PNC TN1420 91-002

Advanced Reactor Safety Research at PNC

1990

FBR & ATR

Power Reactor and Nuclear Fuel Development Corporation

Abstract

An annual plan of FBR safety research had been made since 1986 by Japanese Nuclear Safety Commission.

This plan focuses on the safety philosophy which fully utilizes the intrinsic characteristics of FBRs and which implements the safety technologies of FBRs. Items examined for establishing the safety philosophy includes :

- defense-in-depth
- safety grade classification
- multi-barriers against radioactivity, and
- severe accidents.

In selecting research items, the following four fields of research were identified in relation to the safety philosophy :

1) research relating to safety design and safety evaluation principles,
2) research on accident prevention and mitigation,
3) research on (design basis) accidents evaluation, and
4) research on severe accidents.

ATR safety research plan has been also revisedz. Its three research fields are :

1) research on normal operations and anticipated operational occurrences,
2) research on accident conditions, and
3) research on severe accidents.

The plan focuses on the irradition behavior of Mox fuels, evaluation of operation data of ATR plant and assessment of the safety margin for severe accidents.

This report introduces safety research index, with emphasis on the research progress in 1990, of FBR and ATR that PNC is currently undertaking within the frame of annual safety research plan for nuclear installations established by Japanese Nuclear Safety Commission.

Advanced Reactor Safety Research Classification

I. FBR

II. ATR

I. FBR

PROJECT CODE 1.1	CLASSIFICATION Safety Design and Safety Evaluation Principles Establishment	
TITLE: Development of Database for Safety Design and Evaluation		
COUNTRY: Japan		INITIATED: October, 1984
SPONSOR: S T A		COMPLETED:
ORGANIZATION: P N C		LAST UPDATING: August, 1990
PROJECT LEADER: NAME Y. Kani and T. Nakamura ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: In progress

1. General Aim :

To collect and organize basic data necessary to establish safety design and evaluation policy of large reactors and utilize them to establish appropriate policy.

2. Particular Objectives :

1) Development of FBR components reliability database through collection and analysis of data on operation and maintenance from experimental fast reactor 'Joyo' and various sodium facilities.

2) Development of human reliability database, considering operationability of plant personnel, on systems and components which are important to secure safety in fast breeder reactor plants.

3. Experimental Facilities :

Component Reliability Data Sources:
Experimental Fast Reactor Joyo, 50 MW Steam Genarator Test Facility, 2 MW Sodium Flow Test Facility, Sodium Exposure Test Loop, Control Rod Drive Mechanism Test Loop, and Plant Dynamics Test Loop.

4. Description of Computer Codes :

Computer language	: FORTRAN IV + ASSEMBLER
Computer used	: FACOM M780/VP100
Computer capacity required	: approx. 0.6 MB
Description of codes	: JOSHUA (Database Management System), CREDO, CREST (Statistical Analysis)
Field of application	: Reliability, Availability and Maintainability Analysis.

5. Project Status: Progress to date

1) The data on operation and maintenance experience at 'Joyo' and other sodium facilities at O-arai Engineering Center have been collected and organized and efforts have been made to develop component reliability database for fast reactor.

2) Human factor data on calibration task for plant protection system are being collected and analyzed for 'Joyo'.

6. Project Status: Essential results

1) The operational data were introduced into CREDO database. By combining the data from PNC and the data from experimental fast reactors FFTF and EBR-II and other sodium test facilities in U.S. 22600 component, 1900 failure data and 2.4×10^9 component hour data in total have been accumulated in CREDO database up to now.

2) The probability of miscalibration error obtained from operational experience is one order magnitude smaller than the analytical value based on the THERP method.

7. Next Steps:

Continue collection and arrangement of data on operation at 'Joyo' and other facilities, components failure and human factor. Develop database management system and statistical analysis method and perform analysis of data characteristics.

8. Relation to Other Projects or Codes :

Codes : SETS, FAUST
Other Projects:
1) Application of PSA to an LMFBR Plant
2) Study on PSA methodology for LMFBR plants

9. Reference Document:

Nakai, "Development of FBR Reliabillity Database," Journal of the Atomic Energy Society of Japan, Vol.31, No.5, 1989. (in Japanese)

10. Degree of Availability:

Not for publication except for the above document.

PROJECT CODE 1.1	CLASSIFICATION Safety Design and Safety Evaluation Principles Establishment	
TITLE: Study on Development of Proper Safety Design Policy		
COUNTRY: Japan		INITIATED: April, 1988
SPONSOR: S T A		COMPLETED:
ORGANIZATION: P N C		LAST UPDATING: August, 1990
PROJECT LEADER: NAME Yoshio Kani ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: In progress

1. General Aim :

To analyze the characteristics of operational experiences on FBRs for the purpose of establishing proper safety design policy.

2. Particular Objectives :

1) To analyze the reliability data obtained from the operational experiences of experimental fast reactor JOYO and sodium test facilities.

2) To develop the reliability data for the components and systems of FBR and to examine system reliability requirement.

3. Experimental Facilities :

N.A.

4. Description of Computer Codes :

QUEST	Computer language	: MS-C
	Computer used	: IBM PC/AT, TOSHIBA J-3100
	Computer capacity required	: approx. 640 KB
	Description of code	: Simplified level-1 PSA program

5. Project Status: Progress to date

Reliability data such as failure rates and repair times have been prepared for the major FBR components and compared with those of LWRs. Reliabilities for the various design and/or operational condition are surveyed and their effects are examined.

6. Project Status: Essential results

Failure rates for major FBR components are comparable to those of LWRs which have a large population of data. Recovery effects on the performance of components are relatively large for a decay heat removal system because of its long grace time. Parametric study shows that the reliability for typical active components range from 10^{-2}/d to 10^{-4}/d and those for passive components are smaller than 10^{-4}/d.

7. Next Steps:

Rational reliability allocation will be examined based on the reliability characteristics of the typical FBR systems. Basic materials will be provided for discussing to achieve the specific reliability requirement.

8. Relation to Other Projects or Codes :

Development of Database for Safety Design and Evaluation.

9. Reference Document:

Kurisaka, "Development of FBR reliability database," 1989 Fall Meeting of the Atomic Energy Society of Japan.

10. Degree of Availability:

Not for publication except for the above document.

PROJECT CODE 1.1	CLASSIFICATION Safety Design and Safety Evaluation Principles Establishment	
TITLE: Study on Development of Proper Safety Evaluation Policy		
COUNTRY: Japan		INITIATED: April, 1989
SPONSOR: S T A		COMPLETED:
ORGANIZATION: P N C		LAST UPDATING: August, 1990
PROJECT LEADER: NAME Y. Kani and H. Ninokata ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: In progress

1. General Aim :

To analyze and summarize the operational experience of nuclear power plants as well as the results of safety research for LMFBRs to contribute toward the development of safety evaluation policy.

2. Particular Objectives :

1) Analyze the records of abnormal incidents in various reactor facilities and sodium-related facilities and examine the policy of accident classification.

2) Perform systematic and quantitative evaluation on event sequences based on the results of experiments and analyses in safety research of LMFBRs and strengthen the policy of accident classification.

3. Experimental Facilities :

Not directly applicable

4. Description of Computer Codes :

Not directly applicable

5. Project Status: Progress to date

1) The incident data of LMFBRs have been investigated. CREDO (Centralized REliability Data Organization) data and several published documents are referred for this work. Also examination of those incidents were performed comparing with the events postulated in the safety evaluation.

2) Concerning the core local faults, various factors which could cause fuel failures have been comprehensively identified and classified based on engineering judgments.

6. Project Status: Essential results

1) The incident data of LMFBRs were analyzed in order to obtain insights into characteristics and trends of those incidents. The identified incidents leading to reactor shutdown are enveloped by the events in the safety evaluation for domestic fast reactors, or they are trivial events that do not affect the safety function of the relevant system.

2) Primary causes of initial local fuel failures were identified deductively and examined systematically. Those include mismatch between heat generation and removal, over burnup and random failure.

7. Next Steps:

Will perform preliminary evaluation on the frequency of abnormal events and continue investigation of incident records. Will evaluate the frequency of local faults leading to serious damage in a fuel assembly based on probabilistic methods and engineering judgments.

8. Relation to Other Projects or Codes :

Other Projects:
1) Development of Database for Safety Design and Evaluation.
2) Phenomenological Study on Local Fault in Core.

9. Reference Document:

1) T. Sakuma and Y. Kani, "Analysis on incident data of FBRs," PNC N9410 90-138, Sept. 1990.

10. Degree of Availability:

Not for publication, Official use only.

PRODUCT CODE 1.2	CLASSIFICATION Securing Overall Safety through the Use of PSA
TITLE: Development of Proper Plant Operational & Maintenance Procedures	
COUNTRY: JAPAN	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME S.Terunuma & T.Nakamura ADDRESS PNC-OEC, 4002, Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: in progress

1. General Aim:

The optimum proceduer of plant operation & maintenance is to be developed using the operation support system and the maintenance assistance system to improve plant availability and reliability.

2. Particular Objectives:

1) Operation Support System
With the enhancement of alarm processing and abnormalous diagnostic technique, the ability of abnormaly detection and the propriety of man-machine interface will be evaluated by oprating experience of JOYO and using the operation training simulator.
2) Maintenance Assistance System
Causal relation concerned on machine trouble are put into a database of the maintenance consulting systems. As a result of this, reliability of machinery in plant will be improved and preventive maintenance will be expected.

3. Experimental Facilities:

Experimental Fast Reactor JOYO

4. Description of Computer Codes:

1) Operation Support System

Computer language: C, FORTRAN 77
Computer used: V90-50, H80E (Hitachi), DS600(Toshiba)
Computer capacity required: 5 MB
Description of codes: Inference engine
Field of application: Artificial Intelligence

2) Maintenance Assistance System

Computer language: E SHELL
Computer used: FACOM M780/VP100
Computer capacity required: 5 MB
Description of codes: Inference engine
Field of application: Artificial Intelligence

5. Project Status: Progress to date

1) As a part of the operation support system, the intelligence database was developed for the reactor control system, the primary coolong system and the secondary cooling system. The fundamental software was verified by using the operation training simulator.
2) The maintenance assistance system of the freon cooling system and the electro-magnetic pumps were implemented. The intelligence database was updated by the actual plant experiences.

6. Project Status: Essential results

1) The intelligence database was expanded by adding the primary cooling system and the reactor control system. As for the alarm processing software, its function and appropriateness of man-machine interface were evaluated.
2) The intelligence database of the assistance system has being updated by the operation data obtained from the trial application of the maintenance assistance system to the freon cooling system and the electro-magnetic pumps in JOYO. The upgrading the reasoning method of this system was found out to be feasible.
3) CRT display of the operation support system was installed in the central control room of JOYO.

7. Next Steps:

1) A database related to the electric supply system is added to the operation support system and verified by using the operetion training simulator.
In addition, status signals of the components in the primary cooling system and the reactor control system are inputted to this support system.
2) The intelligence database of the maintenance assistance system is converted to the object-oriented style. In addition, the reasoning method of the maintenance assistance system is upgraded by using higher intelligence.

8. Relation to Other Projects or Codes:

None

9. Reference Documents:

K. Ogura, et al., "Development of JOYO Operation Support System(1)-Present situation and Plan-" G25, Fall Meeting of the Atomic Energy Society of Japan, 1987 (in Japanese)

10. Degree of Availability:

Not for publication except for the above document.

PROJECT CODE 1.2	CLASSIFICATION Securing Overall Safety through the Use of PSA	
TITLE: Study on PSA methodology for LMFBR plants		
COUNTRY: Japan		INITIATED: November, 1982
SPONSOR: S T A		COMPLETED:
ORGANIZATION: P N C		LAST UPDATING: August, 1990
PROJECT LEADER: NAME Yoshio Kani ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: In progress

1. General Aim :

To develop methodology and computational tools for performing systems analysis (level-1 PSA).

2. Particular Objectives :

1) To develop a code network system including fault tree analysis code SETS in order to perform a standard level-1 PSA expeditiously and efficiently.

2) To develop methodology dealing with topical issues such as human reliability evaluation and living-PSA.

3. Experimental Facilities :

N.A.

4. Description of Computer Codes :

1) STES	Computer language	: FORTRAN IV + Assembler
	Computer used	: FACOM M780/VP100, IBM-3081
	Computer capacity required	: approx. 3 MB
	Description of codes	: Fault tree analysis
2) MODESTY	Computer language	: IBM FORTRAN
	Computer used	: IBM PC/AT, TOSHIBA J3100
	Computer capacity required	: approx. 640 KB
	Description of codes	: Modular method-based fault tree construction
3) ETAAS	Computer language	: Turbo-C
	Computer used	: IBM PC/AT, TOSHIBA J3100
	Computer capacity required	: approx. 640 KB
	Description of codes	: Event tree Construction and quantification

4) HURASS	Computer language	: FORTRAN77
	Computer used	: TOSHIBA J3100, IBM PC/AT
	Computer capacity required	: approx. 640 KB
	Description of codes	: Human Reliability Analysis
5) QUEST	Computer language	: MS-C
	Computer used	: IBM PC/AT, TOSHIBA J3100
	Computer capacity required	: approx. 640 KB
	Description of codes	: Simplified level-1 PSA program
6) LIPSAS	Computer language	: C
	Computer used	: Macintosh
	Computer capacity required	: approx. 8 MB
	Description of codes	: Living PSA system

5. Project Status: Progress to date

Event Tree Analysis Assistant System (ETAAS) and Human Reliability Analysis Support System (HURASS) have been developed. The development of systems analysis code network has been completed. The development of Llving PSA System (LIPSAS) is initiated.

6. Project Status: Essential results

Through the above application of the code network system which includes computer codes described in the item 4, it is found that the use of this system can significantly reduce time and manpower and enables consistent system analysis.

7. Next Steps:

Continue to enhance capability of the code network system especially on the field of PSA application to safety design and/or operational safety management. Continue to develop LIPSAS for performing living-PSA.

8. Relation to Other Projects or Codes :

1) Application of PSA to an LMFBR plant.
2) Development of Database for Safety Design and Evaluation.

9. Reference Documents :

1) Hioki, et al., "PC-based Support Programs Coupled with the SETS Code for Large Fault Tree Analysis," PSA'89.
2) Nakai, et al., "A Simplified PSA Program QUEST", PSA'89.
3) Kani, et al., "PC-based Probabilistic Safety Assessment in Japan," International Post-SMiRT 10 Seminar #7, 1989.
4) Nakai, et al., "Development of Living PSA Tool for an LMFBR Plant," 2nd TÜV -Workshop on Living PSA Application, 1990.

10. Degree of Availability:

Not for publication except for the above documents.

PROJECT CODE 1.2	CLASSIFICATION Securing Overall Safety through the Use of PSA	
TITLE: Application of PSA to an LMFBR Plant		
COUNTRY: Japan		INITIATED: November, 1982
SPONSOR: S T A		COMPLETED:
ORGANIZATION: P N C		LAST UPDATING: August, 1990
PROJECT LEADER: NAME Y. Kani and S. Kondo ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: In progress

1. General Aim :

To comprehensively evaluate the design and operation of plant by performing probabilistic safety assessment (PSA) on a fast breeder reactor and utilize the results to establish future safety guidelines and criteria for fast breeder reactors.

2. Particular Objectives :

1) Construct event trees and fault trees models and quantify them. Identify accident sequences which lead to core damage and quantify them. (level-1 PSA)
2) Analyze accident phenomena in the primary system and the containment system and arrange the accident processes with event trees. Evaluate release and transport of radioactive materials. (level-2 PSA)
3) Also consider the effects of external events such as earthquakes.

3. Experimental Facilities :

Not directly applicable

4. Description of Computer Codes :

Computer language : FORTRAN IV + ASSEMBLER
Computer used : FACOM M780/VP100, IBM-3081
Computer capacity required: approx. 3 MB
Description of codes : SETS (Fault Tree Analysis)
Field of application : reliability, availability, physical process and risk analysis.

5. Project Status: Progress to date

1) Event trees and fault trees models have been constructed and modified. The system models and accident sequences with respect to internal initiating events have been quantified using a data base prepared for actual plant evaluation. Also preliminary evaluation on the operational procedures using the probabilistic system models has been performed.
2) The spectra of event progression have been described with pehnomenological event trees for the key core damage sequences identified by the above system analysis. In-vessel and ex-vessel (containment) physical processes have been analyzed and the preliminary evaluation on the release of radioactive materials has been performed.
3) For a seismic event analysis, evaluation of seismic hazard and fragility of main buildings and components are being performed. Also the preliminary seismic system analysis is in progress.

6. Project Status: Essential results

1) The results of systems analysis indicate that overall core damage frequency (CDF) is much lower than 10^{-5}/ry (which is the reference value for future plants proposed by the INSAG of IAEA). There exist no outliner in the failures of systems and components or event sequences which dominates prominently the overall CDF. This shows that a well-balanced safety design is achieved.
2) The event sequences leading to release of radioactive materials were identified and their probabilities and the source terms were quantified. Prospects have been obtained that the probability of mechanical failure of primary boundary is low, the probability of internal pressurization of the containment system which challenges its integrity of the system is extremely low, and hence the risk is kept at a sufficiently low level.

7. Next Steps:

Will perform sensitivity study for various dominant factors and continue evaluation of operational and maintenance procedures of safety systems. Will perform additional evaluation on the in-vessel and ex-vessel event progressions and complete the level-2 PSA. Will quantify the seismic accident sequences.

8. Relation to Other Projects or Codes :

Codes: SAS, SIMMER, CONTAIN, SSC-L etc.
Other Projects:
1) Study on Anticipated-Transient-Without-Scram Accidents
2) Study on Loss-of-Heat-Removal-System accidents
3) Study on Post-Accident-Heat-Removal Phase
4) Source Term and Hazard Analysis

9. Reference Documents:

1) M. Hori, K. Shiba and K. Aizawa, "State-of-the-Art of LMFBR Safety R&D at PNC," Int. Fast Reactor Safety Mtg., Snowbird Utah, August 1990.
2) S. Kondo, et al, "Integrated analysis of In-vessel and Ex-vessel Severe-Accident Sequences," Int. Fast Reactor Safety Mtg., Snowbird Utah, August 1990.

10. Degree of Availability:

Not for publication except for the above documents.

PROJECT CODE 2.1	CLASSIFICATION Prevention of Anomaly Occurrence	
TITLE: Improvement of Reactivity Predication Accuracy		
COUNTRY: Japan		INITIATED: April, 1986
SPONSOR: S T A		COMPLETED: March, 1991
ORGANIZATION: P N C		LAST UPDATING: August, 1990
PROJECT LEADER: NAME Keisho Shirakata ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: In progress

1. General Aim :

To Contribute to LMFBR safety analysis and evaluation

2. Particular Objectives :

To improve prediction accuracies for safety-related nuclear characteristics, such as control rod reactivity, sodium void reactivity, doppler reactivity, overall feedback reactivity, etc.

3. Experimental Facilities :

Fast Experimental Reactor "JOYO" at PNC
Zero Power Physics Reactor "ZPPR" at ANL

4. Description of Computer Codes :

Computer language	: FORTRAN 77
Computer used	: FACOM VP-100 and M-780
Computer capacity required	: 32 MB
Description of codes	: The fast reactor core analysis code system includes the 3-dimensional (3D) diffusion calculation code CITATION, the 3D coarsemesh diffusion calcu lation code MOSES, the 3D transport calculation code TRITAC, cell calculation codes SLAROM and CASUP, etc.
Field of application	: Fast reactor core analysis

5. Project Status:

Nuclear characteristics of the JOYO core, such as feedback reactivities, control rod reactivities and burnup characteristics were analyzed and assessed.
The large core critical experiments on LMFBR (JUPITER program) were analyzed and assessed, and using the C/Es (Calculaton/Experiment) the cross section set JFS-3-J2 was adjusted.
Neuronic calculation codes and models for LMFBR core analysis were developed and improved.

6. Project Status: Essential results

The range of fluctuation in measured control rod reactivity for the six control rods in the JOYO MK-II core was ascertained to be less than 4%.
The C/E for the burnup coefficient of the JOYO MK-I core was 1.02+(7~-13)%.
As the result of cross section adjustment, C/Es were very much improved, and ^{238}U capture cross section was decreased by about 6% for 1keV ~ 1MeV range.
The computing time of the three-dimensional discrete ordinates transport code TRITAC was reduced by a factor of 4 by improving the acceleration method.

7. Next Steps:

Nodal calculation codes will be developed for the analysis of fast reactor core.

8. Relation to Other Projects or Codes :

N.A.

9. Reference Documents :

T. Aoyama, et al., Proc. Fast Reactor Core and Fuel Structural Behaviour, BNES, London, 299 (1990).
T. Sanda et al., J. Atomic Energy Society of Japan, 31, 1324 (1989), (In Japanese).
T. Takeda, et al., Nucl. Sci. Eng., 101, 179 (1989).
T. Takeda, et al., Proc. PHYSOR '90, Marseille, Vol.3, PII-104 (1990).

10. Degree of Availability:

Not for publication except for the above documents.

PROJECT CODE 2.1	CLASSIFICATION Prevention of Anomaly Occurrence
TITLE: Improvement of High-Burnup Core Characteristics Prediction	
COUNTRY: Japan	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME Sakae Shikakura ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim :

To investigate allowable fuel design criteria for large-scale FBR under steady-state and off-normal operating conditions.

2. Particular Objectives :

(1) To analyze swelling and irradiation creep of fuel cladding as well as restructuring and gap conductance of fuel pellet by using fuel and material irradiation data in JOYO and FFTF.

(2) To analyze pin-to-pin interaction and pin bundle-duct interaction by using fuel bundle irradiation data in JOYO, FFTF and Phenix.

(3) To analyze power-to-melt criteria, fuel failure criteria and its mechanism by single and multi operational transient testings of fuel pins in EBR- II and JOYO.

3. Experimental Facilities :

(1) Irradiation Reactor: JOYO, FFTF, EBR-II and Phenix.

(2) PIE Facility : PNC-FMF and MMF in Japan, ANL-HFEF/N, HEDL in USA.

4. Description of Computer Codes :

Computer capacity required: 3 MB (BDI code)
Description of codes : PYTHON-SHADOW code-Thermal hydraulics and pin bundle deformation analysis developed by PNC

5. Project Status: Progress to date

(1) Irradiation behavior of high burnup fuel pin irradiated in Phenix up to 12 at. % BU and sub-assembly life limit of test assembly being irradiated in FFTF have been evaluated.

(2) Operational transient fuel pin behavior has been analyzed by using the Operational Reliability Testing data in EBR-II.

(3) Irradiation test data and out-of-pile material test data of advanced austenitic PNC1520 cladding tube has been analyzed.

6. Project Status: Essential results

(1) It has been experimentaly indicated that several periodic 115% over power events has been acceptable from the view point of fuel pin integrity under the condition of EBR-II operational transient testing.

(2) Material correlations for advanced austenitic core material PNC1520 have been composed.

7. Next Steps:

Fuel pin steady state breach limit and design limit on fuel integrity will be studied based on experimental data and evaluation result of analytical tools.

8. Relation to Other Projects or Codes :

None

9. Reference Documents :

H. Kashihara et. al, "Dimensional stability of FBR fuel pins with modified type 316 stainless steel cladding at high burn up", Fast Reactor Core and Fuel Structural Behavior, Proc. of international conf., 4-6 June 1990, London.
T. Asaga et. al, "Fuel pin behavior during duty cycle testing", International Fast Reactor Safety Meeting, Aug. 12-16 1990, Snowbird, Utah.

10. Degree of Availability:

Available.

PROJECT CODE 2.1	CLASSIFICATION Prevention of Anomaly Occurrence	
TITLE: Study on Shielding		
COUNTRY: Japan		INITIATED: April, 1986
SPONSOR: S T A		COMPLETED: March, 1991
ORGANIZATION: P N C		LAST UPDATING: August, 1990
PROJECT LEADER: NAME Katsuya Kinjo ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: In progress

1. General Aim :

Radiation Distribution Estimation in Reactor Plant

2. Particular Objectives :

Reduction of biological radiation dose of personnel
Optimum design for radiation damage fluence on reactor structures

3. Experimental Facilities :

JOYO
Tower Shielding Facility (ORNL, USA)

4. Description of Computer Codes :

Name : FBR Shielding Analysis System
Contents: Computer code system including nuclear group constant files, group constant handling codes, 1 and 2 dimensional Sn transported codes, and data handling codes for calculated results. Main code is DOT3.5 of ORNL and all codes are not classified.

5. Project Status: Progress to date

(1) Irradiation in JOYO was conducted for the new shielding materials (B_4C and Graphite); these are planned to be used in reactor vessels.
(2) Experimental data from JASPER (Japanese American Shielding Program of Experimental Researches) were analyzed.
(3) JASPER experiments are continued.
(4) Application of 3 dimensional Sn transported code TORT of ORNL code for FBR plant was conducted.

6. Project Status: Essential results

(1) The measurement of axial shield in a fuel assembly was conducted to predict the streaming effect and attenuator effect.
(2) The analysis of JASPER experimental data has made it clear that the B_4C is effective as the shielding material in a reactor vessel, and the analysis system is sufficiently reliable for this material.

7. Next Steps:

(1) Post-irradiation test of shielding structure including B_4C will be performed.
(2) JASPER experiments and data analysis will be continued.

8. Relation to Other Projects or Codes :

None

9. Reference Documents :

(1) T. Kosako et al.: Measurements and Evaluations of Neutron Dose and Spectra at the Reactor Top of the Liquid-Metal Fast Breeder Type Reactor JOYO; Nuclear Technology, 77, June 1987.
(2) N. Ohtani et al., "Benchmark Experiment and Analysis of Neutron Penetration through FBR Radial Shield Mockups", Proc. of 7th Int. Conf. on Radiation Shielding, Vol.2, p.433, Bournemouth, UK (1988).
(3) N. Ohtani, S. Suzuki: Advancements in FBR shielding - Ten Years in Japan; Trans. Am. Nucl. Soc., 62, 441 (1990).

10. Degree of Availability:

Not for publication except for the above documents.

PRODUCT CODE 2.1	CLASSIFICATION Prevention of Anomaly Occurrence
TITLE: Study on Exposure Dose Reduction	
COUNTRY: JAPAN	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME K. Kinjo ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 Telex PNDC J26462	STATUS: in progress

1. General Aim:

To reduce exposure dose of plant personnel in an LMFBR.

2. Particular Objectives:

1) To develop calculation codes for evaluating behavior of radioactive corrosion and fission products and tritium in an LMFBR plant by using the data obtained from both JOYO and the experimental sodium loops.
2) To develop the control measures of corrosion and fission products and the waste management technique, and to verify them at JOYO.

3. Experimental Facilities:

Experimental Fast Reactor JOYO, Fuel Monitoring Facility (FMF) and Activated Material Test Loop (AMTL) in PNC-OEC.

4. Description of Computer Codes:

1) PSYCHE (Program System for Corrosion Hazard Evaluation)
Computer language: FORTRAN 77, Computer used: FACOM M780/VP100
Computer capacity required: 7MB
Description of codes: Evaluation on corrosion product behavior in LMFBR.
Field of application: Analysis of radioactive corrosion product behaviour and resulting radiation fields near the piping and components in a loop type LMFBR primary circuits.
2) TTT (Tritium Transfer and Traps)
Computer language: FORTRAN 77, Computer used: FACOM M780/VP100
Computer capacity required: 6.5MB
Description of codes: Evaluation on tritium distribution in an LMFBR.
Field of application: Analysis of tritium concentration in the coolant, cover gas and atmospher of JOYO and MONJU.

5. Project Status: Progress to date

1) The code validation and refined model development has been underway for both PSYCHE and TTT by using the operational data in JOYO.
2) Following the completion of fundamental tests in AMTL, irradiation tests have been conducted for an in core corrosion product trap and cobalt-free hard facing materials in JOYO.

3) A cesium trap and a cover gas clean-up system was installed in JOYO, and were tested by using a uraniam-nickel fission-product-source.

6. Project Status: Essential results

1) Both values of C/E for corrosion product deposition and dose rate in JOYO primary piping system, using the PSYCHE code, were involved within 0.8 to 1.5.
2) Various nickel alloys have been tested in AMTL to find that the amount of deposited ^{54}Mn is proportional to that of nickel content. Based on the test results, a corrosion product trap made of pure nickel has been fabricated, installed and tested in both upper and lower part of a reflector subassembly of JOYO to validate its function, and their irradiation tests have been conducted to confirm their function.
3) Various carbon materials have been tested in AMTL to show that Reticulated Vitereous Carbon (RVC) is the most effective and suitable material for cesium trap, and the data necessary to design cesium traps, involving the trapping capacity and efficiency of cesium from sodium by the RVC, have been also obtained. A cesium trap has been designed and installed in JOYO primary purification circuit to demonstrate the effectiveness of this system in an LMFBR.
4) The measurement and analyses on the tritium behavior in JOYO have been conducted, those are consisted of the tritium concentration in the primary or secondary sodium or cover gas, tritium distribution in atmosphere of JOYO, and the diffusion rate of tritium through the sodium pipe wall or permeation rate through the themal insulator structure and chemical form analysis of the tritium i.e. gas form : T_2 or TH and water form : THO or T_2O.

7. Next Steps:

1) The code validation and model improvement are continued for both PSYCHE and TTT by using the operational data obtained from JOYO.
2) Irradiation tests of the corrosion product trap and the cobalt-free hard facing materials are conducted for their practical use in an LMFBR.
3) An artificially breached pin is irradiated in JOYO to provide data related to fission product behavior in the core and the primary circuit of JOYO. This especially aims to conduct the FFDL test by sipping method of fission products gas.

8. Relation to Other Projects or Codes:

QAD-CG code was modified to calculate radiation fields in PSYCHE.

9. Reference Documents:

1) K. Iizawa, et. al.,"Calculational Model and Code for Corrosion Products Transfer in Sodium Systems", Proc. Specialist's Meeting on Fission and Corrosion Product Behavior in Primary Circuits of LMFBR's, Karlsruhe (1987) p191.
2) K. Iizawa, S. Suzuki, et. al.,"Study on Radioactive Corrosion Products Behavior in Primary Circuits of JOYO", ibid., p227.
3) T. Odo, et al.,"Review on Development of FFDLs and FP Traps in JOYO", Proc. Specialist's Meeting on Plant Operating Experience of JOYO/KNK-II/ PHENIX/PFR, OEC (1989) p291.
4) T. Odo, M. Kinoshita, et. al.,"Fission-Product-Source Test in JOYO", ibid., p322.
5) K. Iizawa, T. Odo, et. al.,"Summary Report for Radioactive Corrosion Product Behavior and Its Trap Technique in Primary Circuit in JOYO", ibid., p341.

10. Degree of Availability:

Not for publication except for the above document.

PROJECT CODE 2.1	CLASSIFICATION Prevention of Anomaly Occurrence
TITLE : Advancement of Structural and Material Database	
COUNTRY: Japan	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADERS: NAME K. Iwata and Y. Wada ADDRESS PNC-OEC, 4002, Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim:

To make an advanced database of structual materials and simple structural models tests required for the establishment of the elevated temperature structural design guide of LMFBR

2. Particular Objectives:

1) To reflect the database on the development of new design criteria and life prediction methods
2) To provide highly reliable design data on LMFBR structural materials including the environmental effects with emphasis of weldments
3) To verify new evaluation methods on the strength of structures by test results

3. Expermental Facilities:

1) Structural Materials Test Machines In-Air
2) Structural Materials Test Machines In-Sodium
3) Post Irradiated Examination Facilities
4) Buckling Test Equipment for Structure
5) Creep-fatigue Test Equipment for Bellows
6) Thermal Transient Creep- Fatigue Test Facilities by Sodium

4. Description of Computer Codes :

SMAT ---------- FBR Structural Materials Processing System

STAR ---------- Structural test data compilation system

5. Project of Status: Progress to Date

Structural Materials Tests ; since 1986
Simple Structural Models Tests; since 1986

6. Project of Status: Essential Results

1) The advanced evaluation model on normalized and tempered 2-1/4Cr-1Mo steel, which is a typical cyclic softening material, was developed using the structural materials database.
2) Sodium environmental effect on long term creep-fatigue life was evaluated considering carburization behavior.
3) Design allowable stresses on Mod.9Cr-1Mo steel and FBR grade SUS316 were tentatively calculated for trial use in structural design.
4) Strain history effect, strain hold position effect, and multi-axial effect on creep-fatigue of SUS304 were evaluated.
5) Creep-fatigue life evaluation method on SUS304 weldment was presented tentatively.
6) The structural design code for bellows was proposed based on the structural test results.
7) Surface finish effect on thermal transient fatigue by sodium was examined, and it was clarified that this effect is not sensitive, even surface stress is large.
8) Several types of welded structures were tested under cyclic thermal transients.
9) Consolidation of structural database was started using STAR.

7. Next Steps:

1) Development of Data Base Managing System and the extension of functions for structural material data processing system (SMAT)
2) Continuous collection of structural data and input into the database (STAR)
3) Further compilation of test data

9. Reference Documents:

1) Aoto, et al.,ASME PVP Tech.(1987)
2) Aoto et al., ibid.
3) Watashiet al., 9th SMiRT(1987)L
4) Tsukimori et al., 6th ICPVT(1988)
5) Mimura et al.,LIMET '88(1988)
6) Kano et al., ibid.
7) Maruyama et al., ibid.
8) Kasahara et al., 10th SMiRT(1989)E
9) Kawasaki et al., ASME PVP TECH.(1989)
10) Aoto et al., ibid.
11) Wada et al., ibid.
12) Aoto et al.,12th Int.CODATA Conf.(1990)
13) Yamashita et al., Int.Pressure Vessel and Piping 42(1990)
14) Saito et al., J. Pressure Vessel and Piping 44(1990)

10. Degree of Availability:

Not for publication except for the above reports.

PROJECT CODE 2.1	CLASSIFICATION Prevention of Anomaly Occurrence
TITLE : Study on Seismic Analysis Methods of Components and Pipings	
COUNTRY: Japan	INITIATED: April, 1987
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADERS: NAME Koji Iwata ADDRESS PNC-OEC, 4002, Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim:

To enhance reliability of seismic design methods for LMFBR components and pipings.

2. Particular Objectives:

1) To establish dynamic response analysis methods for LMFBR core components and pipings.

2) To verify methods by vibration tests.

3. Experimental Facilities:

N.A.

4. Description of Computer Codes:

Code name	:	FINAS (FInite element Nonlinear structure Analysis System)
Computer language	:	FORTRAN IV
Computer used	:	FACOM M780/VP100
Computer capacity	:	Approximately 2MB
Description of code	:	A general purpose finite element nonlinear analysis
Field of application	:	static, heat transfer, inelastic, large deformation, fracuture mechanics, fluid-structure, and dynamic analysis

5. Project of Status: Progress to Date

1) Seismic safety assessment of an LMFBR core ; since 1986

2) Vibration tests and analysis of bellows piping system ; since 1986

3) Development of linear fluid-structure analysis method ; since 1986

6. Project of Status: Essential Results

1) The cluster vibration analysis method for LMFBR core was established using non-linear gap-spring elements in FINAS computer code.

2) Simplified method for seismic response analysis of bellows piping system was developed.

3) Linear fluid-structure interaction analysis method was established.

7. Next Steps:

Development and further improvement of dynamic response analysis methods are to be continued.

8. Relation to Other Projects or Codes :

None

9. Reference Documents:

1) M.Morishita and K.Iwata, "Seismic Behavior of a Large Free-standing Core", Int. Conf. of 11th SMiRT, 1991

2) M.Morishita, N.Ikahata, and S.Kitamura, "Dynamic Analysis Methods of Bellows Including Fluid-Structure Interaction", 89-PVP-168, ASME, 1989

3) K.Iwata, M.Morishita, and F.Kubo, "Dynamic Fluid-Structure Interaction Analysis Methods in the FINAS Computer Program and Applications to Sloshing Problem", Int. Conf. of Computational Engineering Science, Springer-Verlag, 1988

10. Degree of Availability:

Not for publication except for the above.

PROJECT CODE 2.1	CLASSIFICATION Prevention of Anomaly Occurrence
TITLE: Improvement of High Temperature Structural Analysis Methods	
COUNTRY: Japan	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADERS: NAME K. Iwata and Y. Wada ADDRESS PNC-OEC, 4002, Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim:

To develop more accurate and reliable structural analysis methods for predicting inelastic behavior of high temperature components of fast breeder reactors.

2. Particular Objectives:

1) To enhance the accuracy and the applicability of constitutive equations based on uniaxial and multiaxial test data of structural materials
2) To provide standardized inelastic analysis methods and associated material data
3) To verify inelastic structural analysis methods, based on structural element behavior tests
4) To implement inelastic analysis methods developed above in the existing general purpose structural analysis computer programs such as FINAS

3. Expermental Facilities:

1) Material test facilities
2) Structural element test equipments

4. Description of Computer Codes:

FINAS (Finite Element Nonlinear Structural Analysis System):
A general purpose finite element computer program for the solution of static, dynamic and heat transfer analyses of arbitrary complex structures made of elastic and inelastic materials. Elastic, plastic, creep, swelling, large deformation, large strain, buckling, fracture mechanics analyses can be performed. Modal response, spectrum response and direct integration response analyses considering geometrical gaps and fluid-structure interactions can be performed. Seventy kinds of finite elements are available. The program is developed for FACOM, IBM, CRAY etc. in FORTRAN IV

to run under MVS, COS, NOS, UNIX. The total lines of source program attain to 300,000. Required memory is 1MB. The program has been in use at 20 sites for about ten years.

5. Project of Status: Progress to Date

1) A multiaxial inelastic behavior simulation program (ADMODEL) was developed to assess the constitutive equations.
2) Cyclic plasticity constitutive equations for SUS304 and viscoplastic constitutive equations for Mod. 9Cr-1Mo have been assessed.
3) Basic material properties equations were developed for Mod.9Cr-1Mo.
4) Cyclic behavior verification tests using a notched plate started.
5) Advanced cyclic plasticity constitutive models were implemented in the FINAS program.

6. Project of Status: Essential Results

The non-hardening strain region model was confirmed to be promising as a cyclic plasticity model. A new two-surface cyclic plasticity model was developed. These models were implemented into the FINAS program for practical use.

7. Next Step:

1) Further development of cyclic plasticity model for non-isothermal conditions
2) Assessment of unified constitutive equations
3) Material behavior tests and structural element tests for verification of advanced inelastic analysis methods
4) Enhancement of inelastic analysis capabilities of FINAS

8. Relation to Other Projects or Codes:

None

9. Reference Documents:

1) FINAS User's manual, Ver. 11.0, N9520 89-019, Power Reactor and Nuclear Fuel Development Corporation, 1989.
2) Iwata,K. Int. Conf. of 11th SMiRT, L22/4, 1991
3) Tanaka,E. et al. The 6th Int. Conf. on Mechanical Behavior of Metals, 1991

10. Degree of Availability:

FINAS is available on a commercial basis.
Not available expect for the above document.

PROJECT CODE 2.1	CLASSIFICATION Prevention of Anomaly Occurrence
TITLE: Study on Dynamic Fluid-Structure Interaction Analysis	
COUNTRY: Japan	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: AUGUST, 1991
PROJECT LEADERS: NAME Koji Iwata ADDRESS PNC-OEC, 4002, Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim:

To establish dynamic fluid-structure interaction analysis methods for seismic safety design of LMFBR

2. Particular Objectives:

1) To develop analytical methods for general fluid-structure interaction problems

2) To estimate sloshing, bulsing, and dynamic buckling characteristics of LMFBR components

4. Description of Computer Codes :

Code name	:	FINAS (FInite element Nonlinear structure Analysis System)
Computer language	:	FORTRAN IV
Computer used	:	FACOM M780/VP100
Computer capacity	:	Approximately 2MB
Description of code	:	A general purpose finite element nonlinear analysis
Field of application	:	static, heat transfer, inelastic, large deformation, fracuture mechanics, fluid-structure, and dynamic analysis

5. Project of Status: Progress to Date

1) A practical linear fluid-structure interaction analysis method has been developed and implemented in FINAS code since 1986.

2) A sloshing problem has been analyzed and exprimented since 1987.

3) A bulsing problem has been analyzed and exprimented since 1989.

6. Project of Status: Essential Results

1) A sloshing problem of axisymmetric and general three-dimensional shell structures containing fluid has been analyzed using FINAS code.

2) Analytical capabilities of FINAS code has been shown for fluid-structure system such as modal response, spectrum response, and direct integration response analysis.

3) Vibration tests of thin shell structures has been performed in order to verify FINAS code in the range of significant coupling effect.

4) A satisfactory agreement was obtained between experimental and computed results in terms of natural frequencies, modes,and dynamic response of sloshing and bulsing.

7. Next Step:

Further improvement of the code to apply for more complex system analysis.

9. Reference Documents:

1) K.Iwata, M.Morishita, and F.Kubo, "Dynamic Fluid-Structure Interaction Analysis Methods in the FINAS Computer Program and Applications to Sloshing Problem", Int. Conf. of Computational Engineering Science, Springer-Verlag, 1988

2) S.Kitamura, M.Morishita, and K.Iwata, "Fluid Coupled Vibration of a Circular Plate due to Vertical Excitation", Int. Conf. 11th SMiRT, 1991

10. Degree of Availability:

Not for publication except for the above.

PROJECT CODE 2.1	CLASSIFICATION Prevention of Anomaly Occurrence
TITLE: Study on Seismic Isolation of Reactor Plant Building	
COUNTRY: JAPAN	INITIATED: April, 1987
SPONSOR: S T A	COMPLETED: March, 1990
ORGANIZATION: P N C	LAST UPDATING: March, 1990
PROJECT LEADER: NAME Koji Iwata ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, Japan 311-13 TELEPH. 0292-67-4141 TELEX 0292-67-7173	STATUS: Completed

1. General Aim :

To establish dynamic analysis methods for seismic isolation system of FBR plant.

2. Particular Objectives :

(1) To develop dynamic analysis methods for isolation devices composed of rubber bearings and hysteretic dampers.

(2) To verify the methods by vibration tests.

3. Experimental Facilities :

N.A.

4. Description of Computer Codes :

Code name : FINAS(FInite element Nonlinear structure Analysis System)
Computer language : FORTRAN IV
Computer used : FACOM M780/VP100
Computer capacity : Approximately 2MB
Description of code : A general purpose finite element nonlinear analysis
Field of application: static, heat transfer, inelastic, large deformation, fracture mechanics, fluid-structure, and dynamic analysis

5. Project Status: Progress to date

(1) A detailed analysis method of isolated reactor building has been developed since 1987.

(2) A vibration test using scaled isolator model has been performed to verify the methods since 1988.

6. Project Status: Essential results

(1) Dynamic analysis method for seismic isolation system of FBR plant was developed. Laminated rubber bearing was modeled on elasto-plastic characteristics obtained by static loading test. Steel damper is modeled on hysteretic model (Ramberg-Oswood type).

(2) A series of vibration tests of scaled building model with various isolation systems has been performed on a vibration table, using three-dimensional input waves or large input level. Basic properties and effectiveness of isolation system were observed.

(3) A satisfactory agreement was obtained between experimental results and simulated analysis in terms of the maximum response accelerations and floor response spectra.

7. Next Steps:

None

8. Relation to Other Projects or Codes :

N.A.

9. Reference Documents :

(1) M.Morishita et al, "Investigation of Base Isolation for Fast Breeder Reactor (Part 1 to 3)", Annual Meeting of Atomic Energy Society of Japan, 1987
(2) M.Morishita et al, "Investigation of Base Isolation for Fast Breeder Reactor (Part 4 to 6)", Fall Meeting of Atomic Energy Society of Japan, 1987
(3) M.Morishita et al, "Seismic Base Isolation Tests of Fast Breeder Reactor Building (Part 1 to 3)", Fall Meeting of Atomic Energy Society of Japan, 1988
(4) M.Morishita et al, "Investigation of Seismic Base Isolation for Fast Breeder Reactor Building", Int. Conf. of 10th SMiRT, 1989

10. Degree of Availability:

Not for publication except for the above documents.

PROJECT CODE 2.2	CLASSIFICATION Prohibition of Anomaly Escalation
TITLE: Improvement of Reactor Protection System Reliability	
COUNTRY: Japan	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME Masatoshi Moriyama ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim :

To improve the method of reliability evaluation for the reactor protection system in the large FBR.

2. Particular Objectives :

1) Prepare the method of reliability evaluation for the reactor protection system.

2) Investigate the method of reliability evaluation for the reactor protection with a new shutdown mechanism.

3) Apply the evaluation method of common cause failure for the reactor protection system.

3. Experimental Facilities :

None

4. Description of Computer Codes :

Computer language : FORTRAN IV
Computer used : FACOM M-780
Computer capacity required: 3 MB
Description of codes : SETS
Field of application : Fault tree analysis

5. Project Status: Progress to date

The reliability evaluation method using SETS code was prepared for the reactor shutdown system. The material and environment of the new shutdown mechanism (Curie point electromagnet) were investigated.

6. Project Status: Essential results

Qualitative evaluation was made for the temperature transient of the self-actuated shutdown mechanism at a demand transient, and for the common cause failure applying generic cause approach.

7. Next Steps:

Common cause failures are to be investigated.

8. Relation to Other Projects or Codes :

None

9. Reference Documents :

None

10. Degree of Availability:

Not for publication

PROJECT CODE 2.2	CLASSIFICATION Prohibition of Anomaly Escalation
TITLE: Study on Monitoring and Diagnostic System for Reactor Anomaly	
COUNTRY: Japan	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME Hisashi Nakamura ADDRESS PNC-OEC, 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: in Progress

1. General Aim :

To develop a core surveillance and anomaly detection system for LMFBRs.

2. Particular Objectives :

1) Develop an advanced core surveillance systems with new type sensors.

2) Develop a signal processor with high quality and quick response.

3) Build a surveillance and anomaly detection system with artificial intelligence.

4. Description of Computer Codes :

ALOCAD : "Stand-alone" codes that can process steady and transient data quickly and detect anomalies with several filters.
Computer language: FORTRAN 77
Computer used : XEROX6401(UNIX WS)

5. Project Status : Progress to date

1) Improvement of quick response sensors; thermocouples, flow meters and neutron detectors ---in progress.

2) Improvement of quick response signal processors ---in progress.

3) Systematization of sensors, processors and diagnosis linkage ---in progress

6. Project Status : Essential results

1) A core surveillance and anomaly detection system had developed through in-core instrumentations, quick signal processors and artificial intelligence.

2) The system could detect and identify simple anomalies.

7. Next Steps :

Validation of the core surveillance and anomaly detection system through in-pile tests of experimental fast reactor JOYO.

8. Relation to Other Projects or Codes :

None

9. Reference Documents :

None

PROJECT CODE 2.2	CLASSIFICATION Prohibition of Anomaly Escalation
TITLE: Study on Operation Beyond Cladding Breach	
COUNTRY: Japan	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME Sakae Shikakura ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim :

To study the breached fuel detection and operational limit beyond fuel cladding breach.

2. Particular Objectives :

(1) To establish the diagnostic method for breached fuel pin.

(2) To establish the criteria for reactor operation beyond fuel cladding breach.

3. Experimental Facilities :

(1) Irradiation Reactor: EBR-II of ANL in USA and JOYO of PNC in Japan.

(2) PIE Facility : PNC-FMF and MMF in Japan, ANL-HFEF/N, HEDL in USA, Phenix and CABRI in FRANCE

4. Description of Computer Codes :

Computer language : FORTRAN
Computer used : FACOM
Computer capacity required: 1 MB
Description of codes : TAFF code: Breached pin performance analysis. (in progress)
Field of application : Analysis of fuel behavior at steady state and transient condition.

5. Project Status: Progress to date

Phase-I program of DOE/PNC collaboration in EBR-II has been completed.
Phase-II program of DOE/PNC collaboration in EBR-II was started in 1988.

6. Project Status: Essential results

(1) Pin to pin failure propagation has not been found during 140 days RBCB operation for 37 fuel bundle.

(2) Fuel release from breached pin was proved to be a small quantity in steady state operation.

(3) Thermal evaluation of pre-defected pin was conducted in several power-to-melt tests.

(4) Analytical function to evaluate the deformation behavior by formation of fuel-sodium reaction products were established in breached pin performance analysis code.

7. Next Steps:

Preparations for JOYO RBCB tests will be completed in 1995.

8. Relation to Other Projects or Codes :

None.

9. Reference Documents :

S. Ukai et al: Release Characterization of Delayed Neutron Precursors from Breached FBR Fuel Element; J. Nuclear Science and Technology vol 26, 10 (1989) 33

J. H. Bottcher et al: Long Term RBCB Operation of Mixed Oxide Fuel Subassembly in EBR-II ; ANS Transaction 1989 Winter meeting vol 60, p.311

J. D. B. Lambert: Run-Beyond-Cladding-Breach Oxide Testing in EBR-II , BNES in Inverne ss 4-6 June 1990, P.17

10. Degree of Availability:

Not available except for the above documents.

PROJECT CODE 2.2	CLASSIFICATION Prohibition of Anomaly Escalation
TITLE: Establishment of Database for Structural Integrity	
COUNTRY: Japan	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME K. Iwata and Y. Wada ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX J26462	STATUS: In Progress

1. General Aim:

To establish crack growth and fracture data necessary for structural integrity of the demonstration LMFBR.

2. Particular Objectives:

1) To establish material database of austenitic stainless steel, chromium molybdenum steel and their weldments on Creep, fatigue and creep-fatigue crack growth rate at elevated temperature and fracture toughness at elevated temperature.

2) To clarify crack growth and fracture properties of LMFBR structural model at elevated temperature

3. Experimental Facilities:

Material Crack Growth Test Machines
Sodium Thermal Fatigue Test Loop
Thermal Transient Test Facility for Structures

4. Computer Codes:

N.A.

5. Project Status: Progress to date

1) Material database of crack growth rate and fracture toughness

2) Properties of LMFBR structural model
- Crack growth due to thermal transient in SUS304 pipe and vessel
- Crack growth data of SUS316 bellows:

6. Project Status: Essential results

1) Material database of crack growth rate and fracture toughness

- Fatigue crack-growth rate in SUS304 weldment was almost the same as that in SUS304 base metal.
- Creep, fatigue and creep-fatigue crack-growth rate of Mod.9Cr-1Mo steel was a little lower than that of 2.25Cr-1Mo steel.
- Creep crack growth rate of SUS304 was examined in the region of very slow rate.
- Fracture taughness test on Mod. 9Cr-1Mo steel was carried out.

2) Properties of LMFBR structural model

- Beachmark method and potential drop method are effective for crack-growth measurement in SUS304 structural elements such as plate, pipe, and elbows. Beachmark method and carefull ultrasonic inspection are effective for measurement of crack configuration under thermal loading.
- Fatigue and creep-fatigue crack-growth data of plates, streight pipes and elbows with a aurface crack under mechanical loading. Fatigue crack-growth data of full circumferential crack in a cylindersubjected to thermal loading.

7. Next Steps:

1) Material database of crack growth rate and fracture toughness

- Creep crack growth data under low growth rate condition
- Crack growth data in liquid sodium
- Crack growth data under thermal fatigue condition
- Fracture toughness data

2) Properties of LMFBR structural model

- Creep-fatigue crack growth data of structural elements
- Crack growth data under the condition where mechanical loading and thermal loadind are superposed
- Interference effect of multiple cracks on crack growth rate

8. Relation to Other Projects or Codes:

Fracture Mechanics Analysis and Crack Evaluation

9. Reference Documents:

Nonaka et al., 6th ICPVT(1988)
Watashi et al., Nuclear Engineering and Design 116(1989)
Watashi et al., 7th Post SMIRT Cnference Seminar No.5(1989)
Saito et al., J.Pressure Vessel and Piping 44(1990)
Doi et al., ASME PVP Conference(1990)

10. Degree of Availability:

Not for publication except above reports.

PROJECT CODE 2.2	CLASSIFICATION Prohibition of Anomaly Escalation
TITLE: Fracture Mechanics Analysis and Crack Evaluation	
COUNTRY: Japan	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADERS: NAME I. Nihei and A. Imazu ADDRESS PNC-OEC, 4002, Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: in progress

1. General Aim:

To develop an advanced evaluation method for flaws which is presumed in FBR components for safety assessment.

2. Particular Objectives:

1) To develop the advanced evaluation method and database for crack growth behavior and failure behavior of FBR components.
2) To rationalize the presumed failure for safety assessment by confirmation of LBB character for coolant boundary made of new material and improvement of structural integritry assessment method.

3. Experimental Facilities:

1) Structural Materials Test Machines In-Air
2) Piping Test Rigs In-Air
3) Thermal Transient Fatigue Test Facilities by Air
4) Thermal Transient Fatigue Test Facilities by Sodium

4. Description of Computer Code:

Computer Language: FORTRAN 77 Computer used: CRAY-XMP
Computer capacity required: > 2 MB
Description of codes: 'CANIS' codes (Crack ANalysis In Structures)
1) CANIS-J code: Fracture mechanics parameters are computed from the result of FINAS code.
2) CANIS-G code: Aspect of stable crack growth is calculated using material crack growth data. Aspect of penetrated crack and the number of penetration cycles are main output.
3) CANIS-P code: Probabilistic fracture mechanics assessment code.

5. Project of Status: Progress to date

1) Consolidation of crack analysis method
Selection of fracture mechanics parameter for stable crack growth was completed.
Detail crack analysis method and growth estimation method were developed.
Analysis code 'CANIS-J' and 'CANIS-G' were constructed.
2) Consolidation of criteria for crack growth and failure
Defect assessment method for flawed elbow was developed.

6. Project of Status: Essential results

1) Consolidation of crack analysis method
Stable crack growth estmation method under thermal transient loading was validated by structural failure tests.
2) Consolidation of criteria for crack growth and failure
Distributions of fracture mechanics parameters and crack growth rate of a plate model with a semi-elliptical surface crack were prepared for membrane, bending and their combination stress states. Using these distributions the number of penetration cycles and the aspects of penetrated cracks were estimated.

7. Next Step:

1) Analysis and evaluation method will be improved based on new test data.
2) More accurate evaluation method of unstable crack growth will be constructed.
3) Assessment method based on probabilistic fracture mechanics will be constructed for reinforcement of logic for presumed leak area which was developed from deterministic fracture mechanics.
4) Assessment method for LBB will be consolidated.

8. Relation to other Projects or Codes:

1) Improvement of high temperature structural analysis code
2) Establishment of database for structural integrity

9. Reference Documents:

1) Watashi et al.,Nuclear Engineering and Design 116(1989)423-441
2) Watashi et al.,ASME,PVP-Vol.167(1989)15-23
3) Doi et al.,ASME PVP Conference(1990)
4) Watashi et al.,Preprint of 7th Int. Conf.Inelastic Analysis, Fracture and Life Prediction,Santa Barbara,EdF(1989)
5) Asayama et al., ibid.
6) Furuhashi et al.,PNC Report N9410 90-136(1990)
7) Watashi et al.,11th SMIRT(1991)G30M/4
8) Watashi et al.,11th SMIRT(1991)L11G/3
9) Takenaka et al.,11th SMIRT(1991)L
10) Shimakawa et al., ibid.
11) Iwasaki et al., ibid.

10. Degree of Availability:

Not for publication expect for above reports.

PROJECT CODE 2.2	CLASSIFICATION Prohibition of Anomaly Escalation
TITLE: Improvement of Steam Generator Safety	
COUNTRY: Japan	INITIATED: 1986
SPONSOR: S T A	COMPLETED: Mar. 1991
ORGANIZATION: P N C	LAST UPDATING: Aug. 1991
PROJECT LEADER: NAME Hiroshi Hara ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: in progress

1. General Aim:

1) Improvement of a water leak detection system reliability
2) Clarification of the leak propagation behavior of high-chrome ferritic steels
3) Validation of pressure relief system performance
4) Safety evaluation of a duplex tube steam generator

2. Paticular Objectives:

1) To evaluate availability of an integrated leak detection system consisting hydrogen-meters, acoustic sensors, etc.
2) To select a design basis leak event for large LMFBR from micro-small leak wastage tests and leak propagation tests using high-chrome ferritic steels
3) To evaluate the effects of pressure and temperature on the structures during large scale sodium-water reaction
4) To analyze the behavior of sodium-water reaction products in reactor, and to assess the potential of reactivity change by hydrogen and local fault caused by sodium oxide sodium hydro-oxide, etc.

3. Experimental Facilities:

SWAT-1, SWAT-2, SWAT-3, SWAT-4, SWAT-5 and PEPT

4. Description of Computer Codes:

Computer language:Fortran IV / Fortran 77
Computer used: FACOM M780/VP100
Computer capacity required: 1.1 MB
Description of codes:
(a) SWACS : a pressure propagation analysis code for large leak has a following models :
- Components are assumed to be one-dimendional.

- Pressure wave propagation near a reaction zone is analyzed in the spherical co-ordinate (sphere-cylinder model).
- Pressure wave propagation is analyzed by the explicit method of characteristics in one-dimensional Eulerian co-ordinate.
- Flow-induced force is analyzed by momentrum balance.

(b) LEAP : a leak propagation analysis code
(c) SWAC-10 : a leak detection analysis code

Field of application: Monju and large LMFBR SG safety analysis

5. Project Status: Progress to date

1) Completed
2) The wastage database on three typical high-chrome steels such as modified 9Cr-1Mo steel was constructed. A Series of failure propagation tests is ongoing by use of SWAT-1 under the prototypical condition simulating tube inside cooling.
3) A new model of the initial spike pressure(ISP) calculation module of SWACS was developed for a non-cover gas type steam generator.
4) A basic hydrogen bubble dissolution model was developed to analyze the hydrogen behavior in a reactor system.

6. Project Status: Essential results

2) The wastage resistivity of weld is almost same as that of base metal of the tubes. The actual cooling effect inside tubes lessens the severity of wastage.
3) The new ISP module including a rupture disk dynamic response model was validated using the PEPT data. The analytical result agrees well with the experimental data.
4) The new model can calculate the change of a hydrogen bubble diameter while it flows from a steam generator to a reactor core.

7. Next Steps:

1) None
2) To improve LEAP taking into account the wastage database of high-chrome steels.
3) To improve and validate the SWACS quasi-static calculation module.
4) To analyze the behavior of sodium-water reaction products such as NaOH and Na_2O in the Joyo Mark III plant, where the elimination of IHTS is under consideration.

8. Relation to Other Projects or Codes:

None

9. Reference Documents:

H. Tanabe, et al.,:Analysis of Large Leak Sodium-Water Reaction in Large FBR, Intl Fast Reactor Safety Meeting, Snowbird USA, August 1990.

10. Degree of Availability:

Not for publication except for the above document.

PROJECT CODE 2.3	CLASSIFICATION Containment/Confinement of Abnormal Release of Radioactive Materials	
TITLE: Research on Aerosol Behavior		
COUNTRY: Japan		INITIATED: Apr. 1986
SPONSOR: S T A		COMPLETED: Mar. 1991
ORGANIZATION: P N C		LAST UPDATING: Aug. 1990
PROJECT LEADER: NAME H. Hara and K. Kinjo ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: in progress

1. General Aim:

To construct a database of aerosol behavior including multicomponent aerosol and their chemical characteristics for realistic evaluation of the source term

2. Particular Objectives:

To utilize the results to the assessment of postulated sodium leak accident in future LMFBRs

3. Experimental Facilities:

SAPFIRE facility

4. Description of Computer Codes:

Computer language: Fortran 77 Compute used: FACOM M200
Computer capacity required: 1 MB
Description and field of application:
ABC-INTG code: analysis of aerosol befavior

5. Project Status: Progress to date

1) Test for aerosol leakage through a narrow path
2) Test for multicomponent nuclear aerosol behavior
3) Test for chemical characteristics of sodium aerosol
4) Development of computer code to analyze aerosol behavior under turbulent natural convection of gas
5) Validation and revision of aerosol behavior code using the above mentioned experimental results

6. Project Status: Essential results

1) Sodium aerosol leakage experiments were performed using bent pipes and mesh wire packed pipes of 0.2-10 mm in diameter and annular leakage paths of 0.2-2 mm in the gap size. The following results were obtained;
   - impaction of aerosol particles was a dominant factor of leakage path plugging,
   - the plugging time depended on the product of the aerosol concentration and the gas flow velocity.

2) Sodium pool fire tests were conducted in nitrogen atmosphere containing 3 v/o oxygen to measure the aerosol release rate from the pool. The results reveal that the aerosol release rate depends on the sodium pool temperature and is less than about 10% of the measured sodium burning rate.

3) Experiments of reactor components and electrical instruments were conducted in a sodium aersol atmosphere to confirm their durability and the reliability. The results revealed that the decrease in the heat removal performance of anooling system and the damage of the instruments due to the aerosol deposition and its chemical attack cause no problem.
   A new analysis method using X-ray diffractometry was developed to measure chemical composition and its content ratio of sodium aerosol.

4) A prototypical code was developed and validated for the analysis of turbulent natural convection which dominates aerosol transport behavior.
   Sodium pool fire model was installed into the code.

7. Next Steps:

1) Derivation of a general correlation for leakage path plugging

2) Measurement of concentration, settling velocity and particle size distribution for multicomponent-aerosol

3) Measurement of chemical composition for sodium aerosol

4) Installation of aerosol behavior model to analysis code for turbulent natural convection

8. Relation to Other Project or Codes:

Research on FP release and transport

9. Reference Documents:

1) H. Seino, et al.,"Validation of CONTAIN Code for Sodium Aersol Behavior," International Fast Reactor Safety Meeting, Snowbird, USA, Aug. 1990.
2) S. Ohno, et al.,"Test and Code Development for Evaluation of Sodium Fire Accidents," International Fast Reactor Safety Meeting, Snowbird, USA, Aug. 1990.

10. Degree of Availability:

Not for publication except for the above documents.

PROJECT CODE 3.1	CLASSIFICATION Local Fault in Core
TITLE: Phenomenological Study on Local Fault in Core	
COUNTRY: Japan	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME Y. DAIGO and H. HINOKATA ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim :

To specify accident scenario and key phenomena of local faults in an LMFBR fuel subassembly for a judgement of coolability and detectionability of the accidents.

2. Particular Objectives:

1) To analyze the causes and the failure propagation of local faults.
2) To analyze the SCARABEE in-pile tests for the investigation of a total inlet blockage accident of a fuel subassembly and of the related inter-subassembly failure propagation process.
3) To improve and validate analysis codes through experimental analyses.

3. Experimental Facilities :

SCARABEE(France)

4. Description of Computer Codes :

1) SABENA
Computer language : FORTRAN 77
Computer used : FACOM M780/VP100
Computer capacity required: problem dependent
Description of codes : 2-fluid sub-channel analysis for sodium boiling in a pin-bundle.

2) FUMES
Computer language : FORTRAN IV
Computer used : FACOM M780/VP100
Computer capacity required: approximately 700KB
Description of codes : an analysis code for a boiling pool behavior in a subassembly and the related hexcan melt-through failure.

3) SCION

Computer language	: FORTRAN 77
Computer used	: FACOM M780/VP100
Computer capacity required	: problem dependent
Description of codes	: an analysis code for a molten material behavior in a pin-bundle geometry

5. Project Status: Progress to date

1) Review of domestic and overseas researches : in progress
2) Analysis of the SCARABEE in-pile tests : in progress
3) Code improvement and validation : in progress

6. Project Status: Essential results

Under the simulated condition of the total inlet blockage accident, chronology of failure events in the subassembly was extracted from the SCARABEE/BE+3, PV-A and PI-A test results and analyses.

7. Next Steps:

Review and analysis of the in-pile tests are continued.
Code improvement and validation study are continued.

8. Relation to Other Projects or Codes :

None

9. Reference Documents:

M. Konomura, et al.: The SCARABEE Propagation Test Series PI-A and PV-A, international Fast Reactor Safety Meeting, Snowbird, USA, Aug. 1990.

10. Degree of Availability:

Not available except for the above document.

PROJECT CODE 3.2	CLASSIFICATION Fuel Behavior under Accident Conditions	
TITLE: Fuel Behavior Study under Accidental Conditions		
COUNTRY: Japan		INITIATED: April, 1986
SPONSOR: S T A		COMPLETED: March, 1991
ORGANIZATION: P N C		LAST UPDATING: August, 1990
PROJECT LEADER: NAME Sakae Shikakura ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: In progress

1. General Aim :

To study the realistic failure criteria of high burnup fuel at the accident condition.

2. Particular Objectives :

(1) To evaluate the safety criteria of fast reactor fuel.

(2) To study the fuel performance and fuel failure criteria.

3. Experimental Facilities :

(1) Irradiation Reactor: JOYO in Japan, EBR-II in USA, Phenix and CABRI in France

(2) PIE Facility : PNC-FMF and MMF in Japan, ANL-HFEF/N, HEDL in USA

4. Description of Computer Codes :

Computer language : FORTRAN
Computer used : FACOM
Computer capacity required: 2.5 MB
Description of codes : CEDAR code - Fuel pin behavior at transient over power condition.
Field of application : Analysis of fuel behavior at steady state and transient condition.

5. Project Status: Progress to date

(1) Database for failure criteria was constructed in accordance with the irradiation condition and fuel specification in EBR-II and CABRI operational transient test.
(2) The cladding strength data under the loss of flow enent was obtained by simulating a repid heating burst test of high burnup fuel cladding.
(3) Transient overpower testing was conducted for high power rating (max 800W/cm) irradiated fuel pin and then analyzed from the view point of molten fuel movement through the central void of high burnup pellet.
(4) Code verification for transient fuel pin performance was carried out using above results of EBR-II operational transient test.
(5) Scoping analysis was conducted for CABRI-II project which includes slow ramp tests of high burnup fuels.

6. Project Status: Essential results

(1) Margin for fuel pin failure was clarified under the operational transient event.
(2) The cladding deformation was related to ramp rate, smear density, cladding thickness and molten fuel.

7. Next Steps:

(1) Phase-II of operational transient test in EBR-II including annular pellets and axial heterogeneous pin.
(2) Analysis of rapid heating burst data.
(3) Verification of transient fuel pin performance code.
(4) Design of CABRI-II test matrix and irradiation condition.

8. Relation to Other Projects or Codes :

N.A.

9. Reference Documents :

S. Nagai et al: Fuel Pin Performance and Reliability Analysis Code in PNC; Int. Conf. on Reliable Fuel for LMR, Sept. 7-11, 1986, Arizona
A. Boltax: Fuel Pin Behavior during Cyclic Testing, Int. Fast Ractor Safety Metting, 1990 Snowbird
M. Haessler: The CABRI-II Programme-Overview on Results, ibid.

10. Degree of Availability:

Available

PROJECT CODE 3.3	CLASSIFICATION Decay Heat Removal	
TITLE: Evaluation of Decay Heat Removal by Natural Convection		
COUNTRY: Japan		INITIATED: April, 1985
SPONSOR: S T A		COMPLETED: March, 1991
ORGANIZATION: P N C		LAST UPDATING: August, 1990
PROJECT LEADER: NAME Yoshimichi Daigo ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: In progress

1. General Aim :

To investigate a decay heat removal performance of LMFBRs by means of the natural circulation capability.

2. Particular Objectives :

(1) Confirm forced to natural circulation and decay heat removal capabilities of LMFBR system under the normal shutdown or accident conditions.
(2) Evaluate system performances of various types of decay heat removal systems.
(3) Confirm the integrity of the reactor core after reactor scram under accident conditions.

3. Experimental Facilities :

DELTA (Decay Heat Removal Test Apparatus using water as a working fluid) facility,
PLANDTL (Plant Dynamics Test Loop in Sodium) facility, Core-Component Sodium Test Loop facility.

4. Description of Computer Codes :

N.A.

5. Project Status: Progress to date

(1) DRACS and PRACS model tests using DELTA facility
(2) LOPI simulation tests using the PLANDTL facility
(3) Mixed convection bundle tests on the Core-Component Sodium Test Loop

6. Project Status: Essential results

(1) Thermohydraulics in the reactor vessel during forced to natural circulation transition have been investigated including such phenomena as thermal stratification, pressure drop, heat transfer, and formation of in-vessel flow recirculation path with possible flow reversal in the reflector and blanket regions of the core.
(2) Decay heat removal tests including the effects of natural circulation in the secondary heat transport system.
(3) Experiments simulating a wide spectrum of the LOPI conditions have been carried out and sodium boiling phenomena in the test sections have been investigated with single and parallel bundle arrangements.
(4) The subchannel analysis codes ASFRE and SABENA, and the plant system analysis code SSC have been validated through experimental analyses.

7. Next Steps:

(1) Reconstruct the PLANDTL facility for the tests on decay heat removal by natural circulation.
(2) Conduct intersubassembly heat fransfer experiments and investigate the flow redistribution phenomena in the reactor core.
Develop correlations for the flow reversal and recirculation inside a bundle.

8. Relation to Other Projects or Codes :

Assessment of Heat Transport System under Accident Conditions.

9. Reference Documents :

(1) K. Satoh et al.: Study of decay heat removal by in-vessel natural circulation - water test for the scaled model of a loop type reactor; PNC Report SN9410 87-012 (1987) (in Japanese)
K. Satoh et al.: Thermal hydraulic test on scaled model for pool type FBR decay heat removal - water test -; PNC Report SY9471 87-001 (1) (1987) (in Japanese)
(2) K. Yamaguchi et al.: Decay heat removal by natural circulation; PNC Technical Review, No.62, June 1987 (in Japanese)
(3) H. Ninokata et al.: Distributed resistance modeling of wire-wrapped rod bundles; Ncule. Eng. Des. 104, 93-102 (1987)

10. Degree of Availability:

Available except for the above document (1).

PROJECT CODE 3.3	CLASSIFICATION Decay Heat Removal

TITLE:
Assessment of Heat Transport System under Accidental Conditions

COUNTRY: Japan	INITIATED: April, 1986
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME Yoshimichi Daigo ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim :

To contribute to LMFBR safety analysis and evaluation by means of computer codes.

2. Particular Objectives :

(1) To develop and validate computer codes which simulate the thermal hydraulics under transient and accident conditions of LMFBRs.
(2) To give the best estimate simulation of physical process of transients and design basis accidents as well as the initial phase of severe accidents using the computer codes.
(3) To validated the computer codes using the selected experimental data. Criteria for assuring the reliability and identifying the limitation of the computer codes are to be established.

3. Experimental Facilities :

N.A.

4. Description of Computer Codes :

(1) Plant dynamics analysis: whole plant thermal hydraulic analysis codes for the loop-type LMFBR, SSC-L and the pool-type LMFBR, SSC-P.
(2) In-vessel thermal-hydraulics analysis: Multi-dimensional thermohydraulics analysis code, AQUA.
(3) Fuel assembly thermal-hydraulics analysis: finite element method code for the thermal hydraulic behavior in a fuel assembly, SPIRAL; single phase subchannel analysis code for an LMFBR fuel assembly, ASFRE; sodium boiling in a triangular rod array of an LMFBR fuel assembly, SABENA.

5. Project Status: Progress to date

Development of the computer codes have been completed. Code validation using the selected experimental data and the refinement of the physical models in these codes are underway.

6. Project Status: Essential results

(1) SSC-P and SSC-L: code validation using the LOPI simulation tests data obtained at the PLANDTL facility; accident analysis of the prototype LMFBR MONJU; two dimensional modeling of reactor upper plenum; heat loss modeling from the components in the heat transport system.
(2) AQUA: code validation using thermal-stratification test data in water and sodium; turbulent flow simulation model; higher-order differencing scheme and an Algebraic Stress turbulence Model (ASM).
(3) SPIRAL, ASFRE and SABENA: turbulent flow model for SPIRAL; vectorization to increase computational efficiency for ASFRE and SABENA; noncondensible gas model for SABENA; sensitivity analysis of the empirical correlations employed in SABENA.

7. Next Steps:

(1) The numerical models of the computer codes are to be refined and validated.
(2) A guideline is to be established to clarify to what extent the computer code should be validated and refined and what is the reliability of the codes. Furthermore, experiments are to be proposed, if necessary, to improve the accuracy of the computational results.

8. Relation to Other Projects or Codes :

(1) Evaluation of Decay Heat Removal by Natural Circulation
(2) Probabilistic Risk Assessment (PRA) Study of LMFBR
(3) Study on Loss-of Heart Removal System Accidents
(4) Phenomenological Study on Local Faults in Core

9. Reference Documents :

(1) A. Ymaguchi et al.: Improvement of Intermediate Heat Exchanger Model for SSC and Its Application to the Loss-of-Heat-Sink Accident Analysis in an LMFBR; Proc. of 1988 National Heat Transfer Conference, Houston, July 1988.
(2) I. Maekawa, et al.: Current Status on Thermal Stratification Study; Proc. of 4th Int. Conf. on Liquid Metal Engineering and Technology, Avignon, Oct. 1988.
(3) T. Shimizu, et al.: A Numerical Technique for the Prediction of the Fine Structure of Flow Temperature Fluid in Wire-Wrapped Fuel Pin Bundle Geometries-Development and Testing of the SPIRAL Code; Proc. of the 3rd Int. Symp. on Refined Flow Modeling and Turbulence Measurements, Tokyo, July (1988).
(4) H. Ninokata, et al.: Analysis of Low-Heat-Flux Sodium Boiling Test in a 37-Pin Bundle by the Tow-Fluid Model Computer Code SABENA, Nucl. Eng. Des. 97, 233-246 (1986).
(5) H. Ninokata, et al.: Numerical Simulation of Rod Bundle Sodium Boiling by SABENA; Proc. of 1988 National Heat Transfer Conf., Houston, July 1988.

10. Degree of Availability:

Not for publication except for the above documents.

PROJECT CODE 3.4	CLASSIFICATION Consequences of Sodium Leak	
TITLE: Mitigation of Sodium Leak and Fire Accidents		
COUNTRY: Japan		INITIATED: 1986
SPONSOR: S T A		COMPLETED: Mar. 1991
ORGANIZATION: P N C		LAST UPDATING: Aug. 1990
PROJECT LEADER: NAME Hiroshi Hara ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: in progress

1. General Aim:

To develop more reliable evaluation tools for the response of sodium fire mitigation systems in large fast breeder reactors

2. Particular Objectives:

To improve computer codes and to construct experimental database related to thermal mechanical and chemical consequences of sodium leak and fire accidents in reactor and auxiliary buildings of LMFBR

3. Experimental Facilities:

SAPFIRE facilities

4. Description of Computer Codes:

Computer language: Fortran IV and Fortran 77 , Computer used: FACOM and IBM
Computer capacity required: 1 MB
Description and application of codes:
(1) SOFIRE-M2 and SPM --- analysis of sodium pool fire
(2) SPRAY-3M --- analysis of spray fire
(3) ASSCOPS --- analysis of mixed fire
(4) ABC-INTG --- analysis of aerosol behavior

5. Project status: Progess to date

Conducting the following experiments:
1) Mitigation of energy release at the early stage of a sodium leak accident
2) Localization of sodium leak fire in building
3) Structural integrties of the building during accident
4) Version-up of computer codes

6. Project status: Essential results

1) Columner Combustion.
Six experiments were carried out under conditions of 500 deg-C in Sodium temp.,0.7-2.7kg/s in leak flow rate, 1-4 m in leak height and 5 min. leak duration. Empirical correlation was introduced for columnar leak combustion efficiency. Agreement between the experiments and the correlation is fairly good.

2) Low Temperature Pool Combustion
Rates of combustion and aerosol release have been measured under the conditions of ab. 200 deg-C sodium temperature and 3-21% oxygen concentration.

3) Water Release from Low-Temperature Concrete
Water release rate from concrete and physical properties of concrete below 100 deg-C were measured.

4) New 3-dimensional Sodium Fire Analysis Code,SOLFAS
Computer models on sodium combustion and related phenomena were developed and implemented to the code. Vectorization of the program was also performed to save CPU time.

7. Next Steps:

1) Columnar Combustion; to conduct experiments in order to make sure of the correlation applicable range

2) Water Rlease Code; to develop a computer code for heated concrete, and to validate it by the above experimental data

3) SOLFAS Code; to implement thermal radiation models to the code

8. Relation to Other Projects or Codes:

"Research on Aerosol Behavior"

9. Reference Documents :

S. Ohno: Test and Code Development for Evaluation of Sodium Fire Accidents in the FBR's, ANS 1990 International Fast Reactor Safety Meeting

10. Degree of Availability :

Not for publication except for the above document.

PROJECT CODE 3.5	CLASSIFICATION FP Release & Transport	
TITLE: Research on FP Release & Transport		
COUNTRY: Japan		INITIATED: Apr. 1986
SPONSOR: S T A		COMPLETED: Mar. 1991
ORGANIZATION: P N C		LAST UPDATING: Aug. 1990
PROJECT LEADER: NAME H. HARA and K. KINJO ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: in progress

1. General Aim:

To improve the evaluation accuracy of fission product (FP) release and transport taking account of FBR specific features in order to aid safety assessments of design basis accidents and site evaluation accidents.
To select appropriate requirements to the containment of large LMFBR.

2. Particular Objectives:

1) To quantify the retention effects to FPs by sodium.
2) To construct evaluation models concerning the following FP behavior:
   - FP release from a fuel element to sodium
   - Transport, deposition and retention of FP in sodium
   - FP release from sodium to cover gas or atmosphere
   - FP transport and removal in the atmosphere

3. Experimental Facilities:

SAPFIRE facility with three test rigs, SOLFA-1, SOLFA-2, FRAT-1
START

4. Description of Computer Codes:

Computer language: FORTRAN 77 Computer used: FACOM M200
Computer capacity required: 1 MB
Description of codes: FTAC — a computer code to describe the transport of gaseus and particulate FPs from a rising bubble to sodium pool
Field of application: HCDA bubble behavior analysis
Description of codes: SAFFIRE — a computer code to describe the transport, deposition and retention of FP in primary sodium coolant system

5. Project Status: Progress to date

1) Scoping calculations using CONTAIN have been performed on a sodium leak accidents in the preliminary loop of FBR demo-plant. Prior to the calculation, various mitigation effects were surveyed based on the domestic and foreign studies related to the FP transport from reactor core.
2) Review of papers on FP release from fuel elements.
3) Evaluation of test data on transport, deposition and retention of FPs in sodium.
4) Test on FP release from sodium to cover gas in containment atmosphere.
5) Test on FP transport and removal in atmosphere.

6. Project Status: Essential results

1) The calculation result shows that the mitigation effects to FP release could reduce the environmental source term by one to three orders.
2) Relation between fuel temperature and FP release rate from the fuel was rearranged based on test results.
3) Effects of sodium temperature, sodium flow rate and oxygen concentration in sodium on deposition velocity coefficients of FPs were made clear quantitatively from the results of sodium in-pile loop(FPL-II) tests.
   Deposition velocity coefficients of volatile FPs, decay chain model for short half-life FPs and release model for volatile FPs from sodium to cover gas were installed into the SAFFIRE code.
   Iodine solubility in liquid sodium was measured in the range of 200-800℃ of sodium temperature.
4) Gas-liquid equilibrium partition coefficient Kd and non-equilibrium partition coefficient Kd' of volatile FPs between liquid sodium and the gas phase were obtained in the range of 450-650 ℃ of sodium temperature.
5) A new analysis method using ion-chromatography was developed to measure chemical composition and its content ratio of gaseous iodine in sodium aerosol atmosphere.

7. Next Steps:

1) Measurement of FP release rate from fuel elements and its chemical and physical properties
2) Test on iodine transfer from rising non-condensable gas bubble into sodium pool
3) Sodium pool fire tests to obtain the partition coefficient of volatile FPs dissolved in the pool.
4) To examine whether gaseous iodine exists in sodium aerosol atmosphere.

8. Relation to Other Projects or Codes:

"Mitigation of sodium Leak and Fire Accidents"
"Research on Aerosol Behavior"
"Source Term and Hazard Analysis"

9. Refenence Documents:

1) K.Haga, et al.,"Experimental Study on Equilibrium Partition Coefficient of Volatile Fission Products between Liquid Sodium and the Gas Phase," International fast Reactor Safety Meeting, Snowbird, USA, Aug. 1990.

10. Degree of Availability

Not for publication except for the above document.

PROJECT CODE 4.1	CLASSIFICATION A T W S	
TITLE: Study on Anticipated-Transient-Without-Scram Accidents		
COUNTRY: Japan		INITIATED: April, 1976
SPONSOR: S T A		COMPLETED: March, 1991
ORGANIZATION: P N C		LAST UPDATING: August, 1990
PROJECT LEADER: NAME Hisashi Ninokata ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: In progress

1. General Aim :

To establish systematic, reliable and verified safety assessment tool for analysing the initiating and core-disruption phases of core-disruptive accidents (CDAs).

2. Particular Objectives:

1) To develop, improve and validate the computer codes for whole-core accident analyses. The SAS-series codes for the initiating phase and the SIMMER-series codes for the core-disruption phase of CDAs.
2) To assess spectra of in-vessel accident sequences and consequences using these codes, especially within a framework of probabilistic risk assessment.
3) To provide experimental database to improve the understanding on accident phenomenology and to validate the computer codes. This activity includes the participation in the joint CABRI experiments at CEA and VECTORS energetics-mitigation experiments.

3. Experimental Facilities :

None

4. Description of Computer Codes :

1) SAS3D and SAS4A : A multi-channel whole-core codes for the initiating phase of CDAs.
2) PAPAS-2S : A single-channel code similar to SAS-series but with a more detailed fuel-pin model.
3) SIMMER-II : A two-dimensional coupled neutronics and fluid-dynamics code for the core-disruption phase of CDAs.
4) SAME : An interface between SAS and SIMMER.
5) AFDM : A prototype and test code with an advanced fluid-dynamics model to improve SIMMER. Developed at LANL by an international team (US-Europe-Japan).
6) SIMMER-III : A next-generation SIMMER code being developed.

5. Project Status: Progress to date

1) Whole-core accident analyses have been made mainly using SAS3D and SIMMER-II within a framework of probabilistic risk assessment.
2) The integration and synthesis of the CABRI-I experiments were completed. achievement are in progress.
3) The code validation and improvement studies are in progress through various in-pile and out-of-pile experiments.
4) Improved models validated through the CABRI-I study have been implemented to the revised version of SAS3D. Model improvement of the SAS4A code has been initiated on the long-term materials relocation behavior.
5) The development of SAME-series codes is in progress to couple SIMMER-II with SAS3D and SAS4A.
6) A series of experiments in VECTORS was completed, and the analysis to study energy-mitigation effects of the above-core structures.
7) The AFDM code has been developed as a stand-alone fluid-dynamics code to study basic technologies to be used in future codes such as SIMMER-III

6. Project Status: Essential results

1) An integrated evalution method for in-vessel accident sequences has been established with improved knowledge of key phenomena. At the same time, future R&D area were more clarified in view of the effective risk reduction.
2) The uncertainties in key phenomena relevant to initiating-phase energetics have been significantly reduced. These phenomena include axial expansions, fuel-pin failure and post-failure fuel motion.
3) The development of SIMMER-III was started at a US-Japan joint study initially based on SIMMER-II and AFDM technologies and experiences.

7. Next Steps:

1) The joint CABRI-2 project to study initiating-phase phenomena with high burn-up fuels and to validate SAS and PAPAS codes.
2) Continued improvement and code-validation efforts for SAS-series, and SIMMER-II codes through analyses of various experiments such as CABRI.
3) The development of SIMMER-III will be focussed.
4) A new series of out-of-pile experiments is planned to study the boiling pool behavior in the transition phase.
5) Application of the improved code system to whole-core accident analyses of large LMFBRs.

8. Relation to Other Projects or Codes :

1) Safety Evaluation for the LMFBR Plant Using PRA Techniques
2) Study on Post-Accident-Heat-Removal Phase
3) Study on Source Term and Consequence Evaluation

9. Reference Documents:

1) N. Nonaka, et al.: Improvement of Evalution Method for Initiating-phase Energetics Base on CABRI-I In-Pile Experiments, International Fast Reactor Safety Meeting, Snow-bird, USA, Aug. 1990.
2) S. Kondo, et al.: Integrated Analysis of In-Vessel and Ex-Vessel Severe-Accident Sequences, International Fast Reactor Safety Meeting, Snow-bird, USA, Aug. 1990.

10. Degree of Availability:

Not for publication except for the above documents.

PRODUCT CODE 4.2	CLASSIFICATION L O H R S
TITLE: Study on Loss-of-Heat-Removal-System Accidents	
COUNTRY: Japan	INITIATED: April, 1982
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME Hisashi Ninokata ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 Telex PNDC J26462	STATUS: in progress

1. General Aim:

To establish systematic methods for study on loss-of-heat-removal-system (LOHRS) accidents in LMFBRs postulating reactor shutdown to clarify the inital plant responses and subsequent protected core meltdown (PMD) or recriticality phases.

2. Particular Objectives:

1) To develop, improve and validate the computer codes for analyzing each phase of LOHRS accident. SSC-L code for the initial whole-plant dynamics and APPLOHS-2 code for PMD phase.

2) To assess spectra of plant responses and in-vessel accident sequences/ consequences using these codes, especially within a framework of probabilistic risk assessment (PRA).

4. Description of Computer Codes:

1) SSC-L
Computer language:FORTRAN-77 Computer used:FACOM M780/VP100
Computer capacity required: 3 MB
Description of code: A whole-plant thermo-hydraulic transient analysis code for loop-type LMFBRs. Developed at BNL and modified by PNC for comprising specific heat removal systems in Japanese prototype LMFBR.
Field of application: Operational transients, protected and unprotected accidents including piping rupture.

2) APPLOHS-2
Computer language:FORTRAN-77 Computer used:FACOM M780/VP100
Computer capacity required: 2.2 MB
Description of code: A multi-channel whole-cord code for PMD phase analysis of LOHRS accidents in LMFBRs.
Field of application: Protected loss-of-heat-sink(PLOHS) accidents, protected loss-of-reactor-level (LORL) accidents including loss-of-

pipe-integrity (LOPI).

5. Project Status: Progress to date

1) Whole-plant responses analyses were performed usinfg SSC-L within a framework of PRA.

2) The code validation and impovement studies on SSC-L have been completed through the natural circulation test with JOYO reactor.

3) The development and improvement of APPLOHS-2 are in progress and analyses of PMD phase in LOHRS including recriticality and source term release is under way.

6. Project Status: Essential results

1) The SSC-L analyses clarify the safety characteristics of the plant and also time margin for recovery during accident evolution.

2) Principal models of APPLOHS-2/SIMMER-2 code system have been proved to be reasonable.
Understanding of basic phenomena and source term characteristics in PMD was improved through the analyses using the code system.

7. Next Steps:

1) Continued improvemet and code validation efforts for SSC-L and APPLOHS-2 codes through various experiments.

2) Continued whole-core accident analyses within a framework of probabilistic risk assessment.

8. Relation to Other Projects:

1) Safety evaluation for the LMFBR plant using PRA techniques

2) Study on post-accident-heat-removal phase

3) Study on source term and consequence evaluation

4) Study on natural circulation

9. Reference Documents:

1) A. Yamaguchi, et al.: Plant-wide Thermal Hydraulic Analysis of Natural Circulation test on Joyo with Mk-II Irradidtion Core, NURE 1H-4, Oct. 1989.

2) S. Kondo, et al.:Integrated Analysis of In-Vessel and Ex-Vessel Severe Accident Sequences, Int. Fast Reactor Safety Mtg., Snowbird, USA, Aug. 1990.

10. Degree of Availability:

Not available except for the above reports.

PROJECT CODE 4.3	CLASSIFICATION P A H R
TITLE: Study on Post-Accident-Heat-Removal Phase	
COUNTRY: Japan	INITIATED: April, 1980
SPONSOR: S T A	COMPLETED:
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME Hisashi Ninokata ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim :

To evaluate the coolability, material relocation and reconfiguration behaviors of the core debris after a core disruptive accident for judging the safety margins of LMFBRs.

2. Particular Objectives :

(1) To investigate the key phenomena during relocation process such as thermal interactions between a molten fuel jet released downward from the bottom of the core assembly and the coolant and/or structures.
(2)a) To evaluate the debris coling capability by analyzing the in-pile test data from the international Joint Debris Bed Program (JDBP).
b) To develop an integral evaluation code to assess the capability of post accident heat removal of reactors and to assess safety features on designing reactors.

3. Experimental Facilities :

(1)a) JET-I facility for the molten material jet-coolant/structure interaction experiments at low temperature (max. 500℃).
b) MELT-II facility for the molten material jet coolant/structure interaction experiments at high temperature (max. 2300℃).
(2) PAHR: None (PNC had participated in the JDBP conducted at SNL.)

4. Description of Computer Codes :

(2)a) PRELUDE: To calculate the core debris behavior, coolant flow in vessel and heat transport in primary loops during PAHR phase.
b) DEBRIS-MD: Multi-dimensional heat transfer code to calculate the core debris behavior.

5. Project Status: Progress to date

(1) A series of experiments were performed using combinations of a molten NaCl jet/tin plates, a molten tin jet/tin plates, and a molten stainless steel (S.S.) jet/S.S. plates to study erosion behavior of structures by a molten material jet in the MELT-II facility.
(2) The long-term coolability behavior of the damaged core is parametrically evaluated with the plant dynamics code.

6. Project Status: Essential results

(1) Erosion behavior is significantly affected by the crust and/or molten layer formation at the jet-structure interface, and the erosion rate of a plate was formulated based on the laminar and turbulent heat transfer models with consideration of crust formation and/or molten layer effect.
(2)a) From the JDBP in-pile tests, the relationship between the dryout power of the debris bed and the coolant subcooling is clarified.
b) A debris bed heat transfer model, DEBRIS-MD has been developed which is applicable to a complex cooling boundary condition of real cases.

7. Next Steps:

(1) A series of experiments at low and high temperatures will be continued to develop the theoretical models for analyzing the thermal interactions between a molten fuel jet and the coolant/structures.

8. Relation to Other Projects or Codes :

Codes : SAS-3D, 4A, SIMMER-II, SSC, Contain
Other projects: Study on Anticipated-Transient-Without Scram Accidents, Source Term and Hazard Analysis.

9. Reference Documents :

(1)a) A. Furutani et al.: Erosion Behavior of Solid Plate by a Liquid Jet - Effect of Molten Layer -: 26th National Heat Transfer Conf., Philadelphia, PA, USA, August 1989.
b) M. Saito et al.: Melting Attack of Solid Plates by a High Temperature Liquid Jet - Effect of Crust Formation -: Nucl. Eng. & Des. 121 (1990) 11-23.
(2) H. Nakamura et al.: A Method to Calculate Boiling and Dryout in Debris Bed using a Heat Conduction Code; Int. Topical Meeting on Fast Reactor Safety, Knoxville, TN. USA. 1985.

10. Degree of Availability:

Not for publication except for the above documents.

PROJECT CODE 4.4	CLASSIFICATION Source Term and Hazard Analysis	
TITLE: Source Term and Hazard Analysis		
COUNTRY: Japan		INITIATED: 1986
SPONSOR: S T A		COMPLETED: Mar. 1991
ORGANIZATION: P N C		LAST UPDATING: Aug. 1990
PROJECT LEADER: NAME Hiroshi Hara ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462		STATUS: in progress

1. General Aim:

To improve the methods of source term and off-site consequence evaluation with appropriate conservatism

2. Particular Objectives:

1) Development of integrated safety analysis method for containment system
2) Evaluation of thermal and chemical effects in containment
3) Performing more precise off-site consequence evaluation

3. Experinental Facilities:

SAPFIRE facility
START

4. Description of Computer:

Code name : CONTAIN/LMR
Computer language : FORTARN-77
Computer used : FACOM, IBM or CDC
Computer capacity required : 3000 KB (IBM)
Description of codes :
-developed by SNL
-150,000 lines
-the following phenomena can be treated.
heat and mass transport, chemical reactions, sodium fires, sodium/concrete reaction, aerosol behavior, hydrogen combustion, fission product release, transport and decay
Field of application : Containment analysis

5. Project Status: Progress to date

1) Enhancing and integrating the codes that evaluate an individual phenomena and to improve them based on experimental data
2) Best estimation of source term by taking account of mixed aerosols behavior and retention function of sodium pool for fission products, and heat absorption of building concrete for sodium combustion consequence by taking account of aerosol behavior that is characteristic of FBR

6. Project Status: Essentiel results

1) Development of integrated safety analysis method for containment system (CONTAIN/LMR version)
- A sodium-concrete reaction model(SLAM)has been incorporated to the code.
- A model for sodium-induced hydrogen burn has been developed.
- An upward sodium spray combustion model has been developed.
- A model for sodium vapor condensation to aerosol particles has been developed. Associated FP transfer and chemical reactions can be considered in this model.
- The sodium pool combustion model has been validated with experimental data.
- The aerosol behavior model has been validated with the OECD international benchmark calculation.
- A model for debris bed thermal behavior in an ex-vessel sodium-debris pool has been developed.
2) Thermal and chemical effects evaluation in containment system
- A series of sodium-concrete reaction test has been conducted to investigate FP release behavior during the reaction.
- Preliminary test series for hydrogen combustion in sodium contained atmospheric condition has been conducted.

7. Next Steps:

1) Develop a synthetic model for an ex-vessel sodium-debris pool to treat the sodium-concrete reaction and the debris-concrete interaction concurrently along with associated FP/aerosol release model.
2) Perform comprehensive experiments to investigate FP release behavior from an ex-vessel sodium-debris pool.
3) Investigate hydrogen burn phenomena in a sodium/aerosol contained atmosphere under the postulated FBR accident sequences.

8. Relation to Other Projects or Codes:

Codes: ABC-INTG, CRAC2

9. Reference Documents:

1) Kondo,S. et al., "Integrated Analysis of In-Vessel and Ex-Vessel Severe Accident Sequences", International Fast Reactor Safety Meeting,Snowbird,1990
2) Seino,H. et al.,"Validation of CONTAIN Code for Sodium Aerosol Behavior",ibid.
3) Miyahara,S. et al.,"Rlease Rate of Non-volatile Fission Products during Sodium-Concrete Reaction",ibid.

10. Degree of Availability

Not for publication except for the above documents.

Ⅱ. ATR

PROJECT CODE 1	CLASSIFICATION Normal Operation and Anticipated Operational Occurrrences
TITLE: Analysis and Evaluation of Operation Data of ATR Plant	
COUNTRY: Japan	INITIATED: April, 1989
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME Toshiyuki Furubayashi ADDRESS Fugen Nuclear Power Station 3 Myojin-cho, Tsuruga-shi, Fukui, 914 TELEPH. 0770-26-1221	STATUS: In progress

1. General Aim :

Analysis and evaluation of operation data of a ATR plant.

2. Particular Objectives :

(1) Collection and compilation of operation data of ATR Fugen.

(2) Analysis and evaluation of operation data of ATR Fugen.

3. Experimental Facilities :

Fugen Nuclear Power Station

4. Description of Computer Codes :

Computer language : PL/I
Computer used : FACOM M760/8
Computer capacity required: 2 MB
Field of application : Statistics and Analysis of operation data of the ATR

5. Project Status: Progress to date

(1) Failure data in ATR plant have been collected and compiled.

(2) The data have been analyzed by a statistical computer program.

6. Project Status: Essential results

Operation data have been collected and compiled over ten years in the Fugen. Only ten percent of the malfunctions effect on plant operation. The rest of the malfunctions are mainly failures of utility systems and ventilation systems.

Aging is the main failure cause. It occupies about 50% of the malfunctions. The others are maintenance problem, design problem, external environment and vibration.

7. Next Steps:

Continuous collection, Analysis and evaluation of operation data of a ATR plant.

8. Relation to Other Projects or Codes :

9. Reference Document:

None

10. Degree of Availability:

Not for publication

PROJECT CODE 1	CLASSIFICATION Normal Operation and Anticipated Operational Occurrences
TITLE: Irradiation Tests on Performance & the Behavior of MOX Fuel	
COUNTRY: Japan	INITIATED: 1983
SPONSOR: S T A	COMPLETED:
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME K. Kamimura ADDRESS TOKAI, NAKAGUN, IBARAKI, 319-11 TELEPH. 0292-82-1111 TELEXFAX 0292-87-0391	STATUS: in progress

1. General Aim :

   To comfirm the behaviour of MOX fuel under normal operation, load following and transient conditions.

2. Particular Objective:

   To study pellet-cladding interaction, fission gas release, fuel failure threshold power level and cladding behaviour at high burnup.

3. Experimental Facilities :

   Fugen and HBWR reactors.

5. Project Status: Progress to date

   Base irradiation of segment fuel has been continued in Fugen.

   Load following test of MOX fuel rods has been continued in HBWR.

6. Project Status: Essential results

   Fission gas release rate, irradiation growth and mechanical properties of fuel rod were measured.

   Rod diameter change, rod pressure, fuel center temperature and rod elongation during load following test was measured.

7. Next Steps:

Power ramp tests of irradiated segment fuel rods will be made.

PIE of load following test rods will be conducted.

9. Reference Document:

1) T.Ohtake,M.Kubo,et al.,"Load-Following Experiment for ATR Fuels", HPG Meeting,Stromstad,Sweden,Jun. 1984.

2) R.Yamanaka,T.Kajiyama,et al.,"Post Irradiation Examination of Fugen MOX Fuel",IAEA Specialists' Meeting on Post Irradiation Examination and Experience",Tokyo,Japan,Nov. 1984.

3) T.Abe,K.Kaneda,et al.,"Results from the Load Follow Experiment of MOX Fuel in IFA554/555",Halden Project Seminar on High Burnup Fuel Performance Topics,Fredrikstad,Norway,May 1987.

4) T.Abe,K.Kaneda,et al.,"Behaviour of MOX Fuel Rod During Daily Load Follow Operation",Proceeding of the Atomic Energy Society of Japan Conference in April 1988.

5) K.Tanaka,"PCMI behavior of MOX fuel at middle of fuel life in IFA-554/555",Enlarged Halden Programme Group Meeting on Fuel Performance Experiments and Analysis at BOLKESJφ,Norway 11-16th February,1990.

10. Degree of Availability:

Not ristricted.

PROJECT CODE 1	CLASSIFICATION Normal Operation and Anticipated Operational Occurrences
TITLE: Development of Fault Diagnosis System for ATR Plant	
COUNTRY: Japan	INITIATED: April, 1987
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME Kenshiu Watanabe ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim :

Prevention of escalation of anomalous states in a ATR plant.

2. Particular Objectives:

(1) Developments on an identification method of cause of small anomalous states.
(2) Developments on a computation method for fast-prediction of thermo-hydraulic behaviour during a small anomalous state following a trangient.

3. Experimental Facilities :

(1) Safety Experiment Facility at Oarai Engineering Center, PNC.
(2) FUGEN, the proto-type ATR, 165MWe.

4. Description of Computer Codes :

Computer language: FORTRAN-IV
Computer used: FACOM-M780
Computer capacity required: 0.3 M Word
Description of codes:
(a) Thermo-hydraulic analysis code.
(b) Autoregressive analysis code.
(c) Simulation code of plant operation.
Field of application:
(a) Estimation of thermo-hydraulic response of plant parameters following of a trangient in the ATR plant.
(b) Calculation of coherence function of two plant parameters.
(c) Fast-simulation of thermo-hydraulic nesponse of the ATR plant.

5. Project Status: Progress to date

(1) Development on the identification method:
A simulation data analysis way to make patterns used to an identification of a couse was changed from multivariate model to autoregressive one. Fuzzy technology was used for the couse identification.
(2) Development on the fast-prediction method:
Response of plant parameters not to be measured in a control room was estimated from coupling of measurable plant parameters.
(3) Development on the display method:
A data base of machanical and electrical components in a recirculation system was built to develop a display method based on artificial intelligence techniques.

6. Project Status: Essential results

(1) Development on the identification method:
A total of 75 coherency functions were made by use of the autoregressive analysis. The fuzzy technique could correctly discriminate a direct cause from five kinds of ones leading to a fall in water level in a steam drum.
(2) Development on the fast-prediction method:
About 700 plant parameters not to be measured in a control room were estimated from the coupling of 400 paramaters measured in the control room. These estimation did not affect the fast-prediction of the plant state, less than 1/3 in real time.
(3) Development on the display method:
The data base concerning a few hundreds of the components was made in a relational data base.

7. Next Steps:

(1) Development on the identification method:
The fuzzy identification method will be changed to one based on a neural networks technique to get fast computation time.
(2) Development on the fast-prediction method:
A mam-machine interface using a screen on a cathode ray tube will be made to get interactive design of operator-to-workstation interface.
(3) Development on a display method:
Artificial intelligence technique will be used to build an exepert system using the data base.

8. Relation to Other Projects or Codes :

None

9. Reference Document:

K. Watanabe; OECD NEA CSNI/CEC Specialist Meeting on Trend and Pattern Analyses of Operational Data from NPPs. (1989)

10. Degree of Availability:

Unrestricted distribution

PROJECT CODE 2	CLASSIFICATION Accident Conditions
TITLE : Assessment of Safety Margin for the ATR	
COUNTRY : Japan	INITIATED : April, 1988
SPONSOR : S T A	COMPLETED : March, 1991
ORGANIZATION : P N C	LAST UPDATING : August, 1990
PROJECT LEADER : NAME Yoshitaka Hayamizu ADDRESS PNC-OEC 4002, O-arai, Ibaraki-ken, 311-13 TELEPH 0292-67-4141 TELEX PNDC J26462	STATUS : in progress

1. General Aim

The safety margin evaluation for the FUGEN-type heavy water reactors(ATR).

2. Particular Objectives :

(1) Development of the best estimate codes to predict the operational transient and the LOCA in the ATR.
(2) Development of the probabilistic fuel temperature analysis code under LOCA conditions.
(3) Evaluation of the safety margin for the ATR.

3. Experimental Facilities :

Safety Experimental Facility at O-arai Engineering Center, PNC.
(1) Blowdown test facility,
(2) ECCS water injection test facility,
(3) Reflooding test facility.

4. Description of Computer Codes :

Computer language : FORTRAN-IV
Computer used : FACOM-VP100 and M780
Computer capacity reguired : 2.0MB
Description of codes : SALIAN, LAUNCH, HESTIA, FATRAC.

Field of application : LOCA analysis for the ATR

5. Project Status : progress to date

(1) A new reactivity model and concentric pipe rupture model were incorporated into the best estimate codes.
(2) The statistical method is applied to the fuel temperature analysis code to evaluate the safety margin during LOCA of ATRs.

6. Project Status : essential results

An analytical method is developed which is capable of analyzing deviation in cladding temperature behaior applying a statistics such as manufacturing allowance and error in themal-hydrraulic correlations by using the Monte-Carlo method. Using the code, cladding temperature deviation during a large break LOCA of ATR is assessed.

7. Next Steps :
Applying the statistical method to the best estimate LOCA code as well as the fuel temperature analysis code.

8. Relation to other Projects or Codes :
None

9. Reference Documents :
(1) H.Mochizuki and Y.Hayamizu, "Development of Thermal Hydraulic Models in Low Flow Rate", Preprint 1989 Annual Mtg. of AESJ, in Japanese. D15 (1990)
(2) Y.Hayamizu et al., "Development of Thermalhydraulis Analysis Models for ATR", Journal of the AESJ, 31, 12, (1989), in Japanese.
(3) H.Mochizuki and Y.Hayamizu, "Study of Safety Design Method of Fuel Clad Temperature by Statistical Approach", Preprint 1989 Fall Mtg. of AESJ, in Japanese, E28 (1989).
(4) H.Mochizuki et al., "Characteristic Test of ATR-Type Separator", Proceedings of 68th Annual Mtg. of JSME. 1807 (1990), in Japanese.
(5) H.Mochizuki et al., "Development of Separator Capability Analysis Code for ATR", Proceedings of 68th Annual Mtg. of JSME. 1808 (1990), in japanese.

10. Degree of Availability :

Unrestricted distribution

PROJECT CODE 3	CLASSIFICATION Severe Accidents
TITLE: Study of Severe Accident - Assessment of Core Degradation -	
COUNTRY: Japan	INITIATED: April, 1988
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME Yoshitaka Hayamizu ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim :

To analyze phenomena in core degradation process and degraded core cooling process during severe accidents, and to evaluate the safety margin.

2. Particular Objectives :

(1) Evaluation of core degradation of ATR
(2) Evaluation of Component integrity of primary loop
(3) Experiment of core cooling
(4) Evaluation of source term
(5) Study on man-machine interface

3. Experimental Facilities :

Heavy water cooling facility

4. Description of Computer Codes :

Computer language : FORTRAN 77
Computer used : FACOM-VP100 and M780
Computer capacity required: 2 MB for each
Description of codes : SALIAN, LAUNCH, HEATIA, ATRECS-II , MARCH-ATR
Field of applocation : Severe accident for the ATR

5. Project Status: Progress to date

(1) Evaluation of core degradation of ATR
- A survey of severe accident analysis code for LWRs
- Analysis of plant behaviors during severe accidents such as no water supply after LOCAs

(2) Evaluation of Component integrity of primary loop
- Ballooning experiment with pressure tube material
- Ballooning behavior analysis code development

(3) Experiment of core cooling
- Design of a test facility to measure the capability of core cooling, and construction
- Core cooling tests
- Measurement of contact conductance between a pressure tube and a calandria tube

6. Project Status: Essential results

(1) Evaluation of core degradation of ATR
- The survey of severe accident analysis codes for LWRs has been conducted.
- The March code was modified to be able to analyze the system of pressure tube type reactor.
- Plant behaviors during LOCA without any water supply were analyzed based on the characteristic tests regarding pressure tube ballooning, contact conductance and CHF of calandria tube.

(2) Experiment of core cooling
- Tests to measure the capability of core cooling with heavy water were constructed.
- Contract conductance between pressure tube and calandria tube was measured.

7. Next Steps:

(1) Preparation of reactivity analysis during coolant discharge into the moderator due to pressure tube rupture.
(2) Analysis related to the calandria integrity in coolant discharge and fuel/coolant reaction processes.
(3) Evaluation of contact conductance test

8. Relation to Other Projects or Codes :

None

9. Reference Documents :

(1) Y. Morishita et al,: "Analysis of Pressure Transient with Moving Boundary Caused by Tube Rupture under Thermal Non-equilibrium Conditions", 1st International Conf, Supercomputing in Nuclear Applications, JAERI in Japan, (1990)", Proceedings of the 11th Annual Conference of CNS, (1990).
(2) Y. Morishita et al,: "Development of an Analytical Method to Evaluate the Integrity of a Calandria tube in the case of Pressure Tube Rupture", Proceedings of the 11th Annual Conference of CNS, (1990).
(3) H. Mochizuki and Y. Hayamizu, "Core Coolability by Heavy Water Moderator in Pressure Tube Type Reactor", Preprint 1990 Fall Mtg. of AESJ, in Japanese, F36 (1990).

10. Degree of Availability:

Not for publication except for the above documents

PROJECT CODE 3	CLASSIFICATION Severe Accidents
TITLE: Study of Severe Accident - Assessment of Fuel Behavior -	
COUNTRY: Japan	INITIATED: April, 1988
SPONSOR: S T A	COMPLETED: March, 1991
ORGANIZATION: P N C	LAST UPDATING: August, 1990
PROJECT LEADER: NAME Yoshitaka Hayamizu ADDRESS PNC-OEC 4002 Narita, O-arai, Ibaraki, 311-13 TELEPH. 0292-67-4141 TELEX PNDC J26462	STATUS: In progress

1. General Aim :

To clarify the behavior of fuel degradation process using the results of inpile tests conducted in foreign countries, and to apply it to source term

2. Particular Objectives :

(1) Evaluation of fuel degradation of ATR

(2) Evaluation of FP behavior

3. Experimental Facilities :

None

4. Description of Computer Codes :

Computer language : FORTRAN 77
Computer used : FACOM-VP100 and M780
Computer capacity required: 2 MB for each
Description of codes : SALIAN, HEATIA, CONTAIN
Field of application : Severe accident for the ATR

5. Project Status: Progress to date

(1) Evaluation of fuel degradation of ATR
- Code development about pressure tube rupture analysis

(2) Evaluation of FP behavior
- Preparation of thermal hydraulic analysis in containment vessel of ATR

6. Project Status: Essential results

(1) Evaluation of fuel degradation of ATR
- Development of a structural analysis code of pressure tube during fuel melting was conducted.
- Number of degraded fuel in reactivity accident was evaluated.

(2) Evaluation of FP behavior
- Preparation of thermal hydraulic analysis in containment vessel of ATR was conducted.

7. Next Steps:

(1) To make a method to evaluated source term and FP behavior.

8. Relation to Other Projects or Codes :

CONTAIN code

9. Reference Documents :

(1) H. Mochizuki, "Survey of Source Term Evaluation Codes", PNCT N9420 90-003, (1990).

10. Degree of Availability:

Not for publication except for the above document