag, and In
- Under-estimate Te:
  Difference of initial condition (i.e. amount of release)
Fig. 3  Airborne Iodine element mass concentration at point G

Fig. 4  Airborne Cs element mass concentration at point G

Fig. 5  Airborne Te element mass concentration at point G

Fig. 6  Airborne Ag element mass concentration at point G
Fig. 7  Airborne In element mass concentration at point G

Table 4 Summary of comparison

<table>
<thead>
<tr>
<th>Initial condition</th>
<th>analysis (experiment)</th>
<th>point C</th>
<th>point G</th>
</tr>
</thead>
<tbody>
<tr>
<td>amount of release</td>
<td>form</td>
<td>airborne element mass concentration</td>
<td>form</td>
</tr>
<tr>
<td>Iodine</td>
<td>good agreement</td>
<td>vapor (aerated)</td>
<td>over-estimate</td>
</tr>
<tr>
<td>Cs</td>
<td>under-estimate</td>
<td>vapor (aerated)</td>
<td>good agreement</td>
</tr>
<tr>
<td>Te</td>
<td>under-estimate (~1/10)</td>
<td>aerosol (aerated)</td>
<td>under-estimate (~1/10)</td>
</tr>
<tr>
<td>Ag</td>
<td>under-estimate</td>
<td>aerosol (aerated)</td>
<td>good agreement</td>
</tr>
<tr>
<td>In</td>
<td>under-estimate</td>
<td>aerosol (aerated)</td>
<td>good agreement</td>
</tr>
</tbody>
</table>

*: ref. experiment data TG302

Summary

- Predict well Iodine, Cs, Ag and In
- Under-estimate Te:
  Difference of initial condition (amount of release)
- Depend on release pattern
The Phebus FP program is a world-wide cooperative project designed to address issues associated with the fission product (FP) releases from prototypic irradiated LWR fuel under severe accident conditions, as well as the FP transport and deposition in the reactor circuit and containment. The first two experiments performed with a 1 meter-long fuel bundle composed of 20 fuel rods and one control rod provided information on FP release from the bundle up to its liquefaction at a relatively low temperature due to material interactions. The next experiment FPT4 will fill the lack of understanding of the fission product release during the late phase of the core degradation. At this stage of the accident, the fuel rod geometry can become rubblized forming a debris bed as observed in TMI-2.

The main objective of the test is to study the releases of low volatile fission products and transuranic elements from a rubble bed and the fuel vaporization. A secondary objective is to study the physical transition from debris bed to molten pool.

The in-pile test section is composed of a debris bed enclosed in a Zircaloy canister and layers of melt barriers and thermal insulation. The rubble bed consists of two distinct zones. The upper zone is a mixture of 3.2 Kg of PWR fuel pellet fragments with an average burn-up of around 33 GWd/tU mixed with 0.8 Kg of oxidized Zircaloy cladding shards. The fuel-cladding debris mixture sits on a 12 cm-thick layer of depleted UO$_2$. Six filters are located above the debris bed. They will be operated sequentially and will provide information on releases for various debris bed temperatures. The debris bed instrumentation is composed of W/Re thermocouples and ultrasonic thermometers.

The test package will be subjected to neutronic heating in the Phebus reactor that consists of a set of alternating transient and steady-state periods designed to take the debris bed in steps up to its melting temperature and melt half of the debris bed to form a molten pool. To transport the FP released from the debris bed region to the filters, the debris bed is swept by a gas flow composed of 80\% steam and 20\% hydrogen.

Pretest calculations of the debris bed behavior and FP releases are performed by the various Phebus partners using different codes: DEBRIS, MERIS and VICTORIA codes have been used at the Sandia National Laboratories. IPSN used the ICARE2 V3 Mod0 code to calculate the thermal behavior of the debris bed, the melt progression after the liquefaction onset, the molten pool formation and the temperature field in the test package. This code is developed and funded by IPSN to simulate reactor core degradation under severe accident conditions.

In the ICARE2 reference calculation, a bidimensional porous media model is used to describe the gas and liquid flows in the rubble bed. Heat generation in the fuel region, conduction and radiation heat transfer within the debris bed and the test section, radiative heat transfer between the top of the debris bed to the steam and the upper structures, convective heat transfer associated to the flowing gas are accounted for. After the pool formation, radiative heat transfer in the formed cavity above the pool is taken into account, as well as the enhancement of the heat transport due to buoyancy driven mixing in the molten pool. These models have been assessed against ACRR experiments. The ICARE2 reference and the sensitivity calculations show that the objective of the test can be achieved despite the material properties and model uncertainties.
PHEBUS FPT4:

TEST DESCRIPTION AND PRETEST CALCULATIONS
F. Serre, J-C. Crestia, S. Ederli*, U. Bieder*, F. Fichot

OUTLINE
- TEST OBJECTIVES
- TEST PACKAGE
- TEST SCENARIO
- ICARE2 PRETEST CALCULATIONS
- CONCLUSIONS

* ENEA, * NIS

TEST OBJECTIVES

STUDY OF:
- VOLATILE FISSION PRODUCTS RELEASES FROM A DEBRIS BED
- LOW VOLATILE FISSION PRODUCTS, ACTINIDES AND URANIUM RELEASES FROM A DEBRIS BED
- PHYSICS OF A DEBRIS BED
- TRANSITION DEBRIS BED -> MOLTEN POOL
- FISSION PRODUCT RELEASES FROM A MOLTEN POOL

REQUIREMENTS:
- TIGHT FISSION PRODUCT MASS BALANCE
- GOOD DEBRIS BED TEMPERATURE MEASUREMENT

TEST PACKAGE

- 6 high capacity sequential filters
- External Water Cooling
- Neutronic Heating (Driver Core)
- Debris Bed
- Inlet Gas Injection (Reactor Conditions, and FP Transport)

TEST SECTION

- 3.2 kg irradiated UO₂ (Particle size: 4mm)
- 0.8 kg ZrO₂ Shard
- Depleted UO₂

Debris Bed Instrumentation:
- 3 Ultrasonic Thermometers
- 6 W/Re Thermocouples

Shroud Instrumentation:
- W/Re, Pt/Rh and Cr/Al Thermocouples
**TEST SCENARIO**

*Calibration phase + 3 release plateaux:*

- 2200K: Volatile Fission Product Releases
- 2700K: Non-Volatile Fission Product Releases
- Fission Product Releases from Molten Pool

*Inlet Flow Injection:*

- 0.5g.s⁻¹, 80yo % H₂O, 20yo % H₂

---

**ICARE2 PRETEST CALCULATION:**

**NODALIZATION**

---

**ICARE2 PRETEST CALCULATION:**

**MAIN PHYSICAL MODELS**

- Heat Transfer in the Debris Bed:
  - IMURA-YAGI Correlation
  (Assessed against ACHR MP and DC Series)

- Debris Bed Permeability:
  - CARMAAN Correlation

- Capillary Forces:
  - LEVERETT Function

- 2D Gas Flow Model

- Steam Dissociation

- Radiative Heat Transfers:
  Molten Pool <-> Cavity Walls
  Debris Bed Top <-> Upper Structures
  Steam <-> Walls

- Heat Transfer Molten Pool/Wall:
  - MAYINGER Correlation

---

**ICARE2 RESULTS:**

**DEBRIS BED TEMPERATURES**

- Temperature Objectives Fulfilled
ICARE2 RESULTS:
AXIAL TEMPERATURE PROFILES

- Steam Convection Effect at the Bottom
- At the Top, Cool-Down By Radiative H.T.
- Same Temp. Profiles in Debris Bed and Shroud

360
340
320
300
280
260
240
220
200
180
160
140
120
100
80
60
40
20
0
Z (mm)

FPT4 - REFERENCE CASE
T (K)

ICARE2 RESULTS:
MOLTEN POOL FORMATION

- Downward Pool Progression

FPT4 - PLATEAU P7

ICARE2 RESULTS:
GAS VELOCITY FIELD

- Gas Flow Diversion at the Pool Level

FPT4 - REFERENCE CASE

ICARE2 RESULTS:
AXIAL TEMPERATURE PROFILES

- Axial Temperature Profile Deformation after Pool-Wall Contact
ICARE2 Calculations coupled with FP Release Studies (ELSA code) show that the Proposed Scenario leads to:

- Significant volatile release at temperatures where intact fuel has been well-explored (50% of the initial FP inventory)
- Release at relatively unexplored solid fuel temperatures (actinides, lanthanides)
- Progression to molten pool
- Test conduct possible with test section instrumentation
- Similar Results obtained by SNL with DEBRIS, MERIS and VICTORIA codes
8. Session VI

FCI Experiment

Chairperson: A. Inoue (Tokyo Inst. Tech.)
Co-chairperson: K. Moriyama (JAERI)
8.1 COTELS Fuel Coolant Interaction Tests of UO2 Debris Dropping into Water Pool

Masami Kato, Hideo Nagasaka and Isao Sakaki
Systems Safety Department, NUPEC

ABSTRACT

COTELS project aims at the investigation of ex-vessel debris coolability using a mixture of UO2, Zr, ZrO2 and stainless steel as simulant debris. The project consists of three types of test: Test A to focus on the investigation of a fuel coolant interaction (FCI) phenomena when a molten debris falls into coolant pool on the containment floor, Test B and Test C to focus on FCI and molten core concrete interaction (MCCI), respectively, when coolant is injected onto the molten debris as a severe accident management. This paper presents the results of the first test series, Test A.

The test facility mainly consists of an electrical melting furnace (EMF) and a test vessel which receives the falling debris. EMF has a capability to melt 60 kg of UO2 mixture by induction heating and has a debris falling device to make a 5 cm diameter hole at the bottom of crucible instantaneously. The vessel is 0.8 m inner diameter and about 2.5 m height, and the design pressure is 10 MPa. A concrete floor of 5 cm thickness is used on the melt catcher in order to correctly simulate the effect of the interaction among debris, concrete and water.

The observation of the molten debris jet behavior falling from EMF by high speed cameras showed that it was a continuous jet flow and little break-up to be occurred during falling process in air space. The debris jet velocity at pool surface was estimated to be about 6 m/s.

In Test A, 60 kg of UO2 mixture dropped into water pool under around 0.2 MPa condition. The pool depth ranged from 0.4 m to 0.9 m and water subcooling was from zero to 20 K. Major findings from the experiments are as follows: (1) No pressure spike typical of violent steam explosion was observed in all the experiments. (2) The size of fragmented debris particles was large and ranged from 0.5 to 7 mm. (3) Solidified debris was easily removed from the concrete floor and no ablation was observed on the concrete surface.
COTELS Fuel Coolant Interaction Tests of UO2 Debris Dropping into Water Pool

SARJ-97
October 7, 1997 Pacifico YOKOHAMA
M.Kato, H.Nagasaka and I.Sakaki
Systems Safety Department

Nupec

Features of COTELS

- Ex-Vessel debris cooling test (COTELS project) corresponding to Phase II AM for low pressure vessel failure sequence in collaboration with NNC of republic of Kazakhstan
- FCI and MCCI experiments with conditions derived for the most plausible SA scenarios in LWR's simulating,
  - Real molten debris (UO2/ZrO2/Zr/Steel)
  - Falling debris jet diameter
  - Ambient conditions (pressure, temperature, noncondensible gas partial pressure)
  - Decay heat (MCCI only)
  - Pool water temperature, depth (FCI only)
  - Ratio of accumulated debris diameter to thickness (MCCI only)
  - Flow rate of injecting water

COTELS Test Program

[1] Test 01
- Objective ••• Observation of flow mode of falling debris jet and evaluation of debris dispersion characteristics without FCI
- Test parameters
  - Debris composition and falling velocity
[2] Test A
- Objective ••• Simulation of FCI for debris dropping into pool
- Test parameters
  - Debris mass and composition, and falling velocity
  - Water pool depth, volume and temperature
  - Nitrogen partial pressure
[3] Test B
- Objective ••• Simulation of FCI for water injecting onto debris
- Test parameters
  - Debris composition and falling velocity
  - Geometry of concrete trap
  - Induction heater power
  - Flow rate and flow injection mode
- Objective ••• Simulation of MCCI with overlying water pool
- Test parameters ••• same as Test B

Contents

- Features of COTELS Project
- Test 01: Falling debris jet behavior observation test
- Test A: Fuel coolant interaction test of debris dropping into pool
- Summary
Test Facility of Test 01

- Pressurization of Electrical Melting Furnace (EMF) performance
- Three high speed cameras

Test Conditions of Test 01-1

<table>
<thead>
<tr>
<th>Test Parameter</th>
<th>Conditions</th>
</tr>
</thead>
<tbody>
<tr>
<td>Composition of Debris, % wt</td>
<td>55</td>
</tr>
<tr>
<td>• Uranium dioxide</td>
<td>15</td>
</tr>
<tr>
<td>• Stainless steel</td>
<td>5</td>
</tr>
<tr>
<td>• Zirconium oxide</td>
<td>25</td>
</tr>
<tr>
<td>• Zirconium</td>
<td></td>
</tr>
<tr>
<td>Debris Mass, kg</td>
<td>60</td>
</tr>
<tr>
<td>Integrated heat amount</td>
<td>540</td>
</tr>
<tr>
<td>supplied to EMF (MJ)</td>
<td></td>
</tr>
<tr>
<td>Melt temperature (°C)</td>
<td>2950 (*)</td>
</tr>
<tr>
<td>Medium composition of MR</td>
<td>Ar</td>
</tr>
<tr>
<td>Initial pressure of MR (MPa)</td>
<td>0.1</td>
</tr>
<tr>
<td>Pressure difference between EMF and MR</td>
<td>none</td>
</tr>
<tr>
<td>Diameter of Debris jet (cm)</td>
<td>5</td>
</tr>
</tbody>
</table>

*: Estimated

Design Specifications of Electrical Melting Furnace

- Induction heating: 500kW/2.4kHz
- Max. temperature: 3100 K
- Max. molten mass: 60 kg
- Graphite crucible with TaC sheet
- Multi-layer thermal insulation
- Pressurization function to adjust debris falling velocity
- Graphite dropping plug destroyed by pneumatic mechanism
- Re-closing shutter
Inspection of EMF and Dropped Debris after Test 01-1

- 60 kg of UO₂ debris (UO₂:55%, Zr:25%, ZrO₂:15%, SS:15%) completely melted and dropped from EMF
- Integrity of graphite crucible confirmed
- Solidified stainless steel droplets observed both inside and outside of crucible
- About 2 kg SS (total SS mass = 9 kg) vaporized, condensed and solidified

Dropped Debris
- Scattering of 2 kg debris outside concrete trap
- Almost all dropped debris on concrete floor observed to be continuous
- The continuous debris observed to be very rough with high porosity (~0.4)

Inspection of concrete trap side wall after Test 01-1

- Solidified debris not stuck on concrete trap side wall as well
- Significant ablation not observed on concrete trap side wall as well

Concrete trap side wall surface after removing solidified debris

Inspection of concrete trap base after Test 01-1

- Solidified debris not stuck on concrete trap floor
- No crack observed on concrete surface
- Significant ablation not observed on concrete trap surface
- No damage of thermocouples embedded in concrete trap

Concrete trap surface after removing solidified debris

Analysis of Solidified Debris Composition

<table>
<thead>
<tr>
<th>Element</th>
<th>Cr</th>
<th>Fe</th>
<th>Ni</th>
<th>Zr</th>
<th>U</th>
</tr>
</thead>
<tbody>
<tr>
<td>Upper layer of accumulated debris</td>
<td>10</td>
<td>30</td>
<td>40</td>
<td>60</td>
<td>70</td>
</tr>
<tr>
<td>Middle layer of accumulated debris</td>
<td>20</td>
<td>50</td>
<td>60</td>
<td>80</td>
<td>90</td>
</tr>
</tbody>
</table>

Element Concentration (%)
Test Facility of Test A

- Concrete floor simulating wettability and surface boiling characteristics
- Adjustable pool volume for a given pool depth

Test Parameters of Test A

- Debris Composition: Three Types
  1. UO₂: 55%, Zr: 25%, ZrO₂: 5%, SS: 25%
  2. UO₂: 78%, ZrO₂: 17%, SS: 5%
  3. SS: 100%
- Debris Mass: 30 kg, 60 kg
- Pool Depth: 0.4 m, 0.9 m
- Pool Volume: 0.4 m³, 1.0 m³
- Vessel Pressure: ~0.2 MPa
- Pool Water Temperature: Nearly saturated ~20 K Subcooling
- Vessel Air Space: Steam: 100%, Steam: 75% + N₂: 25%

Pressure and Temperature Response of EMF and Vessel

Test A4: Debris composition (UO₂: 55%, SS: 15%, ZrO₂: 5%, Zr: 25%) dropped into saturated water pool of 40 cm depth.

Pressure Response of Test Vessel (Test A4)

- Evidence of no steam explosion
  - No pressure difference between water and air space
  - No pressure spike observed
  - Large size fragmented debris ranged 0.5 to 7 mm
- Pressure transient characteristics
  - Initial rapid pressure increase due to heat transfer from fragmented debris in water pool (region A)
  - Pressure suppression due to heat-up of subcooled water pool (region B)
  - Gradual increase due to heat transfer from debris bed accumulated on melt catcher (region C)
Dispersed Solidified Debris on Melt Catcher (Test A4)

• Almost all debris fragmented due to FCI different from Test 01
• Diameter of fragmented debris ranged ~0.5 mm to ~7 mm (Implication of no steam explosion)
• About 6.5 kg/m³ average density of debris

Observation of Concrete Surface of Melt Catcher after Test A4

• Solidified debris very easily removed from concrete surface, suggesting the existence of water between the bottom of debris and concrete surface
• No damage observed on concrete surface and almost new (the concrete catcher can be used repeatedly)

Summary

• Test 01 and Test A of COTELS were successfully conducted.
• Result of Test 01 showed,
  - The EMF melted UO2 debris completely and dropped instantaneously
  - Falling debris was continuous jet
• Result of Test A showed,
  - No violent steam explosion was observed
  - Almost all debris was fragmented and concrete surface was well cooled
  - No ablation was observed on concrete floor
  - Detailed analysis is in progress
8.2 The Effect of Coolant Jet Subcooling on the Coolant Injection Mode of Vapor Explosions

Hyun Sun Park, Norihiro Yamano, Yu Maruyama, Kiyofumi Moriyama, Yanhua Yang and Jun Sugimoto
Severe Accident Research Laboratory, Japan Atomic Energy Research Institute
(Phone) +81-029-282-5854, (Fax) +81-029-282-5570

Abstract

The potential of an energetic fuel-coolant interaction (FCI) in light water reactors has been one of primary safety concerns due to its mechanical energy release by dynamic pressures and expansion energetics. The FCI energetics strongly rely on the initial mixing process between molten fuel and coolant. Since the mixing processes are also influenced by the contact geometries between the molten fuel and the coolant, it is of a great importance to investigate the energetics in terms of the contact modes. Comparing with a pouring mode, however, studies on stratified and injection modes of FCIs are limited. Hence, the MUSE (MUIti-configurations in Steam Explosions) tests were initiated at JAERI to study the FCIs in various contact modes by precisely measuring the FCI energetics.

The MUSE facility located inside the ALPHA vessel consists of the interaction tube, expansion tube, melt generator, and injection system. The interaction and expansion tubes with 108 mm ID and 1 and 3.6 m high, respectively. Maximum 5 kg of the molten thermite can be generated in the melt generator located at the bottom of the interaction tube and water from the injection system is directly injected onto the melt. FCI energetics is evaluated by measuring of the piston positions at the expansion tube detected by a series of coils. Pressures and temperatures during the explosion and expansion phases of FCIs are measured. For investigating the coolant subcooling effect on FCIs, five out of seven tests were successfully conducted by varying the water temperature from 55°C to a near saturation, molten thermite mass at 1.4 and 2.8 kg and injection pressure of 0.3 MPa.

The results from the CI002 to CI007 tests clearly showed that the measured mechanical energies of the FCIs increased with the water subcooling increases. The resulted mechanical energies were ranged from a few tenth to several hundreds of Joules. The conversion ratios were obtained by assuming the debris masses participated in the interactions. Debris masses of each tests were chosen by considering the debris sizes of less than 2 mm and 0.2 mm, respectively. These results also showed that the conversion ratios increased with the coolant subcooling increase. The debris mass distributions showed that the mass mean and Sauder mean diameters tended to decrease with the water subcooling increase. They also imply that energetic interactions occurred with smaller debris. It has been recognized that the mixing conditions, especially a coolant jet penetration depth, established by the coolant injection, are key parameters to quantify the FCI energetics in this mode of contact. A computational analyses using a Moving Particle Semi-implicit Scheme (MPS) and a Cubic Interpolated Propagation (CIP) method on the jet penetration behaviors were initiated. The preliminary analysis with MPS method showed that the initial penetration depth of the lighter density liquid in the heavier density liquid, density ratio of about 3.4, was small in an isothermal case.

The MUSE tests are planning to investigate the effects of not only the coolant subcooling, but a jet dynamics and axial constraint on FCI energetics. A series of visualization tests is also under preparation to identify the water jet penetration behaviors into a molten fuel in terms of a jet subcooling, diameter and velocity. These visual data will be used for comparing and verifying the computational analyses.
The Effect of Coolant Jet Subcooling on the Coolant Injection Mode of Vapor Explosions

Presented by
Park, Hyun Sun
and collaborated with
Yamano, N., Maruyama, Y., Moriyama, K., Yang, Y., and Sugimoto, J.

Severe Accident Research Laboratory
Japan Atomic Energy Research Institute

The MUSE project was originated to investigate the likelihood of FCIs in the PHEBUS-FP test facility.

A hypothetical scenario at the facility may establish the stratified contact geometry as water is injected onto a molten test fuel bundle through a breached hole on the test-section.

The scope of the project, however, was expanded to investigate not only a stratified mode but also other contact modes; melt and coolant injection modes with the precise measurement of the FCI energetics.
Objectives

- **MUSE (MUIti-configuration in Steam Explosions)**
  - To verify any possibility of vapor explosions in the PHEBUS-FP test facility.
  - To investigate the vapor explosion phenomena by precisely measuring the energetics of vapor explosions in various contact geometries.
  - To build up the database to verify currently available mechanistic models; e.g., JASMINE (JAERI), etc.

Experimental Program - MUSE Facility

- **Design Specifications**
  - Consistent with other facilities, (KROTOS, WFC1, ZREX), in a geometrical scale and measurement techniques.
  - Flexible to investigate the effect of various geometrical contact modes of FCIs with a minor modification.
    - 108 mm ID
    - 1m long Interaction Tube
    - 3.6 m long Expansion Tube

MUSE - Facility

- **Work measurement**
  - Set of coils wound outside the expansion tube
  - Magnet embedded piston
  - \( CR = \frac{W_{\text{exp}}}{E_i} \sim \frac{M_{E_p}}{E_i} \)
- **Others**
  - Pressures/Temperatures
  - Visual observation
  - Post-test debris distribution

MUSE - Experiments (CI and ST)

- **The coolant injection (CI) mode test**
  - A total of 10 tests have been conducted to investigate the effect of coolant jet subcooling, melt mass (depth) and axial constraint on FCI energetics.
  - Seven out of ten tests were successfully conducted and at present the results of the first five successful tests are presented.

- **The stratified mode (ST) test**
  - The first test was recently conducted.
**MUSE - CI Series: Initial Conditions**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>CI001</th>
<th>CI002</th>
<th>CI005</th>
<th>CI006</th>
<th>CI007</th>
</tr>
</thead>
<tbody>
<tr>
<td>Melt (Thermite)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>( M_i ) kg</td>
<td>1.4</td>
<td>1.4</td>
<td>1.4</td>
<td>1.4</td>
<td>2.8</td>
</tr>
<tr>
<td>( H_f ) mm</td>
<td>50</td>
<td>50</td>
<td>50</td>
<td>50</td>
<td>100</td>
</tr>
<tr>
<td>T K</td>
<td>2800</td>
<td>2800</td>
<td>2800</td>
<td>2800</td>
<td>2800</td>
</tr>
<tr>
<td>Melt (Water)</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>( dT_{\text{th}} ) K</td>
<td>45</td>
<td>42</td>
<td>12</td>
<td>2</td>
<td>12</td>
</tr>
<tr>
<td>D mm</td>
<td>&gt;23.2</td>
<td>23.2</td>
<td>23.2</td>
<td>23.2</td>
<td>23.2</td>
</tr>
<tr>
<td>dP MPa</td>
<td>0.3</td>
<td>0.3</td>
<td>0.3</td>
<td>0.3</td>
<td>0.3</td>
</tr>
<tr>
<td>Injection</td>
<td>Spray</td>
<td>Jet</td>
<td>Jet</td>
<td>Jet</td>
<td>Jet</td>
</tr>
<tr>
<td>Test section</td>
<td>( M_s ) kg</td>
<td>0.7</td>
<td>0.7</td>
<td>0.7</td>
<td>0.7</td>
</tr>
</tbody>
</table>

---

**MUSE - CI Series: Experimental Procedure**

- Thermite Ignition
- \( H_2 \) Gas Close
- Injection Valve Open
- Injection Valve Close

---

**MUSE - CI Series: Pressure History (CI002)**

**MUSE - CI Series: Temperature History (CI002)**
Measured mechanical energies increase with the water jet subcooling increase.

Similar mechanical energies measured in two different melt mass similar coolant jet penetration during the injection.

Lower energetics for the spray injection test.

- Melt mass participated, \( m_r \), in this mode is not clear in the coolant injection mode. Two assumptions for participating melt mass are:
  - Debris with a size less than 2 mm.
  - Debris with a size less than 0.2 mm.

In the CI002 test, no debris catcher was installed – debris escaped through the expansion tube (mostly small in size) were not completely recovered.

The largest mean diameters for the spray injection test.

Instead of the CI002 test, the mean diameters for the CI009 test (same coolant subcooling but higher axial constraint) were used.

Mean diameters decrease with the coolant subcooling increase.
Analysis - AMUSE

- The high speed coolant jet penetration into the melt
  - Moving Particle Semi-implicit (MPS) method
  - Cubic Interpolated Propagation (CIP) method
  - coolant jet penetration depth in terms of a coolant jet speed,
    a jet temperature, a jet diameter and so on.
  - entrained melt particle during the injection
  - evaporation processes

- The MC3D code is planned to use for investigating the stratified mode of FCIs.

- The behaviors of high speed melt jet injection into the coolant are also under investigation using TEXAS code.

MPS Analysis

- The MPS method proposed by Koshizuka et al. (1996 NED).
  - A continuum is represented by moving particles (melt, liquid and vapor).
  - Particle interactions are restricted by weight functions.
  - Simulation of the multiphase, multicomponent flow behaviors without significant numerical diffusion encountered in most FDM schemes (1996 NED, 1997 CSNI-FCI Meeting).

- The behaviors of the coolant jet penetration into the melt (CI series)
  - Collaboration with Prof. Koshizuka and Mr. Ikeda in the University of Tokyo.
  - Preliminary results.

MPS Analysis (...continued)

- The CI series test geometry.
- The calculation conditions;

<table>
<thead>
<tr>
<th>Melt Material</th>
<th>Al₂O₃</th>
</tr>
</thead>
<tbody>
<tr>
<td>Temperature</td>
<td>2800 K</td>
</tr>
<tr>
<td>Density</td>
<td>3400 kg/m³</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Coolant Material</th>
<th>Water</th>
</tr>
</thead>
<tbody>
<tr>
<td>Temperature</td>
<td>300,373 K</td>
</tr>
<tr>
<td>Velocity</td>
<td>8 m/s</td>
</tr>
</tbody>
</table>

- Two evaporation models
  - normal evaporation model.
  - spontaneous nucleation model.

Water Jet Impingement onto the Alumina Melt (Isothermal Case);
\[ \rho_s = 1000 \text{ kg/m}^3 \text{ and } \rho_m = 3400 \text{ kg/m}^3 \]
Conclusions

- The MUSE facility has been successfully verified through a total of ten tests in the CI series.
- The measured mechanical energy of the slug increases as the subcooling increases.
- The qualitative values of the conversion ratios depend on the melt mass participated. However, the trend of the ratios are similar to the one of the mechanical energies.
- Two tests with a subcooling of 12K produced similar energetics even if the melt mass was different (factor of two).
- MPS calculation showed that it was difficult for the water jet to penetrate deeply into the melt to provide appropriate mixing conditions in this type of contact mode.
- Further investigation, improvement and verification of the analysis are needed.

Future Plans

- Studies on the effects of the axial constraint, and the jet dynamics, e.g., a jet speed, are in progress.
- A series of visualization tests to observe the behavior of the water jet penetration into higher density and/or temperature liquids (such as Flourina, wood's metal, Zinc and Tin) is in preparation. It will provide data for; first, identifying the mixing behavior during the CI series of the MUSE tests and second, verifying our computational analysis.
- The facility modification for the melt injection test is also in preparation.
8.3 Deformation and Fragmentation of Molten Zn and Molten Al at Near-Melting Points

Ken-ichiro SUGIYAMA, Ken-ichi MATSUBA, and Tsuyoshi YAMADA

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North 13 West 8, North Ward, Sapporo 060, Japan
Phone & Fax: 81-11-706-6663

The purpose of the present study is to obtain the available information on triggering stage of a vapor explosion. Many researchers believe that crusted melts can not effectively cause intensive thermal interaction resulting in vapor explosion. However, if thermal interaction locally occurs in partially crusted melts, there is a possibility that high pressure, which is capable of spontaneous triggering, is locally produced. We already reported this phenomenon in SARJ-95 and SARJ-96.

We carried out a series of experiments of crusted molten-zinc and molten-aluminum in the present study. The purpose of the present study is to observe the possibility of entrapment or entrainment of water inside the crusted melts. We dropped 20 grams (or 100 grams) of molten-zinc and molten-aluminum at near-melting points into a highly subcooled water. We observed behaviors of deformation and fragmentation of crusted molten-zinc and molten-aluminum by using a high-speed video camera with 400 pictures per second.

The results showed the molten zinc forms a pot-like crust with a open mouth during falling through the water. When a pressure wave hits the crusted molten-zinc, the melt remaining inside effectively interacts with water entrained through the open mouth. This thermal interaction is basically same with spontaneous triggering of molten-zinc in aqueous solutions of salts reported in SARJ-95 except that water is forcedly entrained.

The results showed that molten aluminum forms a closed crust without a mouth in many cases. Some crusted molten-aluminums spontaneously fragment into pieces during falling through the water or when hitting the bottom. The behavior of crusted molten-aluminum appears to be entirely different from that of crusted molten-zinc in aqueous solutions of salts reported in SARJ-95. Although it has not been clear in the present study why molten aluminum, which is used as a composition of fuel element in research reactors, spontaneously fragments into pieces, it is likely that water hydrodynamically entrapped inside the crust produces a high pressure resulting in fragmentation.

DEFORMATION AND FRAGMENTATION OF MOLTEN Zn AND MOLTEN Al AT NEAR-MELTING POINTS

K. Sugiyama, K. Matsuba and T. Yamada
Dep. of Nucl. Eng., Hokkaido Univ.
Sapporo, 060, Japan

BACKGROUND
We are interested in spontaneous trigger in vapor explosion in high temperature melts.
What thermal-hydraulic mechanisms can cause spontaneous triggering in crusted high-temperature melts?
We have been carrying out a series of experiments below minimum heat flux points.
We have already reported that spontaneous triggering in crusted molten Sn or molten Zn occurs due to entainment or entrapment originating in organized motion.

CONTENTS
1. Explaining background of the present study.
2. Reporting deformation of molten Zn falling through the water and forced fragmentation of crusted molten-Zn.
3. Reporting deformation of molten Al falling through the water and spontaneous fragmentation of crusted molten-Al.

CONCLUSIONS
1. Molten Zn forms a pot-like crust with a open mouth in many cases. When a pressure wave hits the crusted molten-Zn, the melt remaining inside effectively interacts with water forcedly entrained through the open mouth.
2. Molten Al forms a closed crust without a mouth in many cases. During falling through the water or after hitting the bottom, the crusted molten-Al spontaneously fragments into pieces.
Coalescence of molten Zn falling through water

Deformation of molten Zn falling through water

Typical appearance of solidified Zn (20 g)
Partial fragmentation of crusted molten Zn due to shock wave

Appearance (left and center) and cross sectional view (right) of solidified Zn (100 g)

Cross sectional view of conical and pot-like crusted Zn which have fragments inside (~20 g)

Appearance of fragmented Zn with thin crust
Typical appearance of solidified Al (20 g)

Cross sectional view of solidified Al

Deformation of molten Al drop falling through water

Coalescence of two molten Al drops falling through water
Appearance of partially fragmented Al (20 g)
9. Session VII

FP Source Term

Chairperson: M. Firnhaber (GRS)
Co-chairperson: A. Watanabe (NUPEC)
9.1 Status of VEGA Fission Product Release Experiment

A. Hidaka, T. Nakamura, Y Harada and J. Sugimoto (JAERI)

ABSTRACT

An experimental program, VEGA (Verification Experiments of Gas/Aerosol Release), was initiated at Japan Atomic Energy Research Institute (JAERI) to study the fission product release behavior from LWR fuels irradiated to high burnups in Japanese power reactors. In the program, fission product release at high temperatures up to 3,000 °C under high pressure of up to 1.0 MPa will be investigated. Re-irradiation of the fuel will be conducted using the Nuclear Safety Research Reactor (NSRR) to generate short life fission products. The fission product transport behavior will also be studied using thermal gradient tubes (TGTs) under well characterized flowing steam/hydrogen/helium atmosphere monitored by on-line oxygen/hydrogen sensors.

Development of the test facility to realize the above mentioned conditions is in progress. A mock-up induction furnace has reached to high temperature of 3,000 °K under an inert atmosphere. Progress of thorium component development to sustain the test fuel under oxidizing and reducing atmosphere at high temperatures in the furnace is also reported. Characterization tests of solid electrolyte oxygen/hydrogen sensors made of Yttria stabilized Zirconia (YSZ) were conducted. Outline and status of the VEGA project is presented.
Status of VEGA Fission Product Release Experiment

A. Hidaka, T. Nakamura, Y. Harada and J. Sugimoto
Japan Atomic Energy Research Institute

Presented at Workshop on Severe Accident Research
Held in Japan (SARJ-97)
October 6-8, 1997, Yokohama, Japan

Objectives
• To obtain FP release data from fuel under severe accident conditions
• To investigate fuel liquefaction process and FP behavior in reactor core and coolant system

Special Targets
• FP release from high temperature fuel (including debris bed)
• Effect of pressure on FP release
• Non-volatile/short-life FP release

Introduction
• Large uncertainties in FP gas/aerosol release from fuel under severe accident conditions
• VEGA (Verification Experiments of Gas / Aerosol Release) program at JAERI to investigate release of FP including non-volatile or short-life radionuclides from Japanese LWR fuel at ~3000°C under high pressure (1.0MPa) condition

Recent Change of Design
• Addition of Thermal Gradient Tube (TGT)
  - 3TGTs (60cm in length, 750-200°C)
  - Temperature control by heater and natural convection
  - Maximum gas flow rate ; 5N ℓ /min

Merits of TGT addition
• Easy evaluation for deposition mass at upstream of main filter
• Most of CsI trapped by main filter
• Possible identification of chemical form by analyzing chemical element deposited onto TGT

• Addition of TGT resulted in movement of shield box from inside to outside of hot cell.

• Susceptor in induction furnace ; W → Graphite
  - W susceptor ; Weakness for integrity under high temperature
Experimental Conditions

- **Test sample**: 6cm long, 100g
  - BWR Tsuruga Unit 1: 26GWd/tU
  - PWR Mihama Unit 2: 39GWd/tU
  - Ohi Unit 1, 2: ~50GWd/tU
  - TMI-2 debris sample
  - PHEBUS/FPT1 RCS sample
- Re-irradiation of test fuel using NSRR to accumulate short-life radionuclides
- **Max, temperature**: 3000°C
  (Isothermal induction heating in ZrO₂, ThO₂, or W tube furnace)

- **Atmospheric pressure**: 0.1~1.0MPa
  - To investigate effect of pressure on gaseous FP diffusion at fuel open pore surface of grain boundary
- **Carrier gas (Max. flow rate**: 5N ℓ /m)
  - Steam (+ Hydrogen)
  - Hydrogen
  - Air
  - Helium

Gaseous FP Diffusion

- Affected by pressure

- UO₂ grain
Measurement

- On-line gamma measurement
  - Filters or Impactors: Cs-137, I-131, Ba-140 (La-140)
  - Charcoal traps: I-131
  - Cooled charcoal: Kr-85, Xe-133
- 3 Thermal Gradient Tubes (750-200°C)
- On-line oxygen/hydrogen measurement
- Off-line gamma spectrometry
- SEM/EPMA, SIMA
- Metallography
- ICP-MS (for solution)

Preliminary Test Matrix

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<tr>
<td>Thoria Development</td>
<td>Fabrication &amp; Thermal Property Study</td>
<td></td>
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<tr>
<td>Heating Test</td>
<td>2400°C (0.1MPa)</td>
<td>2800°C (0.1MPa)</td>
<td>3000°C (0.1MPa)</td>
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</tr>
</tbody>
</table>
  - In Steam or Helium |         |         |         |         |
  - In Hydrogen        |         |         |         |         |
  - In Air             |         |         |         |         |
| Others               |         |         |         |         |

Furnace Heating Tests

- Maximum Temperature of 3000°C has reached with a graphite susceptor furnace.
- Pressure Effect is examined.
Thoria Tube Development

- Powder Preparation (finished)
- Pellet Fabrication Tests
  - Pressing (finished)
  - Slip Casting (finished)
- Tube Fabrication Tests
  - 10cm Tube (finished)
  - Porous Thoria (on going)
  - 30cm Tube (on going)
- Thermal and Mechanical Property Test (planned in Jan./97)

Centrifugal Slip Casting Technique

Flow Chart

- Powder + Water + Refractory
- Slip Mixing
- Centrifugal Casting
- Drying in Mold
- Burn Out
- Sintering
- Casting
- Product

Characteristics

- Sintered at 1600°C in air
- Density of Original Powder vs. Ball Milled Powder
- Fluidity vs. Rotation (rpm)
- Two soft and collapse after drying
- Too hard to be cast
- Flow Chart

Slip Preparation

- Centrifugal Casting Method for Ceramic Body Fitting (JIC)
- Application to Alumina powder

Porosity & Centrifugal Force

IAERI-Cant. 98-009
Summary

- Preparation of the VEGA test facility is progressing and will be finished by June 1998.
- TGTs have added to have better information on FP transport.
- On-line oxygen/hydrogen sensors are ready for atmosphere characterization.
- A mock-up furnace with a graphite susceptor has realized target temperature of 3,000°C.
- Thoria tube fabrication is progressing with slip casting technique.

Thermal and Mechanical Properties

- Thermal Diffusivity (RT-1,400°C)
- Thermal Expansion (RT-1,400°C)
- Hardness (RT-1,400°C)
- Creep (RT-1,400°C)
- Melting Temperature (<3,000°C)
- Strength(RT)
9.2 Results of ISP37

VANAM M3 Experiment on Containment Thermal-Hydraulics and Aerosol Behavior

M. Firnhaber, T. Kanzleiter, G. Weber, S. Schwarz

An OECD/NEA International Standard Problem (ISP) was performed on the experimental basis of containment experiment VANAM M3 performed at the Battelle multi-compartment model containment. The objectives of this experiment was the investigation of the thermal-hydraulic and aerosol behavior of a large dry containment during a severe accident. Measured quantities are pressure, temperature, humidity, aerosol concentration etc. The ISP was conducted as an "open" exercise, e.g. beside boundary conditions all experimental results were made known to the participants prior to their calculation. The ISP attracted wide support. Results to the ISP were submitted by 21 participants from OECD and non-OECD countries using the codes CONTAIN, FIPLOC, MELCOR and others.

Due to the large number of participants the comparisons between experimental and analytical results could be grouped by codes and examined separately. The pressure history has been calculated by most of the participants quite well. The reasons for some minor deviation are air mass distribution, structure and sump temperature and heat transfer. Larger deviations had their origin in input errors. The accuracy of the calculated temperature depends on the correct prediction of stratification in some compartments and of a convection loop between other compartments. With correct stratification and convection only small temperature differences occurred. Reasons for not calculation a correct stratification are unstable temperature distribution and unfavorable nodalization.

Aerosol concentration and depletion results highly depend on the correct calculation of stratification, e.g. only those contribution which delivered correct stratification and convection submitted fairly good results. In particular codes which considered solubility of the aerosols (CONTAIN, FIPLOC) predict the concentration better than codes without solubility. Also, the thermal equilibrium option (MELCOR) gives better results than thermal non-equilibrium.

The large integral codes (CONTAIN, FIPLOC, MELCOR), which calculate both the thermal and aerosol behavior as well, attain a much better degree of agreement with the experimental results than the stand alone aerosol codes (MACRES, MOSAIC, REMOVAL).

In general the thermal hydraulic codes give sufficient accurate results for thermal hydraulic aspect. With respect to aerosols the thermal hydraulic code need some improvement, e.g. wall and volume condensation, humidity. But for better comparison also the experimental technique should be improved through more accurate measurement of humidity, wall condensation and heat transfer. The user had a great influence on the results, mainly be the choice of nodalization and of different code option.

The ISP demonstrated the important of assessments of this kind. It provided a forum for the international community enhancing the experience in performing containment thermal hydraulic and aerosol calculations in comparison with experimental data.
General Objectives of ISP

- International Standard Problems Initiated by OECD/NEA
- Examination of Reliability and Accuracy of Computer Codes Used in Reactor Safety
- Determination of Computer Model Improvements
- Information Exchange Between Code Developer, Code User and Experimentalist
- User Influence

Objectives of ISP37

- State-of-the-art Review on Computer Codes for Containment Analyses, e.g. Thermal-Hydraulics and Aerosol Behavior
- Phenomena to be Investigated
  - Multi-Compartment Geometry
  - Thermal Energy Balance
  - Structural Heat Transfer
  - Wall Condensation
  - Volume Condensation
  - Aerosol Distribution and Settling
  - Hygroscopic Aerosol Component
  - Steam Condensation on Aerosol Component
  - Identification of Experimental Accuracy and Problems
  - etc.

VANAM Test Facility

Low Pressure Path after "Station Blackout"
Test VANAM M3

1-Component Aerosol, Soluble
Dry Aerosol Settling
Wet Aerosol Settling
Volume Condensation

Steam into R3

Steam and Aerosol Injection
Participants and Codes Used in ISP37

- 21 Participants from 12 Countries (incl. 4 non-OECD) Delivered 26 Calculations

- Codes Used:
  - CONTAIN (5 x)
  - MELCOR (10 x)
  - FIPLOC (5 x)
  - ECART (5 x)
  - GOTHIC
  - RALOC
  - FUMO
  - MACRES
  - MOSAIC
  - REMOVAL

---

### Participating Countries

- Korea
- France
- Germany
- Italy
- Japan
- Russia
- Slovakia
- Spain
- Sweden
- The Netherlands
- USA

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### Aerosol Phenomena

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<th>Calculation</th>
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<td>H₂O Droplets</td>
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</table>

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### Boundary Condition

- Pressure
- Temperature
- Injection
- rel. Humidity
- Velocities

---

Fig. 2: Experimental Pressure and Aerosol Concentration History

(Aerosol Concentration History excluding Preconditioning Phase)
Procedure of Evaluation

- Consistency Check
  - Air Mass Distribution
  - Temperature Distribution (Atmosphere, Structures, Sump)
  - Temperature / Air Mass
  - Input Errors
- Reference Calculation for Pressure
  - Simulation of Stratification
  - Use of Proposed Input
  - Agreement with Experimental Flow Velocities
- Pressure Comparison
  - All Reference Calculation
  - Groups According to Codes Used

Conclusions on Pressure

- Pressure
  - Deviation between Experiment and Calculation in General Small (< 0.25 bar)
  - Some Larger Deviation due to Input Peculiarities or Errors
  - Reasons for Deviation
    - Air Mass
    - Structure Temperature
    - Sump Temperature
    - Heat Transfer
Conclusions on Temperature

- Atmospheric Temperature
  - Dome: Only Small Deviations from Experiment (Some Exceptions due to Pressure and Nodalization)
  - R₃: Temperature Indicates Stratification
  - No Stratification: Temperature Close to Experiment
  - R₉₄: Calculated Temperatures Vary Largely (up to 40 K)

Various Reasons:
- Steam Air Content
- Partial Steam Pressure
- Steam/Air Distribution
- Absolute Pressure
- Stratification

Conclusions on Stratification

- Stratification - Convection
  - Experiment
    - Steam Injection into R₅ Forces Inner Convection Loop
    - Steam Injection into R₃ Forces Outer Convection Loop
    - Annulus Atmosphere Stays Stratified
  - Calculations
    - All Calculations Simulate in the Convection Loops (Inner and outer)
    - Most Simulate the Stratification in the Annulus (Exception: Unfavorable Nodalization)
    - Different Behavior in Inner Rooms
      - FIPLOC, CONTAIN, GOTHIC Calculate Stratification
      - 2 MELCOR Calculation Simulate Stratification

Reasons:
- Unstable Temperature Distribution (Temperature Rise in the lower compartments)
- Too Large Time Steps

---

**Fig. 8** NaOH Aerosol Concentration in R₃ (Dome) (MELCOR Results, part 1)

**Fig. 9** NaOH Aerosol Concentration in R₉ (Dome) (CONTAIN Results)

**Fig. 10** NaOH Aerosol Concentration in R₉ (Dome) (Stand Alone Codes)

**Fig. 11** Mass Median Diameter in R₉ (Dome) (FIPLOC Results)
Conclusions on Aerosol Behavior

- Mass Median Diameter (MMD)
  - Measured and Calculated Data In Principle Incomparable
  - Calculations with Solubility Effect (e.g. CONTAIN, FIPLOC) Show a Strong Increase In MMD Starting at Phase 3, 5 and 6
  - MELCOR Calculates Small, Relatively Constant Diameter
- Geometric Standard Deviation
  All Calculated GSD Match the Measured Ones, e.g. the Codes Calculate the Correct Size Distribution

General Conclusions in ISP37

- Thermal Hydraulic Codes in General Sufficient for TH-Aspects
- Improvement Necessary for Aerosol Aspects
  - Wall Condensation
  - Volume Condensation
  - Humidity
  - (Steam Mass, Energy Balance)
- Solubility Models Should be Added
- Numerical Stability (Stratification / Mixing)
- Through Nodalization Great User Influence
- Experimental Uncertainties
- Uncoupled Aerosol ISP
9.3 Fission Product Aerosol Removal Test by Containment Spray under Accident Management Conditions

Hideo NAGASAKA* and Seiichiro YOKOBORI**
*Systems Safety Department, NUPEC,
**Nuclear Engineering Laboratory, Toshiba

ABSTRACT

This paper summarizes the test results of fission product (FP) removal by containment spray simulating accident management (AM) condition. The features of AM conditions concerning FP transport are characterized by (1) low flow spray affecting the steam condensation degradation due to larger water droplets, (2) high humidity condition due to steam generation as a result of debris cooling and (3) continual fresh water supply from outside water source. The objectives of the test program are to provide data demonstrating the effective aerosol removal by the containment spray and to provide the data for qualification of the integral system analysis code such as MELCOR.

The Tests were conducted using full-height simulation containment vessels of GIRAFFE (1/720 volumetric scaling ratio) so that real FP removal phenomena was preserved as in a reactor. Vessel heat loss was compensated by heaters on the outer surface of the vessels. CsI was selected as a typical FP aerosol. Steam generated by decay heat, CsI aerosol and spray water were supplied continuously to the drywell as transient boundary conditions.

Low flow spray droplet distribution was measured at atmospheric condition. CsI diameter and concentration were measured under steam environment by the optical particle counter and the measured values were consistent with typical values determined by analysis.

A system integration test simulating BWR low pressure vessel failure sequence during about 10 hours were successfully accomplished. Even under low spray flow condition, maximum drywell pressure was kept relatively low, though it was a little bit higher than the design pressure. After spray initiation, aerosol concentration decreased rapidly in the entire region of drywell. In the upper drywell, aerosol was removed by diffusiophoresis associated with steam condensation, while in the lower drywell it was removed by impaction. By modifying the FP removal model in the MELCOR, calculated FP concentration transient as well as pressure transient agreed well with test data.
FISSION PRODUCT AEROSOL REMOVAL TEST
BY CONTAINMENT SPRAY
UNDER ACCIDENT MANAGEMENT CONDITIONS

SARJ-97
October 7, 1997 Pacífico YOKOHAMA

H. Nagasaka/S. Yokobori
Systems Safety Department

CONTAINMENT SPRAY AS PHASE II ACCIDENT MANAGEMENT OF BWR

OBJECTIVES

0 TO PROVIDE DATA DEMONSTRATING THE EFFECTIVE AEROSOL FP REMOVAL BY CONTAINMENT SPRAY UNDER THE FOLLOWING PHASE II ACCIDENT MANAGEMENT (AM) CONDITIONS
- LOW FLOW SPRAY
- HIGH HUMIDITY CONDITION DUE TO DEBRIS COOLING BY WATER
- LONG TERM FRESH WATER SUPPLY

0 TO PROVIDE DATA FOR VERIFICATION AND MODIFICATION OF INTEGRAL SYSTEM CODES SUCH AS MELCOR FOR ANALYSIS OF FP TRANSPORT BEHAVIOR

CHARACTERISTIC FEATURES INSIDE CONTAINMENT UNDER AM CONDITIONS

0 LOW FLOW SPRAY
- LOWER CONDENSATION RATE DUE TO LARGER SPRAY DROPLETS
- EFFECT OF THE REDUCED CONDENSATION RATE ON DIFFUSION

0 CONTINUOUS FRESH WATER SPRAY SUPPLIED FROM WATER SOURCE OUTSIDE CONTAINMENT VESSEL
- LARGER FP CONCENTRATION DIFFERENCE BETWEEN DROPLET VOLUME AND CONTAINMENT AIRSPACE

0 HIGH HUMIDITY CONDITION DUE TO STEAM GENERATION AS A RESULT OF DEBRIS COOLING
- PROMOTION OF GRAVITATIONAL SETTLING AS A RESULT OF FP AEROSOL PARTICLE GROWTH

0 INTEGRAL FP REMOVAL EFFECT VIA VARIETY OF DEPOSITION MECHANISM AND COAGULATION MECHANISM

0 TIME VARYING AMBIENT CONDITIONS
- TOTAL PRESSURE / STEAM PARTIAL PRESSURE / TEMPERATURE
- FP AEROSOL GENERATION RATE
TEST FACILITY

- Utilization of Toshiba Giraffe Facility
  - Originally constructed for the development of passive containment cooling system (PCCS) in SBWR

Giraffe Overview
- Full height simulation of BWR
- 1/400~1/720 scaled volume of BWR
- Three separate vessel components simulating drywell / suppression chamber / reactor vessel... used in the PCCS
- Gravity driven core cooling system... not used in the present test

Test Facility Modifications
A - Spray Water Supply System
B - Superheated Steam Supply System
C,D - Cs1 Aerosol Supply / Measurement System
E - Non-condensable gas supply system

Giraffe Vessel Specifications

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<tr>
<th>Vessel</th>
<th>Suppression Chamber (S/C)</th>
<th>Drywell (D/W)</th>
<th>Reactor Pressure Vessel (RPV)</th>
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<td>Height (m)</td>
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<td>Diameter (m)</td>
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<td>Heater Power (KW)</td>
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Instrumentation of Giraffe-FP

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Diagram of Giraffe-FP system with various components labeled.
**SCALING CONSIDERATION OF TEST FACILITY (GIRAFFE-FP)**

- Full Height Simulation of Containment Vessel (CV)
  - Preservation of Vertical Elevations for Falling FP and Spray Droplet
- Usage of Actual Plant Spray Nozzle With Spray Flow Rate Per Nozzle Identical to That of Actual Plant Nozzle
  - Simulation of Droplet Diameter Distribution and Droplet Velocity
  - Simulation of Steam Condensation Rate on Spray Droplets
- Heat Loss Control by Heaters Surrounding CV
  - Minimization of Heat Loss

---

**SPRAY FLOW PATTERN AND DROPLET DIAMETER DISTRIBUTION**

- All Droplets Falling Without Attaching to the Wall
- Test Condition
  - Two Nozzles
  - Spray Flow: 3 (litre/min)
  - Pressure: 1 (bar) Atmosphere

---

**ATOMIZATION TECHNIQUE BY SUPERHEATED STEAM**

- Designed Atomizing Technique Confirmed Visually in Atmosphere
  - Aerosol Diameter Measured Under Steam Environment
  - Typical Atomizing Conditions
    - CsI Solution Concentration: 15 wt%
    - CsI Solution Pressure: 0.2 MPa
    - Superheated Steam Pressure: 0.4 MPa

- Two Fluid Nozzle Used in Giraffe-FP

---

**SCALING CONSIDERATION OF TEST FACILITY (GIRAFFE-FP)—Cont’d**

- Imposing of Several Transient Boundary Conditions to CV
  - Continuous Steam Supply Simulating Steam Generation as a Result of Debris Cooling
  - Continuous Non-Condensable Gas Supply Simulating Gas Generation Due to MCCI and Metal Water Reaction
  - Continuous FP (CsI Aerosol) Supply Simulating FP Reevaporation Inside Pressure Vessel
- Full Height Simulation of Suppression Chamber (BWR Test)
  - Simulation of Pool Scrubbing Effect

**INTEGRAL SIMULATION OF BOTH THERMAL HYDRAULIC TRANSIENT BEHAVIOR AND FP TRANSPORT BEHAVIOR SIMILAR TO THAT IN ACTUAL PLANT CAN BE ACHIEVED IN GIRAFFE-FP**
**CsI MEASUREMENT PROCEDURE**

- **Only one measuring unit used**
  - Optical particle counter
- **Measuring location change achieved by transport of measuring unit up and down**
- **Measurement focused in D/W region after spray initiation**
- **Then measuring unit moved to the D/W bottom**
- **About 30 min of measuring time needed at one location**
  - 3 times repeated measurements at one location

**Typical aerosol condition in lower drywell after RPV failure (TQUV)**

- **Aerosol condition to be imposed as boundary condition in GIRAFFE-FP determined by both MAAP and MELCOR analysis**
  - Typical CsI flow rate as a result of re-evaporation: 1–5 mg/min
  - Typical CsI particle mass mean diameter: ~1 μm
- **Both measured CsI concentration and diameter generated by atomizer nozzle confirmed close to prescribed values**

**System integral test conditions**

- **Test conditions determined from results of plant analysis using Level-2 PSA for dominant low pressure failure of pressure vessel**
  - TQUV
  - LOCA
- **Initial conditions inside containment vessel**
  - Time corresponding to pressure vessel failure by molten core
  - Total pressure / steam partial pressure / non-condensable gas partial pressure
  - Temperature
  - Aerosol concentration
- **Boundary conditions**
  - 1/720 scaled flow rate
  - Spray flow rate
  - Steam generation rate corresponding to debris cooling
  - CsI aerosol injection rate

**Typical test conditions (BWR TQUV case)**

- **Test conditions determined considering both MAAP and MELCOR analysis**

**Drywell**

- Pressure: ~0.30 MPa
- Temperature: ~420 K
- Aerosol concentration: ~10⁻¹⁰ kg/m³

**Suppression chamber**

- Pressure: ~0.30 MPa
- Temperature: ~340 K

**Suppression chamber**

- Aerosol supply (nearly constant)
BWR TQUV TEST RESULTS (CV PRESSURE TRANSIENT)

- Maximum drywell pressure kept relatively low (little higher than design pressure of PCV even under low spray flow condition)
- Effective heat removal of direct contact heat transfer between subcooled water and pure steam
- Drywell maximum pressure observed after 8 hours, when steam condensing capability of spray balancing with steam supply rate

MELCOR ANALYTICAL RESULTS FOR BWR TQUV TEST
- Drywell pressure transient

- Calculated initial pressure increasing rate lower than measured data due to smaller temperature rise above the first vent pipe exit in the analysis
- Consideration of thermal stratification needed

BWR TQUV TEST RESULTS (AEROSOL CONCENTRATION TRANSIENT)

- In the upper drywell, aerosol considered to be removed by diffusiophoresis associated with rapid steam condensation near the nozzle, especially immediately after spray initiation
- In the lower drywell, aerosol considered to be mainly removed by impaction of spray droplets because of saturation temperature condition of water
- In the entire drywell region, aerosol concentration kept lower throughout the test

MELCOR ANALYTICAL RESULTS FOR BWR TQUV TEST
- CsI aerosol concentration in drywell

- Calculated concentration transient agreed well with that of test data
CURRENT STATUS SUMMARY OF GIRAFFE-FP TEST PROGRAM

- Test facility modification of GIRAFFE-FP completed
- Spray droplet distribution under low flow condition measured
- Cs1 aerosol diameter and concentration adjusted to be consistent with analysis condition
- A system integration test simulating BWR TQUV sequence completed
  - Long range tests successfully accomplished
  - Maximum drywell pressure kept relatively low (little higher than design pressure of PCV even under low spray flow condition)
  - Aerosol concentration decreased rapidly after spray initiation

MELCOR ANALYTICAL RESULTS FOR BWR TQUV TEST—Cont’d

- Aerosol concentrations by original and modified MELCOR

- Under pure steam condition, aerosol collection efficiency by diffusiophoresis artificially set to be 1.0 in original MELCOR code
- Due to this inadequate model, calculated concentration by original MELCOR rapidly decreased after spray initiation

TEST SCHEDULE

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9.4 Deposition of CsI Aerosol in Horizontal Straight Pipe under Inert and Superheated Steam Environment

Hiroaki Shibazaki, Minoru Igarashi*, Yu Maruyama, Akio Maeda, Yuhei Harada, Akihide Hidaka, Jun Sugimoto

Japan Atomic Energy Research Institute
* Kawasaki Heavy Industries, Ltd.

ABSTRACT

In a severe accident of an LWR, fission products (FPs) aerosol released from a reactor core region will be deposited on the inner surface of the reactor coolant piping. In such conditions, the piping might be subjected to a thermal load due to decay heat from the deposited FPs. It is very important to quantify the FP aerosol deposition on the piping surfaces. Therefore the FP aerosol behavior in piping is being investigated in the WIND (Wide Range Piping Integrity Demonstration) project at Japan Atomic Energy Research Institute. The objectives of present study are to characterize the aerosol deposition on piping surfaces under various thermal-hydraulic conditions and to obtain insights for the validation of analytical models.

A test facility used in aerosol deposition tests mainly consists of two test sections in series, aerosol generators, an argon gas supply system, a steam generator, an aerosol sampling system and a heat exchanger. The test sections are made from stainless steel pipe of about 100mm in inner diameter and 2000mm in length. Cesium iodide (CsI) aerosol was generated by the aerosol generator using an induction heating technique and introduced into the test section with a carrier gas. Argon gas or a mixture of argon and superheated steam was used as a aerosol carrier gas.

A chemical analysis of the deposited aerosol showed that no evidence was found for the decomposition of CsI under inert and superheated steam environments. The major deposition mechanisms are identified to be the condensation of CsI vapor and the thermophoretic aerosol transportation from the carrier gas to the colder piping surfaces. Thermo-fluiddynamic analyses of the carrier gas with WINDFLOW code implied that a precise prediction is required for the evaluation of the amount and the spatial distribution of the aerosol deposition. Remarkable aerosol deposition onto the floor area and enlargement of the deposited aerosol were observed in the test with a superheated steam environment. An additional test will be shortly performed in order to reconfirm the findings obtained under a superheated steam environment.
Deposition of CsI Aerosol in Horizontal Straight Pipe under Inert and Superheated Steam Environments


Japan Atomic Energy Research Institute
* Kawasaki Heavy Industries, Ltd.

Presented at SARJ-97
October 6 - 8, 1997 Yokohama, Japan

1. INTRODUCTION

icional models

- Aerosol behavior in reactor coolant piping
- Structural integrity of reactor coolant piping under severe thermal-hydraulic conditions

Objectives of Present Study

- To characterize the deposition of CsI aerosol on piping surfaces under various thermal-hydraulic conditions
- To obtain insights for validation of analytical models

Contents

1. Introduction
2. Outline of Aerosol Deposition Test
3. Test Results
4. Conclusions
2. Outline of Aerosol Deposition Test

Flow Diagram of Test Facility

Test Sections

Locations of Aerosol Deposition Coupons
3. TEST RESULTS

Distribution of Deposited Iodine and Cesium

Upstream test section  Connecting pipe  Downstream test section  Total

<table>
<thead>
<tr>
<th>Test No.</th>
<th>Carrier gas</th>
<th>Carrier gas flow rate (GPM)</th>
<th>Axial Temperature distribution at test section inlet (UTS / DTS) (Temp. °C)</th>
<th>Carrier gas inlet velocity (m/s)</th>
<th>% (%I)</th>
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<tr>
<td>WAD4c*</td>
<td>Argon</td>
<td>3.6</td>
<td>1000 - 350/700 - 250</td>
<td>0.71/0.58</td>
<td>874/1008</td>
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<tr>
<td>WAD5</td>
<td>Steam &amp; Argon (15% volume)</td>
<td>2.5</td>
<td>950 - 650/700 - 250</td>
<td>0.74/0.51</td>
<td>854/1015</td>
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</table>

*Thermal-hydraulic conditions of WAD4a and WAD4b are almost identical to WAD4c.
UTS: Upstream Test Section, DTS: Downstream Test Section

Aerosol Diameter Distribution Measured by Cascade Impactor at Inlet of Upstream Test Section

Concentration measured by temporarily installed impaction sampler
WAD4a: 3.5g/m³
WAD4b: 3.9g/m³
WAD4c: 3.6g/m³
WAD5: 1.7g/m³
4. CONCLUSIONS

- No evidence was found for the decomposition of CsI under inert and superheated steam environments.

- Detailed characterization of thermal-hydraulics of gas surrounding aerosol is required for an accurate prediction of the aerosol deposition.

- Remarkable aerosol deposition onto floor area and enlargement of deposited aerosol were observed in the test with a superheated steam environment.

- Further confirmation test is necessary.
9.5 Experimental and Analytical Study on Aerosol Behavior in WIND Project

Akihide HIDAKA, Yu MARUYAMA, Minoru IGARASHI, Kazuichiro HASHIMOTO and Jun SUGIMOTO

Dept. of Reactor Safety Research
Japan Atomic Energy Research Institute

Abstract

The tests on fission product (FP) behavior in piping under severe accidents are being conducted in the WIND Project at JAERI to investigate the piping integrity which may be threatened by decay heat from deposited FPs.

In order to obtain the background information for future WIND experiment and to confirm the analytical capabilities of the FP behavior analysis codes, ART developed by JAERI and VICTORIA by SNL, the FP behavior in safety relief valve (SRV) line of BWR during station blackout without all feed water (TQUX) sequence was analyzed. The analyses showed that the mechanisms that control the FP deposition and transport agreed well between the two codes. However, it was also found that the differences in models such as diffusiophoresis or turbulence, the treatment of chemical forms and aerosol mass distribution may affect the deposition in piping and, consequently, on the source terms.

The WIND experimental analyses were also conducted with the 3-dimensional fluiddynamic code WINDFLOW, ART and an interface module to appropriately couple the fluiddynamics and FP behavior analyses. The analyses showed that the major deposition mechanism for CsI is thermophoresis which depends on the thermal gradient in the gas phase. Accordingly, the coupling analyses was found to be essential to accurately predict the CsI deposition in piping, to which little attention has been paid in the previous studies.
Experimental and Analytical Study on Aerosol Behavior in WIND Project

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1) Japan Atomic Energy Research Institute
2) Kawasaki Heavy Industries, Ltd.
3) Nuclear Power Engineering Corp.

Presented at Workshop on Severe Accident Research held in Japan (SARJ-97)
October 6-8, 1997, Yokohama, Japan

1. Introduction

• WIND (Wide range piping INtegrity Demonstration) experiment to investigate FP aerosol behavior in piping

• SRV line during TQUX sequence of BWR to be simulated in WIND project, was calculated by ART and VICTORIA codes.
  - To confirm the analytical capabilities
  - To obtain background information for future test

• WIND aerosol deposition tests, WAD1 and 2 were analyzed by WINDFLOW, ART and an interface module.
  - To investigate effect of coupling of fluid dynamics and FP aerosol analyses on predictability for FP deposition in piping

2. Outlines of WINDFLOW, ART and VICTORIA Codes

© WINDFLOW (developed at JAERI)
  - 3-dimensional thermo-fluidodynamic analysis
  - Laminar and turbulent flows of compressive fluid
  - Solution by semi-implicit method

• ART (developed at JAERI)
  - Detailed models for FP aerosol behavior in RCS and CV including ESF
  - Fast running capability for PSA
  - Module of THALES-2 for source term analysis

• VICTORIA (developed at USNRC)
  - Mechanistic models for FP release and transport behaviors in RCS
  - Natural deposition mechanisms for FPs in VICTORIA ~ Those of ART

Contents

1. Introduction
2. Outlines of WINDFLOW, ART and VICTORIA Codes
3. Analysis for BWR SRV Line
4. Analysis for WIND Experiment
   (1) WAD1 Test
   (2) WAD2 Test
5. Conclusions
3. Analysis for BWR SRV Line

- Reference plant
  - Browns Ferry (with Mark-I containment)

- Analytical sequence
  - Station blackout without all feed water (TQUX)

- Aerosol behavior in SRV line at 115min
  - Most severe case for SRV integrity during TQUX sequence

- Assumptions (based on MELCOR calculation)
  - Duration for SRV opening : 120s
  - Gas flow velocity : 250m/s
  - CsI and CsOH mass flow rates : $4.81 \times 10^{-3}$, $5.42 \times 10^{-2}$ kg/s
  - Aerosol MMD and GSD : 0.32 $\mu$m, 2.0

Temperature Distribution in SRV Line

- Gas, 800K
- Wall
- Distance (m)
- Temperature (K)
- 800
- 700
- 600
- 500
- 400
- (Steam dome)
- (Suppression pool)

Deposited Mass Distribution (after 120s)

- CsOH, VICTORIA
- CsOH, ART
- CsI, VICTORIA
- CsI, ART
- Cell Number
- Deposited Mass [kg/m²]
Averaged Aerosol Deposition Velocity

- Turbulence
- Diffusion
- Thermophoresis
- Gravitation
- Inertia

Deposition Velocity [m/s]

Cell Number

1 3 5 7 9 11 13 15

Aerosol Number Concentration

- CsOH, VICTORIA
- CsI, VICTORIA
- ART

Mass Class [kg]

Time [s]

10 3600 7200 10800 14400 18000

4. Analysis for WIND Experiment

- Recent WIND aerosol deposition test
  - WAD1 Test; Laminar flow (1.01 m/s)
  - WAD2 Test; Turbulent flow (3.45 m/s)

- Thermohydraulic conditions defined based on THALES-2 calculation for hot-leg during typical PWR severe accidents

- Connection of WINDFLOW and ART with newly developed interface module
  - To reduce uncertainties in FP behavior analyses

Temp. Increase of SRV Line Inlet by Decay Heat

Melting temperature of carbon steel (STS42)

Temperature [K]

0 3600 7200 10800 14400 18000

Time [s]
Aerosol Behavior Test Facility in WIND Exp.

WINDFLOW Nodalization for WAD1 and 2 Tests

(1) WAD1 Test
Temperature Distribution

Gas Velocity and Isotherm in WAD1 Pipe
ART Nodalization for WAD1 and 2 Tests

Averaged Deposition Velocity in WAD1 Test

Csl Deposition Mass in WAD1 Test

(2) WAD2 Test
Temperature Distribution
Csl Deposition Mass in WAD2 Test

5. Conclusions

- FP behavior in SRV line during TQUX was analyzed with ART and VICTORIA codes.
- Mechanisms that control FP deposition and transport agreed well between the two codes.
- However, differences in models such as diffusiophoresis, chemical form and aerosol mass distribution may affect the deposition in piping.
- WIND aerosol deposition test was analyzed with WINDFLOW, ART and an interface module.
- Since major deposition mechanism for Csl is thermophoresis, coupling of fluiddynamic and FP aerosol analyses was found to be essential to accurately predict deposition in piping.
9.6 Steam condensation on spray water drops: experimental results and models

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ABSTRACT

Consequences of an overheating reactor accident can be limited by spraying cold water drops. The spray reduces the pressure and the temperature levels by condensation of steam. It also leads to the washout of the fission products (aerosols and iodine) emitted in the reactor building atmosphere. The present study includes a large program devoted to the evaluation of more realistic washout rates. An experimental device (named CARAIDAS) and a numerical code (named ACACIA) have been developed in order to propose and to qualify models for steam condensation, aerosols collection and gaseous iodine absorption. After a short description of CARAIDAS and ACACIA, experimental and numerical results of steam condensation or evaporation on falling drops are presented.

NOMENCLATURE

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<th>Definition</th>
<th>Unit</th>
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<tr>
<td>C_p</td>
<td>Calorific capacity of water</td>
<td>(J.kg^(-1).K^-1)</td>
</tr>
<tr>
<td>D_d</td>
<td>Drop diameter</td>
<td>(m)</td>
</tr>
<tr>
<td>Dif</td>
<td>Diffusion coefficient of steam in air</td>
<td>(m^2.s^-1)</td>
</tr>
<tr>
<td>dm</td>
<td>Drop mass increase</td>
<td>(kg)</td>
</tr>
<tr>
<td>h_o</td>
<td>Heat transfer coefficient on drop side</td>
<td>(W.m^-2.K^-1)</td>
</tr>
<tr>
<td>h_g</td>
<td>Heat transfer coefficient on gas side</td>
<td>(W.m^-2.K^-1)</td>
</tr>
<tr>
<td>ΔH</td>
<td>Mass enthalpy of steam condensation</td>
<td>(J.kg^-1)</td>
</tr>
<tr>
<td>k_o</td>
<td>Mass transfer coefficient</td>
<td>(m^2.s^-1)</td>
</tr>
<tr>
<td>m</td>
<td>Drop mass</td>
<td>(kg)</td>
</tr>
<tr>
<td>M</td>
<td>Molar mass of water</td>
<td>(kg.mol^-1)</td>
</tr>
<tr>
<td>N</td>
<td>Flux density of steam</td>
<td>(mol.m^-2.s^-1)</td>
</tr>
<tr>
<td>P</td>
<td>Total pressure</td>
<td>(Pa)</td>
</tr>
<tr>
<td>P_s</td>
<td>Steam pressure</td>
<td>(Pa)</td>
</tr>
<tr>
<td>P_v</td>
<td>Saturation vapor pressure</td>
<td>(Pa)</td>
</tr>
<tr>
<td>q</td>
<td>Heat flux density</td>
<td>(W.m^-2)</td>
</tr>
<tr>
<td>q_d</td>
<td>Heat flux density on drop side</td>
<td>(W.m^-2)</td>
</tr>
<tr>
<td>q_g</td>
<td>Heat flux density on gas side</td>
<td>(W.m^-2)</td>
</tr>
<tr>
<td>R</td>
<td>Ideal gas constant</td>
<td>(J.mol^(-1).K^-1)</td>
</tr>
<tr>
<td>S</td>
<td>Steam saturation rate</td>
<td>(-)</td>
</tr>
<tr>
<td>Δt</td>
<td>Time interval</td>
<td>(s)</td>
</tr>
<tr>
<td>T_a</td>
<td>Drop temperature</td>
<td>(K)</td>
</tr>
<tr>
<td>T_g</td>
<td>Gas temperature</td>
<td>(K)</td>
</tr>
<tr>
<td>T_d</td>
<td>Drop-gas interface temperature</td>
<td>(K)</td>
</tr>
<tr>
<td>V</td>
<td>Drop velocity</td>
<td>(m.s^-1)</td>
</tr>
<tr>
<td>z</td>
<td>Falling height</td>
<td>(m)</td>
</tr>
<tr>
<td>λ_g</td>
<td>Thermal conductivity of gas</td>
<td>(W.m^-1.K^-1)</td>
</tr>
<tr>
<td>ρ_g</td>
<td>Density of gas</td>
<td>(kg.m^-3)</td>
</tr>
<tr>
<td>ρ_d</td>
<td>Density of water</td>
<td>(kg.m^-3)</td>
</tr>
<tr>
<td>µ_d</td>
<td>Dynamic viscosity of gas</td>
<td>(Pa.s)</td>
</tr>
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1.- INTRODUCTION

Consequences of a hypothetical overheating reactor accident in a nuclear power plant can be limited by spraying cold water drops into the containment building. The spray reduces the pressure and the temperature levels inside the containment building by steam condensation on drops. Over this thermodynamic function, spray leads to the washout of fission products (aerosols and gaseous iodine) emitted in the reactor building atmosphere. Today, this spray system is taken into account in different safety codes, using washout rates which have been determined in global experiments, so these washout rates provide conservative assumptions. The present study includes a large program devoted to the evaluation of more realistic washout rates. An experimental device (named CARAIDAS) and a numerical code (named ACACIA) have been developed in order to propose and to qualify models for steam condensation, aerosols collection and gaseous iodine absorption. This paper presents experimental results performed on steam condensation or evaporation on drops. The aim of this study is to find out the evolution of the drops diameter along their fall of five meters height and to compare these experimental data with numerical results.

2.- EXPERIMENTAL DEVICE

The experimental device, CARAIDAS (figure 1), is composed of three specific devices for the study of the drops behaviour:

- the experimental enclosure in which thermodynamic conditions could be achieved,
- the monosized drops generator,
- the drops diameter measurements.

Experimental enclosure is a 5 meters high cylinder with an inner diameter of 0.6 meter. The vessel is heated up by circulating a thermodiluent through the double-wall casing. This system is split in three sections to ensure uniform temperatures overall the vessel height. The thermodiluent circuit comprises a pump with a flow rate of 12 m^3/h, an electric heater (40 kW) using a PID regulation to heat up the thermodiluent at a maximum temperature of 160°C and each of the three sections are equipped with one valve and one flow meter. Homogeneous thermodynamic conditions are obtained by using an air steam circulation with a varying flow rate fan (0 to 50 m^3/h), an electric heater (5 kW) to heat up air-steam mixture with a temperature ranging from 20 to 160°C and an absolute pressure PID regulation in the range of 1 to 8 bar by using two valves (one for the pressurized air alimentation and one for the release). Steam is produced by an electric generator.
Steam saturation rate range is from a few percents till 95%. Highest saturation limit is 95% to avoid condensation on the inner surfaces.

When nominal working conditions \((P, V, S)\) are reached, air-steam circulation is stopped and the vessel is isolated by two valves. Several sensors are installed on the vessel to check air-steam mixture homogeneity and evolution during each test:

- five gas temperature sensors (Pt 100, class A),
- three inner vessel wall sensors (Pt 100, class A),
- one pressure transducer (0 - 10 bars),
- three steam saturation rate sensors (one by dew point measurement and two by hygroscopic sensor).

All these experimental data are displayed and saved by using a PC supervisor.

Drops generator is located at the top of the experimental enclosure because this device must be at ambient temperature whatever the vessel temperature. In order to produce monosized drops, the generator (figure 2) is based on a break-up process of a jet to drops by applying a periodic disturbance. With this principle of generation, the space between two successive drops is not large enough to avoid drops coalescence, so an electrostatic sorting out drops is set up. A stream of uniformly charged drops is shaped by applying a potential difference between a ring electrode and the feeding tube. When a short negative impulse is applied to the ring electrode, an uncharged drop is produced. Deflection plates placed downstream, which are under electrical potential difference, create an electric field. So, charged drops passing through this electrical field are deflected and collected. Uncharged drops are not deflected and then are injected into the experimental enclosure. The rate of injection of uncharged drops is set from 1/1 till 1/1000. With these conditions, the device can produce monosized water drops with a diameter ranging from 200 to 600 \(\mu\)m. Drops injection temperature which can be set between 20°C and 80°C by a small electric heater, is measured by a Pt100 sensor on the feeding tube.

After injection into the vessel, the drops diameter is modified by steam condensation or evaporation as function of thermodynamic conditions. Optical measurements of the drops diameter are performed at three elevations: one at the top of the enclosure where the drops are emitted \((z = 0\) m\), a second one at midheight \((z = 2.51\) m\) and the last one at the bottom of the device \((z = 4.39\) m\). These measurements are based on drops shadows axial transmittance. A stroboscopic incoherent light source is placed in front of linear camera (figure 3). When a drop comes in front of the CCD camera, analogic signals from photodiodes are obtained and then numerized. The numerical drop shadow is processed in order to calculate the real drop diameter.

3.- ACACIA CODE

Drops characteristics (temperature, diameter, iodine and particles concentration) are modified during their fall. Three phenomena of transfer between the gas and the drops have to be modelled:

- steam and heat transfer,
- gaseous iodine absorption,
- particles collection.

Because iodine absorption and aerosols collection are coupled with steam and heat transfers, the aim of the first part of the program is to model the condensation phenomena.

In ACACIA code, different assumptions are used:

- gaseous concentrations are homogeneous in the vessel,
- steady state conditions in gas,
- drops are independant.

Steam and heat transfers from the gaseous phase to the liquid one is modelled by the double film theory (figure 4). Steam mass transfer is located in the gaseous film and heat transfer takes place through the whole double film. Beyond this double film, temperatures and concentrations are supposed steady.

The steam flow rate is:

\[
N = k_g \left( \frac{P_g}{R \cdot T_g} - \frac{P_w (T_d)}{R \cdot T_d} \right) \quad \text{(eq.1)}
\]

The transfer coefficient \(k_g\) which is the ratio between the diffusion coefficient of steam in air and the thickness of the gaseous film, is computed thanks to a correlation given by BEARD and PRUPPACHER [1]:

\[
Sh = 1.61 \cdot 0.718 \cdot \text{Re}^{1/3} \cdot \text{Sc}^{1/3}
\]

The heat flow rate \(q_g\) through the gaseous film is:

\[
q_g = h_g (T_g - T_d) \cdot N \cdot M \cdot \Delta H
\]

The heat flow rate \(q_d\) through the drop film is:

\[
q_d = h_d (T_d - T_i) \cdot N \cdot M \cdot \Delta H
\]

As the heat flow rate is the same on the gas side and on the drop side, finally, heat flow rate through the whole double film is:

\[
h_g (T_g - T_d) + h_d (T_d - T_i) = 0 \quad \text{(eq.2)}
\]

So, coupling eq.1 and eq.2, the interface temperature \(T_i\) and the steam flow rate \(N\) are computed by a numerical method (NEWTON-RAPHSON) at level \(z\). Drops characteristics, steam and heat flow rates at level \(z + \Delta z\) allow to compute the drops characteristics evolution at level \(z + 2\Delta z\) (figure 5):
drops diameter:

\[ D_d(z + \Delta z) = D_d(z) \cdot \left( \frac{6 \cdot D_d(z) \cdot \Delta t \cdot N \cdot M}{P_d} \right)^{1/3} \]

with \( \Delta t = \frac{\Delta z}{V} \)

drops temperature:

\[ T_d(z + \Delta z) = T_d(z) - \frac{\pi \cdot D_d(z)^2 \cdot \Delta t \cdot \left[ h_d \cdot T_d - h_f \cdot T_f \right]}{m \cdot Cp} \cdot N \cdot M \cdot \Delta H \]

4. EXPERIMENTAL RESULTS

Experimental results are drops diameters measurements at three levels \( z = 0 \text{ m}, z = 2.51 \text{ m}, z = 4.39 \text{ m} \) in the experimental enclosure for different initial drops diameters and for several thermodynamic conditions \( (P, T_a, S) \) which are representative of conditions that could be expected in containment building during an hypothetical overheating reactor accident.

Two kinds of experiments have been performed, one on evaporation conditions and one on condensation conditions.

In evaporation tests, drops are injected in dry gas, so only evaporation occurs. During these tests, a large range of experimental conditions (table 1) have been achieved in the experimental vessel. For example, absolute pressure is ranging from 1 to 5.4 bar and gas temperature is ranging from 20 to 147°C. The thermodynamic conditions measurements achieved by all the sensors set up on the experimental enclosure show a very good homogeneity all over the five meters height and the steady-state conditions. For each test, mean drop diameter \( (D_{\text{mean}}) \) and standard deviation \( (\sigma) \) of drops at the three measurement levels are presented in table 2.

The drops diameters measurements at the top level \( (z = 0 \text{ m}) \) show that the drops generator produces monosized drops (standard deviation lower than 10 µm) and this with diameters ranging from 295 to 610 µm. For the second and the third levels \( (z = 2.51 \text{ and } z = 4.39 \text{ m}) \), the standard deviation of drops populations are also small \( (\sigma < 15 \text{ µm}) \) even that these optical measurements are more difficult to achieve because of a drops spatial dispersion. Moreover, depending on thermodynamic conditions, drops diameters between the top and the bottom of the enclosure vary from a few percents till a total disappearance of drops. These evaporation tests results are used to qualify steam evaporation model.

In condensation tests, cold water drops are injected in wet gas. Because experiments are performed with steam saturation rate lower than 1 to avoid condensation in the vessel, steam condensation on drops occurs just during the beginning of the drop fall: steam and heat flow rates induce the growth and the heating of drops. When the drops temperature reaches the dew temperature, steam condensation stops. But this temperature is lower than the gas temperature so evaporation appears : there is an equilibrium between the heat flow brought to the drop and the heat lost by evaporation. Thus, the drop temperature is steady but the diameter decreases.

5. MODEL/EXPERIMENT COMPARISON

In order to qualify the steam condensation model included in ACACIA code, experimental and numerical results of the evolution of drops diameters are compared.

ACACIA computations are realized by using experimental thermodynamics conditions (table 1) and initial drops diameter at the top level (table 2). Then experimental and numerical drops diameters at levels \( z = 2.51 \text{ m} \) and \( z = 4.39 \text{ m} \) are compared on the basis of the following ratio \( (Ra) \).

\[ Ra = \frac{D_{\text{CA}} - D_{\text{ex}}}{D_{\text{ex}}} \]

with \( D_{\text{CA}} \) : drops diameter computed by ACACIA
and \( D_{\text{ex}} \) : drops diameter measurements in CARAIDAS

These results (table 3) show a good agreement between experimental data and numerical computations because most of the absolute values of this ratio are smaller than ten percents. This ratio is not computed when drops disappear because in this case the value just could be 100 % or 0 %. For some tests (numbers 18, 21, 22, 23, 24, 25 and 26), differences between code and experiment are greater (> 10 %). Nevertheless, figures 6 and 7 show that the evolutions of drops diameters computed by ACACIA are closed to experimental results even in cases of total evaporation of drops. In fact when the evaporation is very important, drops diameter \( D_{\text{CA}} \) becomes small so the ratio \( Ra \) increases. When the condensation occurs (tests 27 and 28 on figure 7), the drops evolution computed by ACACIA is very closed to the measurements.

6. CONCLUSION

An experimental device and a numerical code have been performed in order to propose and qualify models for steam condensation, aerosols collection and iodine absorption which are the phenomena occurring in spraying systems. The first part of this program (drops behaviour in evaporation and condensation conditions) is ended. The results of numerical simulations of drops evolution show a good agreement with the experimental data acquired.

Now, experimental aerosols tests are going on in order to qualify collection efficiency models.

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<th>Test number</th>
<th>P (bar)</th>
<th>T_{so} (°C)</th>
<th>T_{d} (°C)</th>
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Table 1: Thermodynamic conditions

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<th>D_{min} (μm)</th>
<th>D_{min} (μm)</th>
<th>D_{min} (μm)</th>
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<td>393</td>
<td>363</td>
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<td>591</td>
<td>590</td>
<td>8</td>
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<td>7</td>
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<td>7</td>
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<td>17</td>
<td>302</td>
<td>215</td>
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<td>105*</td>
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<td>14</td>
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<td>7</td>
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<td>324</td>
<td>187</td>
<td>7</td>
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<tr>
<td>22</td>
<td>319</td>
<td>322</td>
<td>D**</td>
<td>-</td>
</tr>
<tr>
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<td>11</td>
<td>D**</td>
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<td>D**</td>
<td>-</td>
</tr>
<tr>
<td>25</td>
<td>295</td>
<td>318</td>
<td>D**</td>
<td>-</td>
</tr>
<tr>
<td>26</td>
<td>303</td>
<td>318</td>
<td>11</td>
<td>D**</td>
</tr>
<tr>
<td>27</td>
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<tr>
<td>28</td>
<td>344</td>
<td>348</td>
<td>301</td>
<td>10</td>
</tr>
</tbody>
</table>

Table 2: Experimental results

* no measurement
** drops disappearance

Table 3: Model/Experiment comparison
Figure 1: Experimental device CARAIDAS

Figure 2: Monosized drops generator

Figure 3: Principle of optical drops diameter measurement

Figure 4: Double-film theory scheme
9.7 Failure Criteria and Fission Products Trapping Effect at Containment Penetrations under Severe Accident Conditions

Atsushi WATANABE *, Takashi HASHIMOTO* and Masahiko OSAKI**
*Systems Safety Department, NUPEC
**Nuclear Engineering Laboratory, Toshiba

ABSTRACT

The purposes of this tests are to demonstrate the integrity of the containment penetrations under severe accident conditions, to investigate the failure criteria and to demonstrate fission products trapping effects along the leakage paths of its degraded penetrations.

Actual modules of the electrical penetration assemblies and actual flange gaskets of the equipment hatches were used for the tests as the containment penetrations. They were installed into the large-scale test apparatus and suffered with high temperature and high pressure steam/air mixture environment.

Twelve electrical module tests (nine low-voltage and three high-voltage modules) and five flange gasket tests have been carried out. No leakage have been found for every test piece less than 200°C. Some leakage was found for seven low-voltage module at 280-320°C. For the high-voltage module, leakage was only found during temperature decreasing process from 380°C to room temperature. In case of flange gasket, the leakage temperature was occurred depending on the system pressure, 330°C at 0.8 MPa and 360°C at 0.4 MPa.

CsI aerosol was injected with carrier air into the damaged test pieces to evaluate aerosol trapping effects along the leakage path. The inlet and the outlet aerosol concentrations were measured by a laser particle counter. Decontamination factor (DF= inlet aerosol concentration / outlet aerosol concentration) was ranged from 10 to more than 1000 for the low-voltage modules and 10 to 30 for the flange gasket.
Failure Criteria and Fission Products Trapping Effect at Containment Penetrations under Severe Accident Conditions

SARJ-97
October 7, 1997 Pacifico YOKOHAMA

A.Watanabe / T.Hashimoto
Systems Safety Department

OUTLINE

1. Background
2. Objectives
3. Containment Penetrations
4. Test Facility
5. Integrity Test
6. Leak Criteria Test
7. Aerosol Trapping Test
8. Conclusion

BACKGROUND

Containment Penetrations with Sealing Compounds made of Organic Seal Materials could be Damaged under Severe Accident Conditions.

To prevent Containment Leakage and Failure, Containment Spray as an Accident Management has been adopted:
- Maximum Temperature $\leq 200^\circ$C
- Maximum Pressure $\leq 2$ Pd

OBJECTIVES

1. Demonstrate the Integrity of the Containment Penetrations under Severe Accident Conditions adopting Accident Management
2. Investigate the Failure Criteria of the Containment Penetrations under Severe Accident Conditions
3. Demonstrate Fission Products Trapping Effects along the Leakage Paths of its Degraded Penetrations
LOCATION OF CONTAINMENT PENETRATIONS
(Typical BWR Containment)

Drywell Main Flange

Gasket Seal (Silicone Rubber)

Containment Vessel

Low voltage Module

(Epoxy Resin)

High Voltage Module

Secondary Seal (EPR)

VARIOUS CONTAINMENT PENETRATIONS

Hatch Flange

Electrical Penetration Assembly (EPA)

Equipment Hatch Flange

Electrical Penetration Assembly

Flange Gaskets

Low Voltage Module

TEST FACILITY

PICTURE OF TEST FACILITY
INTEGRITY TEST

Test Condition
- Test Piece: Low Voltage Module (Cable; 0.9mm²)
- Temperature: 120-200°C (Cycle)
- Pressure: 0.1-0.8 MPa
- Radiation Aging: 800 KGy
- Atmosphere: Steam

Test Results
- Any leak was not observed during and after the test period.
- Temperature/Pressure Profile: Right figure.
- State of test piece after the test: Next photograph.

SUMMARY OF INTEGRITY TEST

<table>
<thead>
<tr>
<th>Test Pieces</th>
<th>Rad.</th>
<th>Steam</th>
<th>Temp. (°C)</th>
<th>Test Duration (hr)</th>
<th>Leak</th>
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<tr>
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<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>0.9mm² (1)</td>
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<td>230</td>
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<td>Yes</td>
<td>200-120</td>
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<td>Yes</td>
<td>220</td>
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<td>No</td>
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<td>200</td>
<td>20</td>
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<tr>
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<td>Yes</td>
<td>200</td>
<td>20</td>
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<td>24</td>
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<td>Yes</td>
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<td>G-Type</td>
<td>Yes</td>
<td>Yes</td>
<td>200-120</td>
<td>54</td>
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Pressure: 0.8 MPa

*Little leak was occurred after cooling down to RT.
LEAK CRITERIA TEST

Test Condition
- Test Piece: Low Voltage Module (Cable; 0.9mm²(T/C))
- Temperature: 200°C - Failure Temp. (20°C escalation/1hr holding)
- Pressure: 0.8MPa
- Radiation Aging: 800KGY
- Atmosphere: Steam

Test Results
- Leak was occurred around 320°C of inlet temperature after 14hrs
- Temperature of outlet sealing material was estimated around 180-200°C
- Temperature/Pressure Profile: Right figure
- State of test piece after the test: Next photograph

LEAK FLOW CHARACTERISTICS

Test Condition
- Test Piece: Damaged low voltage modules (various types)
- Flow Gas: Air
- Temperature: RT-200°C (gas/test piece)
- Inlet Pressure: 0.11-0.9MPa
- Outlet Pressure: 0.1MPa

Test Results
- Leak flow rate is proportional to a square root of the differential pressure under low pressure difference
  - Incompressible flow
  - At higher pressure difference (more than 0.1MPa), the flow rate is nearly proportion to the differential pressure
    - Compressible flow

AEROSOL TRAPPING TEST

Test Condition
- Test Piece: Low Voltage Module (Cable; 0.9mm²(T/C))
- Temperature/Pressure: 100°C/0.13MPa
- Radiation Aging: 800KGY
- Atmosphere: Air
- Aerosol/Concentration: CsI/100mg/m³
- Particle diameter: ~1micron(AMMD)

Test Results
- Inlet and Outlet aerosol concentrations were basically constant during test period
- Decontamination factor (DF*) was estimated more than 100
- Leak flow rate was decreased exponentially during the test period due to aerosol accumulation in the leakage path
  - Note: DF = Inlet Conc./Outlet Conc.
# SUMMARY OF TEST RESULTS

(Leak Criteria Test / Aerosol Trapping Test)

<table>
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<td>Cooling period (230-RT)</td>
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<td>Yes</td>
<td>at 320</td>
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<td>100mm²</td>
<td>Yes</td>
<td>Yes</td>
<td>at 280</td>
<td>100-1000</td>
</tr>
<tr>
<td>2mm²</td>
<td>Yes</td>
<td>Yes</td>
<td>at 290</td>
<td>10-30</td>
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<tr>
<td>High Volt. Module</td>
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<td>K-Type (1)</td>
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<td>No</td>
<td>Cooling period (230-RT)</td>
<td>&lt;100</td>
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<td>Yes</td>
<td>at 330</td>
<td>10-20</td>
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<td>K-Type (3)</td>
<td>Yes</td>
<td>Yes</td>
<td>at 360°</td>
<td>10-30</td>
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</table>

Pressure: 0.8MPa (+ 0.4MPa)

**CONCLUSION**

- All containment penetrations were not leaked under the accident management conditions.
- The failure criteria of the containment penetrations were evaluated and the criteria depend strongly on the ambient temperature rather than the pressure.
- The damaged containment penetrations behave as an aerosol filter, and the decontamination factor (DF) was expected to be more than 10.

*NUPEC*
10. Session VIII

FCI Simulation

Chairperson: K. H. Bang (Korea Maritime Univ.)
Co-chairperson: H. Okada (NUPEC)
10.1 Development of TRACER-II and Application to In-Vessel FCI's

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Tel: +82-51-410-4365, Fax: +82-51-404-3986

Abstract

A vapor explosion is a physical process in which very rapid energy transfer occurs between a hot liquid and a volatile, colder liquid when the two liquids come into a sudden contact. During the course of a severe accident in present day nuclear fission reactors, such a contact of molten fuel and coolant is likely, thus an attention has been paid to the potential hazard of a large-scale vapor explosion such as containment breach by missile generation and dynamic pressure impulse. This paper presents the current status of the computer program TRACER-II development and its application to in-vessel FCI's. The code contains a complete description of vapor explosions: mixing and propagation.

Recently, reactor safety analysts have paid attention to the accident management strategy of retaining the molten degraded core within the reactor vessel, possibly in the vessel lower head. In so doing, the complicated ex-vessel phenomena could be avoided. However, the integrity of vessel lower head may have a potential threat in two ways: thermal attack by the hot molten core and mechanical attack from an in-vessel steam explosion. The fuel-coolant interactions inside the reactor vessel is, therefore, one of the key issues in this in-vessel retention of core melt and needs to be properly modeled and analyzed. The vessel integrity needs to be evaluated against the dynamic pressure impulse from the potential steam explosion. Also the fuel-coolant mixing and melt breakup determines the initial conditions of the thermal interaction between core melt and lower head. This paper reports a preliminary modeling of in-vessel FCI's using TRACER-II and discusses relevant key issues.
Development of TRACER-II and Application to In-Vessel FCI's

K.H. Bang
School of Mechanical Engineering, Korea Maritime University

Workshop on Severe Accident Research held in Japan
Yokohama, Japan
October 6-8, 1997

BACKGROUND AND MOTIVATION

- Fuel-Coolant Interactions (Vapor explosions)
  - Mixing: melt breakup, multi-phase thermal-hydraulics
  - Triggering: local interaction, vapor film collapse
  - Propagation/escalation: fine fragmentation, shock wave
  - Expansion: mechanical work, structural integrity

- Important, but unresolved issue in reactor severe accidents
  - In-vessel retention of core melt: reactor lower head integrity, initial conditions for melt-vessel thermal interaction
  - Ex-vessel coolability: deep, flooded reactor cavity

OUTLINE

- Introduction
- TRACER-II mathematical model
- Analysis of FARO test
- Analysis of KROTOS test
- Preliminary results of LAVA analysis
- Conclusions

MATHEMATICAL MODEL

- Mass, momentum, energy equations: four fluids, three fields (melt, debris, coolant liquid and vapor), 2-D Eulerian
- Melt breakup and fragmentation models
- Interfacial momentum exchange
- Interfacial heat transfer
- Reference models: PM-ALPHA, ESPROSE, CHYMES, CULDESAC, IDEMO, TEXAS, IFCI
**Fuel-Coolant Mixture**

(ALPHA test, JAERI)

**Energy equations**

\[
\frac{\partial}{\partial t} (\rho \varepsilon) + \nabla \cdot (\rho \varepsilon \mathbf{v}) = -\nabla \cdot \mathbf{q} - \rho \mathbf{f}_c
\]

\[
\frac{\partial}{\partial t} (\rho \mu) + \nabla \cdot (\rho \mu \mathbf{v}) = \nabla \cdot \mathbf{E} - \rho \mathbf{f}_c + \mathbf{f}_b
\]

**Phase change**

\[
J = \frac{1}{\rho_a}(K_a(T_a - T_c) + K_c(T_c - T_s))
\]

**Continuity equations**

\[
\frac{\partial}{\partial t} (\rho \varepsilon) + \nabla \cdot (\rho \varepsilon \mathbf{v}) = -\nabla \cdot \mathbf{q} - \rho \mathbf{f}_c
\]

**Momentum equations**

\[
\frac{\partial}{\partial t} (\rho \mathbf{v}) + \nabla \cdot (\rho \mathbf{v} \mathbf{v}) = -\nabla \mu + \nabla \cdot \mathbf{q} + \mathbf{f}_c + \mathbf{f}_b - \rho \mathbf{f}_c
\]

**Interfacial area transport of fuel droplets**

\[
\frac{\partial}{\partial t} (\rho_f) + \nabla \cdot (\rho_f \mathbf{v}) - L_i = T_w - T_f
\]

\[
F_w = -C_w \sqrt{\rho_f \rho_i (T_w - T_i) L_i}
\]

\[
C_w = 0.245 \text{ (Young)}
\]

\[
F_p = \lambda_p \rho_f \sqrt{\rho_i \rho_f (T_w - T_i) L_i}
\]

\[
C_{fr} = 1
\]
Interfacial Exchange Laws

- Flow regime: (Theofanous et al.)
  - fuel \( \alpha_f < 0.3 \) dispersed fuel droplets
  - fuel \( \alpha_f > 0.3 \) porous media
  - liquid-vapor \( \alpha < 0.3 \) bubbly flow
  - \( 0.3 < \alpha < 0.7 \) churn-turbulent
  - \( \alpha > 0.7 \) droplet

- Heat transfer:
  - bubble/droplet convection
  - fuel \( \alpha < 0.7 \) film boiling/radiation
  - fuel \( \alpha > 0.7 \) convection/radiation

Momentum Exchange

- Liquid Continuous \( (\alpha < 0.3, \alpha_f < 0.7) \)
- Vapor Continuous \( (\alpha < 0.3, \alpha_f > 0.7) \)
- Porous Bed with Fuel \( (\alpha > 0.7) \)

Heat Transfer

- Liquid Continuous Phase \( (\alpha < 0.7) \)
- Vapor Continuous Phase \( (\alpha < 0.7) \)

FARO-LWR Experiment

- Molten corium interactions with water
- FARO: direct current corium melter, < 200 kg, 3000°C
- TERMOS: interaction vessel, 0.71 m ID, 3.58 m II, 1-2 m water depth, 10 cm pour dia.
- \( \text{UO}_2 + \text{ZrO}_2 + \text{Zr} \)
- High initial pressure: \(-5 \text{ MPa}\)
- 20 tests
- L-14 test: 125 kg, 2 m water, 5 MPa
Predicted test vessel pressure
Predicted average fuel droplet diameter

Predicted melt downward progression
Predicted average vapor volume fraction

Predicted fuel volume fraction distribution at \( t=0.8 \) s
Predicted vapor volume fraction distribution at \( t=0.8 \) s

Effect of Film Boiling Heat Transfer
Predicted test chamber pressure using fixed fuel diameter
KROTOS Experiment

- Explosion tests at 0.1 MPa
- 10/20 cm ID, 1.1 m water pool
- Melts:
  - Tin, 7.5 kg, 1373 K
  - Al₂O₃, 1.5 kg, 2473-3073 K
  - Corium, 3-5.5 kg, 3073 K
- Melt jet: 30 ~ 50 mm dia.
- Trigger by 12 MPa gas chamber
- >50 tests performed
- KROTOS 28: Al₂O₃, 1.45 kg, 2670 K, 1.1 m 87°C water

KMU Heat Transfer Lab

KROTOS 28

- Predicted fuel volume fraction at 1.8 s of mixing
- Predicted vapor volume fraction at 1.8 s of mixing

KMU Heat Transfer Lab

Predicted explosion pressure traces (cr=1.0)

Predicted explosion pressure traces (cr=4.0)

KMU Heat Transfer Lab

Predicted explosion pressures of KROTOS-28 using uniform fuel and vapor fractions (ωₚ=0.08, φₚ=0.04)
CONCLUSIONS

- TRACER-II has shown a capability of both mixing and propagation calculations of fuel-coolant interactions.
- A better deterministic breakup model is needed for mixing calculation (empiricism of $C_w$).
- The current model seems to have a deficiency in predicting radial momentum gain of fuel droplets, which results in underprediction of KROTOS explosion data (venting effect).
- To enhance the model's prediction capability, each individual relationship needs to separately verify through a careful experimentation (complex phenomena involved in integral test data).
10.2 Development of Computer Code for Expansion Stage in Vapor Explosion

Ken-ichi Sato, Masaki Matsumoto, Kazuhide Takamori*, Akihiko Minato*
Hitachi Works, Hitachi Ltd.
*Power and Industrial Systems R&D Division, Hitachi Ltd.

[ABSTRACT]

A computer code for numerical analysis of two-dimensional compressible flow during expansion stage in vapor explosion has been developed for safety assessment of severe accidents of light water reactors. The field equation is based on a homogeneous compressible model considering distributions of water, steam and core debris. The solution technique is finite volume method using non-staggered mesh scheme with the second order accuracy.

In the initial condition, high temperature core debris spread in mixing area of water pool. Before vapor explosion occurred, core debris was covered with entrained air and heat transfer from the core debris to water was not so large. Contact between core debris and water took place somewhere in the mixing region, and pressure pulse appeared due to rapid evaporation. The pressure pulse propagated in the mixing region and contact between core debris and water began in the pressure pulse front. Transferred heat from core debris to water during $\tau$ after debris-water contact was assumed to be expressed as:

$$Q(\tau) = Q_{\text{total}} \{1 - \exp(-\tau / \tau_0)\}$$

$Q_{\text{total}}$ is the total contribution from stored heat in core debris and $\tau_0$ is a time constant of evaporation.

ALPHA experiments of melt-coolant interaction performed by JAERI were analyzed with the present computer code. The calculated pressure history in the mixing region was compared with the data. The predicted peak and width of pressure pulse were comparable with the data when the contributed fraction of stored heat was 0.5 and the evaporation time constant $\tau_0$ equaled to 2.5 ms.
Development of Basic Numerical Methods for Complex Fluid Dynamics in Severe Accidents

Vapor Explosion
Mixture of debris, water and steam

Molten Debris Spread

Hydrogen Detonation

Unburned gas
Wave front
Burned gas

Features of the Solution Method

<table>
<thead>
<tr>
<th>Discretization</th>
<th>Finite Volume Method</th>
</tr>
</thead>
<tbody>
<tr>
<td>Time Integration</td>
<td>Explicit Euler Method</td>
</tr>
<tr>
<td>Mesh Arrangement</td>
<td>2D Cartesian and Non-Staggered Mesh</td>
</tr>
<tr>
<td>Accuracy</td>
<td>2nd Order Central Difference Scheme</td>
</tr>
<tr>
<td>Other Features</td>
<td>Non-oscillatory Transport Calculation</td>
</tr>
<tr>
<td></td>
<td>Stable Treatment of Large Density Difference between Gas and Liquid (density ratio: up to 10,000)</td>
</tr>
<tr>
<td></td>
<td>Sharp Interface Tracking</td>
</tr>
<tr>
<td></td>
<td>Godunov Scheme for Compressible Flow Dynamics</td>
</tr>
</tbody>
</table>

Mixture Density vs. Pressure Relationship

Pressure P is determined to satisfy the following equation:

$$ P = \frac{1}{\tau_0} \exp\left( -\frac{t}{\tau_0} \right) \Gamma_{\text{total}} $$

Capacity of Control Volume = Gas Volume (P) + Water Volume (P) + Debris Volume (P)

Density of Gas =
  - Air Density (for low pressure)
  - Vapor Density (for high pressure)

Phase Change Rate =
  - $ \frac{1}{\tau_0} \exp\left( -\frac{t}{\tau_0} \right) \Gamma_{\text{total}} $ (from water to gas)

$ \tau_0 $: Time constant (s)
$ \Gamma $: Time after trigger (s)
$ \Gamma_{\text{total}} $: Total evaporation mass (kg/m$^3$)
Stable transport calculation method alleviating numerical diffusion

Stable transport calculation

\[ \phi_{\text{min}} < \phi < \phi_{\text{max}} \]

Unstable transport calculation

\[ \phi_{\text{min}} < \phi < \phi_{\text{max}} \]

Stabilization by introducing minimum upwind difference scheme for overshoot/undershoot disappearance

**Density dependent pressure gradient**

(Stable calculation for density ratio up to 10,000)

Distance

**Additional Velocity by Checker Board Pressure Field**

Finite volume, W

Finite volume, C

Finite volume, E

Pressure

Distance

Velocity at the finite volume boundary related with mass transfer: \( \Delta u \)

\[ \Delta u = \frac{\delta P}{\alpha P} \]

a: Sound velocity (m/s)

\( \rho \): Density (kg/m^3)

Equivalent to Godunov method for compressible flow calculation

**ALPHA Experiments of Vapor Explosion** (Case STX16)

Air

Containment vessel (50m^3)

Aluminum/iron oxides thermite (20kg)

Acryl tank

Water

Calculation Conditions

Tank
2D Cartesian mesh
(41 × 82 = 3362)

Height, 0.9m
Width, 0.45m

Calculation Conditions

Air
Tank
2D Cartesian mesh
(41 × 82 = 3362)

0.25m
Mixed area
(Debris/air
/Water)

0.5m
Water level,
0.6m

Simplified conditions
(1) 2D coordinate
(2) No containment vessel
(3) Smaller water level
(4) No gravity

Comparison between Calculated and Observed Pressure Pulse

Pressure (MPa)

0 10

Time (ms)

0 1 2 3


Conclusions

1. Numerical method for two-dimensional compressible flow during expansion stage in vapor explosion was developed.

2. Calculated pressure history in water pool was compared with the data of ALPHA experiment. The predicted pressure peak was smaller and pulse width was larger than observed ones.

3. The deviation from the data may be attributable to 3D effect of water inertia. Uncertain parameters such as initial void fraction, depth of mixing region, evaporation time delay should be studied.
10.3 Development of FCI Simulation Code JASMINE
(1) Premixing

K. Moriyama, Y. Yang, N. Yamano, Y. Maruyama,
H.-S. Park and J. Sugimoto

Severe Accident Research Laboratory
Japan Atomic Energy Research Institute

JASMINE is an FCI simulation code developed in the framework of ALPHA Program at JAERI. Two modules, JASMINE-pre and -pro, have been developed separately for the simulation of the premixing and explosion processes, respectively. Whole process of steam explosions will be analyzed using the two modules in tandem. The developmental work of the -pre module started in 1994 and presently the model assessment and improvement is under way.

In order to assess the capability of the inter-phase friction and heat transfer models, the QUEOS experiments by Meyer and Schumacher (1996) were referred. In this series of experiments, a cloud of cold and hot particles was dropped into a water pool initially kept at saturation temperature. The material, temperature and size of the particles were systematically changed. In simulating the experiments using zirconia particles, JASMINE-pre reproduced the behavior of the cold particles relatively well, but badly underestimated the sinking velocity of hot particles. Large scale vapor channel experimentally observed was not seen in the simulation result. This result indicated that present modeling of the water-vapor friction and the flow regime needs further examination and improvement to correctly handle the strong counter current flow which is likely to occur and have a strong influence in the premixing process. The steam generation volume was reproduced well with the initial water temperature set at 372.5K as reported, although the result was very sensitive to the initial water temperature setting. The sensitivity of the sinking velocity and steam generation on the drag coefficient for the particles was not strong.

A simple melt jet model was installed in the melt field to handle the conditions where the melt is introduced as a jet. Friction model based on a rough pipe correlation (Schlichting 1979) and a Kelvin-Helmholtz type model for the droplet entrainment from the jet column (Esptein-Fauske 1985) were used in the central nodes where the jet column resides. Simulation of the large scale corium jet-water interaction experiment FARO-L14 at JRC Ispra in 1994 was performed to assess the models. The qualitative behavior of the pressure history was significantly improved by using the jet model compared with the case using the originally installed particle breakup model. Also the results became much less sensitive to the empirical constant in the model. However, leading edge progression, energy distribution to water and vapor etc. were not reproduced well due to immaturity of the water-vapor heat transfer model and also due to the present framework of the code, that is the jet column and entrained droplets are handled in a single field.

As a future plan, extension of the code framework including the introduction of Lagrangian melt component and modification of the gas field to handle non-condensible gas are under consideration.
Development of FCI Simulation code JASMINE (1) Premixing

K. Moriyama, Y. Yang, N. Yamano, Y. Maruyama, H.-S. Park and J. Sugimoto
Severe Accident Research Laboratory
Japan Atomic Energy Research Institute

Presented at the Workshop on Severe Accident Research held in Japan (SARI-97)
Oct. 7, 1997, Pacifico Yokohama

JASMINEx development

Simulation of FCIs by integrating basic physical models on the framework of multi-field thermohydraulic code
Assessment of the influence of FCIs in real scale accident conditions

* Two modules “-pre” and “-pro” are developed in parallel.
-pre: premixing of steam explosions, mild FCIs
-pro: propagation/expansion of steam explosions

OUTLINE

- Introduction
- Model description
- Assessment of the model capability
  Simulation of QUEOS experiments
  Simulation of FARO-L14 experiment
- Summary of present status & future plan

INTRODUCTION

Purpose of JASMINE development
- Simulation of FCIs by integrating basic physical models on the framework of multi-field thermohydraulic code
- Assessment of the influence of FCIs in real scale accident conditions

Schedule
Premixing module: (1994–)
1994– Development of the first version
1995– Debug, Test calculations
1996– Model assessment / improvement
  Participation in ISP-39 (analysis of FARO-L14)
1997– Modification of the code framework

Propagation module: (1995–)
1995– Development of the first version
1996– Debug, Test calculations
1997– Modification of solver, Model assessment

INTRODUCTION (cont’d)
MODEL DESCRIPTION

Summary of the models

<table>
<thead>
<tr>
<th>Phase change of coolant</th>
<th>Propagation module</th>
</tr>
</thead>
<tbody>
<tr>
<td>- evaporation/condensation, heat balance at water-vapor interface</td>
<td>- evaporation/condensation, heat balance at water-vapor interface</td>
</tr>
<tr>
<td>Friction</td>
<td>- drag force in dispersed flow (Ishii-Zuber 1979)</td>
</tr>
<tr>
<td>- rough pipe friction correlation (Schlichting 1979) for jet column</td>
<td>- drag force in dispersed flow (constant drag coefficient)</td>
</tr>
<tr>
<td>- constant friction factor</td>
<td></td>
</tr>
<tr>
<td>Heat transfer</td>
<td>- radiation (Stefan-Boltzmann law)</td>
</tr>
<tr>
<td>- conduction</td>
<td>- conduction</td>
</tr>
<tr>
<td>Breakup/Fragmentation</td>
<td>- generation of fine fragments from melt, entrainment of interaction coolant from water (fine fragment dia. is assumed) (Yuen et al. 1994)</td>
</tr>
<tr>
<td>- melt surface area convection eq.</td>
<td>- conduction</td>
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<tr>
<td>- source term by particle breaking model (Davis-Young 1994, Fitch-Erdmann 1987)</td>
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</tr>
</tbody>
</table>

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ASSESSMENT OF MODEL CAPABILITY

- Simulation of QUEOS Experiments
  (Meyer & Schumacher 1996, FZKA-5612)
  Cold/hot particle cloud-water interaction
  → Capability of friction & HT models were examined.

- Simulation of FARO-L14 Experiment
  (Magallon & Hohmann 1995, NUREG/CP-0142 v.3)
  Corium melt jet-water interaction
  → Overall capability of the code, effectiveness of the jet model were examined.

---

QUEOS Simulation

Experimental conditions

<table>
<thead>
<tr>
<th>particle temp.</th>
<th>particle size</th>
<th>particle mass</th>
<th>common conditions</th>
</tr>
</thead>
<tbody>
<tr>
<td>No.5 cold</td>
<td>5mm</td>
<td>7.0kg</td>
<td>- particle material: zirconia</td>
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<tr>
<td>No.6 cold</td>
<td>10mm</td>
<td>7.0kg</td>
<td>- conditions at water surface contact</td>
</tr>
<tr>
<td>No.9 1300K</td>
<td>5mm</td>
<td>7.0kg</td>
<td>- velocity: 5.12m/s</td>
</tr>
<tr>
<td>No.7 1300K</td>
<td>10mm</td>
<td>7.0kg</td>
<td>- volume fraction: 0.17</td>
</tr>
<tr>
<td>No.10 1800K</td>
<td>10mm</td>
<td>6.3kg</td>
<td>- dia. of cloud: 0.18m</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- condition of water pool</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- temperature: 99.5°C</td>
</tr>
</tbody>
</table>

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QUEOS Simulation (cont'd)

Analytical conditions

- 2D cylindrical grid: 5cm, 8x30, R=0.409m (cross section was matched)
- Water temperature 372.5K-373.0K
- Drag coefficient for particles 0.45-0.22
- Start just before water-particle cloud contact to minimize the influence of numerical diffusion

Compared variables

- Total steam generation volume
- Snapshots of material distribution

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**Total steam generation volume**

- Initial water temperature: $T = 372.5K$
- Graph showing steam generation volume over time.
- Comparison between calculation and experiment.
- Key: Calculation - no9-cal, no7-cal, no10-cal, no9-cal ($T = 373K$), Experiment - no9, no7, no10.

**Leading edge progression**

- Graph showing height of the leading edge over time.
- Comparison between experiment and calculation.
- Key: Experiment - no5: cold, 5mm, no6: cold, 10mm, no7: 1300K, 5mm, no9: 1300K, 10mm, no10: 1600K, 10mm.
- Calculation - no5-cal, no6-cal, no7-cal, no9-cal, no10-cal.

**Effect of drag coefficient on the leading edge progression**

- Graph showing the impact of different drag coefficients ($C_D$).
- Key: no7(cal, $C_D = 0.45$), no6(cal, $C_D = 0.22$), no7(exp).

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Result of QUEOS Simulation

- Total steam generation volume qualitatively & quantitatively agreed with experimental results with initial $T_0$ set at 0.5K subcool, although sensitive to $T_1$ setting.
- Sinking velocity of particles was underestimated. Deviation was exaggerated in hot cases.
- Steam generation & sinking velocity were not sensitive to the drag coefficient for particles in the range of 0.22-0.45.
- What caused velocity underestimation?
  - Strong upward steam flow likely to be responsible. Large scale vapor chimney seen in the experiments were not reproduced in simulations. Flow regime modeling is not enough?
  - Unphysical velocity reduction due to numerical diffusion is another reason?
FARO-L14 Simulation

Experimental conditions
- Melt: corium jet (dia.=92mm, 125kg, 3073K, \( V_{in} = 5\text{m/s} \))
- Coolant: water (5.1MPa, 536K (sat.), 623kg (level 2.05m))
- Vessel: I.D. 0.71m, 2.063m³

Analytical conditions
- Grid: 5x39 2D cylindrical (10cm grid)
- Wall: slip / adiabatic
- Inlet: controlled velocity, constant volume fraction/temp.
- breakup model*: particle breakup / jet breakup
  * used with simple assumption on the agglomeration on the vessel bottom

Pressure history

Jet leading edge position
Result of FARO Simulation

- Pressure history was best matched with exp. result by adjusting empirical factors in the breakup models.
  (Particle model: $C_{PB} = 0.577$ / Jet model: $C_{EF} = 5 \times 10^{-4}$)
- The sensitivity on the empirical factor was significantly weakened by using jet breakup model.
- The jet leading edge progression was qualitatively improved by the jet model, but quantitatively underestimated. Limitation due to the accommodated both jet and droplets in one field.

SUMMARY OF PRESENT STATUS & FUTURE PLAN

- JASMINE-pre can simulate the typical premixing systems involving water, vapor and corium melt with 3-field model. Particle and simplified jet model are available for breakup evaluation.
- Constitutive model assessment by simulation of variety of experiments (cold/hot particles, cold/hot jet) are under way.
- Problems & limitations in the present modeling:
  - Melt jet is handled in the same field with melt droplets
  - Vapor/water friction and heat transfer model still need improvement to handle high vapor generation, high subcooling etc. Interface area tracking is necessary also for water-vapor system?

- Future plan:
  - Modification of steam field to handle non-condensible gas
    (It was experimentally found that $H_2$ generation has a non-negligible effects)
  - Lagrangian component for melt
    (easy to accommodate variety of physics related to the melt: e.g. surface temperature drop, oxidation, jet, debris bed formation)
JASMINE-PRO is a FCI code developed for the simulation of the propagation stage of steam explosion. This code has been developed at JAERI from 1996. At first, the explicit scheme was used. However, this scheme often caused numerical instabilities when there is a sharp spatial changes in pressure. Recently, a semi-implicit scheme has been introduced in order to obtain a more stable solution. The new version of the code is introduced in this presentation.

JASMINE-PRO is a 2 dimensional multi-component code. In order to find the best solution scheme, several schemes were included in the code. Explicit, semi-implicit (SMAC) and implicit schemes (SIMPLE) can be chosen as an option. The solution procedure is arranged as the SIMPLE solution scheme. A series of verifications has been performed. The verification includes calculation of 1D shock tube, nozzle flow, and boiling in pipe. The former two cases are associated with the propagation of pressure, which is an essential aspect in the propagation process, and the last case is associated with heat transfer and evaporation. These verifications showed that JASMINE-PRO code provides the same results as those calculated by other codes with either solution schemes. The calculation for the propagation process in a steam explosion has begun now. The calculations when the semi-implicit or implicit scheme is employed, the code has shown the ability to get a stable and plausible results for a postulated propagation process. However, since the result still not be verified, it will not be shown here. Further verification with the experiments is planned.

In this presentation, the design and structure of the code and design consideration as well as the models used in the code will be introduced. The three cases of the 1D verification calculation are presented. A calculation of the propagation stage is also shown.

The next plan for the code is to verify it with existed experiments. Various models and coefficients in the constitutive relationships will also be confirmed with the experiment results.
Development of FCI Simulation
Code JASMINE (2)

--- Propagation Module ---

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SARJ-97, Oct.6-8, 1997

OUTLINE of the Presentation

INTRODUCTION
   code development history

PROPAGATION MODULE
   concept, code structure, components,
   equations, solution method, constitutive
   relationships, schemes...

VERIFICATION CALCULATION
   Shock tube, nozzle flow and heat in pipe

CONCLUSIONS

History of JASMINE Code Development

Two parts:

JASMINE-PRE: for premixing stage in steam explosions

1994 ~ premixing
   + jet-breakup (in progress)

JASMINE-PRO: for propagation stage in steam explosions

ver. 1 1996 ~ explicit scheme for time (stopped)
ver. 2 1997 ~ semi-implicit scheme for time (in progress)

Outline of Jasmine-Pro Ver.2

Developed from 1997.6 ~

Code design:
   2D, multi-phase (5-phase at current stage)
   solution method: SMAC/SIMPLE

Time: semi-implicit / implicit;
   pressure propagation (0.1-10 ms)
   time step: $10^{-4} \sim 10^{-3}$ ms

Space:
   up-wind differencing scheme; staggered grid;
   TVD scheme (under planning)
   fine grids, (mm cm)
CONCEPT OF PROPAGATION

major phenomena:
• pressure propagation
  shock waves...
• melt fragmentation
• rapid evaporation
• multi-phase interactions

COMPONENTS AND EXCHANGES

INTERFACE
Steam
Water
Melt droplet
Fragment
Interaction Coolant
- Small part of coolant participates in 'explosion'
- Equilibrium two-phase fluid is assumed

Solution of System Equations

In each time step:
[Diagram showing flow of solution]

SIMPLE
Equations of State
Mass Equations/ Unity of Volume fraction
Energy Equations
Momentum Equations
Pressure Correction Equation
Velocity Modification

Convergence

 constituents in JASMINE-PRO

Heat Transfer Models: d: dispersed, c: continuous fluid

\[ h_{cd} = \alpha_d A_d h_{\text{con}} + (1-\gamma_d) \alpha_d A_d h_{\text{cond}} \]

where:
- convection part:
  \[ h_{c,d\rightarrow c} = \frac{\lambda}{d_d} (2 + 0.6 K e^{1/2} r^{-1/3}) \]
  \[ K_{c,d} = \frac{B_{e,d} d_{c,d}^{1/3} (\gamma_{c,d} - \gamma_d)}{\mu_c} \]
- conduction part:
  \[ h_{c,d\rightarrow c} = 2 \frac{\lambda_{d\rightarrow c}}{d_d} \]

Radiation (F = 0.8):

\[ h_{\text{rad}} = F \sigma_{\text{sb}} \frac{T_m^4 - T_i^4}{T_m - T_i} \]
### Fragmentation Models

- Hydrodynamic fragmentation model (Yuen & Theofanous):

\[
\Gamma_f = \frac{6 \sigma_m}{\pi d_m^2} \frac{dM}{dt}; \quad \frac{dM}{dt} = \frac{\pi d_m^2 |v_m - v_f|}{6 t^*} (\rho_m \rho_f);
\]

\[
t^* = 13.7 \text{Bo}^{1/4}; \quad \text{Bo}(t) = \frac{3 C_s \rho_d d_m(t) (v_m - v_f)^2}{16 \sigma};
\]

- Interacting coolant production \( f_e = 1.0 \)

\[
\Gamma_f = f_e \Gamma_f \frac{\rho_i}{\rho_m}
\]

### Verification 1: Shock Tube

**Rigid Wall**

- **AIR**
  - \( p = 3 \text{ bar} \)
  - \( \rho = 1 \text{ kg/m}^3 \)

- **AIR (99% by Volume) + water (1% by Volume)**
  - \( p = 1 \text{ bar} \)
  - \( \rho = 1 \text{ kg/m}^3 \cdot \text{AIR} + 1000 \text{ kg/m}^3 \cdot \text{Water} \)

**Constitutive models**

\[
f_1 = -f_2 = c_f \rho_i \gamma_b |u_1 - u_2| (u_2 - u_1)
\]
**Verification 2: Nozzle Flow**

Constitutive models

\[ f_{12} = c_r r_2 \rho_1 (u_2 - u_1) \]

Inlet boundary conditions:

- \( W_1 = 1.5 \text{kg/s.m}^2 \)
- \( W_2 = 0.5 \text{kg/s.m}^2 \)

---

**Volume Fraction of Liquid**

**Velocity of \( u_i \) (m/s)**

**Velocity of \( u_i \) (m/s)**

**pressure (N/m²)**

**Nozzle Flow Case 4**

(c=0.6, P=1000 Pa, Part=0.5)
Nozzle Flow Case 5
\(c=0.0; p_2=10.0; p_{exit}=0.5\)

Nozzle Flow Case 8
\(c=1.0E3; p_2=10.0; p_{exit}=0.5\)
CONCLUSIONS

- For the conditions in the propagation stage of a steam explosion, numerical instabilities easily occur when an explicit difference scheme is used.
- To obtain a more stable solution, a new version of JASMINE-pro code based on semi-implicit/implicit schemes (SMAC/SIMPLE) is under development.
- Test calculations were performed to verify the ability of the code by simulating
  - Pressure shock wave behaviors in the shock tube and nozzle.
  - Boiling phenomena in a tube.
CONCLUSIONS (conti.)

- Most of cases showed good agreements with the results of other codes.
- Other constitutive relations, such as a drag, etc. will be verified in the next step.
- Fragmentation calculation will be performed in the next step.
11. Session IX

Computer Simulations

Chairperson: K. Sugiyama (Hokkaido Univ.)
Co-chairperson: M. Kajimoto (NUPEC)
NUCLEAR REACTOR THERMAL-HYDRAULIC ACTIVITIES AT MINT TRIGA REACTOR

Mohammad Suhaimi KASSIM, Adnan BOKHARI
Malaysian Institute for Nuclear Technology Research (MINT)
http://www.mint.gov.my
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Paper to be presented at:
Eighth International Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-8) Kyoto, September 30 - October 4, 1997
And
Workshop on Severe Accident Research in Japan, 1997 (SARJ-97)
Yokohama, October 6-8, 1997.

ABSTRACT
MINT TRIGA Reactor is a 1-MW swimming pool nuclear research reactor commissioned in 1982. In 1993, a project was initiated to upgrade the thermal power to 2 MW. The IAEA assistance was sought to assist the thermal-hydraulic analysis required to safely upgrade the cooling system. The IAEA has provided expert assistance to introduce the PARET code in the calculations. Later, the help of a local university for assistance in flow modeling using the FLUENT code was sought. In 1996, this writer underwent a six-month training in the use of the RELAP5 code. However, in the middle of 1997, MINT has decided to change the scope of the project to safety upgrading of the MINT Reactor.

INTRODUCTION
MINT TRIGA Reactor is a 1-MW swimming pool TRIGA Mark II nuclear research reactor commissioned in 1982 [Fig. 1]. The reactor is located in Bangi, about 35 km south of Kuala Lumpur, the capital of Malaysia. After almost 15 years of operation and maintenance, a proposal was forwarded to the then Nuclear Energy Unit management to upgrade the thermal power of the reactor from 1 MW to 2 MW so as to better meet the ever-increasing demand of the reactor.

2. THE POWER UPGRADING PROJECT
In 1993, a project was initiated to upgrade the thermal power to 2 MW. The Malaysian Government has approved a budget of about US$1.0 million dollars for this project. The IAEA assistance was sought to assist the thermal-hydraulic analysis required to safely upgrade the cooling system. The IAEA has approved a technical assistance consisting of fellowship training, expert missions and scientific visits for a two-year period from 1995 to 1996. The first expert mission was a two-week visit by Mr. Byung Jin Jun of KAERI to outline the activities that need to be done to complete the power-upgrading project[1]. Later a neutronics expert from Slovenia was sent to MINT to assist with various neutronics codes such as WIMS-D/4, EXTERMINATOR, ORIGEN, ANISN, TRIGAP, XSWOUT and TRISTAN [2]. Fig. 2 shows some of the activities that has been done and ongoing.
3. USE OF PARET CODE

The IAEA sent Professor Tunc Aldemir of the Ohio State University to introduce the PARET code in the calculations. The PARET code is a one-dimensional, coupled thermal-hydraulic and point-kinetics code, which was originally developed for the analysis of SPERT-1 transients and later adapted for the analysis of transient behaviour in research reactors [3]. Due to its relative simplicity, it is widely used especially in research reactors with limited computational capabilities.

The PARET code is normally used in forced flow thermal-hydraulic calculations. Originally, however, in this mission, Ibrahim et al. has succeeded to utilize the PARET code to simulate natural convection flow calculation [4,5]. This capability is useful in assessing the feasibility and safety of various power increase options while maintaining core-cooling mode. Fig. 3 shows a comparison of some of the experimental and PARET results. The neutronic characteristic of the core have been determined using the WIMS-D/4 [6] code and the TRIGAP code[7].

4. USE OF FLUENT CODE

In 1995, the staff seeks the help of a local university for assistance in flow modeling using the FLUENT code. A 5-day workshop was run and 8 participants from MINT were introduced to preparing the input and analysing the results. However, no input specifically related to the MINT reactor was prepared. Later this year, MINT will procure the FLUENT code so that MINT personnel may use the code to do flow calculations for both the MINT Reactor and the MINT radioactive waste incinerator pilot plant. For the MINT reactor, the various in-pool components will be modeled and analysed using FLUENT code [8].

5. USE OF RELAP5/MOD3.2 CODE

In 1996, this writer underwent a six-month training in the use of the RELAP5 code. During the stint, this writer was able to install and compile the RELAP5 code on a SGI Indy workstation at the Mechanical Engineering Department of the Ohio State University. He was also able to prepare the input for the core section of the MINT TRIGA Reactor.

Further work at MINT has to be postponed until the RELAP5 code is obtained from the USNRC. It is envisaged that the complete cooling system will be modeled including both the recently refurbished secondary cooling system and the current primary cooling system[9].

6. THE SAFETY UPGRADING PROJECT

However, in the middle of 1997, MINT has decided to change the scope of the project to safety upgrading of the MINT Reactor. Fig. 4 incorporates some of the changes to the project schedule considering various technical and personnel limitations.

ACKNOWLEDGEMENT

This writer wishes to thank MINT, the Malaysian Ministry of Science, Technology and the Environment, and of course the Malaysian Government for the opportunity to undertake this reactor upgrading project. Thanks also to the Asian Researcher Invitation Program of JAERI for this opportunity to attend this NURETH-8 conference, this Workshop on Severe Accident Research in Japan in 1997 (SARJ-97).

REFERENCE

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Fig. 6: NPP Reactor Upgrading Activities (1995-2000)
11.2 Use of MAAP in the Assessment of Containment Integrity Following the Severe Accident

J. I. Yun and C. S. Kang, Seoul National University

The purpose of this study is to assess the potential use of MAAP in the evaluation of containment integrity following the severe accident, and to recommend design modifications for its performance improvement against severe accident conditions. Version of 4.0b of MAAP is used for this study, and as a case study, DCH of Kori Unit 1 is evaluated.

For Kori Unit 1, the results of study show that TMLB is the controlling accident in DCH. Under the station blackout situation, when the accidents like PCP seal failure or PORV malfunction occur, the depressurization of primary coolant would not be sufficient enough to actuate the inflow of accumulator injection since the primary system still maintains a substantially high pressure due to lack of feedwater supply to steam generators. DCH could occur with high pressure melt injection if the reactor pressure vessel ruptures during this condition, and could eventually induce the early failure of containment. The core melt would fall into the cavity and interact with coolant or concrete, the pressure of containment atmosphere would build up, and its subsequent bleach could occur. In this case, the main source of non-volatile fission products would come from CCI. The pressure increase due to DCH and CCI is computed along with its fission product releases in this study.

The results of study show that the containment integrity of Kori Unit 1 is actually threatened by DCH following TMLB accidents, which suggests several design modifications for Kori Unit 1 to prevent DCH and reduce CCI. First, in order to design against DCH, it is found that a certain design changes are recommended in the geometry of containment cavity such as outlet cross-sectional area, cavity floor area, and cavity volume. The effects of outlet cross-sectional area are found to be very small on DCH, while the cavity floor area is substantially important in spreading the molten core and reducing CCI. Second, the reduction of release amount of fission products from the vessel is the major concern. The existence of water in the cavity causes an instantaneous pressure increase, but helps to reduce the total amount of release of fission products. The geometry change is also suggested to ensure the sufficient flow of water from sump to cavity since the prompt cooling of corium by the water allows the substantial slowdown of accident propagation within the containment. Third, the early opening of PORV is solicited to allow the reduction of primary coolant pressure under the set-pressure of accumulator injection so that the timely inflow of accumulator injection could prevent the high pressure melt injection.
Use of MAAP in the Assessment of Containment Integrity Following the Severe Accident

Name: J. I. Yun & C. S. Kang
Date: 1997. 10. 08.
Place: Yokohama, Japan

1. The Descriptions of Kori Unit 1

- Containment
  - Large dry type
  - Inner steel containment: dome (hemispherical dome), bottom (torispherical), and cylindrical part
  - Outer reinforced concrete shield building: protect the steel cont. from external turbine missile and adverse atmospheric conditions

- Primary system
  - 2 loops
  - 1 Hot leg, 2 cold legs, 1 S/G, 2 coolant pumps / loop
  - Pressurizer (2 safety valve, 2 relief valve)
  - 2 Accumulator (volume = 56.63 m, 700 psig)

Description of S/G and ESFs

- Steam Generator
  - U tube (material: inconel)
  - 5 safety valves, 1 relief valve
  - New type S/G which will be changed is considered

- Engineered Safety Features
  - Safety Injection system is activated at 12.66 Mpa
  - RHRS is activated at 1.066 MPa
  - Containment spray (upper train only)
  - Cooling Fans
  - Not existed Ignitor
2. The Result of PSA Level 1

+ System Reliability
  - System reliability is very low during the station blackout accident.
  - i.e., the failure frequency of AFWS is $1.1 \times 10^{-2}$ but $1.2 \times 10^{-1}$ at the station blackout.

+ The Frequency of Core Melt
  - Domestic frequency of the offsite power loss = 0.17 yr
    > common frequency = 0.08 yr
  - The frequency due to the station blackout = 54% of the total frequency $8.3 \times 10^{-5}$ yr

The Design Parameter of Kori Unit 1

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Core Heat Output</td>
<td>1728.5 Mw</td>
</tr>
<tr>
<td>Core</td>
<td>121 assemblies, 179 Pins</td>
</tr>
<tr>
<td>Reactor Coolant Pump</td>
<td>89,000 gpm, centrifugal type</td>
</tr>
<tr>
<td>PZR Relief Tank Volume</td>
<td>31.15 m$^3$</td>
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<tr>
<td>PZR Free Volume</td>
<td>28.32 m$^3$</td>
</tr>
<tr>
<td>Cont. Free Volume</td>
<td>1.565E4 m$^3$</td>
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<tr>
<td>Cont. design Pressure</td>
<td>38.7 psig</td>
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<tr>
<td>Cont. structure</td>
<td>Inner steel containment</td>
</tr>
<tr>
<td></td>
<td>Concrete Shield Building</td>
</tr>
</tbody>
</table>

3. Case Study 1: Containment Modeling

+ Previous Model
  - The steel containment is considered to be a steel liner contacted with concrete shield building.

+ Improved Model
  - The steel containment is considered to be a steel liner separated with concrete shield building.
  - Gap resistance between the steel containment with the concrete shield building is considered (gap size = 5 ft).

The Result of Analysis

+ The Response of Containment Building
  - Early pressure and temperature in containment:
    Improved model < Previous model.
  - Peak pressure and temperature:
    Improved < Previous model.
  - The steel containment played an important role in heat sink during the accident early.
  - Late containment pressure and temperature:
    Improved model > Previous model.
  - Gap resistance effect.
4. Case Study II : TMLB

**Assumptions**
- DC power for PZR PORV is limited to half hour from the beginning of accident
- TMLB with pump seal LOCA = 1st case
- TMLB without pump seal LOCA = 2nd case
- The gap resistance is considered
- The used tool for analysis is MAAP ver. 4.0b

**Reference plant : Kewaunee**
- Plant design is similar with Kori unit 1
- The event of Kori 1 is compared with Kewaunee's event
- The cont. model of Kewaunee is previous model

---

**Event Comparison I with Kewaunee : TMLB**

<table>
<thead>
<tr>
<th>Event</th>
<th>Kori unit 1</th>
<th>Kewaunee</th>
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<tbody>
<tr>
<td>Loss of AC Power</td>
<td>0 sec</td>
<td>0 sec</td>
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<tr>
<td>RCS Break Failed</td>
<td>2700 sec</td>
<td>2700 sec</td>
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<tr>
<td>S/G Dry</td>
<td>11982 sec</td>
<td>12787 sec</td>
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<tr>
<td>Core Uncover</td>
<td>13145 sec</td>
<td>14312 sec</td>
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<tr>
<td>Hot Leg Rupture</td>
<td>19935 sec</td>
<td>21098 sec</td>
</tr>
<tr>
<td>RV Bottom Failure</td>
<td>33930 sec</td>
<td>37607 sec</td>
</tr>
<tr>
<td>Containment Failure</td>
<td>219577 sec</td>
<td>244301 sec</td>
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</tbody>
</table>

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**The Results of Analysis**

**The Effect of RCP Seal LOCA**
- All Events are slightly faster by RCP seal LOCA
- Temperature and pressure in containment : not nearly affected by RCP seal LOCA

**The effect of PZR PORV**
- The Operation of PORV is needed to mitigate accident
- Temperature and pressure in containment : decrease by opened PORV

**The Effect of Cavity Floor Area**
- The containment failure was not delayed by increment of cavity floor area in this analysis
The Suggestion

- The Suggestion and conclusion

- DC power for PZR PORV would be supplied > 1 HR (Kori unit 1 <= 0.5 hr)

- The increment of containment pressure by DCH (1–2 % of UO2 is participated) is not detected in this analysis

- The affection of CCI by cavity floor area would be further studied

Department of Nuclear Engineering
AHF LARGE CL LOCA, 0.41806 M2, NO RECIRC

Containment Response: No Gap Effect

KORI TMLB: SEAL LOCA, HOT LEG CREEP RUPTURE

Containment Response
KORI TMLB: SEAL LOCA, HOT LEG CREEP RUPTURE

Containment Response (with RCP Seal LOCA)

Containment Response (Without RCP Seal LOCA)
KORI1 TMLB: SEAL LOCA, HOT LEG CREEP RUPTURE

Containment Response (1.5 Multiplied Cavity Area)

Containment Response (Limited DC Power)
KORI1 TMLB: SEAL LOCA, HOT LEG CREEP RUPTURE

Containtment Response (not limited DC power)
11.3 A Scoping Analysis of Containment Venting at a BWR using THALES-2

J. ISHIKAWA*, K. MURAMATSU* and T. SAKAMOTO**

* Japan Atomic Energy Research Institute
2-4 Shirakata Shirane Tokai-mura, Naka-gun, Ibaraki-ken, 319-11, Japan
** Toshiba Advanced Systems Co.
1-2-4 Isogo, Kawasaki-ku, Kawasaki-shi, Kanagawa-ken, 210, Japan

Containment venting is one of the important accident management measures for BWR plants where venting is used to prevent containment failure due to overpressure. Containment venting would be conducted in two stages, prior to core damage and after core damage. If conducted after core damage, venting through suppression would reduce the fission product source terms by scrubbing effect.

The integrated severe accident analysis code THALES-2 code, developed at the Japan Atomic energy Research Institute (JAERI) for use in Probabilistic Safety Assessment (PSA), is now being used for analysis of accident management measures. Merits of THALES-2 are that it is relatively fast running and that it can simulate various system effects. System effects here means the effects of complex interactions between various plant systems or between plant systems and physical phenomena on accident progression and fission product (FP) source terms.

Extensive studies were performed in 1980's in several countries on the effectiveness of containment venting to reduce the risk of various types of plants. These studies were mainly on the effectiveness of venting for reducing the frequencies of core damage and large failure of containment. Relatively simple models were used for source term analysis in such studies. In recent years, however, significant progress has been achieved in the development of computer codes for predicting source terms, such as the coupling of thermal-hydraulics and FP behavior models as well as validation and improvements of FP behavior models based on experiments. It is considered therefore to be important to re-examine the source terms of containment venting using improved codes.

A series of scoping analyses on accident progression and FP source terms were performed aiming at obtaining better understanding on the controlling factors of the source terms of containment venting and demonstrating the usefulness of THALES-2 for planning or reviewing of containment venting strategies. The accident conditions assumed were the TQUV sequence (loss of feedwater transient followed by failure of high pressure and low pressure injection systems) at a BWR with a Mark-I containment. Containment sprays were assumed to be failed. The containment venting was assumed to be done with a hardened vent line from the wetwell. Two sets of sensitivity calculations were made for examining the effect of scrubbing model and the effect of revaporization of volatile species on the source terms.

The calculation results confirmed that the containment venting through the wetwell reduces source terms. The sensitivity calculations on scrubbing model indicated that the original model of the ART module of THALES-2, which is similar to the theoretical model of the SPARC code developed in US, gives more conservative estimation than other two models based on experiments performed in Japan. These sensitivity calculations highlighted the strong dependency of the efficiency of the pool scrubbing on aerosol size and suggested that models that influence aerosol size should be carefully examined and be validated by experimental data as far as possible.
A Scoping Analysis for Containment Venting at a BWR Using THALES-2

Jun ISHIKAWA*, Ken MURAMATSU* and Toru SAKAMOTO**
(*) Japan Atomic Energy Research Institute
(**) Toshiba Advanced Systems Co

Presented at the 8th Workshop on SARJ-97
October 6-8, 1997, Yokohama, Japan

Contents
1. Background and objectives
2. Plant and accident scenario
3. Models for venting calculation
4. Calculated release with and without venting
5. Sensitivity calculation
   - Effect of Scrubbing Model
   - Effect of Revaporization
6. Concluding remarks

Background and Objectives

Background
- Containment venting is an important AM measure
- Extensive studies in 1980s using simple codes
- Significant progress after then in code development
  - Coupling of T/H and FP transport models (THALES-2, MELCOR, MAAP, ESCADRE)
  - Improvement of FP models based on experiments

Objectives
- Perform analysis with THALES-2
  - to obtain better understanding factors
  - to identify factors that need further study

Plant and Accident Scenario

- BWR4 with Mark-I containment (Based on Browns Ferry plant)
- TQUV sequence (Loss of feed water + no injection to core + no containment spray)
- Venting from Wetwell at 1.5 x Design Pressure
- Successful closure of vent line after depressurization

Mark-I containment
Control Volume Model in THALES-2

Fission Product Transport Models of the ART Module

Scrubbing Models

A: Model by Kaneko et al. based on experiments by Japanese electric utility group (Used as the base case in this work)

B: Model of the original ART code

C: Constant value based on EPSI experiments at JAERI DF=2000

References

Calculation Cases

Comparison of Venting and No-Venting cases
Case1: No-Venting. Containment fails in the drywell
Case2(base case): Successful venting. Scrubbing model A
Case2A: Same as Case2, Venting of failure pressure

Sensitivity Calculations
Case3: Same as Case2 except use of scrubbing model B
Case4: Same as Case2 except use of scrubbing model C
Case5: Same as Case2 except ignoring of revaporation of deposited aerosol
Comparison of Venting and No-Venting Cases

Accident Progression

1. Containment failure:
   - No Venting
   - Venting at Design Press x 1.5
   - Venting at Failure Press

Amount and Forms of CsI in Wetwell (CASE2)

Amount and Forms of CsI in Steam Dome (CASE2)

Mass Median Diameter of Aerosol
Comparison of CsI Released to the Environment with and without Venting

Effect of Scrubbing Model on Aerosol Mass in Wetwell and Source Terms

Effect of Revaporization Model
Concluding Remarks

This scoping calculation showed that:

- Venting significantly reduces source terms because of scrubbing in the suppression pool.
- Timing of venting has significant effect.
- Sensitivity calculations on modeling for scrubbing and revaporization did not change the source terms very significantly.

Recommendation for Further study

- Further improvement / confirmation of models
  - iodine chemistry model
  - Aerosol size (effect of hygroscopicity, etc)
  - Comparison with other codes / tests
- Applications to examination of various parameters:
  - Set point pressure for opening
  - Set point pressure for closing
  - Dependency to accident sequences
Abstract

MAAP, MELCOR, and SCDAP are designed to predict the behavior of a nuclear power plant during a severe accident. MAAP and MELCOR are parametric codes designed to support risk assessment studies. Both codes treat the in-vessel and containment phases of a severe accident. SCDAP is a detailed, mechanistic code designed to (a) benchmark the more simplified codes, (b) perform detailed calculations to support the resolution of specific technical issues, and (c) support experimental programs. SCDAP/RELAP5, the version of SCDAP being developed by the US Nuclear Regulatory Commission, is limited to the in-vessel phase of the accident. SCDAPSIM, a simulator version being developed by an international consortium, will treat both the in-vessel and containment phases.

Although the MAAP and MELCOR contain a large number of user-specified modeling parameters that can influence the conservatism or non-conservatism of their plant calculations, it is possible to estimate the important trends in the MAAP and MELCOR calculations by comparisons with SCDAP. In particular, comparisons between the codes indicate that MAAP and MELCOR may, relative to SCDAP, underpredict the rate of core uncoverage, hydrogen production (particularly during reflood), and time of core collapse and vessel failure under non-flooded conditions. Under flooded conditions, MAAP and MELCOR may predict the successful termination of the accident, when in fact, the core material may continue to heat up and the vessel may fail. In this paper, the differences between important models, the influence of selected user-specified modeling parameters in MAAP and MELCOR, and the results of a limited number of plant calculations where the results of MAAP, MELCOR, and SCDAP have been compared will be discussed.
Comparisons between MAAP, MELCOR and SCDAP/RELAP5

SCDAP/RELAP5's Design Objectives are Different from Those of MAAP and MELCOR

• MELCOR and MAAP
  - Fast running, simplified risk assessment codes
  - Design specific versions (BWR, PWR...)
  - Integral RCS and containment analysis
  - Extensive user dials

• SCDAP/RELAP5 (SCDAPSIM)
  - Detailed mechanistic codes
  - Design specific features limited
  - User parameters limited to few critical processes
  - SCDAP/RELAP5 limited to RCS

Design Philosophy

• SCDAP/RELAP5 is designed to facilitate model assessment and validation of plant models
  - Building block approach for system thermal-hydraulics and structures
  - Control system/trip logic
  - Representative 2D fuel rods, control rods/blades, and structures
  - Lumped parameter and 2D finite element models for debris/structures

Comparisons between MAAP, MELCOR and SCDAP/RELAP5

• Design philosophy
• User parameters
• Important models
• Code-to-code comparisons
• Summary and conclusions
MAAP and MELCOR are designed specifically for full plant calculations
- Model validation against experiments require special models or versions
- BWR/PWR specific systems with relatively fixed thermal-hydraulic system representations
- Control system/trip logic functions
- Lumped parameter models

MAAP and MELCOR make extensive use of model parameters to allow user sensitivity studies and enhanced user control of results
- User parameters can be used to tune codes to match experimental results or other more detailed calculations
- User parameters can be used to control progression of accident

SCDAP/RELAP5 user selected parameters are intentionally minimized
- TH nodalization and selection of representative core components
- TH modeling parameters used to control flow regimes or select alternative correlations or models
- Damage progression parameters limited to critical areas of modeling uncertainty
  - Defaults are recommended for user applications
  - Defaults are set to best estimate values through code-to-data comparisons

System thermal-hydraulics models have strong impact on initial core uncover/heatup
- MAAP and MELCOR use quasi-equilibrium models for stratified and well mixed flow regimes
  - Predefined natural circulation patterns
  - Treatment of flow blockages version and user dependent
- SCDAP/RELAP5 uses two-fluid, non-equilibrium model
  - 2D hydrodynamics typically used in the vessel to predict flow patterns associated with natural circulation and changes in geometry
  - Empirical models developed for hot leg natural circulation used typically in PWR calculations
Core structures models influence the early phases of the accident and the effectiveness of core reflood

- **MAAP core structures models depend on the version of the code used**
  - MAAP 3B lumps core materials and structures together
  - MAAP 4 treats the heating and melting of fuel, cladding, and control rods/blades separately

- **MELCOR lumps core materials together but treats fuel rods and control rods separately**

---

Late phase models influence RPV failure and the effectiveness of vessel flooding

- **MAAP’s late phase models were significantly improved in MAAP 4**
  - MAAP 3B does not treat molten pool formation, growth, and bypass melt relocation (TMI-2-like progression)
  - MAAP 4 has lumped parameter models conceptually similar to those of SCDAP/RELAP5
  - MAAP 4 assumes optimum debris configuration for cooling of lower plenum debris and vessel

- **MELCOR does not treat molten pool formation, growth, and bypass melt relocation in core region**
  - Core collapse occurs as zircaloy melting temperatures are reached locally (default temperature)
  - Melt relocation into lower plenum occurs only with core plate failure

- **MELCOR does not treat molten pool natural circulation behavior in lower plenum**
  - Debris configuration and heat transfer likely to provide early failure and complete melt drainage
Late phase models influence RPV failure and the effectiveness of vessel flooding

- SCDAP/RELAP5 treats the in-core formation, growth, and collapse of molten pools, debris/melt/structural interactions using detailed lumped parameter models
- SCDAP/RELAP5 treats formation and growth of molten pools (including natural circulation) in lower plenum using detailed 2D finite element model
  - Debris configuration dependent on accident progression
- SCDAP/RELAP5/MOD3.2 represents a significant improvement over previous versions for flooded vessel and cavity conditions
  - Debris configuration considered to be best estimate

Code-to-Code Comparisons

<table>
<thead>
<tr>
<th>Plant/Transient</th>
<th>MAAP (Version)</th>
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<td>3.0(7af)</td>
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<tr>
<td>Browns Ferry STSB</td>
<td>-</td>
<td>1.8.3</td>
<td>3.1e</td>
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<tr>
<td>Spanish 3 Loop W SB</td>
<td>-</td>
<td>1.8.2</td>
<td>3.0(7af)</td>
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<tr>
<td>Krsko 2 loop W LBLOCA</td>
<td>3B</td>
<td>-</td>
<td>(RELAP5/MOD2)</td>
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</tbody>
</table>

* SB - Station Blackout; STSB - Short Term Station Blackout; MSLB - Main Steam Line Break

MAAP is unable to predict core heatup for Krsko NPP LBLOCA (relative to RELAP5/MOD2)

![Graph showing core heatup comparison]

*20% Break Size*

**Primary System Water Mass (Kg)**

**Maximum Core Temperature (K)***
Variation in Predicted Event Timing

MAAP underpredicts hydrogen production relative to the other codes

MELCOR underpredicts peak core temperatures relative to the other codes

Predicted RPV response with reflood varied from no-failure (MAAP) to failure (SCDAP/MELCOR)

- TVO SB with core flooding
  - MAAP 4.0  No failures
  - MELCOR 1.8.3  Failure predicted for flooding starting at peak temperatures of 2000 K and above

- AP600 with vessel and core flooding
  - MAAP 4.0  No failure predicted through ~24 hrs
  - SCDAP/RELAP5 MOD3.2  Failure predicted after ~4 hrs
Summary and Conclusions

- Code-to-code comparisons show wide variation in predicted results
  - Modeling capabilities
  - System nodalization
  - Users parameters in MAAP and MELCOR
  - MAAP4 versus MAAP3B

- MAAP and MELCOR appear to be adequate for initial core uncovery with two notable exceptions
  - MAAP predicted uncovery rates were significantly delayed relative to other codes for several transients, particularly LBLOCA
  - Initial water levels were significantly overpredicted (MAAP 3B and MAAP 4) in transients in AP600 and Krsko

- MAAP, in particular MAAP 3B, appears to underpredict total hydrogen production relative to other codes
  - MAAP 3B does not predict the enhanced hydrogen production and melting observed in reflood experiments and predicted by SCDAP/RELAP5
  - MAAP 4 predicts significantly more hydrogen during reflood but is still noticeably lower than the other codes

- MELCOR predicts core collapse near zircaloy melting temperature, significantly earlier than for MAAP or SCDAP/RELAP5
  - Peak core temperatures are several hundred K lower than other codes

- Use of MAAP 3B or MAAP 4 for flooding conditions questionable
  - Neither version predicted significant core slumping or RPV failure even under the most extreme of conditions (Flooding delayed to temperatures in excess of 2000 K)
  - Both MELCOR and SCDAP/RELAP5 predicted RPV failure in extreme cases
  - Optimum debris configuration used to enhance coolability

- Use of MELCOR for flooding conditions questionable
  - Code lacks molten pool models so heat transfer to surrounding water or adjacent structures may be unrealistic

- Use of SCDAP/RELAP5/MOD3.2 recommended for flooding conditions
  - SCDAP/RELAP5/MOD3.2 includes models specifically developed for flooded vessel and cavity conditions
11.5 Hypothetical Accidents in Accelerator-driven Subcritical and Lead-cooled Fast Systems

H.U. Wider, JRC Ispra, Italy

(Present by A. V. Jones, JRJ Ispra, Italy)

Accelerator-driven systems (ADSs) with a fast subcritical core and with lead as coolant and flowing target for the impinging proton beam, are getting a great deal of attention because of their potential as nuclear waste burners. In Spain the LAESA consortium has recently formed to build a 100 MWt pilot plant as proposed by Prof. Rubbia of CERN.

It is therefore important to consider already now all remote possibilities for severe accidents in such systems so that they can be considered in the design. Loss-of-Coolant Accidents (LOCAs) are not a problem since the lead coolant in an ADS is at a low pressure and the primary coolant is in one vessel (pool design) and a guard vessel is foreseen.

Reactivity Induced Accidents (RIAs) have been investigated with the EAC2 fast reactor accident code and an upgraded point kinetics into which a neutron source has been inserted. These calculations show that both fast and slow reactivity insertions, that are smaller than the subcriticality of the ADS, lead to mild power rises with maximum increases of a few tenths of nominal. However, these increased powers will eventually lead to a limited number of pin failures if the accelerator beam is not interrupted or switched off. Fast reactivity insertions that are larger than the subcriticality can lead to more pin failures which reduce the reactivity through fuel sweepout. An important reason for not getting high powers in the cases with lead coolant is the low void worth of lead.

Loss-of-Flow (LOF) or Loss-of-Heat-Sink (LOHS) accidents lead to an undercooling of the core and voiding (in a sodium cooled core) and fuel and clad melting if the accelerator beam is not interrupted or switched off. However, even in the case of an LOF accident in a sodium cooled system, no high power is calculated due to the sodium voiding. However, strong negative reactivity effects due to fuel dispersal do not reduce the power much below nominal and therefore most of the core would melt if the accelerator beam was not interrupted. However, in the course of such an accident the evacuated guide tube of the proton beam will eventually melt at some location and be flooded by the lead. Passive measures to flood this guide tube before any fuel or clad melting starts should be be developed. Active measures to shut off the current to the accelerator in an accident condition are also important. EAC2 calculations show that a switching off of the beam reduces the power very rapidly to decay heat levels.

If a timely beam shutoff is not achieved in a LOF or LOHS type accident, cladding and (metal) fuel melting will occur before lead boiling started. The metal fuel will probably mix with the lead coolant and will possibly be rapidly dissolved in the molten lead. If this were not true, a fuel agglomeration on the vessel bottom could occur and lead to thermal loads and criticality problems. This could be prevented with an in-vessel core catcher.
Other potential causes for accidents could be lead freezing (MP 327°C) due to an overcooling of a rather fresh core during decay heat removal. Lead freezing could lead to frozen chunks that could block the core inlet. Problems with the emergency decay heat removal could also lead to problems. Due to feedback effects a doubling of the beam power (which is rather unlikely) was found to result in less than a doubling of the power.
Hypothetical Accidents in Accelerator-driven Subcritical and Lead-cooled Fast Systems

H. U. Wider
JRC Ispra, ISIS Institute

A. Low-probability Accident Scenario in a Rubbia-type Waste Burner


C. Lead freezing problems

D. Emergency decay heat removal

E. Proposed Molten Lead Experiment to test preventative measures for B.) and understand phenomena in C.) and D.)

Low-probability Accidents in a Rubbia-type Waste Burner

No significant problems with Loss-of-Coolant Accidents (LOCAs) because of low pressure coolant, all primary coolant in one vessel (pool design) and the use of a guard vessel.

No significant problems with reactivity insertion accidents because of the subcritical design. Also beam-power increase accidents pose no significant problem. However, the beam should eventually be interrupted or shut-off to avoid local damage.

Severe Coolant Disturbances such as Loss-of-Flow or Loss-of-Heat Sink Accidents could lead to core melts if the accelerator beam is not switched off or interrupted. Several active and a few passive means to do this are being proposed.

Emergency decay heat removal is a challenge for all reactor types. A passive cooling of the outside of the vessel with natural air convection is a very promising approach.

A special problem of lead-cooled systems (mostly during decay heat removal) could be an overcooling of the lead, causing lead freezing (M.P. 327°C) and coolant flow obstructions. Design measures to prevent this (electrical heater rods, dumping of secondary coolant, limiting the air cooling) are possible.

The bending magnets which turn the accelerator beam downwards into the waste burner may fail (e.g. due to a local loss of electricity). The proton beam would then generate a dangerous skyshine. Design measures such as an effective beam stopper and/or having the beam underground and also an interlocking system that allows running of the accelerator only when there is electricity to the bending magnets, are possible.
ADS under Consideration

800 MWe Na-cooled paper design used for WAC comparative calculations for FBRs

Point kinetics in EAC-2 modified to include a constant neutron source and insertion of a constant negative reactivity contribution into the code to simulate the subcriticality

European Accident Code-2 includes

TRANSURANUS pin mechanics and heat conduction model
CFEM single and two-phase boiling model
MDYN in-pin and channel fuel motion, fuel coolant interaction and coolant voiding
PKIN point-kinetics neutron flux

HEXPRT Neutronics

3D nodal transport calculation with higher order 1-D boundary source method

Capability to generate reactivity worth curves

New X-section data for thorium-lead system through contract with FZK Germany

Fig.13 Power history in 10 cent/s TOP case with source and -10$ subcriticality

Fig.13a Reactivities causing the power history in Fig.13
2.0 Expt: TOP-6$/s with source

Power history in 6 $/s TOP case with source and -5$ subcriticality

Fig. 11 Power history in 6 $/s TOP case with source and -5$ subcriticality

Power during accidental doubling of the beam strength in 1s. Lead simulated with Na with a boiling point of 1700 deg C and 1/10 of the sodium void worth.

Fig. 11a Reactivities causing the power history in Fig. 11

Reactivity histories for the above power. Assumed subcriticality -5$
Extreme Reactivity accident in an ADS with a total insertion of 6S with 6S/s. Lead coolant simulated by Na with a boiling point of 1700 deg.C and 1/10 of its regular void worth.

Reactivities for the above power history. Initial subcriticality -3S.

Fig. 1a Power History of LOF case with source and -10S subcriticality

Fig. 1b Reactivity Histories in above LOF case
Damage Propagation in a Severe Coolant Disturbance Accident without Beam Shut-off - if the decay heat didn't get removed properly, a similar but much more slowly propagating scenario could develop.

An overheated lead coolant would first melt the cladding and probably carry it away. If the proton beam is still not interrupted, metal fuel will melt before the lead boils. Chances are that a significant amount of metallic fuel would be carried away by or dissolved in the lead coolant. If this were not the case, accumulations of fuel on the bottom of the vessel could occur - an in-vessel core catcher might have to be considered.

Lead Freezing Investigations

Ph.D. thesis on lead freezing problems in a Waste Burner started recently. As a tool the US code FLOW3D is used.

Objectives:

1.) Search and investigate all possibilities for lead freezing in the primary system - particularly for decay heat removal from low burn-up cores and cooldown of the secondary loop.

2.) Investigate the consequences of lead freezing in the primary system - particularly in the primary heat exchanger, the core inlet regions and near the walls for emergency decay heat removal.

3.) Examine different prevention measures such as electrical heater rods, dumping the coolant of the secondary loop, reduce emergency air cooling.
Objectives of first Molten Lead Natural Circulation Experiment

1.) Testing passive beam interruption devices - important for all accidents leading to lead overheating

2.) Investigating the vessel outside cooling with natural circulation of air - emergency decay heat removal

3.) Investigation of lead freezing aspects during emergency decay heat removal

4.) Investigation of the behavior of lead activation products such as mercury (or polonium - but only at end of test series) or spallation products such as tritium (to be simulated with hydrogen)
12. Session X

Melt Behavior and Accident Management

Chairperson: S.B. Kim (KAERI)
Co-chairperson: M. Kato (NUPEC)
12.1 EXPERIMENTAL COMPARISON OF VVER AND PWR FUEL UNDER SEVERE ACCIDENT CONDITIONS

(Abstract)

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The core of Soviet-designed VVER nuclear power plants is made of similar, but not exactly the same materials as Western PWRs. In order to point out the differences in materials behaviour and the effect of hexagonal arrangement of fuel rods in the assembly an experimental programme has been launched in KFKI Atomic Energy Research Institute. The experimental programme consisted of separate effect and integral tests.

In separate effect tests the interactions of core materials and the behaviour of Zr1%Nb cladding were investigated in the following experiments:
- UO\textsubscript{2} pellet - Zr1%Nb cladding interaction in 1000-1300 °C range,
- Stainless steel spacer - Zr1%Nb cladding interaction in 1000-1250 °C range,
- Boron steel - Zr1%Nb cladding interaction in 1080-1235 °C range,
- Effect of H\textsubscript{2} uptake on the mechanical behaviour of Zr1%Nb cladding in 900-1200 °C,
- Ballooning tests with normal and preoxidized Zr1%Nb cladding samples in 700-1200 °C range.

Each type of experiments were preceded by similar tests using PWR materials. The results of separate effect tests showed that the interaction of VVER materials is similar to PWR materials, however the interaction rate between cladding and steel and pellet is slightly higher. The hydrogen uptake results in earlier embrittlement of Zr1%Nb cladding compared to Zircaloy-4. The rupture of cladding due to ballooning process takes place in case of Zr1%Nb at lower temperatures as well.

The integral tests were performed on the CODEX facility with 7-rod hexagonal VVER fuel assemblies. During the heat-up period of the tests the temperature escalation and H\textsubscript{2} production due to cladding-steam interaction was investigated. The cooling down periods were performed with argon or water. The post-test examination of the cuts of the experimental section provided detailed information on the destruction of fuel assembly in conditions up to 2400 °C. The results of CODEX test showed that in the early phase of the core degradation in VVER reactors the same phenomena takes place as in PWRs.

The CODEX experimental programme will be continued in the framework of CEC/OPSA project with "air ingression" type tests using PWR fuel assembly.
EXPERIMENTAL COMPARISON OF VVER AND PWR FUEL UNDER SEVERE ACCIDENT CONDITIONS

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INTRODUCTION

There are four VVER-440 units in operation at Paks site in Hungary. The core of Soviet-designed VVER nuclear power plants is made of similar, but not exactly the same materials as Western PWRs. The VVER fuel rods are placed on a triangular lattice, the fuel assembly has hexagonal form. The VVER-440 fuel assembly is covered by an external shroud. The clad material is Zr1%Nb alloy. The spacer grids are made of stainless steel, some new designs use Zr1%Nb. The VVER-440 has control assemblies with boron steel absorbers.

<table>
<thead>
<tr>
<th></th>
<th>PWR</th>
<th>VVER</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel geometry</td>
<td>square</td>
<td>hexagonal with shroud</td>
</tr>
<tr>
<td>Clad material</td>
<td>Zircaloy</td>
<td>Zr1%Nb</td>
</tr>
<tr>
<td>Spacer grid</td>
<td>inconel/Zircaloy</td>
<td>stainless steel/Zr1%Nb</td>
</tr>
<tr>
<td>Absorber</td>
<td>AIC</td>
<td>boron steel</td>
</tr>
</tbody>
</table>

Table 1. Comparison of PWR and VVER fuel

In order to point out the differences in materials behaviour and the effect of hexagonal arrangement of fuel rods an experimental programme has been launched in KFKI Atomic Energy Research Institute. The experimental programme consisted of separate effect and integral tests.

SEPARATE EFFECT TESTS

Cladding - pellet interaction

During accidental situations at high temperature the clad material reduces the \( \text{UO}_2 \) pellet. The interaction results in the formation of two \( \alpha-Zr(O) \) layers and between them a molten \( (\text{U,Zr}) \) layer occurs.

The objective of the test was to determine the total interaction layer thickness, for it should be considered in the evaluation of clad performance. In the experiment VVER fuel samples with Zr1%Nb clad, containing three pieces of original pellets were treated at different temperatures. Both end of the fuel were closed hermetically by welding in Ar gas at low pressure. Then the specimens were put into an autoclave, which was heated by a high frequency generator. During the test the autoclave was filled with pure Ar gas. The 40 bar pressure, which was applied in the test was sufficient to provide the necessary contact between cladding and pellet. The temperature of the specimen was measured by thermocouples and a pyrometer. The test were performed in 1000-1300 °C range. After the treatment the fuel samples were cut and polished for metallographical investigation.

On the basis of experimental results the total interaction layer thickness of \( \text{UO}_2-Zr \) interaction was determined as function of reaction time for different temperatures. For code calculations the coefficients of Arrhenius equation were derived.

Another series of experiments was performed in steam atmosphere. The specimens were analysed by optical microscope using image analyser. The results of these tests provided information not only on the growth of different layers, but on the time of complete oxidation as well.
Cladding - spacer interaction

The VVER-440 fuel assemblies have stainless steel spacer element at each 250 mm of length. In accidental conditions the high temperature interaction between the clad and spacer results in eutectic formation. The eutectic has much lower melting point, than that of the two metals.

The objective of the test was to determine dependence of eutectic formation on the reaction time and reaction temperature.

A special equipment was developed for the measurement of the rate of eutectic formation between pairs of metal pieces. A mobile high temperature furnace was previously heated up to the required temperature and was moved to include the sample holder device. In this device the metal discs are heated to the furnace temperature without being in contact. At the beginning of the test a weigh produces the necessary force to push the metal pieces together. From that time the movement of the fore transmitting ceramic rod is indicated by an electronic micrometer. The signal recorded by the micrometer is proportional to the amount of liquefied eutectics pressed out of the interaction zone between metal pieces. To avoid oxide layer formation vacuum is kept in the sample holder device. The tests were performed in 1000-1250°C range.

Based on the experimental results the melt-through of the 0.7 mm thick clad at different temperatures was calculated.

<table>
<thead>
<tr>
<th>Temperature, °C</th>
<th>Time, s</th>
</tr>
</thead>
<tbody>
<tr>
<td>1060</td>
<td>445</td>
</tr>
<tr>
<td>1080</td>
<td>260</td>
</tr>
<tr>
<td>1090</td>
<td>120</td>
</tr>
<tr>
<td>1140</td>
<td>34</td>
</tr>
<tr>
<td>1180</td>
<td>15</td>
</tr>
</tbody>
</table>

Table 2. Time of clad melt-through

Cladding - boron steel interaction

The absorber material is in contact with Zr1%Nb component. In high temperature conditions the same eutectic problem occurs for these materials as for stainless steel and clad. The measurements were carried out similarly to spacer-cladding tests, using the same methodology. The results were converted into Arrhenius equation coefficients. The test were performed in 1080-1235°C range.

![Fig. 3. Equivalent thickness of Zr1%Nb cladding - stainless steel (OX18H10T) interaction](image)

![Fig. 4. Equivalent thickness of Zr1%Nb - boron steel interaction](image)
During the oxidation of zirconium alloy cladding in steam atmosphere some part of the generated hydrogen is absorbed in the cladding. The hydrogen content results in embrittlement of the cladding and the absorption process affects the total hydrogen release from reactor vessel.

The objective of these tests was to determine the amount of absorbed hydrogen. The hydrogen uptake was measured at Zr1%Nb cladding samples, which had been previously oxidised at 900 and 1200 °C in steam. After oxidation the specimens were cooled down and the oxygen uptake was determined by the measurement of mass gain. Subsequently the hydrogen content was measured by hot extraction method.

The results showed that the hydrogen concentration increases with increasing mass gain, which is proportional to the oxygen content of the cladding. Measurement at 900 °C reveals, that the solubility of hydrogen limits the absorption. When the solubility limit is reached the hydrogen concentration begins to decrease. This solubility limit was not reached in the case of higher oxidation temperature.

The absolute values of hydrogen uptake for Zr1%Nb was found approximately four times higher than for Zircaloy.

In order to study the embrittlement effect the samples after the oxidation were taken to ring rupture tests. The rupture of cladding took place at lower deformation rate for samples with thicker oxide layers.

The design criteria of Zircaloy cladding (<17% local oxidation) was checked for Zr1%Nb cladding, it was found that total embrittlement of Zr1%Nb cladding takes place already at 5-10% local oxidation.

The Zircaloy-4 and Zr1%Nb cladding tubes were tested in an appropriate experimental facility providing linear pressurisation of the specimens under isothermal conditions between 700-1200 °C. The specimen was placed in a quartz test tube filled with inert gas (Ar), then vacuumed and heated up in an electrical furnace. The pressure of the inert gas in the quartz tube was kept at constant 1 bar by means of a buffer volume. After ca. 1000 s heat up the sample was pressurised with argon gas at a constant pressure gradient provided by choke with a capillary tube. Different pressurisation rates between 0.01-0.1 bars could be applied by different size of capillary tubes. The furnace temperature and the cladding inner pressure were recorded.

The specimens were 50 mm long slices of original PWR and VVER claddings with the identical inner and outer diameter of 9.3/10.75 and 7.8/9.2 mm respectively. The samples were closed with Zircaloy-4 end-plugs welded to the cladding in argon atmosphere. The pressurisation was performed through a 200 mm long and 2.2 mm diameter Zircaloy-4 pipe attached to on end of the specimen. In order to investigate the effect of corrosion on mechanical strength of Zr1%Nb some samples were treated in steam or iodine atmosphere before the ballooning tests. Samples with outer oxide layer of 0-50 μm or iodine concentration of 0-18 mg/cm³ were investigated.

On the basis of ballooning tests, performed at different temperatures in the range of 700 - 1200 °C with 50 °C steps, the following conclusions have been drawn:

The experiments revealed that in the temperature range of 800-1000 °C the mechanical strength of the Zr1%Nb cladding is lower than that of the Zircaloy-4 tube, since the α-β phase transition temperature is different for VVER and PWR cladding materials.

The coolant side oxidation has a significant effect on the mechanical strength of the cladding. The strength of Zr1%Nb increases up to 10 μm oxide layer thickness, but decreases with further oxidation. Decreasing deformation with increasing ZrO₂ layer have also been observed.

The iodine treatment did not influence the mechanical behaviour significantly: only a small increase of high temperature strength and a small decrease of deformation have been observed.
INTEGRAL TESTS

The integral tests were performed on the CODEX (COre Degradation EXperiment) facility with 7-rod hexagonal VVER fuel assemblies.

The CODEX integral test facility represents the main geometrical arrangement of a VVER-440 reactor fuel assembly and has been constructed of VVER materials. The basic part of the facility is the test section comprising a seven rod bundle with electrically heated rods. The heating tungsten bars are surrounded with ring-shaped UO₂ pellets and enclosed in industrially fabricated Zr1%Nb alloy cladding. The bundle is fixed by three spacer grids and placed into a hexagonal shroud. This shroud is covered by thermal insulation layers. The steam generator and superheater section of the facility provides argon and steam flow for the test section during heating-up and cooling-down phases. Up to now three experiments were executed on the facility, one with fuel pellet simulant and two with real fuel.

<table>
<thead>
<tr>
<th>Test</th>
<th>Pellet</th>
<th>Cooling mode</th>
<th>Coolant</th>
</tr>
</thead>
<tbody>
<tr>
<td>CODEX-1</td>
<td>Al₂O₃</td>
<td>slow</td>
<td>argon</td>
</tr>
<tr>
<td>CODEX-2</td>
<td>UO₂</td>
<td>slow</td>
<td>argon</td>
</tr>
<tr>
<td>CODEX-3</td>
<td>UO₂</td>
<td>quick</td>
<td>water</td>
</tr>
</tbody>
</table>

Table 3. Main parameters of CODEX test matrix

The first experiment with UO₂ pellets was performed at the end of 1995. During the first phase of the experiment the bundle was preheated by argon inlet flow without electrical heating. The second phase started with switching on electrical rod heating, and steam injection was added to the argon flow resulting in additional heat generation. The electrical power was linearly increased. After 1200 s, rod temperature escalation was observed. The second phase lasted 1800 s, then the electrical heating was switched off, the steam injection was stopped and the facility was cooled down slowly by argon. The measured data showed that the highest temperature (2400 °C) was reached close to the top of the bundle. Subsequent to the experiment the bundle was filled up with artificial resin and afterwards cut into slices in order to facilitate the further investigation.
Fig. 5. Temperature measurements in CODEX-2 test:

- Temperature profiles for various test conditions:
  - Temperature vs. distance from the core boundary
  - Temperature vs. time for different coolants
  - Temperature vs. pressure for various coolant flow rates

Fig. 4: Cross section of CODEX bundle, showing:
- Core structure
- Coolant flow paths
- Instrumentation and control systems

Fig. 3: Layout of the CODEX facility:
- Experimental setup
- Safety measures
- Control and monitoring systems
The CODEX-3 experiment with quick water cooling has been performed in two steps. In the first step similar procedures were taken as in CODEX-2 experiment, but the cooling down phase was performed with water and it started at 1200 °C. In that case the temperature was too low reaching escalation conditions, however a strong oxide layer was created. After the first step a visual observation was carried out. It was found that there was no damage on the rod bundle and the oxide layer on the external surface was approximately 60 μm.

The second step consisted also three phases, but the cooling down was initiated at higher temperature (1600 °C). Only a slight temperature escalation was observed before the bundle was cooled down. The post-test examination showed that there were some signs of high temperature material interactions, but no similar damage to CODEX-2 case was observed.

During the next years “air ingression” type experiments will be carried out on the CODEX facility for western type PWR bundles 3x3 in the framework of CEC OPSA project. The objective of the tests are to study the increase of air ingress oxidation reactions during core degradation. These tests will be used as a basis for a later PHEBUS test with air ingress. The first CODEX air ingress test will be performed in 1998.

CONCLUSIONS

The differences between VVER and PWR materials were studied under severe accidents conditions. In separate effect tests the interactions of core materials and the behaviour of Zr1%Nb cladding were investigated. Each type of experiments were preceded by similar tests using PWR materials. The integral tests were compared to similar tests on CORA facility. The findings of comparison are summarised in Table 4.

<table>
<thead>
<tr>
<th>Phenomena</th>
<th>Relation of VVER to PWR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Interaction rate between cladding and core</td>
<td>slightly higher</td>
</tr>
<tr>
<td>materials</td>
<td></td>
</tr>
<tr>
<td>Hydrogen uptake</td>
<td>higher</td>
</tr>
<tr>
<td>Embrittlement due to oxidation</td>
<td>at lower oxidation rate</td>
</tr>
<tr>
<td>Cladding rupture due to ballooning</td>
<td>at similar or lower temperature</td>
</tr>
<tr>
<td>Integral test phenomena at the early phase of</td>
<td>similar</td>
</tr>
<tr>
<td>core degradation</td>
<td></td>
</tr>
</tbody>
</table>

Table 4. Comparison of high temperature phenomena for VVER and PWR

The basic conclusion of the investigations was that some phenomena during severe accidents in VVERs takes place at different conditions than in PWRs. In spite of these differences the general picture of early phase of core degradation seems to be similar in VVER and PWR reactors.

REFERENCES

ABSTRACT

The main safety tasks of the Accident Confinement System (ACS) of a RBMK-1500 plant are to provide confinement of steam, water and radioactive products released by a break in the primary coolant circuit or by opening of the main steam relief valves or fast-acting pressure-reducing valves, to prevent excessive pressure building up within section of confinement system, to provide adequate structural strength of each compartment to withstand the pressure and temperatures which may occur in a loss-of-coolant-accident (LOCA) and to limit release of radioactivity to the environment in a design basis accidents to specified limits.

The results of the thermohydraulic and structural response of the ACS of Ignalina NPP to LOCAs is presented. This is the first analysis that considers both short and long term responses of the ACS for full range of LOCAs employing state-of-the-art best estimate methods. An adequate modeling of complex multi-compartment and multi-suppression pool system of the Ignalina NPP ACS is performed using new C11AF version of the CONTAIN code. The analysis employs mass and energy source from calculations performed using RELAP5 code. The calculated atmospheric temperatures and pressure inside ACS compartments as well as temperatures of structures are used as input data for structural analyses of the ACS.

For strength evaluation of the ACS of Ignalina NPP the maximal design basis accident - a rupture of the pressure header of the main circulation pumps - is considered. Most of the compartments at the Ignalina NPP have rectangular parallelepiped configurations. Their walls could be subjected to high pressure and temperature and also to compression of the weight of the structures from above. Different models are discussed for an evaluation of the strength of such structures of concrete. The simulation of reinforced concrete carried out using sandwich elements. The model assumes elastic deformation of both the reinforcement and the concrete. Thickness of the concrete under compression is evaluated by either allowable stress or by limit analysis techniques for concrete. The part of the concrete in tension is replaced by a substance of a very low modulus of elasticity, the shear resistance of which equals that of concrete. Thickness of reinforcing layers is found by uniform distribution of actual steel cross-sections in a given direction. Metal framework in both vertical and horizontal walls are simulated by finite elements. Reinforcements in the monolithic blocks are uniformly distributed in the width of a block, or the monolithic blocks are evaluated like beam elements. The model using reduced plate finite elements of an orthotropic material is discussed. In this manner the temperature-induced gradual heating of both vertical and horizontal walls can be evaluated.

The results of a finite element analysis are compared with those of the traditional technique of limit analysis. The bending moment and axial force of the cross-section are founded from the analytical expression for plates subjected to bending and under different support conditions. The study covered the evaluation of performance parameters: loads initiating the cracks; sizes of cracks initiated; possibilities of closing the cracks. Numerical results of strength evaluation for most loaded compartments are presented.
Evaluation of the RBMK-1500 Accident Confinement System

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3035 Kaunas
Lithuania

1. INTRODUCTION

The Ignalina NPP does not possess the conventional Western external containment, most of which have either cylindrical or spherical protection shells. The components of the RBMK-1500 reactor cooling systems are confined within rectilinear compartments separated from each other by different openings. Walls of reinforced concrete are used for the compartments and for the larger part of the main cooling loop. The confinement that protects both the staff and the environment in case of an emergency in the multiple circulation circuit is the accident confinement system [1]. This system at the Ignalina NPP operates as one of the above protection shells; in emergency it reduces the internal pressure by means of the steam condensation device. The walls and floors/ceilings of the ACS are of the steel-reinforced concrete, with internal walls in the second reactor unit covered by steel liners. The internal lining is incomplete in the first reactor unit [1]. To insure proper sealing, the walls of concrete are coated with epoxy-silicate films. Figure 1 is an overall view of the ACS of the Ignalina NPP.

The dependability of the ACS for RBMK-1500 plants has not been independently verified at the design and construction stages. There was a number of reasons to suspect that the design characteristics do not meet Western safety standards. These include:

- the simplistic analysis models used during the ACS project development and the incompleteness of the design information;
- on-site inspection evidence that discrepancies exist between the facility as designed and as installed.

Therefore, independent assessment and analysis of the capabilities of the ACS has been started by the Ignalina Safety Analysis Group at Lithuanian Energy Institute. This includes on-site inspection, thermal-hydraulic evaluation of ACS response to a broad range of LOCA events and a variety of break locations. State-of-the-art technique has been used for analyses. The present paper provides an overview of results of these investigations.

2. THERMAL-HYDRAULIC EVALUATION OF ACS RESPONSE

Best-estimate methods have been employed to evaluate the response of the RBMK-1500 plant to a broad range of LOCA events. The response of the primary system was analysed using RELAP5/MOD3.2. The mass/energy break flow rates obtained from these calculations were subsequently employed to determine both the short-term and long-term loads imposed on the ACS using CONTAIN Version C11AE. The ACS of the RBMK-1500 plant contains a total of ten suppression pools, thus for its proper analysis a version of the code was required which can model multiple suppression pools.

Up to the present, analytical results available in the open literature regarding the response of RBMK plants to LOCA events have been limited to relatively brief time periods. The present study is the first one which analyses the long-term aspect of LOCA transients for the primary system and the ACS. The analysis of the ACS considers both the case when the CTCS heat exchangers perform as designed, and when they fail.

The LOCA transients analyzed in this study were chosen so that they would cover the range of break sizes (maximum design basis LOCA to small break) and a variety of break locations. The main purpose of the RELAP5 analysis of the primary system was to generate best-estimate mass/energy source terms for the subsequent assessment of the ACS. The obtained results showed that the mass/energy source term presented in the TOB documentation [2]...
which was used in most earlier studies, was based on bounding and in some respects simplistic and non-physical assumptions. For example, in the TOB estimate, the break flow out of the ruptured pressure header is assumed to become pure steam within 14 sec. Such a result can be obtained only by neglecting most two-phase flow phenomena. The analogous RELAP5 computation shows that the break flow remains two-phase during the entire transient. A best-estimate treatment of the design basis LOCA event produces changes not only in the rate and quality of the break flow, but also in the response of the primary system.

For most transients the RELAP5 computations were carried out for a time period of one to several hours. This is sufficient to de-pressurize the primary system and/or to approach a quasi-equilibrium state. The subsequent CONTAIN calculations were extended out to a time period of up to 24 hours. This made it necessary to develop a mass/energy balance method for generating containment source terms for the extended analysis time.

Analysis conducted is the first study presenting analytical results regarding the long-term ACS response to LOCA events. The performed computations show that the pressure loads imposed on the ACS can be divided into two time segments. A 'short-term' pressure peak which occurs within several minutes after initiation of the LOCA, and a 'long-term' pressure rise which reaches a broad maximum in several hours. The short-term pressure increase is terminated by the rapidly decreasing rate of break flow as the primary system de-pressurizes. The subsequent 'long-term' peak is generated because the temperature of the condenser tray water increases and the rate of energy removal into structures falls more rapidly then the mass/energy addition rate due to break flow. As the decay energy diminishes, and the stored thermal energy in the graphite block is dissipated (the later is an important energy source for RBMK plants) the balance between energy source and loss terms is restored and eventually the atmospheric pressure begins to decrease. For most LOCA events, the long-term pressure rise achieves a broad maximum in the 3 to 6 hour time span, the magnitude of the peak in most cases stays well below the pressures imposed during the initial phase of the transient.

The magnitude of the long-term pressure increase depends upon two factors. The first and most obvious one is the time dependent balance between the rate of mass/energy addition to the ACS atmospheres and the energy/mass removal rate by the various available heat sinks. The sign of this atmospheric energy balance determines whether the pressure will be increasing or decreasing. The magnitude of the eventual pressure increase depends also on the amount of non-condensibles that remain in the ACS atmospheres. The RBMK-1500 ACS is provided with a controlled venting feature. During the initial stages of a LOCA event (for a time period of ~5 min), non-condensibles are vented to the outside atmosphere. The time period available for venting remains essentially constant, the amount that is vented then depends on the initial pressure build-up and thus on the nature of the transient. Differences in the amount of non-condensibles vented influence the subsequent long-term pressure rise in the design basis LOCA and the MRV malfunction events. A third significant influence, which applies only to LOCA events which direct their mass/energy flow to the upper plate of a single ACS tower, is the volume of the ACS into which the non-condensibles are compressed.

The heat exchangers of the Condenser Tray Cooling System constitute one of the principal long-term energy sinks. For this reason this study considered the question how would the ACS respond without the availability of CTCS. An estimate of the probability that the CTCS would fail to perform as designed are shown to be in the 0.003 to 0.007 per demand range. The assumption of CTCS failure thus represents a low probability multiple-failure scenario and as such belongs in the 'severe accident' spectrum. It is included in this study for completeness. However, the importance of CTCS cooling should not be over-stated. Energy balances performed for different transient time spans show that even when the CTCS functions at full capacity, other energy removal mechanisms...
remain relevant. The condenser tray water and the massive concrete structures of the RBMK-1500 ACS have a very large heat capacity and they absorb a significant fraction of the released energy.

For the design basis LOCA two long-term (24 hr) computations were carried out, one with a fully functioning CTCS, the other for the case where the CTCS is assumed to fail. For the case when the CTCS functions as designed, the secondary pressure peak is considerably lower than the short-term peak and thus does not pose an additional challenge to the integrity of the ACS, Figure 2. Significantly and somewhat surprisingly, the calculated results for the scenario with failed CTCS, show that even without the assistance of external cooling the pressure rise remains bounded, Figure 3. The energy removed by the very large heat capacities of the RBMK-1500 ACS water pools and concrete structures plus the energy required to heat the subcooled ECCS water to saturation limits the pressure rise to a total of ~2 bar in the break compartment, and to ~0.9 bar over-pressure in the compartments beyond the condensing trays. The design pressure of these compartments is 0.8 bar and blow-out panels are provided which are set to open at a pressure difference of ~0.8 bar. Thus the computed result indicates that for a limited time period the computed pressure transient lies above the design limits. However, the margin is sufficiently close to design values that no catastrophic failures are anticipated.

The results summarised above deserve additional comment. Note that they apply for a period of 24 hours. During this time the massive concrete structures still have not reached thermal equilibrium while the core decay energy has decreased substantially and the stored energy in the graphite block has been entirely dissipated. A significant fraction of the decay energy has been employed to saturate the ECCS coolant flow which has a source independent from the CTCS. If the analysis were to be extended beyond 24 hours and limitations were imposed on ECCS flow, then the concrete structures would approach thermal saturation (in ~3 days) and the pressure would start to increase once more. The second comment concerns the blow-out panels. According to design specifications, the pressure difference setting for their opening coincides with the design over-pressure of the ACS compartments beyond the condensing trays. This implies that the designers either deliberately did not want to take advantage of the sizeable margin which is normally available between the 'design pressure' and the 'failure pressure' of a structure, or that there is confusion in terminology.

A summary conclusion of this study is that the analysis of primary system and ACS response of the RBMK-1500 plant to LOCA events employing best-estimate methodology has demonstrated both the complexity and the resilience of these systems. The inherent complexity requires the development of models which push at the limits of currently employed analytical methods. Part of the complexity is produced by the high degree of redundancy and is thus safety related. This becomes especially apparent in the analysis of long-term transients for which the number of alternative options and thus alternative scenarios increases. In most cases a long-term transient analysis will thus not be unique but will depend on the scenario (the main component of variability being operator action) chosen by the analyst. Though, as noted, the methods employed in this study are 'best-estimate' an effort was made to choose 'conservative' scenarios. In this respect the results also have a conservative slant. It is shown in this study that for the broad range of LOCA events analysed design loads on the ACS are approached or marginally exceeded only for those cases where multiple failure of safety systems is assumed.
3. STRUCTURAL ANALYSIS OF THE ACS

Structural analysis of the RBMK-1500 ACS was performed using state-of-the-art ALGOR code. The case considered is the largest hypothetical accident of the project: a double-end break of the pressure header of the main circulation pumps. Description of the ACS model and preliminary results are presented elsewhere [3]. The analytical results for one of the compartments (identified by 1) shown in Figure 4 are presented. This compartment is located in a top part of ACS. There are homed pressure and suction headers of main circulation pumps.

Most of the compartments at the Ignalina NPP have rectangular parallelepiped configurations. Their walls could be subjected to high pressure and temperature and also to compression of the weight of the structures from above. Different models can be applied to test the strength of such structures of concrete. We used in our analysis of the ACS of the Ignalina NPP the finite elements analysis together with the conventional structural calculations of the standards [4,5]. A detail description of structural analysis methods used is presented in [6].

The wall strength of one compartment is analysed for the worst hypothetical accident: a double-end break of the pressure header located in compartment 117 [1]. In such an accident simulation the maximum pressure of 0.4 MPa is achieved in 13 seconds. It then drops and after 50 s the pressure becomes stable at 0.12 MPa. The temperature rises in 10 s up to 143°C, then drops to 120°C at 38 s and rises again, so that in 50 s the temperature is 160°C. Compartment 407 is directly connected to compartment 117, so that equal temperatures in both was assumed. The wall inside surfaces reaches 143°C in 80 s. The pressure-temperature behaviour is shown in Figure 5. Our computation starts at the maximum pressure, that is after 13 s when the wall surface is heated to 60°C [7]. The temperature distribution through the walls is given in reference.
9. CONCLUSIONS

Best-estimate methods have been employed to evaluate the thermal-hydraulic response of the RBMK-1500 plant to a broad range of loss of coolant accidents. The response of the primary system was analyzed using RELAP5/MOD3.2. The mass/energy break flow rates obtained from these calculations were subsequently employed to determine both the short-term and long-term loads imposed on the Accident Confinement System using CONTAIN Version C11AE.

A summary conclusion of thermal-hydraulic study is that the analysis of primary system and ACS response of the RBMK-1500 plant to LOCA events employing best-estimate methodology has demonstrated both the complexity and the resilience of these systems. The inherent complexity requires the development of models which push at the limits of currently employed analytical methods. Part of the complexity is produced by the high degree of redundancy and is thus safety related. This becomes especially apparent in the analysis of long-term transients for which the number of alternative options and thus alternative scenarios increases. In most cases a long-term transient analysis will thus not be unique but will depend on the scenario (the main component of variability being operator action) chosen by the analyst. Though, as noted, the methods employed in this study are "best-estimate" an effort was made to choose "conservative" scenarios. In this respect the results also have a conservative slant. It is shown in this study that for the broad range of LOCA events analyzed design loads on the ACS are approached or marginally exceeded only for those cases where multiple failure of safety systems is assumed.

The results of the study were applied to the resistance model and investigation of strength of the ACS steel-reinforced structure of the Ignalina NPP. Data on the geometry of ACS and on reinforcement of the walls were accumulated and processed. Construction standards were applied in strength predictions of reinforced concrete to evaluate the carrying-ability of the structure and its performance by the technique of limit analysis. The ACS analysis with finite elements of several types (ALGOR element library) yielded instantaneous reserve strength factors for static loads. Because of the relatively slow rate of loading, the investigation has confirmed that dynamic effects do not contribute additional insight to regular static analysis. Furthermore, because of the internal heating of the confinement structure, thermal effects on the walls would tend to reduce stresses imposed by internal pressure. Thus, neglecting thermal effects would make the static analysis more conservative.

ABBREVIATIONS

ACS Accident Confinement System
CTCS Condenser Tray Cooling System
ECCS Emergency Core Cooling System
LOCA Loss Of Coolant Accident
MRV Main Relief Valve
NPP Nuclear Power Plant
RBMK Russian Acronym for “Large Power Channel Reactor”
TOL Russian Acronym for “Technical Safety Report of Ignalina NPP with RBMK-1500 Type Reactors”

REFERENCES

5. Norms and Roots for Building Structures: SNiP 2.03.01-84, SNiP 2.03.04-84. 1988. (In Russian).
12.3 EXPERIMENTAL INVESTIGATIONS OF MELT SPREADING AND SCALING ANALYSIS

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V.A. Bui, W.H. Leung, A. Gabaldonik, A.T. Dinh

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ABSTRACT

This paper presents results of experimental studies conducted at the Royal Institute of Technology, Division of Nuclear Power Safety (RIT/NPS) on melt spreading. A series of low and intermediate temperature experiments were performed, using different pairs of simulant fluids for melt and coolant. Significant results were obtained from three high-temperature experiments conducted with the corium melt simulant, CaO–B2O3, in a 3.5 meter long by 0.2 meter wide channel. The basemat upon which the melt spread was in one case concrete and in the other two cases a 10mm thick steel plate. Roughly 12 liters of the binary oxide melt at 1200°C was employed.

A scaling relationship between dimensionless spread depth and characteristic spreading time was derived. Experimental measurements from simulant material experiments are presented along with comparisons to the scaling analysis.

1. INTRODUCTION AND BACKGROUND

Severe accident management may play a large role in the design and licensing of the next generation of nuclear power plants. In the case of a reactor pressure vessel (RPV) melt-through during a postulated severe accident, measures to stabilize the core melt within the containment may have to be implemented. One promising measure is to provide a large spreading area for the melt during a postulated severe accident, measures to stabilize the core melt within the containment. In the case of a reactor pressure vessel (R.P.V) melt-through with comparisons to the scaling analysis.

2. EXPERIMENTAL STUDY

The Nuclear Power Safety division at KTH is a partner in the shared cost research project "Corium Spreading and Coolability" (CSC) sponsored by European Commission. The RIT/NPS experimental program in this area is directed to develop a sound scientific and technical understanding of the melt spreading process in order to evaluate the severe accident management design proposed for EPR.

2.1. Experimental Arrangement

A facility was designed for high temperature binary-oxidic melt spreading tests. The geometry of the concrete channel can be seen in Figure 1.

Oxidic melt is generated by inductive heating in a Si-C crucible and, then, remotely poured into a collection device with a 28mm diameter tube at its bottom. The melt exits this tube by gravity flow onto a short inclined section (2m long and 0.5m high) and then flows into the 1-dimensional channel of 2mm width and 3.5m length. The base of the channel is steel-reinforced concrete of 0.8m thickness. Likewise, the sidewalls of the channel are 0.6m thick and 0.2m in height. In each test channel, K-type thermocouples were arranged to detect the progression of the spreading front.

2.2. Experimental Results

Besides the melt physical properties, the melt spreading depends on melt superheat, melt pour rate, and might be affected by conditions of the floor and substrate, e.g., dry or wet floor, different substrate materials. Therefore, melt superheat (from 0 to 100K), melt volume (from 5 to 20 liters) and coolant temperature (from 3°C to 85°C) were varied in the S3E experiments. In total, 45 low and intermediate temperature experiments were performed to date.

Extensive data base was obtained on spreading dynamics, final spread-melt thickness, spreading distances for dry spreading, spreading of melt in liquid coolant.

<table>
<thead>
<tr>
<th>No</th>
<th>Spreading fluid pairs</th>
<th>Coolant</th>
<th>Channel substrate</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Water</td>
<td>None</td>
<td>Steel</td>
</tr>
<tr>
<td>2</td>
<td>Paraffin oil</td>
<td>None</td>
<td>Steel</td>
</tr>
<tr>
<td>3</td>
<td>Cerrobend</td>
<td>None</td>
<td>Steel</td>
</tr>
<tr>
<td>4</td>
<td>Cerrobend</td>
<td>Water</td>
<td>Steel</td>
</tr>
<tr>
<td>5</td>
<td>Eutectic binary salt</td>
<td>None</td>
<td>Steel</td>
</tr>
<tr>
<td>6</td>
<td>Non-eutectic binary salt (30%CaO–70%B2O3)</td>
<td>None</td>
<td>Steel</td>
</tr>
<tr>
<td>7</td>
<td>Eutectic binary salt</td>
<td>Paraffin oil</td>
<td>Steel</td>
</tr>
<tr>
<td>8</td>
<td>Non-eutectic binary salt</td>
<td>Paraffin oil</td>
<td>Steel</td>
</tr>
<tr>
<td>9</td>
<td>Non-eutectic oxide melt (30%CaO–70%B2O3)</td>
<td>None</td>
<td>Concrete</td>
</tr>
<tr>
<td>10</td>
<td>Non-eutectic oxide melt (30%CaO–70%B2O3)</td>
<td>None</td>
<td>Steel</td>
</tr>
</tbody>
</table>

Table 1: Simulant fluids employed in RIT/S3E program.

All the experiments performed, to date, were conducted in open atmosphere. The experimental program named S3E (Scaled Simulant Spreading Experiments) employs simulant materials as melt. Table 1 lists combinations of melt, coolant and substrate materials in the current S3E program.
In 'wet' spreading tests (spreading under coolant) it was observed that initial interactions between melt flow and coolant cause fragmentation of the melt, which cools fast and forms a porous layer of lower density. Subsequent melt flow lifted the porous layer and carried it to a distance determined by flow rate (inertial forces). It was observed that high heat removal rate, in tests with cold water led to slow-down of the spreading front progression. Spreading into a coolant significantly decreased the spreading velocity, especially for a melt with low superheat (above the liquidus temperature).

2.3. Oxide melt spreading

To date three high-temperature spreading experiments were performed, employing a binary-oxide CaO-B₂O₃ melt as working fluid. Substrate materials were 5cm-thick common-sand concrete or 1cm-thick steel. Thermocouples were embedded in the concrete and steel substrates to measure the thermal response.

Experimental conditions of oxide-melt spreading tests can be seen in Tables 2 and 3.

Test 3MDC-Ox-1 and Test 3MDS-Ox-1 are similar, but employing different materials (concrete vs. steel) for the substrate. It was found that the spreading process on the steel substrate occurred smoothly. Instead, the spreading process on the concrete substrate encountered violent interactions between the high-temperature melt and concrete. As the melt front progressed into the concrete channel it was clear that substantial gas generation occurred due to the heating of the concrete. Both unbound and bound water in the concrete was driven out and it was clearly observed in the videotapes that the viscous glassy-like melt had substantial gas passing through it.

Builtup of vapor pressure due to the evaporation of the free and bound water in the concrete, occurred at several locations along the bottom of the concrete channel during the course of this experiment. It resulted in significant disturbance in the spreading melt layer and a small amount of the melt was splashed outside the channel. Gas percolation and collapse of the bubbles on the melt surface caused porosity in the melt layer.

The spreading distances were almost the same in the two spreading tests 3MDC-Ox-1 and 3MDS-Ox-1. Probably, the termination of the spreading process is largely governed by the solidification of the melt leading edge.

Test 3MDS-Ox-1 and Test 3MDS-Ox-2 were performed to investigate the influence of melt superheat on the spreading distance. The pre-test calculations showed that the melt superheat does not affect the spreading characteristics significantly even in the viscous-spreading regime. Change of superheat from 150K to 50K superheat changed the spreading distance from 2.9 to 2.2 m. This is because the sensible heat is only a fraction of the total heat removal needed to immobilize the spreading melt.

Experimental results of the oxide-melt spreading test on concrete substrate are shown on Figs. 2-4. It can be seen that the concrete temperature was maintained at 300°C until the water in the neighborhood of the thermocouple had completely vaporized.

3. SCALING ANALYSIS

Two major phenomena which govern the spreading process are (i) hydrodynamic motion of the spreading liquid (melt); and (ii) solidification of melt during the spreading process. The first phenomenon is affected by gravitational, inertial, viscous and surface forces. The second phenomenon is affected by heat transfer from the melt to surrounding media, i.e. downward $q_{down}$ and upward $q_{up}$, heat removal rates, internal heat generation rate $q_{i}$.
The above phenomena are also affected by physical properties and phase change behavior of the melt. In addition during spreading of melt in another (lighter) liquid coolant, molten fuel-coolant interactions may influence the spreading dynamics.

It should be noted that the spreading process is strongly coupled with the ex-vessel debris long-term coolability issue. The thinner the debris bed or melt pool, the easier it is to cool with the accident management scheme of water flooding. Thus, from the reactor safety standpoint the most useful scaling parameter characterizing the spreading process is the terminal thickness of the spread melt (debris), \( \delta_{sp} \).

Such a view to the spreading process does not rule out the role of the spreading dynamics. Rather, it fosters a focused approach to the analysis of the spreading data. It is likely that the terminal spreading film thickness is not sensitive to experimental design-induced factors, e.g., gate opening mechanisms etc. The present scaling concept does not follow the previous modeling and analysis approaches whose focus was placed on the spreading-dynamics parameters.

A dimensionless length scale of the spreading process, with melt solidification, is introduced:

\[
L = \frac{\delta_{sp}}{\varepsilon_{cap}}
\]  

where \( \delta_{sp} \) is the average height of the spread melt and \( \varepsilon_{cap} \) is capillary thickness in a purely hydrodynamic spreading regime.

Dimensionless time scale is defined as the ratio of competing characteristic times of the hydrodynamic spreading \( \tau_{conv} \) and melt solidification \( \tau_{solid} \):

\[
T = \frac{\tau_{conv}}{\tau_{solid}}
\]  

Based on the solutions of momentum and energy-conservation equation, we obtain the following relations for \( T \) in gravity-inertia regime:

\[
T = \frac{V_{jet}^2}{D \cdot \varepsilon_{cap} \cdot \left( \frac{2 \eta}{G} \right)^{1/2}} \cdot \rho_m \cdot \varepsilon_{cap} \cdot \left( C_{pm} \cdot \Delta T_{solid} + \eta \cdot H_{fusion} \right)
\]  

and in the gravity-viscous regime:

\[
T = \frac{V_{jet}^2}{D \cdot \varepsilon_{cap} \cdot \left( \frac{2 \eta}{G} \right)^{1/2}} \cdot \rho_m \cdot \varepsilon_{cap} \cdot \left( C_{pm} \cdot \Delta T_{solid} + \eta \cdot H_{fusion} \right)
\]  

It should be noted that in this analysis the downward heat transfer coefficient \( h_{down} \) is evaluated by using RIT correlation \( (Nu = 0.0077 \cdot Pr) \) for convective heat transfer to a phase change or rough boundary, \( \eta \) is the fraction of the latent heat of fusion which needs to be removed until the melt becomes immobile. It is found that \( \eta = 0.5 \) provides the best fit to experimental data available.

It can be seen in the above equations that the dimensionless time scale combines the main parameters of the spreading melt \( \{V_{jet}, G, \Delta T_{solid}\} \), geometry \( D \), boundary conditions \( \{h_{conv}, T_{conv}\} \) and the melt physical properties \( \{H_{fusion}, C_{pm}, \rho_m, \eta\} \). The dimensionless length scale \( L \) represents a measurable (and highly reproducible) result of the spreading process (terminal spreading melt thickness, \( \delta_{sp} \)). In order to obtain a scaling law we need to relate the dimensionless length scale \( L \) to the dimensionless time scale \( T \).
Based on the mass conservation equation, we obtain the following scaling relationship for inviscid spreading:

\[ L = C \cdot T^{1/2} \]  

Since the proportionality coefficients in all the equations employed are orders of unity, coefficient \( C \) should be about 1 (\( C \cong 1 \)). This is then confirmed by comparing to experimental data available.

For the gravity-viscous regime, we obtain

\[ L = C_v \cdot T^{1/2} \cdot N^{1/3} \]  

where coefficient \( C_v \) is of order of unity, but likely \( C_v < 1 \), and \( N \) is a viscosity number (\( N > 1 \)) defined as

\[ N = \frac{U_\text{lab}^{1/2} \cdot L_\text{lab}^{1/3}}{D \cdot \rho \cdot \nu} \]  

where \( U_\text{lab}, D, \rho, \) and \( \nu \) are kinematic viscosity, total melt volume, gravitational acceleration coefficient, width of the spreading channel and melt supply flow rate, respectively. It can be seen that higher the flow rate \( G \) or lower the fluid viscosity \( \nu \), the smaller the viscosity number \( N \). If \( N < 1 \), we take \( N = 1 \), and Eq.(6) becomes Eq.(5) obtained for the gravity-inertia spreading regime.

The performance of the scaling relationship was examined against data on spreading distance obtained from spreading experiments performed in the past (SPREAD [1], KATS-5, 6, 7, [2] and COMAS-5a [3]). It is found that the present relationship can describe the dependence of the spreading distance on melt superheat, melt volume and melt supply flow rate. The measured values conform to the inviscid model for the KATS experiments, since the melts employed were at very high superheat and of low viscosity. The COMAS experiment employs a melt with low superheat and substantial viscosity.

Table 4: Analysis of FzK/KATS and COMAS-5a spreading experiments.

<table>
<thead>
<tr>
<th>Experiment</th>
<th>Melt</th>
<th>( T )</th>
<th>( C_{\text{inv}} )</th>
<th>( C_v \cdot N^{1/3} )</th>
<th>( C_v \cdot T^{1/2} \cdot N^{1/3} )</th>
</tr>
</thead>
<tbody>
<tr>
<td>KATS-6</td>
<td>iron</td>
<td>1.48</td>
<td>1.48</td>
<td>2.13</td>
<td>2.13</td>
</tr>
<tr>
<td>KATS-7</td>
<td>iron</td>
<td>3.00</td>
<td>1.53</td>
<td>1.73</td>
<td>2.58</td>
</tr>
<tr>
<td>KATS-S</td>
<td>oxide</td>
<td>1.00</td>
<td>1.38</td>
<td>1.40</td>
<td>1.40</td>
</tr>
<tr>
<td>KATS-T</td>
<td>oxide</td>
<td>3.27</td>
<td>1.75</td>
<td>1.80</td>
<td>3.06</td>
</tr>
<tr>
<td>COMAS-5a</td>
<td>H-corium</td>
<td>4.61</td>
<td>3.27</td>
<td>2.15</td>
<td>3.01</td>
</tr>
</tbody>
</table>

Scaling analysis of the wet spreading tests performed with low and intermediate temperature melts in the RIT S3E program is also shown in Fig.5.

4. SUMMARY AND CONCLUDING REMARKS

In this work, the focus is placed on the final configuration of the melt spreading process, i.e. melt layer thickness and spreading distance, rather than on the spreading dynamics. This allows development of a unique integral scaling relationship for 1D spreading, with solidification, in both gravity-inertia and gravity-viscous regimes.

Scaled simulant spreading experiments were performed, employing a variety of simulant materials as working fluids and under different (temperature, cooling, substrate, flow regime) experimental conditions. It is found that the scaling relationship, developed here, predicts the spreading distance and the terminal spread-melt thickness very well.
12.4 MOLTEN CORE - ZIRCONIA CERAMICS INTERACTION EXPERIMENTS


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Abstract

The article contains some data about zirconia ceramics interaction (~20% pores) with corium components: FeO, TiO₂, NiO, Cr₂O₃ and metallic iron in oxidizing and neutral atmosphere and also with corium imitator in neutral atmosphere. It is showed that the main factor in the intensity of the ferrum oxides penetration in the zirconia ceramics is their reduction degree.

The main process in the melt penetration into ceramic body is capillary impregnation and oxidation vapour transfer.

Introduction

Under conditions of nuclear reactor severe accident with melting of its active zone the composition of the arising corium will be determined by the reactor destruction degree, whether its active zone was melted fully or partially, the reactor body was damaged or not, what is the oxidizing degree of metallic components, especially zirconium. For example, the share of zirconia can vary from 16 to 40 wt% and in some parts of corium can reach 100%. The active zone of reactor VVER-1000 contains: UO₂- ZrO₂, Cr₂O₃, FeO, NiO and some Cr and Ni oxides. According to the calculations the corium temperature can reach 2500-2600 K and more. For this reason the perspective material for core catcher is zirconia stabilized ceramics. Its melting point is 2970 K.

The phase diagrams of ZrO₂-ferrum oxides, ZrO₂-Cr₂O₃, ZrO₂-TiO₂ show that ferrum oxides form eutectics with zirconia with melting point 1600 K (for 97 wt% FeO), 1800 K (for 80 wt% FeO₂), Cr₂O₃ forms eutectic with ZrO₂ at 2130 K (6 mol% ZrO₂); and TiO₂ forms chemical compound Zr₃TiO₇ (Tm=2200 K) and eutectic at 2030 K (80 mol% TiO₂)

In this paper was investigated zirconia ceramics interaction with melts: 1) some oxides, which can arise by steel oxidation (ferrum, nickel, titanium, chromium oxides) and with boron oxide, as in the air as in neutral atmosphere; 2) metallic iron in the same conditions, 3) mixture, which is imitate corium compositions, in neutral atmosphere.

Experimental technique

For investigation of interaction of ceramics with melts there were used samples of the grain structure ceramics (as in [2]) in the form of disks with diameter of 20 mm and height-5-11 mm and crucibles with diameter and height-20 mm. The crucibles had axial deepening with diameter and height-8 mm. The ceramic porosity was 20-22%. Mostly for tests was used ceramic composition 89.5 mol% ZrO₂-10.5 mol% Y₂O₃. In the tests with Fe and Ti oxides also were used ceramics with CaO stabilizer (13.2 mol% CaO). The main phase in two types of ceramics was cubic phase, but ZrO₂-CaO also had 12 mol% of monoclinic phase. This composition have a very high thermo - shock resistance. In two tests there were used dense samples (porosity < 1%) from fused ZrO₂- Y₂O₃-Nd₂O₃ composition.

The powders of the oxides were pressed as tablets with 8 mm diameter, 3-4 mm height and weight about 0.4 g. The same size had iron samples, but their mass was about 1 g.

Experiments with the corium imitator were carried out in the following composition: 25.6 wt% UO₂+10.9 wt% ZrO₂+6 wt% Zr+25.5 wt% steel. This composition was sifted before experiments.

Pressed tablets of oxides and steel samples were placed on the butt-ends of ceramic disks, and pieces of corium imitator were placed into crucibles. The samples were usually weight before and after the tests.

For tests were used vacuum furnaces with atmosphere of argon which contained up to 0.5% oxygen admixture, as well as furnaces of radiation heating in air [4]. In this furnace the heating of samples was made by light flux from xenon lamp of 10 KWT power. In this case the heating was gradient, whereas in vacuum furnace isothermal.

The temperature in vacuum furnaces was measured by W-Re thermocouples and pyrometer. In furnace of radiation heating the temperature was measured by optical pyrometer via rotated modular for excluding reflected from samples light [4]. With radiation heating usual temperature gradient in the sample was about 100°/mm, and heat flux was 0.6-0.8 MW/m². Tests on oxides and steel were made at 2270 K with exposure at this temperature during 1 h. Tests with corium imitator were made at temperatures 2520 K and 2650 K at 1 h exposure, and 2100 K at 4 h exposure. In all cases, besides TiO₂ there were made 2-3 tests. After these experiments the samples were tested by microscopic and x-ray microanalysis.

Experimental results

After the tests in grain structure ceramics arose a hole in the place of contact with oxide tablet. Around the hole there was impregnated zone. Pictures of typical thin section are given at Fig 1-3, and sizes of impregnated zones are in Table 1.
The ceramics by melt of ferrum and titanium oxides. Chromium oxide in argon atmosphere. T1O2 penetrated practically on the full depth of samples and changed the practically was not changed. Another picture was after isothermal testing in argon atmosphere with ceramics was not so intensive as in "hot" layer. In more "cold" areas ceramics the layer of the same depth where initial grain structure was safe, but the density of was accompanied by recrystallization of ceramic grains. After the "hot" layer there .was rather high. The comparison, of original fractional composition of ceramic with depth of 2 mm consisted from particles of 20-50 mkm size. The content of TiO2 was lower subzone the structure was not changed. The destruction area had depth not more 5 mm.

Interaction of ceramics with TiO2, Cr2O3, NiO shows, that in air the impregnation was much less, than it was with ferrum oxide. For example, the investigation of hot part of the ceramic samples after interaction with TiO2 showed, that there was dense layer with depth of 2 mm consisted from particles of 20-50 mkm size. The content of TiO2 was rather high. The comparison, of original fractional composition of ceramic with fractional composition in these zone showed that interaction process was intensive and was accompanied by recrystallization of ceramic grains. After the "hot" layer there was the layer of the same depth where initial grain structure was safe, but the density of ceramic grains laying considerably increased (porosity decreased). Interaction of TiO2 with ceramics was not so intensive as in "hot" layer. In more "cold" areas ceramics practically was not changed. Another picture was after isothermal testing in argon atmosphere. TiO2 penetrated practically on the full depth of samples and changed the colour because of partially reducing process. Chromium oxide in argon atmosphere reduces to metal and although the Cr2O3 tablet was not melted during the experiment due to not very high temperature (2430 K), the oxide penetrating was never the less observed. Experiments with dense ZrO2 samples showed absence of impregnating trace of the ceramics by melt of ferrum and titanium oxides.

Tests with corium imitator showed that visible erosion of the ceramics was absent.

Discussion

The hole in ceramic samples could be formed as a result of zirconia dissolving in the tested oxides and shrinkage of the ceramics impregnated with the melt. Which of these mechanisms was the main, is difficult to determine. But for practical purposes there can be made an assumption in favor of the first one.

Knowing the sizes and weight of oxides, it is possible to estimate the ratio of components in the melts, supposing, that the hole was formed as a result of dissolving of the ceramics in oxides.

![Table 1: Measured and calculated sizes of interaction zones.](image)

<table>
<thead>
<tr>
<th>Ceramic composition</th>
<th>Oxides</th>
<th>Height of samples, mm</th>
<th>Depth of hole, mm</th>
<th>Depth of impregnated zone, mm</th>
<th>Volume of hole, mm³</th>
<th>Volume of impregnated zone, mm³</th>
</tr>
</thead>
<tbody>
<tr>
<td>ZrO2</td>
<td>Fe2O3</td>
<td>10</td>
<td>0.5</td>
<td>7.0</td>
<td>5.4</td>
<td>108</td>
</tr>
<tr>
<td>+Y2O3</td>
<td></td>
<td>10</td>
<td>2.0</td>
<td>4.5</td>
<td>1.0</td>
<td>82</td>
</tr>
<tr>
<td>TiO2</td>
<td></td>
<td>10</td>
<td>2.0</td>
<td>4.6</td>
<td>7.3</td>
<td>140</td>
</tr>
<tr>
<td>Cr2O3</td>
<td></td>
<td>10</td>
<td>0.8</td>
<td>3.0</td>
<td>5.0</td>
<td>82</td>
</tr>
<tr>
<td>B2O3</td>
<td></td>
<td>10</td>
<td>0.3</td>
<td>3.5</td>
<td>10</td>
<td>190</td>
</tr>
<tr>
<td>ZrO2 +Cr2O</td>
<td>Fe2O3</td>
<td>10</td>
<td>0.5</td>
<td>7.0</td>
<td>5.4</td>
<td>99</td>
</tr>
<tr>
<td></td>
<td></td>
<td>10</td>
<td>1.6</td>
<td>3.7</td>
<td></td>
<td>143</td>
</tr>
</tbody>
</table>

(1-gradient heating, in air; 2-isothermal heating, argon atmosphere)

Obtained ratio $W_{oxide}/W_{zirconia}$ in the case of ferrum and titanium oxides are close to literature data for eutectics and chemical compound. But it needs further investigations. High ratio $W_{oxide}/W_{zirconia}$ can mean weak interaction between zirconia ceramics and chromium oxide in air.

It was supposed that the main mechanism of ferrum oxides penetration into zirconia ceramics was capillary impregnation.

In the grain structure zirconia ceramics all porosity is open porosity. If we suppose that all pores can be filled by liquid we can estimate volume and depth of impregnate zone. But real impregnate zone is 2-2.5 times bigger than estimated one, and the porous area are filled maximum up to 50%. At capillary impregnation there are formed gaseous congestion, and also regrouping and tighten of ZrO2 grains takes place. All these slightly increase impregnated area. The lower "cold" zone consist of dark ZrO2 grains, practically not changed, but containing up to 0.09% ferrum oxide. Evidently Fe2O3 was transferred in vapour phase and diffusional saturation of ZrO2 grains.

It is possible to declare that capillary impregnation to the zone bottom can not be and liquid divided in fragments must stay unmoved or move in high temperature region [3].

For practical purposes it is expedient to use the ratio of ceramics weight to oxide weight in the destruction zone. In the ferrum oxides case it is 5:1.

The offered mechanism of impregnation for gradient heating of ferrum oxides is probably true also for other oxides.

The main factor in impregnation and interaction intensively of ferrum oxides with zirconia ceramics is their reduction degree. Absence of any erosion of the ceramics in the tests with corium imitator can be explained by presence of up to 6 wt% Zr in corium. At high temperatures (>1800 K) Zr reduces some oxides, for example Fe2O3 to metallic iron. For this reason activity of corium impregnation and its interaction with ceramics sharply decreases.
The big difference revealed between the Fe and Ti oxides impregnation intensity in porous and dense zirconia ceramics confirms that capillar impregnation and vapour transfer of melt into the ceramic body is dominating process. So reduction of the melt impregnation into ceramics can be reached by modifying the character and value of porosity.

This investigation did not reveal difference between the reaction of zirconia ceramics with stabilizer Y₂O₃ and CaO to resistance with ferrum and titanium oxides melts (the interaction time was 1 hour).

Conclusion

From the results of this investigation there can be made conclusion that interaction intensively of the zirconia ceramics at temperature level~2300 K depends on atmosphere composition (partial pressure of oxygen in gaseous medium), as well as on oxide and metallic phase ratio in the corium, and the ceramic porosity character.

The achieved results showed up also the influence of temperature distribution in the ceramic samples on the size of the interaction zone. Indeed, the NiO, TiO₂, Cr₂O₃ impregnation zones in the ceramics under isothermal conditions are bigger than the same zone at temperature gradient 100°/mm.

The obtained results can be used for valuation of ceramic catcher mass in severe accident in air medium. In this case the ceramic erosion determining factors are ferrum oxides.

Our data shows that the ceramic catcher mass must be not less than 5 times more than that of the ferrum oxides in corium.

Important is the fact that ceramic samples with \( \Phi 20 \text{ mm} \) and height of \( 10 \text{ mm} \) do not destroy at temperature gradient 100°/mm with 200°/min heating speed and natural cooling.

References

12.5 Investigation of Alternative Solutions for Severe Accident Management in Future Reactors

P. Raymond, I. Szabo, P. Richard, P. Marsault
CEA/DRN/DER/Innovative Reactor Concepts Service

1. Introduction
Since 1991, the CEA/DRN "Innovations - Future Reactors" Program (IFRP) has been developed in order to elaborate, evaluate and validate technical options which can be of interest for future reactors.

The main objectives of this program are to improve both the safety and cost of future nuclear power plants, to optimize the fuel cycle and the management of nuclear materials.

For these purposes, the IFRP covers five R&D investigation fields:

- The follow-up and evaluation of some specific reactor systems implemented in next-generation reactors, e.g., the AP-600 from Westinghouse and the Advanced Boiling Water Reactor. The CEA/DRN is also conducting preliminary reactor feasibility studies for the different uses of nuclear energy such as the MAPS spatial propulsion reactor, in collaboration with the Centre National d'Études Spatiales (CNES), and the future industrial irradiation facility of the CEA: the Jules Horowitz Reactor.

- The systems and technology studies, including: new safety engineered systems devoted to remove the reactor residual heat under accident conditions, new structural materials and potential outcomes from the development of data and information processing technologies;

- The innovative severe accident research program (ISARP), aimed at reducing various risks, from the development of data and information processing technologies;

- The core and fuel cycle studies, aimed at improving the flexibility of reactor operation and the follow-up and evaluation of some specific reactor systems implemented in new-generation reactors.

The CEA/DRN actually performs in the frame of cooperative actions with French partners EDF and Framatome, several specific R&D related to the European Pressurized Reactor. Other CEA/DRN "generic" R&D concerning next-generation reactor are developed in the frame of Coordinated and Shared Action coordinated by the European Union in the 3rd and 4th Euratom Framework Programs. They are summarized in Table 1. Most of the results of both these specific and "generic" experimental programs and codes, developed at CEA/DRN, can be used for both existing reactors and next-generation reactors.

2. Framework and objectives of ISARP
After the TMI-2 accident occurred, it was recommended internationally by Nuclear Safety Authorities, more particularly, by the French Nuclear Safety Authority [2], that reactor core melt accidents have to be taken into account when designing future nuclear reactors. As a result, numerous Severe Accident R&D Program have been developed worldwide. They are mainly related to both existing reactors and next-generation Advanced Light Water Reactors.

Table 1 - Theoretical and Experimental Severe Accident R&D at CEA/DRN

<table>
<thead>
<tr>
<th>Name/Location</th>
<th>Main Characteristics</th>
<th>Main Objectives</th>
</tr>
</thead>
<tbody>
<tr>
<td>VULCANO Test facility</td>
<td>Prototypic corium</td>
<td>High Temperature ~ 300 °C</td>
</tr>
<tr>
<td>CEA Grenoble</td>
<td>Corium simulant</td>
<td>3D code</td>
</tr>
<tr>
<td>AEROSTAT CEA Grenoble</td>
<td>Corium simulant</td>
<td>Corium spreading calculations</td>
</tr>
<tr>
<td>TOLBIAC CEA Grenoble</td>
<td>Large structures</td>
<td>Steam Explosion</td>
</tr>
<tr>
<td>SULTAN CEA Grenoble</td>
<td>High Heat Fluxes, Low pressure</td>
<td>Condensation in presence of incondensable</td>
</tr>
<tr>
<td>COPAIN CEA Grenoble</td>
<td>Analitical tests</td>
<td></td>
</tr>
<tr>
<td>BILLEAU CEA Grenoble</td>
<td>Corium Jet simulated by hot sphere</td>
<td>Pre-mixing phase in Steam Explosion (MCID experimental validation)</td>
</tr>
<tr>
<td>MISTRAS CEA Grenoble</td>
<td>Multi-compartment containment</td>
<td>Hydrogen distribution</td>
</tr>
<tr>
<td>DYNA2 CEA Saclay</td>
<td>Spray - droplets simulated by spheres</td>
<td>Multi D test Dynamic aspects of spray</td>
</tr>
<tr>
<td>FRUCTIDOR CEA Cadarache</td>
<td>Two components Argon, Air, Vapor</td>
<td>Transport of gases between two compartments</td>
</tr>
<tr>
<td>KALI-H2 CEA Cadarache</td>
<td>Single component helium</td>
<td>Hydrogen mitigation</td>
</tr>
</tbody>
</table>

The main topics of the ISARP are organized in five investigation fields:

- The development of the technology of advanced fuels, absorbers and burnable poisons for future reactors
- The innovative severe accident research program (ISARP), aimed at reducing various risks, from the development of data and information processing technologies;
- The core and fuel cycle studies, aimed at improving the flexibility of reactor operation and the follow-up and evaluation of some specific reactor systems implemented in new-generation reactors;
- The systems and technology studies, including: new safety engineered systems devoted to remove the reactor residual heat under accident conditions, new structural materials and potential outcomes from the development of data and information processing technologies;
- The follow-up and evaluation of some specific reactor systems implemented in next-generation reactors.
Complementedly to the above severe accident R&D, some of which are very closely related to well-defined Next-Generation Reactors and performed in respective frames, CEA/DRN has developed, since 1991, the "Innovations-Severe Accident Research" to investigate alternative solutions, aimed at solving - by design - issues pertinent to severe accident management (SAM) in conceptual future reactors, the characteristics of which need only to be globally defined.

As a matter of fact, it is generally assumed in most of the "conceptual" studies of ISARP, that - compared to the existing or next-generation reactors - components of the "future reactors" considered can be modified within a reasonable extra cost (which can only be estimated accurately in an integrated fashion at a later and in-deep design stage). Although the cost impact evaluation is uncertain at the preliminary conceptual studies stage, possible solutions theoretically attractive can obviously induce prohibitive extra cost and hence, are to be rejected and not investigated further. Thus cost estimates, even roughly achieved, appeared to be useful in ISARP as one of the most demanding criteria for selecting amongst different "innovative" solutions.

The main results expected from ISARP are:

1. A comprehensive view of the current state of the art on severe accident phenomenology and SAM strategies currently proposed for existing and next-generation reactors,
2. Proposals of alternative solutions and evaluation of their potentiality for application in future reactors,
3. Identification and ranking the R&D needs for demonstration purpose of the feasibility of the most promising "innovative" concepts.

3 - Approach and Methodology:

The approach pursued in order to fulfill the ISARP program objectives consists of:

- Analysis of severe accident scenarios and determination of a collection of plausible "generic" situations, i.e., which can be found in a large number of scenarios. Identifying and listing the phenomena and then rank them for their relative importance (PIRT elaboration) with regard to the remaining issues in severe accident management (SAM),
- Analysis of solutions proposed for the resolution of these remaining issues,
- Identification of possible alternative solutions and selection of the most promising of them,
- Conceptual studies of the selected management strategy or mitigation system, with a view to demonstrate their feasibility, their operability and their potential efficiency.

4 - Examples of ISARP Studies:

The very first step, both necessary and useful for the development of innovative activities in SAM, is to critically survey the state of the art and to identify thoroughly the remaining problems or unresolved issues. This was done in ISARP through literature surveys [1, 4] and thorough analysis of the phenomenology involved in the numerous severe accident scenarios.

4.1 Scenario and Uncertainty Analyses:

Prior to the development of new mitigation concepts/strategies in the frame of ISARP, the evaluation of the state of knowledge as well as still unresolved issues and remaining uncertainties actually consists of:

- Analyzing, as thoroughly as possible, the severe accident progression in different constitutive parts (vessel, primary circuit, containment,...) not only of existing reactors but also, of future reactors, possibly equipped with mitigation devices (in- and ex-vessel core catchers, hydrogen igniters and recombines, etc.),
- Identifying and ranking the key phenomena involved in these scenarios: to accurately give the order of priority of the R&D needs.

Analyses of the severe accident progression phases in-vessel, ex-vessel and in the containment, were performed within the framework of the GAREC (Groupe d'Analyse des besoins de R&D liés à la Réception du Corium) and of the GIC (Groupe Innovation Confainment).

The methodology used, similar to ISTIR [5], includes the elaboration by CEA/DRN specialists of:

- PIRTs (Phenomena Identification and Ranking Tables), and
- Associated RDNIRTs (R&D Needs Identification and Ranking Tables) which would indicate the relative priority of the R&D needed for demonstration purpose of the feasibility of a concept as well as its operability and efficiency. The "weight" used for ranking the R&D needs must take into both the current state of the art and results expected from the ongoing international R&D related to the same concept/strategy or the same technical "issue".

These severe accident scenarios and uncertainty analyses aimed in providing pertinent results on all aspects of a severe accident. PIRTs and RDNIRTs resulting from these critical analyses provide deeper insights in the threats and tools to be taken into account in the study and the possible development of prevention and mitigation means. Generally, a single severe accident sequence or the entire scenario is evaluated using various computational tools from simple analytical calculations to the use of integrated codes, such as MAAP 4 [6].

During the last two years, the scenario analyses, conducted at the CEA/DRN concerned mainly the "generic" sequences of the release and spreading of corium onto the EPR core-catcher area [7].

In-vessel scenarios are presently analysed, using the GAREC approach, with the view to evaluate the possibility to retain the corium in vessel through external cooling in the case of a 1 400 MW reactor [8].

In the frame of ISARP, an alternative in-vessel retention "dual" strategy, based on the combined use of an in-vessel core-catcher and reactor cavity flooding, has been recently investigated [9]. Scenario related to a future reactor vessel equipped with an in-vessel core catcher are being investigated. These analyses are carried out so as to identify the problems raised by this alternative candidate strategy.

Concerning the reactor containment integrity, specific attention is given to the analysis of hydrogen production and distribution scenarios. PIRTs of representative sequences (in-vessel, ex-vessel) are being elaborated with a view to identify and rank the key phenomena. The main objective is to evaluate how to reduce the hydrogen, water and gas release production and kinetics are compatible with current mitigative measures, performance and also possible high local hydrogen concentrations in containment compartments. The RDNIRTs results expected would improve "probabilistic" evaluation of mitigation devices and test matrix to be conducted in the experimental facilities in relation to this theme. They are also useful to optimize the mitigation device number and location in containment and, finally to envisage - for future reactor- containment building architecture which integrates the very early design stage hydrogen risk mitigation strategy.

The "phenomenological" analyses of the consequences of generic situations are completed by a "probabilistic" evaluation of the effects of uncertainties which could affect the initial and boundary conditions these situations depend on. As a tentative and very rough assumption, the ROAAM (Risk-Oriented Accident Analysis Methodology of T. G. Theofanous [10]) was used. For illustrative purposes, the straightforward transposition of ROAAM to the EPR "melt core" analysis is shown on...
As stated above, scenario calculations concern either the entire scenario or only a few specific sequences. A "two rice" approach is generally pursued in code utilization, i.e., fast-running integrated code for the complete scenario calculations and detailed computer codes for sequences analyses. Both integrated codes and detailed "mechanistic" codes are actually available at CEA/DRN.

Reflection about the most convenient way to efficiently use the existing theoretical models and codes developed by or available at CEA/DRN for scenario and uncertainty analyses has resulted in a preliminary set of "requirements for a severe accident scenario evaluation code", dedicated to severe accident scenarios/sequences and their associated uncertainties calculations. More in-depth discussions about requirements will be pursued in 1998 at CEA/DRN, involving scenario phenomenology analysts, model/code development teams, code users, and specialists of advanced computing techniques, etc.

4.2 Ex-vessel corium core catchers

One of the main issues still to be solved is ex-vessel corium coolability and retention inside the containment building without radioactivity release to the environment. To achieve this, different concepts which allow the melt core to be retained either outside or inside the reactor vessel are studied.

From a thorough survey and analysis of the proposed ex-vessel core-catcher concepts [4], a categorization in two types of strategies was made, based upon the way the cooling water is used to cool the corium down, namely the "wet" and the "dry" strategy. Wet strategy is applied preferably in recent next-generation reactors.

At the CEA/DRN, the application of the alternative "dry" strategy leads to the elaboration of two concepts: the multi-crucible core catcher [11] and the flat corium catcher with promotion of radiative heat transfer [12]. Both these core catchers use containers to collect and cool the corium. They only differ in their way of achieving an exchange surface sufficient to remove the corium residual heat. In the first concept (Figure 2), the shape, a very long cylinder, which was adapted for the crucibles, provides a large surface to volume ratio which greatly helps the cooling process. In the flat corium catcher (Figure 3), corium residual power is removed, on the one hand, by conduction through the bottom of the core catcher vessel, on which the corium has spread, and on the other hand, by radiation on other metallic parts of the same vessel, entirely cooled from outside by natural water circulation.

The multi-crucible core catcher (Figure 2) consists of:
- a corium collector surrounding the vessel;
- several dozen vertical crucibles attached to the bottom of the collector and placed right under the bottom of the vessel;
- a passive cooling system.

Figure 2: Functional diagram of multi-crucible core catcher
The flat catcher with promotion of radiative heat transfer is different from the first one in its use of a single container to collect and cool the corium (large vessel with a flat bottom, the lateral walls of which are designed to receive most of the radiative heat from the free surface of the corium spread at the bottom).

![Flat corium catcher with promotion of radiative heat transfer](image)

**Figure 3**: Flat corium catcher with promotion of radiative heat transfer

The two core-catchers have similar operation modes:

- In the case of a severe accident, the liquid and solid debris ejected from the vessel are intercepted by the walls of the corium collector and accumulate at the bottom of the latter. They ablate the water tight metallic liner, melt the plugs at the crucible inlet and flow down into the crucibles (in the vessel in the case of the flat corium catcher).

- Some time before reactor vessel rupture, the automatic opening - upon an emergency shutdown signal or preferably a core meltdown detection signal - of the feed valve (cf. Figure 2) allows the coolant to penetrate in the space provided outside the corium containers (crucible assembly or single large vessel). In the multi-crucible core catcher concept, the residual heat is entirely removed by conduction through the walls whereas in the flat catcher with promotion of radiative heat transfer, it is removed both by conduction in the upper and lower parts via the surfaces designed to receive the radiated energy. The steam resulting from coolant evaporation condenses in the containment compartments and draining flow paths are provided to allow the condensed water to return to the feed water tank. Natural circulation of water and steam can be then established in different containment compartments. Ultimate residual heat removal means are foreseen in order to ensure the long term cooling.

An overview of the R&D performed with a view to demonstrate the feasibility of the CEA/DRN dry core-catcher concepts were presented at SARJ’96 [13]. Various aspects of the overall thermal hydraulic behavior of the multi-crucible core catcher, as predicted by coupling existing CEA/DRN codes are illustrated on figure 4.

![Calculation scheme used for multi-crucible core catcher studies](image)

**Figure 4**: Calculation scheme used for multi-crucible core catcher studies.
4.3 Study of in-vessel corium retention

In-vessel retention strategies are considered as possible alternative solutions to ex-vessel retention strategies.

In-vessel retention studies, performed in the frame of ISARP include:
- the definition of the functional specifications required by in-vessel retention system which must moreover remain compatible with normal reactor operation,
- the investigation of concepts meeting these specifications,
- the feasibility study of the most promising solutions.

At CEA/DRN, the very first in-vessel retention strategy investigated, namely the “dual” strategy, presented in detail at this workshop [9], is briefly summarized hereafter.

The dual strategy is based upon the combined use of:
- water injection in-vessel, at the core-melt onset detection, to cool down either a badly degraded core or a large mass of debris collected in an internal core-catcher (figure 5)
- cavity flooding for ex-vessel cooling purposes,

and is proposed as a candidate strategy for in-vessel corium retention applicable for high operating power-range LWRs (~ 1400 MWe).

Preliminary feasibility of the strategy will be pursued, using almost the same experimental database and computing tools as shown on figure 6. The thermo-mechanical design is achieved using analytical calculations from the CASTEM 2000 code [14]. Primary circuit and corium pool thermal-hydraulics are evaluated by coupling the CATHARE [15] and TOLBIAC [16, 17] codes. Core degradation scenarios are calculated using the MAAP4 code.

2.4 Studies on innovative containment concepts

The containment is the last barrier which prevents the release of fission products to the environment. By design, the containment is built to resist to conditions resulting from design basis accidents. For future reactors, the following points must be taken into account in the design of the containment:
- the pressure and temperature loads resulting from a core meltdown accident,
- the management of the hydrogen generated,
- the implementation - in the reactor containment - of new systems devoted to preventing and mitigating of the consequences of severe accidents.

Such mitigative systems have been investigated at CEA/DRN in the framework of the “Innovations - Future Reactors” Program and described in reference [18]:
- the use of a fast depressurization system prevents core meltdown accidents under high RCS (Reactor Coolant System) pressure,
- primary and/or secondary residual heat removal systems,
- the design of corium catcher systems,
- the evaluation and management of in-containment coolant resources,
- the study of means allowing the hydrogen explosion risk to be minimized.

All these systems must operate in a coherent and complementary way in the case of severe accidents in order to minimize the loads on the containment and releases of fission products.
With this in mind, the CEA has developed the ROSALIE containment concept [19] for PWRs, of which the basic design principle integrates both the safety and economic aspects. The studies on severe accidents have provided more detailed insights to the problems linked to the coherent use of different mitigation means but have also permitted a first optimization of the concept, thus resulting in the minimization of volumes and costs, by sharing resources.

The basic principles of this concept are the following (Figure 7):
- separation of the containment into two large primary and secondary compartments (drywell and wetwell), thus allowing, as in Boiling Water Reactors, the pressure increase to be limited in the containment by means of a pressure suppression pool;
- possibility of inerting the primary containment,
- the annular pressure suppression pool contains 1,000 m³ of water. It can also be used for the cooling of the ex-vessel core catcher and for aerosol scrubbling,
- the entire primary circuit is covered by an internal metallic tight containment, which however allows the reloading, maintenance and replacement of large components (such as Steam Generators) to be performed,
- management of steam and hydrogen flowpaths in the primary compartment of the containment in order to condense the former and to better recombine the latter in the secondary compartment so as to prevent any hydrogen explosion.

The evaluation and optimization of the ROSALIE containment concept was performed using the CONTAIN code [20] and by taking into account « generic » phases of severe accident scenarios:
- the depressurization phase of a large break LOCA allows the pressure loads to be predicted depending on the design of the drywell,
- the in-vessel core meltdown accident phase, during which the oxidized zircaloy produces high temperature hydrogen which is released in the containment. This phase was used to verify the efficiency of the foreseen hydrogen mitigation means,
- the ex-vessel core melt accident phase in which the removal and proper cooling of the corium are verified. The possible interactions of this corium with water are also taken into account at RPV failure, which leads to additional production of steam and hydrogen.

Variation of the gas mixture concentration in a reference scenario is shown on figure 8 for illustration purpose. Plots on the Shapiro diagram do indicate that the use of mitigation devices actually prevent the gas mixture « curve » to « enter » the deflagration zone during the whole scenario.

3. Conclusions
The objectives of the CEA/DRN "Innovations - Severe Accident Research " Program (ISARP) are:
- To improve the knowledge on core meltdown accident scenarios and on accident management so as to better identify and rank the important phenomena and take into account, in the design, severe accident prevention and mitigation means,
- To conduct studies on systems and concepts thus meeting the demands in safety improvement for future reactors, particularly by integrating severe accidents on the conceptual stage by using prevention and mitigation means which allow the third barrier, which is the containment, to preserve its integrity.

These studies, which began a few years ago, result in a better understanding of the physical phenomena to be modeled and in the elaboration of technical solutions for the problems raised. These technical solutions, of which the pertinence is validated versus the assigned objectives, must also be optimized and evaluated economically.

They contribute to a better orientation of theoretical and experimental R & D programs. The organization of these studies based on the analyses made by groups such as the GAREC and the GIC has entailed the pooling of the competence and expertise necessary to the solving of the problems linked to severe accident management and thus to increase the synergy among a large number of laboratories of the CEA/DRN.

REFERENCES


13. Panel Discussion

Severe Accident Research for Future Reactors

Chairperson: K. Abe of JAERI, Japan
Panelist: B. Clement of IPSN, France
H. H. Hennies of FZK, Germany
A. Omoto of Tokyo Electric Co., Japan
C. S. Kang of Seoul National University, Korea
A. Merzliakov of RRCKI, Russia
A. Behbahani of USNRC, USA
Panelists: B. Clement (CEA/IPSN, France)
           H. H. Hennies (FZK, Germany)
           A. Omoto (TEPCO, Japan)
           C. S. Kang (Seoul National Univ., Korea)
           A. Merzliakov (RRCKI, Russia)
           A. Behbahani (USNRC, USA)

Chairperson: K. Abe (JAERI, Japan)

1. Opening

Abe opened the session of Panel Discussion. He briefly introduced himself and showed the theme of the Panel Discussion. Each Panelist then introduced himself on his background and current activities concerning severe accident.

2. Overall Procedures

Abe showed the agenda of the Discussion. It consists of the following items to be firstly presented by each Panelist:
   (1) Basic strategy for the development of future reactors
   (2) Safety objectives for future reactors
   (3) Severe accident measures for future reactors
   (4) Important severe accident research for future reactors
   (5) Others

Abe then indicated the following issues to be thereafter discussed among Panelist and participants on the floor:
   (1) What safety level is sought in designing future reactors?
       Tolerable but improvement requested if reasonable, or
       Widely acceptable and no more improvement required
   (2) Are there special design characteristics in proposed future reactors for preventing or mitigating severe accidents initiated by external causes?
   (3) For designing future reactors, what further severe accident researches are needed?
   (4) Is there a need of other severe accident research even with achievement of widely acceptable safety level by design? If yes, what researches are needed and why?
3. Panelists' View on Selected Items

Clement presented ISPN severe accident research priorities to prevent and mitigate the accident in compliance with the defense-in-depth strategy in three major areas; core degradation and melt progression, fission products behavior, and containment behavior. He also presented the important issues for EPR Project, such as methodologies with new features, safety demonstration and improvement of containment function. He emphasized that the code validation is essential for both current and future reactors, and that both separate effect and integral experiments are important.

Hennies mentioned that German Atomic Low requires that future reactors be designed so that evacuation should not be needed outside of the plant. Severe accident research has been widely conducted since TMI and restarted after Chernobyl accident in Karlsruhe. Based on these studies, several measures have been taken, such as containment venting to cope with long-term pressurization of containment. For future reactor developments, mostly for EPR, hydrogen and steam explosion issues have been investigated.

Omoto discussed containment performance targets with focus on the severe accident. Japanese utilities and nuclear industry are currently developing self-regulatory guidance document for the design of the containment for future LWRs. This guidance document utilizes both deterministic and probabilistic approaches so that the designer can confirm the compliance to the probabilistic safety criteria and the containment capability to withstand certain severe accident loads under the provisions of Safety Margin Basis or Design Extension Conditions. He showed the important future research areas, such as hydrogen including radiolysis, in-vessel retention, and melt coolability during core concrete interaction.

Kang presented five basic strategies of developing future reactors; greater visibility of safety, better economics, greater use, more reliable services, and better use of uranium resources. He then showed the safety objectives for future reactors in terms of defense-in-depth with several examples in both preventive and mitigative measures for KNGR. He concluded that the safety improvement should have two principles; newly-introduced systems will not noticeably increase existing societal risk, and enhancement of safety should be discretely implemented based upon cost-benefit.

Merzliakov indicated three main goals of nuclear safety research; 1) protection of peoples, society and environment, 2) reduction of total risk from NPP, and 3) improvement of public opinion. By reviewing the historical methodology of the safety evaluation from engineering approach, PSA, to hazard management, he mentioned that the required knowledge base has been largely increased. Concerning core melt retention, he pointed out that corium stratification observed in RASPLAV Project draws new several questions.
Behbahani presented the ex-vessel cooling issue for future reactors, including AP600, as a desirable mean of heat removal. Since the lower head integrity is an important phenomena, research on critical heat flux on a downward facing surface is being conducted at Penn States University supported by USNRC. He emphasized that the associated phenomena should be investigated separately at one hand, and also new phenomena may arise from integral experiments, such as RASPLAV.

Abe thanked all the Panelists for their presentation. He then asked the Panelists if there is any probability target for future reactors in relation with evacuation.

Kang mentioned that the KNGR safety goal is that the total core damage frequency (without earthquake and sabotages) be less than 10^{-5}/ry, and containment failure frequency be less than 10^{-6}/ry. The design objectives are 10 times lower, respectively, by consideration of cost-benefit.

Hennies mentioned that German Atomic Low requires that future reactors be designed basically in deterministic way to avoid evacuation. But there is a freedom for the interpretation as complementary use of the probabilistic method, which has been clarified through recent research. He emphasized that the decision should be made by regulatory authorities and research side should provide scientific base.

Clement supported Hennies' last points that the safety research should provide useful information and regulatory authorities should make a compromise between the two.

Omoto mentioned, as an engineer in utilities, that the design target should be developed. The target value of 10^{-7}/ry is tentatively set for the avoidance of evacuation from early containment failure. Also 50mSv is set for the level of no/limited emergency planning.

Merzliakov mentioned that the probabilistic assessment largely depends on the models, which usually valid within limited parameter ranges. He also said that the different initial and boundary conditions may alter the results, since the severe accident involves so many complex phenomena.

Abe asked the Panelists if there is any further questions or comments among Panelists.

Hennies commented on the future research from risk perspectives and consideration of operational conditions. He mentioned that EPR with large generated power may have large risk for in-vessel coolability, and that ex-vessel cooling, including the use of core catcher, may be needed. He said that the timely operators' action for the ex-vessel cooling is essential.

Kang commented on the ongoing new safety licensing rule in Korea for KNGR development,
which consists of 1) design basis accident analysis, 2) deterministic severe accident analysis, and 3) probabilistic safety assessment. He emphasized that the good balance between scientific curiosity and economy should be maintained.

Abe asked additional comments from the participants on the floor.

Firnhaber (GRS, Germany) commented on the research activities, especially on the code development, such as ATHLET, COCOSYS, RALOC and ASTEC, ongoing in GRS with France. He emphasized that this is important not only from research interest but also required from licensing authority, and that the work can only be accomplished with best-estimate codes.

Szabo (CEA/DRN, France) commented on the alternative solutions for severe accident management in future reactors. The methodology used consists of analyses of severe accident scenarios/sequences; analyses of alternative solutions for mitigating their consequences; identification and ranking of the R&D topics still needed for demonstrating the feasibility of a selected solution and its possible application to future reactors. As shown in the previous session, he said that the method employed, especially in elaborating the "dual strategy" for in-vessel corium retention purposes, was found to be effective.

4. Discussion among Panelists and Floor

Abe opened the discussions among Panelists and participants on the floor. He showed an example of internal and external risks, and raised a question on possible design features for reducing the risk from external events in future reactors.

Kang mentioned that the criteria of the seismic design of KNGR is increased from 0.2 G to 0.3 G. The military sabotage from the North Korea is the biggest problem, but he hopes that the four-party negotiation will come out a good solution.

Omoto commented on the research area and possible several new features to cope with external events. He emphasized that the total (internal + external) risk should be less than the acceptable certain fraction of societal risk, although the uncertainties associated with external events may not be largely reduced in future.

Clement mentioned that the geological research on the historical earthquakes is being conducted for the possible effect on EPR design.

Omoto commented that the quantitative determination of the hazard curve is really difficult, and that both historical records and tectonics features are utilized for the evaluation of local earthquakes in
Hennies commented on the consideration of the seismic design in Germany. He emphasized that people concerned about earthquakes usually do not care of severe accidents.

Jones (JRC Ispra, Italy) mentioned that severe accident researchers pay little attention because earthquake is just one of initiators, but not a severe accident itself.

Abe then moved on the issue of further research needs for future reactors or remaining issues. He gave some examples, such as in-vessel retention and hydrogen control.

Clement mentioned that in-vessel retention should be properly addressed in in-depth manner; if the in-vessel retention cannot be achieved, ex-vessel retention in containment should complimentarily be considered.

Omoto commented on research areas for in-vessel retention. For PWR external flooding may be feasible, but for BWR other measures such as spray of lower vessel, control rod drive hydraulic system injection in the bottom, or pouring water into reactor vessel may be practical.

Nagasaka (NUPEC, Japan) explained the new in-vessel retention method. This utilizes a lower head outer surface cooling combined with in-vessel cooling, which enables the immediate cooling after coolant injection. Even if this cooling method could not attain in-vessel retention, the accumulated saturated water could be used for ex-vessel debris cooling with low possibility of steam explosion. Hence the activation of water injection by operators will be warranted. He noted that this concept is similar to those proposed by France and Korea.

Allison (ISS, USA) commented that in-vessel or ex-vessel retention is not always possible, and that TMI accident or AP600 with small power may be a special case. He emphasized that the combination of both in and ex-vessel flooding will be needed, especially for large core reactors.

Clement commented on the IPSN hydrogen work for the catalytic recombiner development, which involves complex phenomena, such as physical and chemical poisoning effects of aerosols.

Hennies mentioned that the hydrogen distribution in containment is also a complex and complicated problem. Even for existing reactors hydrogen control is not a resolved issue in Germany.

Omoto commented on the safety margin basis for hydrogen in three areas, deflagration to detonation transition (DDT) evaluation, combustible gas control system design, and containment pressurization. Radiolysis and hydrogen absorption technology are important research issues.
Firnhaber commented on the safety requirements on hydrogen for future PWRs from German and French high level experts committee; global hydrogen detonation must be practically eliminated and provision must be taken with respect to local detonation and possible DDTs.

Mneev (IVTAN, Russia) mentioned that the military technologies are being utilized for designing explosive-resistant multi-barrier containment in Russia. Initial experimental results for steel and concrete containment have been obtained and he suggested a proposal to create an international science and technology commission for the assessment of the study and future applications.

Abe thanked this interesting proposal and mentioned that this will be treated with suitable people. Abe asked attendants for any other comments for any issues.

Sugimoto (JAERI, Japan) commented that steam explosion is still an open issue even for future reactors as identified at OECD Specialists Meeting on FCI in Tokai in May 1997. This will be further important if in-vessel or ex-vessel flooding may be failed. He pointed out that some fission product behaviors, such as revaporization and resuspension, are also unresolved important issues for future reactors.

Allison commented on the possible large effect of high burnup fuel on the core melt, debris bed and fission products behaviors.

Omoto mentioned that in the case of reactivity initiated accident, the code calculations with the high burnup fuel model look mild, provided the pressure boundary is preserved.

Allison agreed that the results will be benign in the design base accidents. He mentioned, however, that the consequences may be largely different for severe accident conditions mostly due to large difference of material properties.

5. Summary and Conclusions

Abe summarized the highlights of the Panel Discussion; we had a good discussion on the safety target, research needs for future reactors, specific research topics, such as in-vessel retention, hydrogen and some other issues.

Lastly he thanked all the Panelists and participants for their active contribution to the Panel Discussion.
ADOPTED PROCEDURE FOR THE DEFINITION:

TO DERIVE THE DESIGN OF THE UNITS FROM THE DESIGN OF EXISTING PLANTS IN AN "EVOLUTIONARY" WAY.

TAKING INTO ACCOUNT:

- THE OPERATING EXPERIENCE,
- THE IN-DEPTH STUDIES,
- PROBABILISTIC SAFETY ASSESSMENT.

THE INTRODUCTION OF NEW FEATURES MUST BE TAKEN INTO ACCOUNT:

- THE PREVENTION OF SEVERE ACCIDENTS,
- THE MITIGATION OF THEIR CONSEQUENCES.
THE OBJECTIVE OF A SIGNIFICANT REDUCTION OF THE RELEASES OF RADIOACTIVE SUBSTANCES IMPLIES A SUBSTANTIAL IMPROVEMENT OF THE CONTAINMENT FUNCTION:

- THE ADVANTAGES AND THE DISADVANTAGES OF A SPRAY SYSTEM INSIDE THE CONTAINMENT MUST BE CAREFULLY EXAMINED AS REGARDS THE H2 RISK,

- THE RESIDUAL HEAT MUST BE REMOVED FROM THE CONTAINMENT BUILDING WITHOUT VENTING DEVICE,

- THE CONTAINMENT BUILDING MUST BE DESIGNED SO AS TO WITHSTAND BOTH A GLOBAL DEFLAGRATION OF THE AMOUNT OF HYDROGEN AND A REPRESENTATIVE FAST DEFLAGRATION,

- THE PENETRATION OF THE BASEMAT OF THE CONTAINMENT BUILDING BY A CORIUM MUST BE AVOIDED (CORE-CATCHER ?).

SAFETY DEMONSTRATION:

- SINGLE INITIATING EVENTS HAVE TO BE « EXCLUDED » OR « DEAL WITH »,

- TO « PRACTICALLY ELIMINATE » ACCIDENT WHICH COULD LEAD TO LARGE EARLY RELEASES.

IMPLICATIONS FOR THE « OUT OF VESSEL »:

- ACCIDENT SEQUENCES INVOLVING CONTAINMENT BYPASS AND PREVENTING FAILURES MUST BE « PRACTICALLY ELIMINATED » BY DESIGN PROVISIONS,

- THE HIGH PRESSURE REACTOR CORE MELTDOWN SITUATIONS MUST BE « EXCLUDED » BY PREVENTING MEASURES,

- GLOBAL HYDROGEN DETONATIONS AND STEAM EXPLOSIONS INSIDE AND OUTSIDE THE VESSEL WHICH MAY THREATEN THE CONTAINMENT INTEGRITY MUST BE « PRACTICALLY ELIMINATED ». 
A. Omoto (TEPCO)

**ABWR design**

Insight from PSA of then-current 1100Mwe BWR5 design:

1) to enhance high pressure ECCS system
2) to enhance the residual heat removal system
3) to diversify the driving force for Control Rod Drive

Later in 90's

4) to provide additional capabilities for SAM to use all available on-site resources

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**Comparison of Core Damage Frequencies**

Source: GE paper NUREG-1150 by Redding

Japanese Activities for the ALWRs

- ABWR-II (Evolutionary)
- JSBWR (Natural circulation, SBWR uprate)
- RBWR (ABWR-Breeder/High Converter)
- APWR-> NEW PWR 21 (Evolutionary, Hybrid)
- SPWR (1000Mwe, AP600 uprate)
- SPWR (by JAERI)

- Super-critical LWR

Next Generation LWR Program

**GOAL** (industry projects) in essence
Economics, User-friendliness, Safety
Plant Constructable in 2010's
Plant Output: 1500Mwe (stretchable to 1700Mwe for ABWR-II)

**ABWR-II**

Phase I (FY 91-92) Phase II (FY 93-95)
- Plant Concept - nuclear boiler focus

Phase III (FY 96-98)
- Entire Plant Concept
- T & D Program
**Next Generation ABWR**

**Design features**
- Large Fuel Bundle
- Reduction in Refueling Time & numbers of CR
- Uprating
- Advanced Safeguard
  - Active + Passive Containment Cooling
  - Advanced RCIC
- Higher conversion
  - Short core: a possible option

**Safety of the next-generation LWR**

*Probabilistic & Deterministic approach*

1. Probabilistic safety targets
   - IAEA INSAG-3: $CDF < 10^{-5}/Ry$
   - Release beyond the level req'ng evacuation $< 10^{-6}/Ry$
   - Generation of energetic load that may lead to early containment failure $< 10^{-7}/Ry$

2. Confirm capability of containment to cope with mechanical loads associated with severe core damage
   - Safety Margin Basis (Design Extension Condition)
   - Reduce operator burden associated w/AM
Safety Margin Basis
• Industry’s Self-regulatory Documents
• FY 1996-97
• Procedure to evaluate & give design/SAM considerations if necessary
  Hydrogen (DDT evaluation, CGCS design, Pressurization)
  DCH
  Ex-vessel FCI
  Core-concrete reaction
  Recriticality
  Over-pressure & -temperature
  Bypass & leakage
  IVR

Safety Margin Basis - Hydrogen -

DDT evaluation
Less than 13\% under MW reaction of 75\%AFC (dry, local)
Compartment effect: Nupec tests

CGCS design
5-95\% confidence level in hydrogen production in analysis
  PWR 100\%AFC
  BWR w/inert containment 10\% AFC

Radiolysis: Halogen(50\%) particle (1 \%) in water
G-value by experiments, boiling/non-boiling

Containment pressurization
Service level C stress limit or other appropriate
Best estimate in load combination
Safety Margin Basis - Core-concrete reaction -

Evaluation basis

Participating debris volume: Full core  Timing: Best estimate

Acceptance criteria

1) No significant concrete ablation: Use EPRI criteria tentatively
   - surface area of 0.02 sq. m/Mwth
   - equivalent to 0.5 Mw/sq. m surface heat flux from debris
   - MACE & other tests not supportive to EPRI criteria so far
     Scale effect: Does stable crust persist?
     Physical process: slumping/water layer not fully included

2) Limited non-condensible gas generation
   BWR with small containment: 500 k mol (ABWR, 2xPd)

3) Use of basalt concrete

Safety Margin Basis - Recriticality -

1) Evaluate recriticality
   in-vessel & ex-vessel

2) Confirm if the time window is very small or the configuration makes recriticality difficult

3) Consideration in SAM to add boron solution in case resuming in-vessel injection
**Safety Goal / Performance Target**

**Offsite dose target**
--> Performance target
& Release target

**Issues:**
1) Definition of "cut-off probability" (avoidance level of early containment failure etc)
--> Tentative setting: $10^{-7}$/Ry
2) Definition of the level for "No / Limited Emergency Planning"
--> Tentative setting: 50 mSv

---

**Cut-off probability**

**Current domestic licensing**

Turbine missile, airplane crash $< 10^{-7}$/Ry
Load combination: Seismic + Accident

**International**

$10^{-6}$-$10^{-7}$ risk level in
IAEA Safety Series 89, ICRP Pub. 46, USNCRP 87
URD: No CV analysis required for sequences $< 10^{-7}$/Ry
AECB: Neglect consequential event below $10^{-7}$/Ry

**Toxic pollutant regulation**

$10^{-5}$ /lifetime for pollutant without threshold $\rightarrow 1.4E(-7)/Y$
WHO drinking water guide
Areas for research <In-vessel Retention>
- Can be a substitute for a robust containment?

- PWR: **External flooding**
  IVR test also by Japanese PWR Utility/vendor

- BWR: **Not practical to consider external flooding**
  A "spray" of lower vessel from outside may be practical
  Massive heat sink in the bottom
  Low core power density
  Many stainless steel penetrations in lower head
  CRD-Hydraulic System injection in the bottom
  Pour water into RPV rather than to cavity

Areas for research <Core-concrete Reaction>

**MACE & other test results:**
Not fully supportive to EPRI criteria (0.02 Sq. m/Mwth)
Translation into real plant phenomenon
  Scale effect: Does stable crust persist?
  Physical process: slumping/water layer not fully included

**Establish Design Criteria**
Complex physical process
Expect simplicity such as area & water volume/Mwth
Areas for research  <Hydrogen>

1) Radiolysis
   <Importance of radiolysis for a small containment>
   - Combination of H2(M-W reaction) with O2 (radiolysis)
   - Current criteria for CGCS(such as R.G. 1.7): need change
   <Research subjects>
   - Radiolysis in submerged debris bed
     (not included in the current source term)
   - G-value for metallic components (Effect of Halogen: known)
   - Combination methodology

2) Hydrogen gettering technology
   - Important pressurizing source for a small containment
Severe Accident Consideration and Research in Future Reactors

October 1997

C.S. Kang
Professor
Seoul National University

basic strategy of developing future reactors

- greater visibility of safety: risk within acceptable level
- better economics: cost advantage over conventional thru standardization (KNGR) & modularization (SMART)
- greater use: district heating, cogeneration, desalination (SMART)
- more reliable services: diversification of reactor types (PWR, PHWR)
- better use of U resources: Liquid Metal Reactor (KALIMER)
safety objectives for future reactors
Defense in Depth

- prevention of accidents: reduction of CDF
  - measures taken at different levels of defense
  - strengthening their independencies
- mitigation of accidents: containment of fission product releases
  - strengthening containment functions
    - (large hydrogen detonations, large steam explosions)
- large early releases to be "practically eliminated"

Possible means to achieve these objectives:
- improved man-machine interface
- increased thermal inertia
- decreased complexity
- increased use of inherently safe and passive elements
- systematic consideration of accident management

PREVENTIVE MEASURES FOR KNGR

- Low coolant temperature, low power density, & large S/G water inventory
- SDVs (safety depressurization valves) on PZR for feed-and-bleed
- DVI: 4 trains of safety injection electrically connected to 2 divisions
- Separation of safety injection and shutdown cooling
- EFWS (emergency feedwater system) is 4-train system
  - 2 100% motor-driven and 2 100% turbine-driven pumps
  - 2 independent emergency feedwater storage tanks
- Fluidic device of vortex valve at the discharge of SITs (safety
  injection tank)
- PSCS (passive secondary condensing system) through SGs
Mitigation Measures for KNGR (DOUBLE CONTAINMENT)

- Hydrogen control (keep less than 10%)
  - glow plug type igniters 50% & catalytic igniters 50%
- Debris coolability
  - sufficient reactor cavity floor area
  - passive reactor cavity flooding system (fuse plug)
- High pressure melt ejection (HPME)
  - SDS against rapid RCS depressurization to prevent DCH
- Equipment survivability during severe accident conditions
  - SBO, earthquake, pressure, temperature, radiation, etc.
- Containment Structural Integrity (for 24 hours following accidents)
  - ASME Section III Service Level C Limit (steel containment)
  - Factored Load Category (concrete containment)
- Accident Management Program (AMP)

safety improvement principles

- newly-introduced systems will not noticeably increase existing societal risk
- enhancement of safety discretely implemented based upon cost-benefit
The main goals of nuclear safety research are

- protect peoples, society and environment from radiological danger
- decrease total risk from NPP lower than total risk from other sources of energy.
- improve public opinion about safety of nuclear energy generation
Methodology of safety estimations

- "Engineering" approach (1950-1960s)
  Provision of safety during normal operation and design-basis accidents
- Probabilic safety assessment (1970-1990s)
  Safety analysis, including severe accident
- Hazard management (1990s-...)
  Guaranteed management, demonstration of the controllability and residual risk assessment

Required Knowledge Base

Main goal of severe accident research - expand the knowledge base of processes and raise its reliability
Core Melt Retention

A lot of experiments with corium simulants
  - homogeneous liquid
  - homogeneous heat release

- Conclusion
  Retention by ex-vessel cooling is possible

Corium stratification was observed
- New questions without answers
  - Does stratification exist in molten corium or in ingot only?
  - Does stratification suppress natural convection?
  - Is heat release homogeneous? (FP distribution)
  - Physical properties distribution in stratified corium?
  - Rheology of partially melted corium
  - ...
EX-VESSLE COOLING
Presentation to SARJ-97

By
Ali Behbahani
Accident Evaluation Branch
Division of Systems Research

October 8, 1997

ISSUES FOR FUTURE REACTOR

• In-vessel melt retention
• Ex-vessel phenomena

Will address only the ex-vessel cooling aspect of in-vessel melt retention
LOWER HEAD INTEGRITY - IMPORTANT PHENOMENA

• Internal heat flux distribution on RPV lower head
  - Mass and composition of melt relocation to lower plenum
  - Molten pool natural circulation
  - Stratification of melt/focusing effect of metallic layer
  - Melt/RPV interactions
  - Cooling in gaps between crust and RPV wall

• External heat removal/ex-vessel flooding
  - Ex-vessel boiling/critical heat flux
  - Effect of insulation

• RPV creep failure under high temperature and pressure conditions

EX-VEssel COOLING

• Ex-Vessel flooding is an important concept which has been considered as a desirable mean of heat removal generated by the decay heat in the relocated molten corium in the lower head.

  - This concept can be an important measure for accident mitigation for future reactors such as AP600
CRITICAL HEAT FLUX ON A DOWNWARD FACING SURFACE  
(Penn State)

Objective: To demonstrate heat generated in the debris contained in the reactor vessel lower head can be effectively dissipated by boiling on the outer surface of the vessel.

- Perform heat transfer measurement of downward-facing surfaces (i.e., hemispherical and toroidal geometries with various diameters)
- Obtain database for CHF on downward-facing curved surfaces

CRITICAL HEAT FLUX ON A DOWNWARD FACING SURFACE  
(Continued)

- Develop a comprehensive model for downward-facing boiling on curved surface and validate the model against experimental data
- Establish a proper scaling law and develop a design correlation that can be used to predict the rate of downward-facing boiling and the value of CHF on external bottom surfaces of commercial-size reactor vessels.
- Perform experiments on the effect of thermal insulation on the distribution of the heat flux distribution on outer surface of a heated hemispherical vessel.
CRITICAL HEAT FLUX ON A DOWNWARD FACING SURFACE
(Continued)

Accomplishments:

- Designed and built the subscale boundary layer boiling test facility
- Observed characteristic features of two-phase boundary layers in transient quenching and steady-state boiling experiments.
- Conducted heat transfer measurements under transient and steady state conditions for downward facing hemispherical surface at saturated and subcooled conditions.
CRITICAL HEAT FLUX ON A DOWNWARD FACING SURFACE
(Continued)

• Developed scaling law for the CHF on the external surface of a heated hemispherical vessel to estimate theoretically the spatial variation of the CHF and vessel size effect in the pool boiling.

• Designed and built the insulation structure around external surface of the hemispherical vessel.

• NUREG/CR-6507, CHF Phenomena on a Downward Facing Curved Surface (June 1997)

Plans For FY98

• Observe characteristic features of boiling in transient quenching and steady-state boiling experiments with insulation.

• Conduct heat transfer measurements under transient and steady state conditions for downward facing hemispherical surface with insulation at saturated and subcooled conditions.

• Develop scaling law for the CHF on the external surface of a heated hemispherical vessel with insulation to estimate theoretically the spatial variation of the CHF and vessel size effect.

• NUREG-CR Report on CHF phenomena and modeling on downward-facing hemispherical surfaces with insulation (April 1998)
Nucleate Boiling Data at the Bottom Center Location - With Insulation
Nucleate Boiling Data at an Off-Center Location - With Insulation
M. Firnhaber, GRS

- Beside the experimental work, already described, you need best estimate computer codes to transfer the result to a real reactor case.
- at GRS, Germany, extensive work in ongoing on code development, assessment and application
  - e.g. ATHLET
  - ATHLET/CD
  - COCOSYS
  - RALOC
together with France
  - future: ASTEC
- not only research interest, but also required from licensing authority
- example: next viewgraph
- this work can only be performed and assessed with best estimate codes, like the above mentioned or others.
GPR / RSK proposal on hydrogen management:

Global hydrogen detonation must be "practically eliminated". Besides, provisions must be taken with respect to local detonations and to possibilities of DDT events, which might jeopardize the containment and its internal structures.

The containment must be designed to withstand a global deflagration of the maximum amount of hydrogen, which could be contained in this building during core melt accidents and a representative fast local deflagration.

Specific requirements:

- limitation of the local concentration of combustible gases by design of the internal structures, use of catalytic devices and igniters (an alternative: inertization of the containment)

- hydrogen production corresponding to 100 % fuel clad metal-water reaction has to be taken into account (time dependent release rates !)

- local high hydrogen concentrations must be prevented. If it is not possible to demonstrate, that local concentrations remain below 10 Vol.-% specific provisions must be implemented (inertisation or reinforced walls)
I. Szabo (CEA/DRN)

1-Background and Framework

- Core melt accidents to be considered at the design stage
- Ultimate objective: Corium in containment
- Most R&D program devoted to: Existing NPP or Next-Generation ALWRs
- CT/DRN program:
  - Cooperative actions with EDF, Framatome
  - Concerted actions or Share Cost Action EU framework
- I.S.A.R.P.
  - Investigation of alternative solutions for future reactors
  - Assumed possibility to modify reactor components

SAR'97, Yokohama, Oct. 97

2-Methodology

- Analyses of severe accident scenario/sequences:
  - In reactor main components
  - Possibly equipped with mitigative features
  - Identifying and ranking key phenomena
    - Elaboration of PIRTs
- Analyses of alternative solutions
  - Identification and ranking of R&D needs
    - RDNRTs
  - Selection of the most promising solution
- Feasibility study

SAR'97, Yokohama, Oct. 97
Example of probabilistic analysis of a sequence

4 - CONCLUSIONS (Continued)

Examples of conceptual studies performed in the frame of ISART concern three main known:

In-vessel retention issue
In-vessel retention issue
Hydrogen issue

The alternative solutions presented here are related to helium reactors. The components of which are assumed to possibly be modified to some extent, or wholly. These solutions are to be discussed on a reactor basis for the defined reactors design.
GENERAL FEATURES OF KNGR

<table>
<thead>
<tr>
<th>GENERAL FEATURES</th>
<th>KSNPP</th>
<th>KNGR</th>
</tr>
</thead>
<tbody>
<tr>
<td>Capacity</td>
<td>2,815 MWt</td>
<td>4,000 MWt</td>
</tr>
<tr>
<td>Unit Generation Cost</td>
<td>3% less than coal-fired</td>
<td>20% less than coal</td>
</tr>
<tr>
<td>Plant Life-time</td>
<td>40 years</td>
<td>60 years</td>
</tr>
<tr>
<td>Seismic Design</td>
<td>SSE 0.2g</td>
<td>SSE 0.3g</td>
</tr>
<tr>
<td>Design Criteria</td>
<td>design basis accidents</td>
<td>DBA + severe accidents</td>
</tr>
</tbody>
</table>

- KNGR for 10 years as the standard PWR for electricity generation
- Burnup of 50,000-55,000 MWD/MTU with Gd or Eb as burnable poison
- Soluble boron minimized in reactivity control for load-follow maneuver
A. Omoto (TEPCO)

**Design to cope with "external" events**

**Research**

Expect researchers to reduce uncertainty / subjectivity in seismic/ tsunami hazard curve

What law governs low probability high consequence events

Linear theory in Log(magnitude)-Log(probability) ?

**Possible New Features**

Reduction in weight and resultant movement by steel R/B, Steel-concrete structure

Seismic isolation

Premature shutdown

---

**Acceptable level (certain fraction of societal risk)**

Internal Events + External Events

As Technology progress and Experiences accumulated ->
Areas for research \textit{<In-vessel Retension>}

- Can be a substitute for a robust containment?

- PWR: \textit{External flooding}
  IVR test also by Japanese PWR Utility/vendor

- BWR: \textit{Not practical to consider external flooding}
  A "spray" of lower vessel from outside may be practical
  Massive heat sink in the bottom
  Low core power density
  Many stainless steel penetrations in lower head
  CRD-Hydraulic System injection in the bottom
  \textit{Pour water into RPV rather than to cavity}
H. Nagasaka (NUPEC)

Pursuit of In-Vessel Retention (IVR)

- Pursuit of IVR
  - Difficulty in taking credit of IVR for any SA scenario without AM for IVR
  - Pursuit of IVR adopting a special AM for IVR

- Favorable AM for IVR
  - Applicable to both PWR and BWR
  - Immediate cooling after coolant injection
  - Applicable to ex-vessel debris cooling as well

- Newly proposed AM for IVR (cf. Fig.1)
  - Lower head outer surface cooling combined with in-vessel cooling
  - Water injection into the clearance between outer surface of lower head and the top plate of radiation shield
  - Fin cooling effect of no-evaporating falling water via penetrations
  - Accumulation of saturated water in the lower part of containment vessel

Features of Newly Proposed AM for IVR

- Immediate cooling after coolant injection
  - Much time required for cooling initiation after coolant injection for pressure vessel flooding AM
  - Pursuit of IVR as early as possible

- Effective usage of the accumulated no-evaporating water in the lower part of containment for ex-vessel debris cooling in case of pressure vessel failure
  - Low probability of steam explosion occurrence due to saturation temperature of the accumulated water
  - Effective prevention of debris spreading and molten core-concrete interaction due to the existence of water above the concrete floor in case of pressure vessel failure

- Less load on operator concerning the water injection timing for PhaseII AM of ex-vessel debris cooling
図1  Lower Head外面冷却の概念図
A. Omoto (TEPCO)

Safety Margin Basis - Hydrogen -

DDT evaluation
Less than 13% under MW reaction of 75%AFC (dry, local)
Compartment effect : Nupec tests

CGCS design
5-95% confidence level in hydrogen production in analysis
  PWR 100%AFC
  BWR w/inert containment 10% AFC

Radiolysis: Halogen (50%) particle (1%) in water
  G-value by experiments, boiling/non-boiling

Containment pressurization
Service level C stress limit or other appropriate
Best estimate in load combination

Areas for research <Hydrogen>

1) Radiolysis
   <Importance of radiolysis for a small containment>
   - Combination of H2(M-W reaction) with O2 (radiolysis)
   - Current criteria for CGCS (such as R.G. 1.7): need change
   <Research subjects>
   - Radiolysis in submerged debris bed
     (not included in the current source term)
   - G-value for metallic components (Effect of Halogen: known)
   - Combination methodology

2) Hydrogen gettering technology
   - Important pressurizing source for a small containment TEPCo
Closing Remarks

H. Ogasawara
Director and General Manager, Systems Safety Department
Nuclear Power Engineering Corporation

In closing this Workshop, I am very much obliged to all of the participants here, for your contributions of valuable information for your eager discussions, including today's final panel discussion. On behalf of JAERI and NUPEC, the Government of Japan, I especially appreciate the contribution of foreign participants.

Severe accident research and development activities have been continuing for more than ten years throughout the world. As seen in the SARJ-97 Meeting, there are several technical concerns which are now approaching resolution from the many years of developmental efforts. However, uncertainties remain relating to ex-vessel and in-vessel debris cooling mechanisms, the source term problem, and fission product behavior. In this workshop, the current status of leading international programs, including PHEBUS, Rasplav, RUT, EC/CEA project and containment testing at SNL were presented. Significant progress in computer code development and verification in many countries were also reported. Finally, future trends in severe accident research were indicated by the latest information on ambitious programs of the countries represented here. Moreover we have a new proposal from DRN/CEA, the DIVER system, Dual strategy, which will become a future item of discussions.

I think it will be important after ten years of research activities to find a strategy to apply the results of severe accident studies towards the safety of nuclear plants. As is well known, there is an international movement to assure nuclear safety even under severe accident conditions. Appropriately, today's panel discussion addressed the relation between severe accident research and the future reactor designs. In my opinion, this kind of international discussion may be the first step towards application of severe accident results. Some accident management ideas were proposed relating to future plant designs, and several technical concerns were discussed. However some differences were apparent in the panelists opinions, due to differences in national situations. "How safe is safe enough" is still an issue to be solved. It is very important to balance safety and economy.

I sincerely appreciate all of the people here who presented papers, cooperated in discussions and supported the meeting with preparation work and management. Finally I must thank Mr. Yamano for his contributions to severe accident studies in Japan. As already announced at the opening session, he is not with us anymore. I express my deep and sincere sympathy to his family.

Thank you very much for your attention, and see you again next year in Japan.
Appendix A  Final Program for SARJ-97

SARJ-97

The Workshop on
Severe Accident Research held in Japan

Pacifico Yokohama
Yokohama, Japan
Phone: 045-223-6089, Fax: 045-223-6090

October 6 - 8, 1997

Organized by

Japan Atomic Energy Research Institute (JAERI)
and
Nuclear Power Engineering Corporation (NUPEC/MITI)

Chairperson  M. Maeda (JAERI)
Co-chairperson  H. Ogasawara (NUPEC)

WWW Home Page at http://sarl.tokai.jaeri.go.jp/SARJ/

Notice: Some of materials to be presented at this Workshop are preliminary in nature and contain proprietary information which is made available only for the participants. All participants are expected to observe the restriction on use of such information other than for their own use.
October 6, 1997
Room: 431, 432

Registration Starts at 8:30

M. Maeda (JAERI)

Plenary Session: Overview of Research Activities
Chairperson: H. Nariai (Tsukuba Univ.), Co-chairperson: J. Sugimoto (JAERI)

9:10 Overview of Severe Accident Research at JAERI  J. Sugimoto (JAERI)
9:40 Present Status of Containment Integrity Tests at NUPEC  H. Nagasaka (NUPEC)
10:10 Overview of Severe Accident Research at USNRC  A. Behbahani (USNRC)
10:40 Coffee Break
11:00 Reactor Safety Research at IPSN  J. Bardelay (CEA/IPSN)
11:30 Overview of Severe Accident Research at KAERI  S. B. Kim (KAERI)
12:00 Lunch Break

Session I: In-Vessel Retention 1
Chairperson: K.Y. Suh (Seoul National Univ.), Co-chairperson: Y. Abe (Yamagata Univ.)

13:30 In-Vessel Corium Retention: Proposal of a "Dual" Strategy  I. Szabo (CEA/DRN)
14:00 Thermomechanical Analysis for an Advanced In-Vessel Retention Design  K. Y. Suh (Seoul National Univ.)
14:30 Modelling Lower Plenum Core Debris  S. K. Wong (City Univ. of Hong Kong)
15:00 Study on the One-Dimensional Flow Characteristics of the Counter-Current Flow in Debris Beds  H. Isurugi (Yamagata Univ.)
15:30 Coffee Break
16:00 OECD RASPLAV Project: Phase 1 Results  A. Merzhakov (RRC KI)

Session I: In-Vessel Retention 2
Chairperson: A. V. Jones (JRC Ispra), Co-chairperson: A. Serizawa (Kyoto Univ.)

16:30 Experiment and Analysis on In-Vessel Debris Coolability in ALPHA Program  Y. Maruyama (JAERI)
17:00 SONATA-IV Experiments on In-vessel Debris Coolability and Retention  J. H. Jeong (KAERI)
17:30 Molten Material Heat Transport Tests with Coolant Boiling  K. Y. Suh (Seoul National Univ.)
18:00 Lattice Gas Automata Simulations of Flow through Porous Media  Y. Matsukuma (Yamagata Univ.)
18:30 Adjourn
October 6, 1997

Session II: Computer Code Development
Chairperson: C. Allison (Innovative Systems Software), Co-chairperson: K. Muramatsu (JAERI)

13:30 Simulation of the Arrival and Evolution of Debris in a PWR Lower Head with the SFD ICARE2 Code
F. Fichot (CEA/IPSN)

14:00 COCOSYS (Containment Code System) - A Detailed Approach to Analyze Containment Behavior During Severe Accidents
H. J. Allelein (GRS)

14:30 Development of Super Simulator "IMPACT" Part (1) IMPACT System Configuration
N. Sato (NUPEC)

15:00 Development of Super Simulator "IMPACT" Part (2) Physical Phenomena Modeling of Severe Accident and Some Modules Verification Tests
K. Miyagi (NUPEC)

15:30 Coffee Break

Session III: Hydrogen Behavior
Chairperson: V. Sidorov (RRCKI), Co-chairperson: T. Hashimoto (NUPEC)

16:00 Large-Scale Experiment and Scaling of DDT Conditions in Hydrogen-Air-Steam Mixtures - An Overview
V. Sidorov (RRCKI)

16:30 Numerical Investigation of Missiles Acceleration by Hydrogen Explosion
A. Efimenko (RRCKI)

17:00 Analyses of NUPEC's Large Scale Hydrogen Mixing in a Reactor Containment Vessel
J. Fermandjian (NUPEC/CEA)

17:30 Summary of Hydrogen Combustion Tests Results at NUPEC
T. Hashimoto (NUPEC)

18:00 Adjourn
October 7, 1997

Session IV : Structural Integrity
Chairperson: Vincent Luk (SNL), Co-chairperson: Y. Maruyama (JAERI)

8:30 Pressurization Test on a Full Scale Equipment Hatch Model S. Arai (NUPEC)
9:00 Pre-Test Analysis on the SCV Model Test K. Komine (NUPEC)
9:30 Pressurization Test of a 1/10 Steel Containment Vessel Model T. Matsumoto (NUPEC)
10:00 Coffee Break
10:30 Preliminary Analysis and Instrumentation Planning of a Prestressed Concrete Containment Vessel Model D. Pace (SNL)
11:00 Analytical Study on Change of Tendon Tension Force Distribution during the Pressurization Process of Pre-stressed Containment Vessel T. Kashiwase (NUPEC)
11:30 Studies on Reactor Piping Integrity during Severe Accident in WIND Project A. Maeda (JAERI)
12:00 Lunch Break
13:30 Metallurgical Examination of Piping Failed at High Pressure and High Temperature in WIND Project Y. Harada (JAERI)

Session VII : FP Source Term
Chairperson: M. Firnhaber (GRS), Co-chairperson: A. Watanabe (NUPEC)

14:00 Status of VEGA Fission Product Release Experiment A. Hidaka / T. Nakamura (JAERI)
14:30 Results of ISP37 : VANAM M3 Experiment on Containment Thermal - Hydraulics and Aerosol Behavior M. Firnhaber (GRS)
15:00 Fission Products Aerosol Removal Test by Containment Spray under Accident Management Conditions H. Nagasaka (NUPEC)
15:30 Coffee Break
16:00 Deposition of CsI Aerosol in Horizontal Straight Pipe under Inert and Superheated Steam Environment H. Shibazaki (JAERI)
16:30 Experimental and Analytical Study on Aerosol Behavior in WIND Project A. Hidaka (JAERI)
17:00 Steam Condensation on Spray Water Drops : Experimental Results and Models D. Ducret (IPSN/DPEA/SERAC)
17:30 Failure Criteria and Fission Products Trapping Effect at Containment Penetrations under Severe Accident Conditions A. Watanabe (NUPEC)
18:00 Adjourn

18:30 - 20:30 Reception at 3F Lounge
October 7, 1997

Session V: PHEBUS/FP Program
Chairperson: J. Fermandjian (NUPEC/IPSN), Co-chairperson: A. Hidaka (JAERI)

8:30 Fission Product Release, Transport and Chemistry Indications from the First Two PHEBUS-FP Tests
   B. Clement (CEA/IPSN)
9:00 The PHEBUS Experiment FPT1
   J. Furlan (CEA/IPSN)
9:30 Analysis of Bundle Degradation Behaviour in the First Two PHEBUS FP Tests
   A. V. Jones (CEC/JRC/Ispra)

10:00 Coffee Break
10:30 Evaluation of Fission Products Release and Transport in the Circuit of PHEBUS FP Test by MACRES Code
   Y. Kawada (NUPEC)
11:00 Phebus FPT4: Test Description and Pretest Calculation
   F. Fichot (CEA/IPSN)

Session VI: FCI Experiment

11:30 COTELS Fuel Coolant Interaction Tests of UO₂ Debris Dropping into Water Pool
   M. Kato (NUPEC)
12:00 Lunch Break
14:00 The Effect of Coolant Jet Subcooling on the Coolant Injection Mode of Vapor Explosions
   H. S. Park (JAERI)
14:30 Deformation and Fragmentation of Molten Zn and Al near Melting Points
   K. Sugiyama (Hokkaido Univ.)

Session VIII: FCI Simulation
Chairperson: K. H. Bang (Korea Maritime Univ.), Co-chairperson: H. Okada (NUPEC)

15:00 Development of TRACER-II and Application to In-Vessel FCI’s
   K. H. Bang (Korea Maritime Univ.)
15:30 Coffee Break
16:00 Development of Computer Code for Expansion Stage in Vapor Explosion
   A. Minato (Hitachi)
16:30 Development of FCI simulation code JASMINE (1) Premixing
   K. Moriyama (JAERI)
17:00 Development of FCI simulation code JASMINE (2) Propagation
   Y. Yang (JAERI)
17:30 Adjourn

18:30 - 20:30 Reception at 3F Lounge
October 8, 1997

Session IX : Computer Simulations
Chairperson: K. Sugiyama (Hokkaido Univ.), Co-chairperson: M. Kajimoto (NUPEC)

9:30 Nuclear Reactor Thermal-Hydraulic Activities at MINT TRIG A Reactor
    M. S. Kassim (Malaysian Institute for Nuclear Technology Research)
10:00 Use of MAAP in the Assessment of Containment Integrity Following the Severe Accident
    J. I. Yun (Seoul National Univ.)
10:30 Coffee Break
10:45 A Scoping Analysis for Containment Venting at BWR using THALES-2
    J. Ishikawa (JAERI)
11:15 Comparison between MAAP, MELCOR and SCDAP
    C. M. Allison (Innovative Systems Software)
11:45 Hypothetical Accident in Accelerator-Driven Subcritical and Lead-Cooled Fast Systems
    H. U. Wider (CEC/JRC/Ispra)
12:15 Lunch Break

Room : 433, 434

Session X : Melt Behavior and Accident Management
Chairperson: S.B. Kim (KAERI), Co-chairperson: M. Kato (NUPEC)

9:30 Experimental Comparison of VVER and PWR Fuel Under Severe Accident Conditions
    Z. Hozer (KFKI Atomic Energy Research Institute)
10:00 Evaluation of the RBMK-1500 Accident Confinement System
    E. Uspuras (Lithuanian Energy Institute)
10:30 Coffee Break
10:45 Experimental Investigations on Melt Spreading and Scaling Analysis
    V. A. Bui (Royal Institute of Technology)
11:15 Molten Core - Zircon Ceramics Interaction Experiments
    V. N. Mineev (IVTAN)
11:45 Investigation of Alternative Solutions for Severe Accident Management in Future Reactors
    I. Szabo (CEA/DRN)
12:15 Lunch Break
October 8, 1997

13:30  **Panel Discussion**

"Severe Accident Research for Future Reactors"

Chairperson:  K. Abe of JAERI, Japan
Panelist:  B. Clement of IPSN, France
          H. H. Hennies of FZK, Germany
          A. Omoto of Tokyo Electric Co., Japan
          C. S. Kang of Seoul National University, Korea
          A. Merzliakov of RRCKI, Russia
          A. Behbahani of USNRC, USA

16:20  Closing Remarks  H. Ogasawara (NUPEC)
16:30  Adjourn
### October 6, 1997

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<th>Session II: Computer Code Development</th>
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<th>Session V: PHEBUS/FP Program</th>
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<td>Session VI: FCI Experiment</td>
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## Appendix B  List of Participants

**SARJ-97**

*The Workshop on Severe Accident Research held in Japan*

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<tr>
<th>Name</th>
<th>Organization</th>
<th>Country</th>
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<tr>
<td>Kiyoharu Abe</td>
<td>Japan Atomic Energy Research Institute (JAERI)</td>
<td>JAPAN</td>
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<tr>
<td>Yutaka Abe</td>
<td>Yamagata University</td>
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<td>Toshiba Corporation (Toshiba)</td>
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<td>Mamoru Akiyama</td>
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<tr>
<td>Hans Josef Allelein</td>
<td>Gesellschaft fur Reaktorsicherheit (GRS) mbH</td>
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<td>Chris M. Allison</td>
<td>Innovative Systems Software</td>
<td>U.S.A.</td>
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<td>Youichi Amano</td>
<td>Nuclear Power Engineering Corporation (NUPEC)</td>
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<td>Yasumasa Andoh</td>
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<td>Shoji Arai</td>
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<td>Izuo Aya</td>
<td>SRI</td>
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<td>Kwang-Hyun Bang</td>
<td>KOREA Maritime University</td>
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<td>Joel Bardelay</td>
<td>CEA/CEN</td>
<td>FRANCE</td>
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<td>Alireza Bchbahani</td>
<td>U.S. Nuclear Regulatory Commission (USNRC)</td>
<td>U.S.A.</td>
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<td>Viet Anh Bui</td>
<td>Royal Institute of Technology</td>
<td>SWEDEN</td>
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<td>Keiko Chitose</td>
<td>Mitsubishi Heavy Industries,LTD. (MHI)</td>
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<td>ANATECH Corp.</td>
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<td>Fernando J. Doria</td>
<td>Atomic Energy of Canada Limited</td>
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<td>Didier Ducret</td>
<td>IPSN/DPEA/SERAC</td>
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<td>Yuji Furukawa</td>
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<tr>
<td>Katsubori Goda</td>
<td>Kansai Electric Power Co., Inc. (KEPCO)</td>
<td>JAPAN</td>
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<tr>
<td>Valentina Golovko</td>
<td>Institute of Atomic Energy of National Nuclear Center (IAENNC)</td>
<td>KAZAKSTAN</td>
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<tr>
<td>Masashi Goto</td>
<td>Toshiba</td>
<td>JAPAN</td>
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<tr>
<td>Mr. Gregory Gromov</td>
<td>Scientific and Technical Center</td>
<td>UKRAINE</td>
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<tr>
<td>Satoshi Haba</td>
<td>Electric Power Development Company (EPDC)</td>
<td>JAPAN</td>
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<tr>
<td>Toshiyuki Hamanaka</td>
<td>Nuclear Safety Research Association (NSRA)</td>
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<td>Hans Heenning Hennies</td>
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<td>Michael Hessheimer</td>
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<td>Akihide Hidaka</td>
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<tr>
<td>Masataka Hidaka</td>
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Appendix C Questionnaire to the Participants to SARJ-97
(Please circle the number or make comments, and forward this to Secretariats)

A. Where do you think is your preferred place for the next Workshop?

Comments or proposals
• Tokyo
• A spa
• Yokohama
• Hokkaido
• Okinawa
• Kyoto
• Not Yokohama

B. How do you rate the presentations of the Workshop?

1. Excellent
2. Good
3. Average
4. Poor

Comments or proposals
• Many important and useful papers were presented.
• In order to keep the time for the discussion, limiting the time for 30 min/ presentation was very appropriate.
• Nice mixture of local and foreign papers.
• Some presentation were excellent, some were poor, The overall level was good!

C. What do you think about the topic of the Panel Discussion?

1. Timely
2. Out of date
3. Too early
4. Interesting
5. Not interesting

Comments or proposals
• Not enough difference of opinion for an exciting discussion.

D. Have you accessed WWW server for the information of SARJ?

Yes
No

If Yes, how do you rate the information on WWW server?

1. Very helpful
2. Helpful
3. Average
4. Poor

Comments or proposals
• How to get there was not very helpful.
• It took long time to download information.
• Request electronic copies from the authors.
• Put out papers there, too.
E. How do you rate the work (preparation, meeting etc.) of Secretariats of the Workshop?
   1. Excellent  ⬜⬜⬜⬜⬜
   2. Good       ⬜⬜⬜⬜
   3. Average
   4. Poor
Comments or proposals
   • 3 days is too compressed - long days and little time for discussion.

F. The place of SARJ-97 has been changed from Tokyo. How do you rate the place this years?
   1. Far better
   2. Better     ⬜⬜⬜
   3. Same       ⬜⬜⬜⬜⬜
   4. Worse      ⬜⬜
   5. Very worse
Comments or proposals
   • Tokyo is more convenient.
   • See A Hotel too far from conference.
   • I did not attend the previous SARJ, but Yokohama was excellent!
   • Meeting room too small. [viewgraphs difficult to read for persons far from the screen]

G. Have you communicated with experts participating in another conferences (GLOBAL'97, KJPSA)?
   1. Yes        ⬜⬜⬜⬜⬜
   2. No         ⬜⬜⬜⬜⬜⬜⬜
   3. Others
Comments or proposals
   • Linkage with JKPSA is very important for participants and dose for JKPSA organizer.
   • Security measures make contact difficult.

H. How do you rate the Reception of the Workshop?
   1. Enjoyed very much  ⬜⬜⬜⬜
   2. Enjoyed           ⬜⬜⬜⬜
   3. Average
   4. Poor
Comments or proposals
   • Very sorry that I could not attend it.

I. Do you have any other comments or proposals? (Please use the opposite side of the paper if the space is not enough.)
   • The time was too short for the material to be presented - 2 1/2 days should become 3 1/2 or evokes papers should be wasted out. Speakers should focus on what is new since the last SARJ. The reception was appreciated as was the efficiency of the secretariats.
   • See above about the electronial copies.
Appendix D  Reference Paper

The following paper was not presented at the SARJ-97 Workshop due to unavoidable circumstances. However since the paper was originally planned to be presented, this is attached as references.
SIMULATION OF THE LOSS OF COOLANT ACCIDENTS AT A NUCLEAR POWER PLANT WITH WWER-1000 USING THE MELCOR CODE

J.M. Petrovskii, B.I. Nigmatullin, K.K. Khasanov
Electrotechnical Research and Engineering Center, LWR Safety, 142530 Elektrogorsk, Moscow Region, Russia
E-mail: psmi3@npp.mpei.ac.ru or: sergey@cc-cnsmi.com, Fax: (095) 361-16-26

INTRODUCTION
The severe accidents analysis presented includes major accident sequences initiated by breaks (with equivalent diameters of 80, 50 and 25 mm) in the primary coolant system accompanied by a station blackout and failure of the diesel-generators used to supply electrical power to ECCS pumps and the SG emergency feedwater pumps. A severe accident analysis of the reactor core with WWER-1000 has been performed using the MELCOR code. This study includes a wide spectrum of severe accident transients, from normal operating conditions to core reactor cavities melt-through. These data should be used for the further study of probabilistic safety analysis issues. This work is performed in the framework of the Program of Severe Accident investigations undertaken at the Electrotechnical Research and Engineering Center, LWR Safety, Russia.

ABSTRACT
The severe accidents analysis presented includes major accident sequences initiated by breaks (with equivalent diameters of 80, 50 and 25 mm) in the primary coolant system accompanied by a station blackout and failure of the diesel-generators used to supply electrical power to ECCS pumps and the SG emergency feedwater pumps. A severe accident analysis of the reactor core with WWER-1000 has been performed using the MELCOR code. This study includes a wide spectrum of severe accident transients, from normal operating conditions to core reactor cavities melt-through. These data should be used for the further study of probabilistic safety analysis issues. This work is performed in the framework of the Program of Severe Accident investigations undertaken at the Electrotechnical Research and Engineering Center, LWR Safety, Russia.

MAIN NPP INPUT DATA
The main NPP input data are presented in Table 1.

RESULTS OF ANALYSIS
The main results of the SBLOCA calculations for 80, 50 and 25 mm breaks are similar to the 80 mm case but are somewhat slower. The chronological event sequences for SBLOCA at a NPP with WWER-1000 (V-1200) are presented in Table 2.

CONCLUSIONS
The accident sequences for the breaks with diameters of 50 and 25 mm can be compared to the 80 mm case, but are somewhat slower. The chronological event sequences for SBLOCA at a NPP with WWER-1000 (V-1200) are presented in Table 2.

The cyclical character of water injection from the HA to the RCS may be explained by the fact that after water penetrates the core, the vigorous steam generation begins

The break is not large enough to allow release of all of the water into the RCS, and the pressure increases. The check valve in the "HA-RPV" line closes and the injection of water from the ECCS stops.

Because of the mass losses through the leak and gradual vaporization of the primary coolant by the decay heat, the reactor scrams (with a 5 s delay), the shutdown of the main reactor coolant pumps begins, the main turbine stop valve closes and the turbine also stops its rotation, and the feedwater supply to the SGs stops.

Because of the primary system pressure decrease below the ECCS HA emergency setpoint, water is injected into the RCS via the hydroaccumulators entering the RPV lower and upper heads.

Heat transfer between the primary and secondary systems (via the SG tube walls) causes the pressure increase in the secondary system and initiation of the blowdown of the steam-water mixture via the SG safety relief valve.

Loss of the primary coolant causes a decrease of the steam-water mixture level in the RPV, leading to core heatup initiation. The fuel cladding-water vapor reaction provides additional core heating that leads to core melting.

After the heat-up initiation, core fragments relocate on the lower RPV structure. This results in the blowdown of the steam-water mixture from the SGs and release of water from the reactor core nodalization scheme is presented in fig. 1. The metallic and concrete portions of the RCS and containment are represented by 26 "block models.

The containment is represented by two control volumes: the reactor cavity and the free space in the reactor building above the reactor cavity. The metallic walls of the containment are modeled by corresponding heat structures. It is assumed that if the reactor cavity pressure is about 5.8 MPa, hydrogen accumulates in the reactor cavity. The RCS is modeled with four-reactor independent dimensions, and the containment gas release system fails. This causes the containment melt relocation to the reactor bottom and creates the containment melt relocation (at the maximum pressure is 0.71 MPa). The accident sequences for the breaks with diameters of 50 and 25 mm are similar to the 80 mm case but are somewhat slower. The chronological event sequences for SBLOCA at a NPP with WWER-1000 (V-1200) are presented in Table 2.

CONCLUSIONS
The accident sequences presented include major accident sequences initiated by breaks (with equivalent diameters of 80, 50 and 25 mm) in the primary coolant system accompanied by a station blackout and failure of the diesel-generators used to supply electrical power to the ECCS pumps and the SG emergency feedwater pumps.

After the break initiation and the loss of off-site power, the condenser of the MTPs begins, SG feedwater flow stops, and the MTTPs clouse. Reactor scram is initiated at 2.5 s by the electrical power. The cooldown level in the reactor core decreases in the accident period but increases after the initiation of water injection from the hydroaccumulators to the RPV at 1100 s.

It is assumed that the system for a vapor-gas release from the containment fails, this assumption allows an estimate of the maximum pressure and gas composition in the containment.

MELCOR NODALIZATION SCHEME
This nodalization scheme used in this study has no serious limitations in this case. The input data files for the NPP unit with WWER-1000 were prepared according to the ABSTRACT. The four loops of the RCS are modeled as two loops, one loop representing itself (containing the pressurizer) and the other representing the remaining three loops. Control volumes representing the NPP equipment have the same volumes and elevations at the NPP equipment but do not exactly reproduce the RCS shape. For example, a detailed representation of the loop seals is not included because the loop seal breakdown is not expected to be important in this accident sequence. The loop seal failure on the pressure vessel will be considered in the future analysis.

The reactor core (hotated part) is represented by 9 concentric radial rings and 10 axial nodes. The lower head of the reactor is represented by 2 control volumes. The reactor core nodalization scheme is presented in fig. 2. The metallic and concrete portions of the RCS and containment are represented by 26 "block models.

The containment is represented by two control volumes: the reactor cavity and the free space in the reactor building above the reactor cavity. The metallic walls of the containment are modeled by corresponding heat structures. It is assumed that if the reactor cavity pressure is about 5.8 MPa, hydrogen accumulates in the reactor cavity. The RCS is modeled with four-reactor independent dimensions, and the containment gas release system fails. This causes the containment melt relocation to the reactor bottom and creates the containment melt relocation (at the maximum pressure is 0.71 MPa). The accident sequences for the breaks with diameters of 50 and 25 mm are similar to the 80 mm case but are somewhat slower. The chronological event sequences for SBLOCA at a NPP with WWER-1000 (V-1200) are presented in Table 2.

CONCLUSIONS
The accident sequences presented include major accident sequences initiated by breaks (with equivalent diameters of 80, 50 and 25 mm) in the primary coolant system accompanied by a station blackout and failure of the diesel-generators used to supply electrical power to the ECCS pumps and the SG emergency feedwater pumps. The calculations show the characteristic times for such transients are presented in figs 3-17.

The main results of the SBLOCA calculations are presented in Table 1. It is assumed that reactor scram and closing of the MTSPs begins after the loss of off-site power. It is also assumed that all the SG control valves on the primary system accompany the station blackout and failure of the diesel-generators used to supply electrical power to the ECCS pumps and the SG emergency feedwater pumps. It is also assumed that all the SG control valves on the primary system accompany the station blackout and failure of the diesel-generators used to supply electrical power to the ECCS pumps and the SG emergency feedwater pumps. It is also assumed that all the SG control valves on the primary system accompany the station blackout and failure of the diesel-generators used to supply electrical power to the ECCS pumps and the SG emergency feedwater pumps.
The chronologial event sequences of SBLOCA at a NPP with WWER-1000 (V-320)

<table>
<thead>
<tr>
<th>Event/Time, s</th>
<th>Time, s</th>
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<tr>
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<tr>
<td>Loss of off-site power</td>
<td>0</td>
</tr>
<tr>
<td>MCP coastdown initiation</td>
<td>2.3</td>
</tr>
<tr>
<td>ECCS HA water injection initiation</td>
<td>3710</td>
</tr>
<tr>
<td>ECCS HA water terminated</td>
<td>16700</td>
</tr>
<tr>
<td>Core melting initiation</td>
<td>10500</td>
</tr>
<tr>
<td>Molten core-reactor bottom interaction initiation</td>
<td>16622</td>
</tr>
<tr>
<td>Water full evaporation in the reactor cavity</td>
<td>45984</td>
</tr>
<tr>
<td>Water full evaporation in the secondary side</td>
<td>45984</td>
</tr>
<tr>
<td>Reactor cavity bottom melt-through</td>
<td>45984</td>
</tr>
</tbody>
</table>

Main characteristics of the NPP with WWER-1000 (V-320)

<table>
<thead>
<tr>
<th>Parameter</th>
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<tbody>
<tr>
<td>Thermal power of reactor, MW</td>
<td>3000</td>
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<tr>
<td>Coolant flowrate through reactor, m³/s</td>
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<tr>
<td>Coolant bulk temperature at the reactor inlet, °C</td>
<td>293.8</td>
</tr>
<tr>
<td>Coolant pressure at the core outlet, MPa</td>
<td>15.7</td>
</tr>
<tr>
<td>Crack diameter, mm</td>
<td>7.8</td>
</tr>
<tr>
<td>Break initiatiion</td>
<td>0</td>
</tr>
<tr>
<td>Loss of off-site power</td>
<td>0</td>
</tr>
<tr>
<td>MCP coastdown initiation</td>
<td>2.3</td>
</tr>
<tr>
<td>ECCS HA water injection initiation</td>
<td>3710</td>
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<tr>
<td>ECCS HA water terminated</td>
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<tr>
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<tr>
<td>Molten core-reactor bottom interaction initiation</td>
<td>16622</td>
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<tr>
<td>Water full evaporation in the reactor cavity</td>
<td>45984</td>
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<tr>
<td>Water full evaporation in the secondary side</td>
<td>45984</td>
</tr>
<tr>
<td>Reactor cavity bottom melt-through</td>
<td>45984</td>
</tr>
</tbody>
</table>

REFERENCES

3. Input Data for Thermalhydraulic Calculations of the NPP with WWER-1000 (V-320). Informational Package issued by VNIIEAES, Moscow, Russia.
### 表1 SI基本単位および補助単位

<table>
<thead>
<tr>
<th>單位</th>
<th>名称</th>
<th>記号</th>
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<td>長さ</td>
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<tr>
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<td>キログラム</td>
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</tr>
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<td>時間</td>
<td>秒</td>
<td>s</td>
</tr>
<tr>
<td>電流</td>
<td>アンペア</td>
<td>A</td>
</tr>
<tr>
<td>熱力学温度</td>
<td>ケルビン</td>
<td>K</td>
</tr>
<tr>
<td>物質量</td>
<td>キログラムモル</td>
<td>mol</td>
</tr>
<tr>
<td>磁場強さ</td>
<td>カルガリ</td>
<td>cd</td>
</tr>
<tr>
<td>平面角</td>
<td>ラジアン</td>
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<tr>
<td>立体角</td>
<td>ステララジアン</td>
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### 表2 SIと用いられる単位

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<td>min、h、d</td>
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<td>度、分、秒</td>
<td>°、′、″</td>
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<td>L</td>
</tr>
<tr>
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<td>トン</td>
<td>t</td>
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<td>電子ボルト</td>
<td>eV</td>
</tr>
<tr>
<td>原子質量単位</td>
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### 表3 国内外の名称をもつSI単位

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<tr>
<td>重力</td>
<td>ワット</td>
<td>W</td>
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<td>体積</td>
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<td>低温</td>
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<td>Ω</td>
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<td>円周率</td>
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<td>ボケラル</td>
<td>Bq</td>
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<td>Gy</td>
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<td>線量</td>
<td>セルボルト</td>
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### 表4 SIと日本単位との対応

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<td>ガル</td>
<td>Gal</td>
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<td>キュリ</td>
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<tr>
<td>レントゲン</td>
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<td>フェルト</td>
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### 表5 SI機関

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<td>ヒュンガル</td>
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<td>レーム</td>
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### 変換表

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<th>lbf</th>
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<th>100 Pa·s (N·s/m²)</th>
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### 静電気

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<th>kW·h</th>
<th>cal (計算)</th>
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### 放射能

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