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Severe Accident Approach – Final Report Evaluation of Design Measures for Severe Accident Prevention and Consequence Mitigation

Nuclear Engineering Division

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Evaluation of Design Measures for Severe Accident Prevention and Consequence Mitigation

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CONTENTS

1. Introduction 1

2. Background 2
 2.1 The Role of CDA in Design and Licensing 2
 2.2 Reactivity Feedback in Fast Reactors and Inherent Safety Design 3
  2.2.1 Reactivity Feedbacks in an SFR in the Absence of Material Phase Changes 4
  2.2.2 Reactivity Feedbacks in an SFR if Material Phase Changes Occur 8
   2.2.2.1 Reactivity Feedback due to Sodium Boiling 9
   2.2.2.2 Reactivity Feedback due to Cladding Melting and Relocation 9
   2.2.2.3 Reactivity Feedback due to Fuel Melting and Relocation 9
 2.3 Potential CDA Initiators and Accident Paths 11
  2.3.1 Prevention of Severe Accidents 11
  2.3.2 Limitation of Core Damage 11
   2.3.2.1 Accident Paths That Can Lead to CDA and Early Fuel Relocation 13
   2.3.2.2 Whole Core Loss of Flow (LOF) 14
   2.3.2.3 Transient Over-Power (TOP) 15
   2.3.2.4 Fuel Failure Propagation 16
   2.3.2.5 Loss of Heat Sink 17
  2.3.3 Containment of Accidents Inside the Primary System 18
   2.3.3.1 Extended Fuel Motion: Freezing and Plugging 18
   2.3.3.2 Particulate Debris Bed Formation and Coolability 19
   2.3.3.3 Vessel Failure Mechanisms 20
  2.3.4 Radiological Consequences 22

3. Design Measures for Severe Accident Prevention and Mitigation 24
 3.1 Prevention of Irreversible Core Geometry or Structural Changes 24
  3.1.1 Self-Actuated Shutdown Devices 24
   3.1.1.1 SASS using Magnetic or Fuse Devices 24
   3.1.1.2 SASS using Hydraulically Supported Absorber Balls 28
  3.1.2 GEMS 32
  3.1.3 Natural Circulation Decay Heat Removal 33
   3.1.3.1 Direct Reactor Auxiliary Cooling System 34
   3.1.3.2 Reactor Vessel Decay Heat Removal System 37
  3.1.4 Core Restraint System Design for Negative Reactivity Feedback 39
  3.1.5 Control Rod Driveline Design for Negative Reactivity Feedback 40
  3.1.6 The Impact of Sodium ‘Void’ Worth 41
 3.2 Mitigation of Accident Consequences after the Onset of Irreversible Core Changes 45
  3.2.1 In-Pin Fuel Relocation Prior to Cladding Failure in Transients Without Scram 45
  3.2.2 Modified Fuel Assembly Design for Enhanced Fuel Relocation 48
  3.2.3 Core Catchers 51

3.2.3.1 In-Vessel Core Catcher 51
3.2.3.2 Ex-Vessel Core Catcher 53
3.2.4 Inerted Head Cavity Isolation Domes 54
3.2.5 Filtered/Vented Containment; Leak Detection and Fire Suppression Systems 54

4. Review of Approaches to SFR Severe Accidents 57
4.1 Review of Japanese Approach – JSFR 57
4.1.1 Safety Targets and Design Principles 57
4.1.2 The Role of Severe Accidents 60
4.1.2.1 Controlled Material Relocation Concept 61
4.1.3 Core Disruptive Accident Analysis 66
4.1.4 Summary 67
4.2 Review of the French (AREVA/CEA) SFR Safety Approach 67
4.2.1 Safety Approach 67
4.2.2 The Role of Severe Accidents 69
4.2.3 Core Disruptive Accident Analysis 69
4.2.4 Summary 70
4.3 Review of the US Approach – IFR 70
4.3.1 Safety Approach 70
4.3.2 The Role of Severe Accidents 71
4.3.3 Unprotected Accident Analysis 75
4.3.4 Summary 76

5. Evaluation of the Impact of Selected Design Measures on Fuel Melting Accident Prevention 77
5.1 Major Codes for Evaluation of CDA Energetics 78
5.1.1 The SAS4A Code System 78
5.1.1.1 The SAS4A Metal Fuel Version 80
5.1.2 The SIMMER-III Code 81
5.2 Evaluation of CDA Scenarios in Japan 83
5.2.1 Initiating Phase Analysis 84
5.2.1.1 Initiating Phase Model Enhancements 84
5.2.1.2 Initiating Phase Analysis Results 85
5.2.2 Transition Phase Analysis 86
5.2.2.1 Transition Phase Model Enhancements 87
5.2.2.2 Transition from SAS4A IP Analysis to SIMMER-III TP Analysis 87
5.2.2.3 Transition Phase Analysis Results 88
5.2.3 Integrated LOF Analysis Summary 89
5.3 Evaluation of CDA Scenarios in France 90
5.4 Evaluation of Severe Accident Scenarios in US 93

6. Conclusions 96

References 98

FIGURES

Figure 1 Simplified Path Structure of CDA Initiators and the Lines of Assurance 13
Figure 2 Conceptual Design of a Ferromagnetic Curie-Point-Operated Device.......................27
Figure 3 Hydraulically Supported Absorber Ball Assembly in a SFR......................................30
Figure 4 Absorber Ball Assembly Conceptual Design 31
Figure 5 EBR-II Primary System 35
Figure 6 ABTR Pool-Type Heat Transport System Configuration...36
Figure 7 JSFR Loop-Type Heat Transport System Configuration..37
Figure 8 Schematic Overview of the NSTF 38
Figure 9 Cavity Formation during the Initial Accident Phase...47
Figure 10 Molten Fuel Relocation Modes...47
Figure 11 Post-Accident Material Relocation with CMR...49
Figure 12 Schematic View of JSFR with Multi-Layered Debris Tray..52
Figure 13 Core Catcher with Neutron Absorbers..53
Figure 14 Safety Assurance Strategy for JSFR...58
Figure 15 JSFR Plant View 59
Figure 16 Schematic View of the Modified JSFR Heat Transport System 59
Figure 17 Schematic View of the Original JSFR Heat Transport System 60
Figure 18 Self-Actuated Shutdown System 62
Figure 19 Concept Fuel Assemblies for Enhanced Molten Fuel Escape from the Core.............62
Figure 20 FAIDUS - Fuel Assembly Concepts...63
Figure 21 Modified-FAIDUS Fuel Assembly Design 64
Figure 22 The Role of Early Fuel Discharge during the Transition Phase 65
Figure 23 JSFR Reactor Vessel and Internal Structures 66
Figure 24 Schematic View of the European Sodium Fast Reactor 68
Figure 25 PRISM Reactor Module..73
Figure 26 Figure 26 Schematic Representation of the RVACS..74
Figure 27 GEM Assembly Locations in the PRISM Core 74
Figure 28 Ultimate Shutdown System Assembly..75
Figure 29 Typical LEVITATE Configuration..79
Figure 30 PINACLE Configuration 80
Figure 31 Overall Framework of the SIMMER-III Code 82
Figure 32 Multiphase, Multi-Component Fluid Dynamics Model in SIMMER-III 82
Figure 33 Total IP Reactivity and Power History for the Nominal LOF Case 86
Figure 34 Reactivity Components during the IP Nominal LOF case..86
Figure 35 Power and Reactivity Histories for the TP Nominal Case..88
Figure 36 Geometry of Simmer-III Analysis of CRGT Fuel Removal.......................................89
Figure 37 Phoenix Core Configuration and Location of Postulated Blockage 91
Figure 38 SIMMER-III Geometry of Assemblies Considered in the TIB Analysis....................92
Figure 39 Power Generated in the Blocked Assembly during the Postulated TIB Accident.......92
Figure 40 Reactor Power and Peak Coolant Temperature Change Histories for ULOF 93
Figure 41 Power and Reactivity Histories for a Rapid Unprotected LOF 95

TABLES

Table 1 Primary Design Characteristics of SASS..26
Table 2 Oxide and Metallic Fuel Characteristics..71

ACRONYMS

ABLE - Axial Blanket Elimination
ABTR – Advanced Burner Test Reactor
ACS – Auxiliary Cooling System
ALMR - Advanced Liquid Metal Reactor
ALARP - As Low as Reasonably Practicable
ATWS - Anticipated Transient without Scram
BDBE - Beyond Design Basis Events
CDA - Core Disruptive Accidents
CDF - Core Damage Frequency
CMR - Controlled Material Relocation
COTS - Commercial off the Shelf
CRBR - Clinch River Breeder Reactor
CRDL - Control Rod Drive-Line
CRGT - Control Rod Guide Tube
CRSS - Control Rod Stop System
CSC - Corium Spreading and Coolability
DBA - Design Basis Accidents
DBE - Design Basis Events
DEC - Design Extension Conditions
DFBR - Demonstration Fast Breeder Reactor
DHR - Decay Heat Removal
DHRS - Decay Heat Removal System
DRACS - Direct Reactor Auxiliary Cooling System
EAGLE - Experimental Acquisition of Generalized Logic to Eliminate Recriticalities
EFR - European Fast Reactor
FA - Fuel Assembly
FR - Fast Reactor
FACT - Fast Reactor Cycle Technology
FAIDUS - Fuel Assembly with Inner Duct Structure
FCI - Fuel Coolant Interaction
FFTF - Fast Flux Test Facility
FS - Feasibility Study
FSA - Fuel Subassembly
GEM - Gas Expansion Modules
HCDA - Hypothetical Core Disruptive Accidents
IAEA - International Atomic Energy Agency
IFR - Integral Fast Reactor
IHX - Intermediate Heat Exchanger
INPRO - International Project on Innovative Nuclear Reactors and Fuel Cycles
IP - Initiating Phase
IRACS - Intermediate Reactor Auxiliary Cooling System
ISS - Inherent Shutdown System
IVR - In-Vessel Retention

JAEA - Japan Atomic Energy Agency
JAPC - Japan Atomic Power Company
JSFR - Japanese Sodium Fast Reactor
LMR - Liquid Metal Reactor
LOF - Loss of Flow
LOHS - Loss of Heat Sink
LOPI - Loss of Pipe Integrity
LORL - Loss of Reactor Level
LWR - Light Water Reactor
MOX - Mixed Oxide
NDHX – Natural Draft Heat Exchanger
NPR - New Production Reactor
NRC - Nuclear Regulatory Commission
NSSS - Nuclear Steam Supply System
NSTF - Natural Convection Shutdown Heat Removal Test Facility
PAHR - Post-Accident Heat Removal
PAMR - Post-Accident Material Relocation
PLOHS - Protected Loss of Heat Sink
PPS - Plant Protection System
PRACS - Primary Reactor Auxiliary Cooling System
PSA - Probabilistic Safety Assessment
RSS - Reactivity Shutdown System
RV - Reactor Vessel
RVACS - Reactor Vessel Auxiliary Cooling System
RVDHRS - Reactor Vessel Decay Heat Removal System
SA - Subassembly
SASS - Self-Actuated Shutdown System
SFR - Sodium Fast Reactor
SHTS - Sodium Heat Transport System
SMFR - Small Modular Fast Reactor
SG - Steam Generator
TIB - Total Instantaneous Blockage
TOP - Transient Over-Power
TP - Transition Phase
ULOF - Unprotected Loss of Flow
ULOHS - Unprotected Loss of Heat Sink
USS - Ultimate Shutdown System
UTOP - Unprotected Transient Over-Power

SUMMARY

An important goal of the US DOE reactor development program is to conceptualize advanced safety design features for a demonstration Sodium Fast Reactor (SFR). The treatment of severe accidents is one of the key safety issues in the design approach for advanced SFR systems. It is necessary to develop an in-depth understanding of the risk of severe accidents for the SFR so that appropriate risk management measures can be implemented early in the design process.

This report presents the results of a review of the SFR features and phenomena that directly influence the sequence of events during a postulated severe accident. The report identifies the safety features used or proposed for various SFR designs in the US and worldwide for the prevention and/or mitigation of Core Disruptive Accidents (CDA). The report provides an overview of the current SFR safety approaches and the role of severe accidents. Mutual understanding of these design features and safety approaches is necessary for future collaborations between the US and its international partners as part of the GEN IV program.

The report also reviews the basis for an integrated safety approach to severe accidents for the SFR that reflects the safety design knowledge gained in the US during the Advanced Liquid Metal Reactor (ALMR) and Integral Fast Reactor (IFR) programs. This approach relies on inherent reactor and plant safety performance characteristics to provide additional safety margins. The goal of this approach is to prevent development of severe accident conditions, even in the event of initiators with safety system failures previously recognized to lead directly to reactor damage.

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By

A. M. Tentner

1. INTRODUCTION

An important goal of the US DOE Gen IV program is to develop a demonstration Sodium Fast Reactor (SFR). The treatment of severe accidents is one of the key issues of R&D plans for the advanced reactor systems in general, and for SFR in particular. Despite the lack of an unambiguous definition of the safety approach applicable for severe accidents, there is an emerging consensus on the need for their consideration in the design. In the aftermath of the Three-Mile-Island accident, it was the judgement of the NRC staff in 1985 that extremely unlikely severe accidents constituted the major risk to the public associated with potential radioactive releases from nuclear power plants. The U.S. Nuclear Regulatory Commission's Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants [1-1] states that the Commission expects the new plants to achieve a higher standard of severe accident safety performance than prior designs. Thus it is important to develop an in-depth understanding of the risk of severe accidents for the SFR proposed so that appropriate risk management can be undertaken early in the design process.

The evaluation of severe accidents has played an important role in the safety analysis of SFRs. The U.S. SFR program actively studied the potential scenarios and consequences of Hypothetical Core Disruptive Accidents (HCDA) for SFRs with oxide fuel during the Clinch River Breeder Reactor (CRBR) program in the 70s and 80s. These studies included both experimental studies and the development and validation of mechanistic computer codes that can calculate in detail the sequence of events that determines the outcome of a postulated accident initiator. Later, the focus of the US SFR safety R&D activities shifted to the prevention of severe accident consequences through passive safety features of SFRs utilizing metal fuel.

A significant amount of experience in the design and safety analysis of SFRs using oxide fuel has been developed in both Japan and France during the last few decades. Extensive experience in the design and safety analysis of SFRs using oxide and metal fuel has been also accumulated in the US. A key objective of the metal-fueled SFR development program has been to design reactors that can inherently avoid damage during postulated unprotected accidents such as inadvertent reactivity insertion or loss of coolant flow without reactor scram.

The goal of this report is to provide a review of the SFR features and phenomena that influence directly the sequence of events during a postulated severe accident, to review the safety features used or proposed for various SFR designs in the US and worldwide for the prevention and/or mitigation of CDAs, and provide an overview of the current SFR safety approach and the role of severe accidents in Japan, France, and the US.

2. BACKGROUND

2.1 The Role of CDA in Design and Licensing

The role of the CDA in SFR safety evaluation has not always been well defined and the suggested CDA role in the licensing process has ranged from: a) very little consideration of the CDA consequences based on low probability of occurrence, supported by demonstrated adequate reliability of plant components, shutdown systems, and heat-removal systems to b) treatment of CDA as design-basis accidents (DBA) in the classic sense.

In light of the extensive safety evaluation of SFRs conducted in the US, Japan, France, and other countries over more than 30 years, the above two approaches to licensing have been recognized to represent extreme solutions, and a more balanced approach, outlined in Ref. [2-1], appears to have been accepted by the designers and regulators. This approach includes the following elements:

1) The probability of the initiators leading to potential core meltdown should be made sufficiently small ($<10^{-6}$ per reactor-year) so that CDAs do not have to be included in the conservative safety approach taken in the licensing process for DBAs

2) A further reduction in risk can be accomplished by performing best-estimate analyses that demonstrate the low probability ($<10^{-2}$ per reactor-year) of a postulated core disruptive accident leading to substantial energetics that would challenge the integrity of the primary system. Thus the combined probability of such energetic events would be $<10^{-8}$ per reactor-year.

3) The design should be driven by functional requirements based on mechanistic analyses such that any weak links identified should be upgraded to provide an overall optimum system in terms of potential energy release and absorption of this energy, rather than using arbitrary CDA energetics.

4) The objective should be to demonstrate, on the basis of best-estimate analysis, the long term capability to contain the fuel debris following a postulated core-meltdown accident. As noted in [2-1], there is substantial evidence that, even in the absence an engineered fuel retention system of any kind, the inherent response of the reactor structures and mitigating features, including containment-atmosphere venting and cleanup systems, would keep the radioactive release within tolerable limits.

The U.S. Nuclear Regulatory Commission's Policy Statement on Severe Accidents [2-11] states that the Commission expects new plants to achieve a higher standard of severe accident safety performance than prior designs. The focus on severe accident issues in the Policy Statement was prompted by the NRC staff's judgment that accidents of this class, which are beyond the substantial coverage of design basis events, constitute the major risk to the public associated with nuclear power plant accidents. A fundamental objective of the Commission's severe accident policy is that the Commission intends to take all reasonable steps to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur. The Commission recognizes the need for striking a balance between accident prevention and consequence mitigation. It is clear that core-melt accident evaluations and containment failure evaluations should continue to be performed for all future plant designs. The Commission also recognizes the importance of such potential contributors to severe accident risk as human performance and sabotage. The issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent applicable, will be emphasized as special considerations in the design and in the operating procedures developed for new plants. A balanced focus will be paid to the negative impact of human performance on severe accident risk as well as its potentially positive contribution to halting or limiting the consequences of severe accident progression. Design features should be emphasized that reduce the risk of early containment failure, thus providing more time for the positive contributions of operator performance in curtailing severe accident consequences. Also, design features should be given special attention that serve to decrease the role of human error in the sequence of events leading to the initiation or aggravation of core degradation.

The US NRC perspective on the research needed for advanced reactor licensing is outlined in [2-12]. The SFRs are fundamentally different from LWRs, and existing regulatory tools (codes, data) are not directly applicable to SFR designs. The development of new tools needed for SFR evaluation is constrained by reduced research budgets. The regulators must decide: a) what are the key safety and risk issues for the design, b) how to assure that all issues have been identified, and c) which issues require additional experimental data. The applicant testing programs may not answer all the questions, and remaining questions usually include scaling questions, inability to simulate all the components, and beyond design basis performance. In the past the regulators have conducted independent research to address these questions, and this practice is expected to continue in the SFR licensing. The regulators will not have the same amount of data for the evaluation of the SFR as is available for LWRs, and it is unlikely that they will have the opportunity to develop their own analysis tools in time to support the applications. As a result, the regulators will depend on applicants as in the past, and conservatism will be needed on issues where uncertainties exist. The NRC supports an international cooperative research as the only practical way to proceed in identifying the needed capabilities and tools [2-12].

2.2 Reactivity Feedback in Fast Reactors and Inherent Safety Design

Fast neutron reactors are sensitive to changes in operating conditions which affect the reactivity of the reactor and therefore the power, including both temperature and geometry. The response of the reactor to such changes in operating conditions, especially in off-normal or accident conditions, determines the level of risk that the reactor would represent. For example, in the case of the failure of one or more safety-grade systems combined with the failure to scram the reactor, the inherent change in the reactivity of the reactor and the magnitude and timing of the resulting changes in net core reactivity will determine whether or not reactor power and thus reactor temperatures will be kept within acceptable limits. It is essential to understand the basis for the changes in reactor power in response to multiple reactivity feedbacks, as this can be used to guide development of fast reactors that would have superior safety characteristics and lower levels of risk. In the discussion below we divide the reactivity feedbacks into two broad categories:

a) Reactivity feedbacks due to changes in the temperature of the core materials which lead to reversible changes in density and neutronic properties, but do not result in phase changes, e.g. sodium boiling or fuel melting;

b) Reactivity feedbacks due to large changes in the temperature of the core materials, which result in phase change of one or more of the core materials and irreversible changes in the core geometry.

2.2.1 Reactivity Feedbacks in an SFR in the Absence of Material Phase Changes

The change in reactivity from an equilibrium state for a sodium-cooled fast reactor is determined by the following balance of core reactivity, showing all of the major reactivity feedback mechanisms in the absence of material phase changes:

$$
\begin{aligned}
\Delta\rho(t) = \quad & \alpha_{Doppler} \{ \ln [T_{F,avg}(t) / T_{F,avg}(0)] \} / \beta_{eff} \\
& + \alpha_{Na} [T_{Na,avg}(t) - T_{Na,avg}(0)] \\
& + \alpha_{Axial\ Exp.} [T_{Clad,avg}(t) - T_{Clad,avg}(0)] \\
& + \alpha_{Radial\ Exp.} \{ [T_{in}(t) - T_{in}(0)] \\
& \qquad + XMC/XAC [(T_{out}(t) - T_{out}(0)) - (T_{in}(t) - T_{in}(0))] \} \\
& + \alpha_{Control\ Rod} \{ aL_{CRD} [T_{CRD}(t) - T_{CRD}(0)] \\
& \qquad - bL_{VHP} [T_{VHP}(t) - T_{VHP}(0)] - bL_{VCP} [T_{VCP}(t) - T_{VCP}(0)] \} \\
& + \Delta\rho_{Ext.} \qquad (1)
\end{aligned}
$$

where:

$\Delta\rho(t)$	= change in reactor core reactivity
$\alpha_{Doppler}$	= Doppler coefficient, T dκ/dT
$T_{F,avg}$	= average fuel temperature in the core, K
β_{eff}	= beta-effective (delayed neutron fraction)
α_{Na}	= sodium density coefficient, $/K
$T_{Na,avg}$	= average sodium coolant temperature in the core, K
$\alpha_{Axial\ Exp.}$	= fuel axial expansion coefficient, $/K
$T_{Clad,avg}$	= average cladding temperature in the core, K
$\alpha_{Radial\ Exp.}$	= core radial expansion coefficient, $/K
T_{in}	= core inlet coolant temperature, K
XMC/XAC	= grid plate to core midplane distance / grid plate to above-core load plane distance, m
T_{out}	= core outlet coolant temperature, K
$\alpha_{Control\ Rod}$	= control rod driveline expansion coefficient, $/m
a	= thermal expansion coefficient of the control rod driveline, 1/K
L_{CRD}	= length of control rod driveline in contact with the hot pool, m
T_{CRD}	= control rod driveline temperature, K
B	= thermal expansion coefficient of the reactor vessel wall, 1/K
L_{VHP}	= length of reactor vessel wall in contact with the hot pool, m
T_{VHP}	= reactor vessel wall temperature in the hot pool region, K
L_{VCP}	= length of reactor vessel wall in contact with the cold pool, m
T_{VCP}	= reactor vessel wall temperature in the cold pool region, K
$\Delta\rho_{Ext.}$	= externally applied means to change reactivity, $

Notes:
1. Fuel axial expansion or contraction is assumed to be controlled by the expansion or contraction of the cladding.

2. Fuel assemblies are assumed to remain straight with radial core expansion (no bending or bowing).

It is important to note that while this expression is useful to show the grouping of the reactivity feedback coefficients and their sensitivities to various operating conditions, it should be cautioned that this equation only applies at an equilibrium state, and does not provide any information about the transient response of the reactor during upset conditions and cannot provide any insight as to the expected maximum power and temperatures that may occur. That information can only be obtained from detailed analyses considering the heat transfer, coolant flow, structural temperatures, and by using time dependent kinetics calculations to determine the transient power level.

As the equation shows, starting from an equilibrium state, the reactivity change of the reactor is determined by a large number of factors (coefficients), almost all of which are driven by temperatures in the reactor core, either directly or indirectly such as through thermal expansion effects. The sign of each coefficient depends on whether the feedback tends to increase reactivity with an increase in temperature, power, etc., in which case the sign is positive, or if the feedback tends to decrease reactivity, in which case the sign is negative.

To make the dependence of reactivity on core operating conditions clearer, it is useful to rewrite the core reactivity balance equation as follows [2-13], where it is assumed that the changes in power and reactivity occur simultaneously:

$$\boldsymbol{\Delta\rho(t) = \quad [P(t)\text{-}1]\, A + [P(t)/F(t)\text{-}1]\, B + \delta Tin(t)\, C + \Delta\rho_{Ext.}} \qquad \textbf{(2)}$$

where

$\Delta\rho(t)$ = change in reactivity

$P(t)$ = normalized reactor power

$F(t)$ = normalized reactor flow

$A = \alpha_{Doppler}\Delta T_F(0)$

$B = [\alpha_{Doppler} + \alpha_{Na} + \alpha_{Axial\ Exp.} + 2\,(XMC/XAC)\,\alpha_{Radial\ Exp.}$
$\quad + 2\,\alpha_{Control\ Rod}\,(\,aL_{CRD} - bL_{VHP}\,)\,]\,\Delta T_C(0)/2$

$C = \alpha_{Doppler} + \alpha_{Na} + \alpha_{Axial\ Exp.} + \alpha_{Radial\ Exp.}$
$\quad + \alpha_{Control\ Rod}\,(aL_{CRD} - bL_{VHP} - bL_{VCP})$

$\Delta T_F(0) = T_{F,avg}(0) - T_{Na,avg}(0)$

$\Delta T_C(0) = T_{out}(0) - T_{in}(0)$

$\delta Tin(t) = T_{in}(t) - T_{in}(0)$

$\Delta\rho_{Ext.}$ = externally applied means to change reactivity, \$

Notes:

1. This expression shows that the change in reactivity is determined by the changes in power, power-to-flow ratio, and the core inlet temperature, along with any externally applied reactivity. This partially explains why the standard severe accident initiators for a sodium-cooled fast reactor are the unprotected (unscrammed) loss-of-flow (ULOF), loss-of-heat-sink (ULOHS), and transient overpower (UTOP, control rod withdrawal).

2. This expression also shows why it is the groups of reactivity feedback coefficients (A, B, and C) that are important for determining transient response, not any individual component.

By introducing the normalized core power and core flow, i.e., power and flow at any time as compared to the power and flow at normal steady-state operating conditions, it can be seen that the change in core reactivity is dependent on the change in only four operating conditions:

1. reactor power
2. the ratio of reactor power to the coolant flow through the reactor core
3. the coolant temperature entering the core
4. any change in reactivity caused by external means

This observation has resulted in the establishment of three general classes of reactor accidents for fast reactors:

1. loss of forced coolant flow, i.e., change in the ratio of reactor power and coolant flow
2. loss of normal heat removal paths, i.e., change in the coolant inlet temperature
3. inadvertent movement of reactor control rods, i.e., external means to change reactivity

Reactor power only changes as a result of the changes in core reactivity, and as such is not an independent variable in this expression. For off-normal conditions or accidents where the reactor protection systems are activated and successfully scram the reactor in response to the developing changes in core operating conditions such as a loss of forced coolant flow, the change in core reactivity is determined by the reduction in core reactivity caused by the insertion of the control rods. However, for lower-probability accidents where the reactor protection systems fail to activate, the change in core reactivity and thus core power is determined by the reactivity feedback generated as a result of the changing core temperatures, as shown in equation 2. It should be recognized that these changes in reactivity will arise only from mechanisms that do not require any active system to function, i.e., they are the result of physical phenomena that occur as an inherent property of the materials involved. This becomes the basis for the concept of "inherent safety" for sodium-cooled fast reactors, a concept that was developed and demonstrated as part of the DOE ALMR program in the 1980s and early 1990s that has the potential to reduce the risk of severe accidents in fast reactors.

A very important observation from equation 2 is that most of the reactivity feedback components do not function independently but in groups, as for factors 'B' and 'C'. The Doppler feedback is the only feedback component that contributes to a change in core reactivity as a result of a change in reactor power only (factor 'A'), but it is important to remember that reactor power at any given time is not an independent variable in this equation, being a function of the temporal change in reactivity prior to that time.

There is a large group of reactivity feedback components that contributes to the change in core reactivity as a result of a change in the power-to-flow ratio in the reactor, factor 'B', including Doppler, density of the sodium coolant, axial expansion of the fuel, radial expansion of the core, and axial expansion of the control rod drive mechanisms. It is important to realize that the category of "power-to-flow" ratio includes cases such as an increase in power without changing the flow, a decrease in flow without changing the power, etc. Note that a case where power changes but flow also changes in proportion would produce no reactivity feedback from this grouping of reactivity feedback components. Such a case would cause fuel temperature to rise, but that is accounted for by the first term in equation 2. Note that if the assumption of having axial fuel expansion controlled by the cladding temperature is removed, then any increase in power would also cause the fuel to expand, and there would be an additional term in the expression for factor ‘A’. A similar group is also responsible for changing core reactivity in response to a change in the core coolant inlet temperature, factor ‘C’.

Doppler

The Doppler reactivity effect is primarily due to the broadening of the fuel neutron absorption resonances in response to an increase in the fuel temperature, which in turn leads a reduction in the neutron flux and thus a negative reactivity feedback. During a postulated severe accident an increase in reactor power will cause an increase in the fuel temperature which, through the Doppler negative reactivity feedback will tend to reduce the core reactivity and limit the power increase.

Sodium Density

A decrease in the sodium density will decrease the number of scattering collisions the fast neutrons undergo in the core and thus increase the fast neutron flux and the core reactivity. However, a decrease in the sodium density can also allow more neutrons to leave the core, leading to decrease of the core reactivity. The net reactivity effect of a sodium density decrease depends on the core dimensions and the core location where the density change occurs. In the central regions of the core the net reactivity effect of a sodium density decrease is to increase the core reactivity, while near the core periphery the net reactivity effect of a sodium density decrease can become negative. In large SFR cores the net reactivity feedback due to sodium density decrease remains positive over much of the core. The sodium density decreases in response to increases in the sodium temperature during postulated severe accidents. As long as the sodium remains in liquid phase the density changes are relatively small and the associated positive reactivity feedback is easily compensated by the other negative reactivity feedbacks discussed in this section. Sodium boiling, if it occurs, can lead to larger and more rapid density changes which are discussed in Section 2.2.2 below.

Axial Fuel Expansion

The axial expansion of the fuel pins leads to a reduction in the fuel density in the active core region, and thus has a negative reactivity effect. In general, the fuel pin axial expansion can be caused by an increase in the fuel and cladding temperatures. As both these temperatures increase during a postulated severe accident, the axial fuel expansion provides an inherent safety mechanism. The magnitude of the negative reactivity feedback due to axial fuel expansion depends on the interaction between the fuel and the cladding. While for oxide fuel this interaction is mainly caused by the contact force between the fuel and cladding, in the case of metallic fuel the fuel can interact chemically with the cladding to form a metallurgical ‘bond’. Both of these effects become more pronounced at higher burnup and may require a minimum burnup to be present. If there is little interaction between the fuel and the cladding, the fuel can expand freely in the axial direction and the axial fuel expansion is determined by the fuel temperatures. If the fuel has expanded radially however, as occurs with increased irradiation, the fuel interaction with the cladding can be significant, and the axial fuel expansion is determined by balancing the axial forces that act on fuel and cladding. The time response of the axial fuel expansion reactivity feedback during a transient also depends on the conductivity of the fuel. Because the metal fuel has a higher conductivity than the oxide fuel, the negative axial expansion reactivity feedback will tend to respond faster during a transient in a metal fueled core than in an oxide fueled core.

Radial Core Expansion

The reactivity feedback due to the radial expansion of the reactor core is typically one of the largest reactivity feedback components due to the sensitivity of fast reactor cores to geometry changes caused by changes in the core material temperatures. This reactivity feedback can be either positive or negative, depending on design details for horizontal positioning the reactor core assemblies. To explain the details of this reactivity feedback mechanism, it is essential to examine the core assembly structure and the means provided in the design to ensure appropriate core geometry during operation. The impact of the core restraint system design on the radial core expansion reactivity feedback is discussed further in Section 3.1.4.

Control Rod Mechanism Expansion

The reactivity feedback due to the control rod mechanism axial expansion is another reactivity feedback component caused by the core structure geometry changes in response to material temperature changes. Because the control rod mechanism is washed by the coolant that leaves the core, an increase in the temperature of the sodium leaving the core will cause the control rod mechanism to expand and insert the rod further into the core, providing a negative reactivity feedback. However, the magnitude of this feedback also depends on the way the control rod drives are supported in the reactor vessel. The impact of the control rod mechanism design on the control rod expansion reactivity feedback is discussed further in Section 3.1.5. There is also a relation between the control rod mechanism expansion and the axial fuel expansion discussed above. As the fuel expands upward, this effectively causes an insertion of the control rod, and this effect can dominate the base axial fuel expansion coefficient if the control rod worth is large.

2.2.2 Reactivity Feedbacks in an SFR if Material Phase Changes Occur

This section discusses the reactivity changes due to a phase change in one or more of the materials present in an SFR core. These events are associated with very low probability events, which would require the simultaneous occurrence of several adverse conditions. The materials present in an SFR core include liquid sodium, fuel, and cladding. Under normal operating conditions and all design basis accident conditions the sodium is in liquid state, while the fuel and cladding are in solid state. The most important reactivity feedbacks due to material phase changes are those associated with the sodium boiling and fuel melting and relocation.

2.2.2.1 Reactivity Feedback due to Sodium Boiling

An important reactivity feedback mechanism is the reactivity change that would be introduced as a result of coolant boiling, or ‘voiding’, where the sodium coolant temperature becomes high enough to boil and the liquid sodium is displaced by sodium vapor. This reactivity feedback should only be encountered as a result of extremely severe accident conditions, if ever. Such accidents would have a very low probability of occurrence. Given the compact nature of a fast reactor core, the effect of the presence of sodium on the neutron flux, and the much lower density of sodium vapor as compared to liquid sodium, the reactivity effect of replacing the liquid sodium with sodium vapor can be very large, on the order of 6-8$ of reactivity for the entire reactor core. The onset of sodium boiling, if it occurs, will tend to increase the core reactivity and power until the occurrence of fuel melting and dispersal, discussed below, which tends to reverse this trend by introducing a substantial amount of negative reactivity. Because various fuel assemblies are exposed to different power and flow conditions, the timing of boiling onset, if it occurs, will be different in various assemblies. In more heterogeneous cores the time delay between boiling initiation in various power-to-flow ratio assemblies will be higher, allowing more time for the negative fuel relocation reactivity in high power-to flow assemblies to reduce the reactivity and power before boiling can start in the lower power-to-flow fuel assemblies. It is important to remember that even under such conditions, the other reactivity feedback mechanisms are still functioning, and play a role in determining the response of reactor to the transient conditions. Sodium boiling leads to reduced heat transfer from the fuel pins to the coolant, and the reactivity and power increase associated with sodium boiling usually would lead to cladding and fuel melting and relocation, resulting in irreversible core geometry changes. A more detailed examination of the sodium void worth impact on SFR safety and performance is presented Section 3.1.6.

2.2.2.2 Reactivity Feedback due to Cladding Melting and Relocation

The decrease of the amount of cladding material in the central core regions due to cladding melting and relocation will lead to an increase in the fast neutron flux and a positive reactivity feedback. However, the reactivity effect due to cladding relocation is substantially smaller than the reactivity feedback due to fuel melting and relocation. As the cladding relocation tends to occur after the onset of the fuel relocation, especially in metal fuel cores, the cladding relocation feedback is generally dominated by the negative fuel reactivity feedback discussed below. If the cladding melting and relocation is initiated shortly before the fuel pin failure and fuel relocation, as could be the case in some scenarios in oxide fuel cores, the positive reactivity feedback due to cladding relocation will combine with the sodium voiding feedback and all the other negative feedbacks described in section 2.2.1 to determine the core conditions at the time of fuel pin failure.

2.2.2.3 Reactivity Feedback due to Fuel Melting and Relocation

The melting and relocation of the fuel has a complex effect on the core reactivity, which has been studied extensively in the US and abroad by research organizations involved in the SFR safety analysis. The fuel is the source of fast neutrons needed to maintain the chain reaction, and the removal of fuel from the central core region will have a strong negative reactivity effect, while moving fuel from the core periphery towards the core center will introduce positive reactivity. The complexity of the fuel relocation feedback is due to the fact that the molten fuel can relocate both inside the fuel pin cladding and outside, in the coolant channel. The timing of these two relocation phenomena, and the location of the cladding failure which couples them by providing the path for the in-pin fuel to enter the coolant channel, combine under different accident scenarios to lead to a complex fuel relocation reactivity feedback described below.

As the power increases during a postulated severe accident, the inside of the fuel pin begins to melt leading to the formation of an internal cavity. This cavity is filled with a mixture of molten fuel and fission gas, and expands continuously, both radially and axially, due to fuel melting. While the formation of the molten fuel cavity can occur in both metal and oxide fuel pins, the location and fuel melting progression are different due to the thermo-physical properties of these fuels. In metal fuel pins, with a higher thermal conductivity, the axial temperature profile peaks near the top of the active core, and the molten cavity tends to develop near the top of the pin. Thus in metal fuel cores it is likely that the molten cavity will reach the top of the fuel column prior to the occurrence of cladding failure. This can also occur in oxide cores, where the molten fuel cavity tends to develop closer to the core mid-plane, when subjected to slow ramp TOPs. If no blanket pellets are present, the pressurized molten fuel in the cavity can relocate rapidly to the lower pressure upper plenum, introducing a substantial amount of negative reactivity and causing an associated power decrease. Thus, if cladding failure occurs after the initiation of the in-pin molten fuel relocation, it is likely to happen at lower reactivity and power levels, an important safety advantage during the early stages of molten fuel relocation. If, on the other hand, the cladding failure occurs prior to the onset of in-pin fuel relocation, it will lead to a rapid cavity depressurization and thus prevent a later fuel ejection in the space above the fuel column. In both cases, the initial cladding failure location plays an important role in determining the reactivity feedback due to the early post-failure fuel relocation. After the occurrence of cladding failure the fuel reactivity feedback is the net result of the in-pin and ex-pin fuel relocation events. The early post-failure fuel relocation is dominated by the rapid acceleration of the in-pin molten fuel towards the failure location. If the failure location is near the core mid-plane, the in-pin molten fuel is relocated towards the higher reactivity region, and can lead to a temporary net positive fuel reactivity feedback. If the initial cladding failure is located further above the core midplane the initial in-pin fuel motion tend to move at least some of the fuel towards regions of lower reactivity and the net fuel reactivity feedback due to the early fuel relocation can become negative. A significant safety advantage of the metal fuel cores is that the physical properties of the metal fuel, in particular the high thermal conductivity, lead to an initial cladding failure located above the core midplane for a wide range of postulated severe accident situations. The molten fuel ejected into the coolant channel is generally relocated towards the lower reactivity regions of the core by the pressure gradient and thus provides a negative reactivity feedback. Shortly after the cladding failure, once enough molten fuel has been ejected into the coolant channel and has been accelerated towards the core periphery, the negative reactivity feedback due to the coolant channel fuel dispersal begin to dominate, and the net fuel relocation reactivity feedback becomes strongly negative.

2.3 Potential CDA initiators and Accident Paths

To ensure reactor safety, the design of SFRs in the US has relied on a defense-in-depth approach, including: 1) reliability of normal operations, 2) protective features that limit the consequences of potential malfunctions, and 3) additional margins for protection against unforeseen and unexpected circumstances. For severe accidents this approach has led to four lines of assurance against the consequences of a reactor malfunction:

1. Prevention of accidents
2. Limitation of core damage
3. Containment of accidents inside the primary system
4. Attenuation of radiological products release

2.3.1 Prevention of Severe Accidents

The prevention of accidents that can lead to fuel melting can be achieved to a large extent through high reliability and quality assurance for components, plant-protection and shut-down heat-removal systems, and incorporation of design features that promote safety. The SFR has unique advantages in terms of inherent safety, such as the liquid metal cooling for natural circulation heat removal, and in the case of metal fueled reactors an inherent negative reactivity feedback due to the pre-failure in-pin fuel relocation. Inherent safety features can also be added through a judicious design which promotes negative reactivity feedbacks during abnormal situations. Such inherent safety features which take advantage of the core geometry changes due to abnormal temperature changes are described in Section 3.1 of this report. This approach was shown to lead to SFR designs that prevent core damage in double fault accident sequences that could lead to CDAs in previous SFR designs (e.g. CRBR, MONJU, SPHENIX), thus reducing the probability of a severe accident that can have a significant impact on the public to a very low level. Although some still believe that the first line of assurance can be made sufficiently secure so that no further consideration of CDAs is necessary, the general consensus today is that a more balanced approach is needed for SFRs which also includes consequence limiting features in addition to accident prevention features. The role of the consequence-limiting features is to protect the public against the effect of an accident beyond the capacity of the normal protective systems, as discussed below in Section 2.2.2. It is noted that some of the severe accident analyses discussed in this report neglect the role of designed inherent safety mechanisms such as negative reactivity feedbacks due to the core restraint system design or the control rod driveline design.

2.3.2 Limitation of Core Damage

It has long been recognized in the study of postulated Core Disruptive Accidents (CDA) that an important characteristic of the Sodium cooled Fast Reactors (SFR) is that the intact SFR core is not in its most reactive configuration. This causes the SFRs to react differently from the Light Water Reactors (LWR) to the relocation of core materials or dimensional changes that may occur during a postulated CDA. While it is theoretically possible that postulated core geometry changes due to material relocation could lead to prompt-critical reactivity excursions, as first discussed by Bethe and Tait as early as 1956 [2-2], numerous physical processes come into play that limit the material relocation that could lead to such postulated geometries and thus provide inherent safety barriers that can prevent or mitigate energetic reactivity excursions. No attempt was made in the Bethe and Tait analysis to establish physically possible initial conditions: the analysis assumed a completely molten core in its original geometry slumping under the acceleration of gravity. The recognition that the arbitrary assumption of coherent core collapse used in Bethe and Tait analysis leads to results that are much too conservative has led to the development of a mechanistic approach to the analysis of CDAs. Instead of using postulated core geometry changes to evaluate the accident outcome, the mechanistic approach postulates an initiating event and attempts to analyze mechanistically the subsequent accident sequence in order to determine the physically possible core material relocation and geometry changes. The mechanistic approach relies on the use of complex computer codes that describe the physical phenomena relevant for CDAs. These codes must be validated through analyses of separate-effects experiments as well as integrated experiments, in order to ensure that the mechanistic analysis yields a conservative description of the CDA sequence of events. The results of the mechanistic analysis are used to determine the time-dependent energy release during the accident, and are used to evaluate the post-accident fuel coolability and radiological consequences. A simplified accident path structure due to potential CDA initiators is illustrated in Figure 1 [2-1].

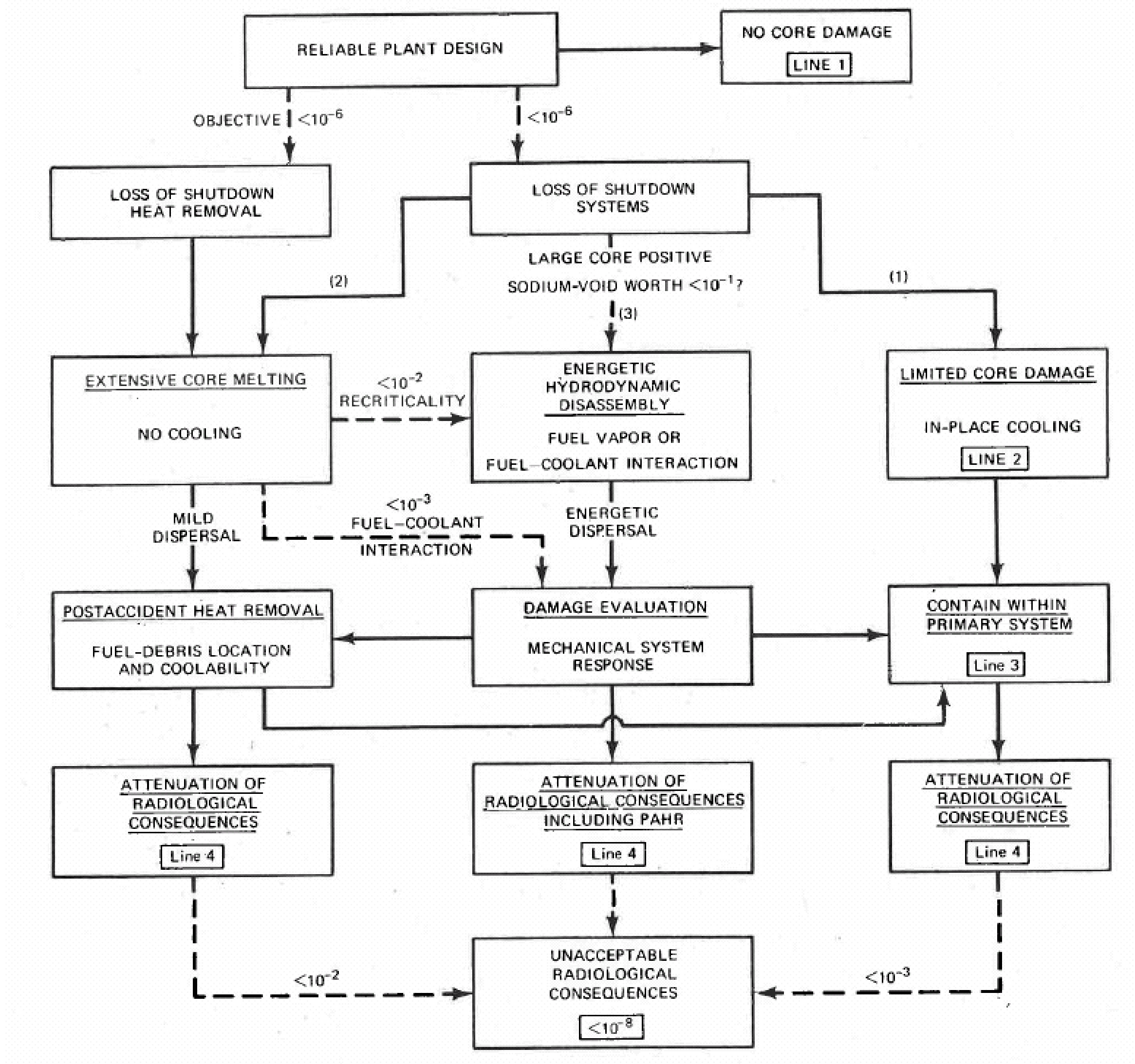

Figure 1 Simplified Path Structure of CDA Initiators and the Lines of Assurance [2-1]

2.3.2.1 Accident Paths That Can Lead to CDA and Early Fuel Relocation

Classes of possible initial conditions that can lead to fuel melting and relocation have been identified as follows:

1) Events leading to a reactivity insertion at a rate high enough so that the reactor plant protection system would be unable to respond efficiently. These events include:

a) Gas-bubble intake
b) Failure of core support structure
c) Failure of core restraint system

2) Malfunctions within the design basis of the reactor plant protection system combined with the failure of the plant protection system. These malfunctions are usually divided into:

a) Loss of flow
b) Transient overpower
c) Fuel-failure propagation

3) Malfunctions leading to the interruption of heat removal capability even after shutdown, such as:

a) Severe pipe break
b) Loss of heat sink.

It is usually accepted [2-1] that the first class of initiating events can be effectively precluded by design, and therefore only the second and third classes are discussed below.

2.3.2.2 Loss of Flow (LOF)

A postulated LOF accident with failure to scram in an oxide-fueled SFR will lead to coolant boiling and associated core voiding. The detailed voiding pattern and subsequent accident sequence are largely dependent on the reactor size and design. For small oxide-fueled reactors such as the Fast Flux Test Facility (FFTF) and intermediate size reactors such as CRBR the reactivity effects associated with coolant voiding are not sufficient to result in prompt-burst conditions at the time of fuel pin failure [2-3, 2-4]. Since little cladding melting and relocation is expected, the post-pin-failure power changes are determined by the reactivity changes due to fuel relocation. Because the cladding failures in a LOF tend to occur near the core center in high void worth oxide-fueled reactors, the early fuel relocation tends to introduce positive reactivity due to the in-pin molten fuel relocation toward the failure location. This early effect is followed by a significant negative reactivity insertion due to the fuel dispersal driven by fission gas and sodium vapor pressures. In a small oxide-fueled reactor where the pin failure occurs at reactivity levels well below prompt critical the post-failure reactivity remains below prompt critical during the early fuel relocation events and then decreases rapidly due to axial fuel dispersal, leading to a gradual core meltdown. In the case of the CRBR, however, the magnitude of the positive sodium-void worth associated with the fissile fuel region is such that the fuel pin failure can occur at reactivity levels closer to prompt critical, and subsequent early fuel relocation effects can lead to a temporary prompt critical condition and associated power increase. Extensive mechanistic analyses performed with the SAS4A code have shown that the duration and power level of such events is limited by the axial fuel dispersal in the coolant channels which rapidly introduces large amounts of negative reactivity. The potential problems associated with positive sodium-void worth are enhanced in a large reactor where the maximum positive sodium-void reactivity may be 8-10 dollars if other negative reactivity mechanisms are not present. In this case the fuel pin failure can occur at a high power level with the reactor near prompt critical due to sodium voiding alone, with a substantial fraction of the core still not voided. Following pin failures in voided and non-voided channels, the reactivity addition due the early fuel relocation and accelerated sodium voiding can lead to prompt critical conditions and an associated rapid power increase. This power increase is limited and then reversed by the negative reactivity introduced by the axial fuel dispersal in the coolant channels. Freezing of the molten fuel and cladding mixture can occur as it reaches the colder sections of the fuel assembly and, depending on the geometry of the fuel assembly and the thermo-physical characteristics of the fuel and cladding, can lead to channel plugging. The coolant channel plugging can prevent the fuel escape from the core and potentially lead to the subassembly wall melting, which marks the beginning of the transition phase. The selection of subassembly designs that ensure that the axial escape path for the molten fuel is maintained during the initiating phase of the accident can avoid the occurrence of the transition phase and provide an accident mitigation approach that is discussed later in this report in Section 3.2.

In metal-fueled reactors the above LOF scenario is influenced by the thermo-physical properties of the metal fuel. The lower melting point and high thermal conductivity of the metal fuel favors the formation of molten fuel region that extends to the top of the fuel pin and allows the onset of in-pin fuel relocation prior to the cladding failure. This early negative reactivity insertion plays an important role in reducing the reactivity and power at the time of cladding failure, as shown in the LOF analyses described in [2-5] which studied a rapid flow coast-down that would occur in the case of all pump seizure. For reactors with moderate to high void worth reactivity the relatively coherent voiding of the core can drive the reactivity near prompt critical, but Doppler and the fuel axial expansion feedbacks will prevent the reactivity from reaching prompt critical. The high overpower that develops causes the fuel melting and the onset of rapid in-pin fuel relocation prior to cladding melting. The negative reactivity feedback associated with the pre-failure in-pin fuel relocation will reduce the reactivity below critical, bringing the power back to near nominal or at most a few times nominal levels. Because of the pre-failure in-pin fuel motion, the short-lived positive reactivity feedback that occurs during the early phase of post-failure fuel relocation before the fuel dispersal becomes dominant will not drive the reactor prompt critical. In addition, because the power level is relatively low when cladding failure occurs, there will be more temporal incoherence of the failures than might otherwise be the case.

It is noted that a combination of reactor design features and inherent negative reactivity feedbacks affect the reactivity and power levels at the time of fuel pin failure during a postulated LOF with failure to scram and thus the expected energetics of the LOF initiating phase. These design features which include limited sodium-void reactivity, radial core expansion, control rod driveline expansion, etc, and their specific effect of the accident sequence are discussed later in this report.

2.3.2.3 Transient Over-Power (TOP)

Analyses of low-ramp-rate TOP transients considered for FFTF and CRBR [2-6] show that the accident terminates due to the upward transport of fuel in the coolant channel following the occurrence of pin failure in limited number of subassemblies. The analyses were performed for reactivity insertion rates up to 0.1 \$/s, which is four times the design maximum control rod withdrawal rate. The results indicated cladding failures occurring near the top of the core and dispersal of the fuel ejected from the fuel pin by the flowing sodium. Due to the failure location, both the in-pin and ex-pin fuel motion introduce negative reactivity, and the net fuel reactivity contribution is negative immediately after the cladding failure, leading to neutronic shutdown. Once failure occurs in one or more assemblies, flow diversion from the disrupted channels and the decrease in reactivity combine to cause a decrease of the fuel temperature in the intact subassemblies. In this scenario, the intact subassemblies can be cooled in place. At higher reactivity insertion rates the fuel pin failure location would tend to move downwards toward the core centerline, causing the early fuel relocation to introduce less negative reactivity or even to have a net positive reactivity contribution, leading to a temporary power increase followed by fuel dispersal in the coolant channel and a rapid reactivity and power decrease. If, however, fuel plugging and blockages should develop in conjunction with fuel sweep-out under certain conditions, this could lead to flow starvation and large scale meltdown similar to the postulated LOF accident (path 2 in Fig. 1).

In metal fueled reactors the in-pin fuel relocation occurs before the cladding failure and introduces enough negative reactivity to prevent the initial positive reactivity due to early post-failure fuel relocation from driving the reactor prompt critical. TOP analyses performed with SAS4A show that even at the extreme reactivity insertion rate of 10 $/s the accident consequences are relatively benign. Although in this extreme case there is significant core damage, the energy release is limited and there is no immediate threat to the reactor vessel.

2.3.2.4 Fuel Failure Propagation

Extensive work has been conducted over several decades to study the local faults within a fuel assembly and their potential for propagating throughout the assembly. This research has not revealed any sequence of events which could result in subassembly disruption prior to detection and shutdown of the reactor. This conclusion applies to oxide as well as metal fueled reactors [2-7, 2-8].

For oxide fuel, if there is a failure of the cladding, the oxide fuel can then contact the liquid sodium coolant. Oxide fuel chemically interacts with sodium, forming degradation products. Experiments using fuel pins designed to have a cladding failure have shown that these degradation products form on the outside of the fuel pin around the failure location. While in principle, it would be possible for the deposits to grow to a size that would start to significantly impede coolant flow, which could lead to overheating of the fuel pins in the vicinity of the initial failure, experiments have also shown that the deposits grow slowly, providing ample time for detection of the fuel pin failure by the delayed neutron detectors in the primary system. As a result, reactor shutdown and removal of the assembly with the failed fuel pin can be easily accommodated, limiting the damage to the initial fuel pin failure. For metal fuel, the sequence of events is different. Since the metal fuel does not chemically interact with the liquid sodium, the only consequence of a cladding failure is that some of the fission products are able to escape into the primary sodium coolant. Experiments using fuel pins designed to have a cladding failure have shown that this condition can be tolerated indefinitely without causing any further consequences other than activation of the sodium coolant. There is ample time to accomplish reactor shutdown and removal of the affected fuel assembly in this case as well.

In order to examine the potential for whole core severe accidents it has become customary to consider the consequences associated with a postulated total instantaneous inlet blockage of one fuel assembly at power as is believed to have occurred for at least one assembly in Fermi-1. In this case, if the reactor is at operating power, the complete loss of coolant flow in that assembly would result in failure of the fuel pins and fuel melting.

For an oxide-fueled reactor the inlet blockage quickly leads to sodium boiling, followed by dry-out of the fuel pin. Cladding melting follows in a few seconds and the cladding relocation is influence by sodium vapor streaming, with both steel draining and flooding being possible. The penetration of the molten cladding material into the lower as well as upper blanket regions represents a source for potential blockage formation and complete upper and lower blockages may occur several seconds after the initiation of clad relocation. Fuel melting occurs a few seconds following clad relocation and the early fuel motion may be downward as well as upward, depending on the amount of sodium streaming and extent of cladding blockage. In time, a boiling fuel-steel mixture largely confined to the active fuel zone will evolve, with a fuel crust layer protecting the hexcan walls. If the fuel-steel mixture remains confined to the active fuel zone by the axial blockages the fuel crust on the hexcan walls will eventually re-melt, allowing melt-through of the assembly wall and further propagation to the neighboring assemblies.

In a metal-fueled reactor sodium boiling and pin dry-out is followed by fuel melting and pin failure, while the cladding is still intact. This is likely to be the case since the rate of eutectic formation upon exceeding the eutectic temperature for the metal fuel-clad alloy is relatively slow compared to the time scale for fuel melting. In contrast to the oxide fuel case, development of blockages that could prevent fuel dispersal is not expected for the metallic fuel case. The cladding is still intact at the time the fuel dispersal begins and the absence of the upper blanket region would allow the fuel dispersal directly into the upper fission gas plenum region, which has a low heat capacity and thus a limited capability to lead to fuel freezing. Because the time for fuel ejection into the upper plenum is short compared with the time for eutectic penetration of the hexcan wall the propagation to the adjacent hexcans can be ruled out.

However, in the aftermath of the Fermi-1 accident, fast reactor assemblies are designed with multiple inlets to prevent such an occurrence. Combined with online monitoring and cleaning of the sodium coolant to avoid any buildup of impurities, these measures have made the inlet blockage event not credible to occur, as there is no mechanistic sequence that would result in a total blockage of the assembly inlet. If one considers that a partial inlet blockage may be possible, this occurrence would be detected as it was in Fermi-1, either due to increased coolant temperature exiting this assembly if thermocouples are present, or due to the delayed neutron signal in the event that the resulting fuel overheating eventually causes one or more pins to fail. In this case there would be ample time for reactor shutdown and removal of the affected assembly, since with multiple inlets, the extent of the blockage would only cause a small reduction in coolant flow, depending on the number of inlets used.

2.3.2.5 Loss of Heat Sink

In contrast to the TOP and LOF accidents where fuel melting and relocation can begin within tens of seconds following the accident initiation if the accident initiators are rapid enough, the sequence of events in a degraded decay heat removal transient develops over a period extending from several hours to several days. If the auxiliary decay heat removal system functions as designed, fuel melting will not occur. However, if the reactor vessel decay heat removal system function is also substantially degraded, gradual melting of the core will eventually occur after boil-off of the sodium. Early power excursions are precluded in this case since the reactor is highly subcritical, and the possibility of energetic secondary re-criticality events caused by fuel relocation and compaction are the main concern.

2.3.3 Containment of Accidents inside the Primary System

2.3.3.1 Extended Fuel Motion: Freezing and Plugging

The axial fuel relocation in the coolant channels, away from the core region, during a severe accident is the main negative reactivity contributor that reduces the reactivity and power levels. The extended fuel relocation from the core region to the lower and upper plenum is strongly influence by freezing of the molten cladding and fuel that can lead to plugging of the coolant channels and obstruct the axial fuel relocation. Upon cladding failure in an unprotected LOF transient, for example, molten fuel released near the core mid-plane will disperse bi-axially through the coolant channels toward the core ends. The blockage formation, if it occurs, delays the axial fuel dispersal, allowing more time for the molten fuel to heat and ablate the hexcan walls, which eventually can be breached, allowing the fuel to enter the inter-assembly space and eventually some of the neighboring subassemblies. The blockage formation during the extended fuel relocation is dependent on both the fuel type and the core design, especially the length and diameter of the coolant channels above and below the core and the heat capacity of the materials present in these regions.

For oxide fuel, the coolant channels are likely to be ‘voided’, i.e., there is no liquid sodium in the core, but sodium vapor would be streaming upwards in the coolant channels. The oxide fuel, with a melting temperature approximately 1500 C above that of steel, typically has been found to melt and entrain a layer of cladding as the front passes, forming a slug of molten cladding traveling ahead of the molten fuel. Fuel pin cladding within the core region may be hot enough to preclude re-freezing, but in the cooler regions above and below the core re-freezing of the molten fuel-cladding mixture can occur. Plugging then can occur by re-freezing this molten cladding to form a blockage farther along the channel, upward or downwards. This would delay further fuel motion until the blockages had been remelted.

In contrast, the metallic fuel has a melting temperature below that of steel. In addition, as metal fuel is overheated it tends to form an alloy with the steel cladding that has an even lower melting point, below the boiling point of the sodium coolant. The process of alloy formation begins while the fuel is still inside the pin cladding and may continue at a reduced rate while the molten fuel is moving in the coolant channel. At the time of fuel pin failure, the coolant channel may still contain liquid sodium, depending on the specific conditions of the accident initiator. The fuel relocation in a metal-fueled reactor usually begins before cladding melting, and the fuel is not expected to become hot enough to melt and entrain any cladding material near the flow front. The alloying interaction process between the fuel and steel continues at a reduced rate as the molten fuel/steel alloy that leaves the fuel pin moves along the coolant channels. At the low temperatures near the flow front, the contact temperatures also would be below rapid eutectic dissolution temperatures. The blockage formation in this case, if it occurs, would be caused by the re-freezing of the fuel/steel mixture and will be determined by the temperature of the structures outside of the core region. Due to the lower melting temperature of the molten fuel/steel alloy the blockage formation is less likely to occur in the metal-fuel case, and complete blockage formation may be avoided altogether, depending on the length and diameter of the coolant channels above and below the core and the heat capacity of the materials present in these regions, as described below.

For upward fuel motion, the upper part of the assemblies will also be at an elevated temperature, favoring extended fuel relocation and limited freezing. Initial TOP experiments in the TREAT reactor with fuel pins that had an upper plenum length of approximately 25 cm, indicate that the temperatures above the core are high enough to preclude large blockage formation, and that fuel motion would continue up and out of the assembly, effectively removing fuel from the core, especially if there is fission gas present in the fuel/steel alloy. For fuel pins with a considerably longer gas plenum, larger blockages could form in the upper region of the assembly, delaying further fuel motion.

For downward fuel motion, it is more likely that there will be blockage formation in the region below the core due to the much lower temperatures in that region, which would delay the axial fuel dispersal in that direction. However, analyses of an Advanced Liquid Metal Reactor (ALMR) design with a metel-fueled core concluded that the complete fuel penetration of a shorter lower-shielding region, with a length of approximately 50 cm, is highly likely if the coolant channel geometry below the core is based on the larger diameter typical for IFR metal-fuel designs. These results indicate that in a metal-fueled SFR a shorter lower-shield region with larger diameter coolant channels could provide an effective path for the fuel escape from the core, possibly preventing or limiting the molten fuel penetration of the assembly wall and the radial propagation of the disrupted fuel region to the neighboring fuel assemblies.

2.3.3.2 Particulate Debris Bed Formation and Coolability

There are two limiting scenarios that describe the molten core debris flow downward toward the lower plenum during a LOF event. If the structure below the core has large diameter coolant channels, freezing and plugging would not substantially obstruct the downward flow and the molten core material could flow rapidly from the core, reaching the lower core structures in a short time after cladding failure. If the lower core structure, however, has very small diameter channels with a substantial heat capacity then the core debris could freeze and plug the channels.

For the case with large diameter channels and if the coolant channels in the core have been voided, the melt can descend rapidly into the sodium filled areas below the core, driven by fission gas pressure and, in oxide fuelled reactors, steel vapor pressure. The result of the melt-sodium contact is dependent the vigor of this contact and on the temperature of the molten core materials at the time of contact with liquid sodium. In the case of oxide fuel, which has a much higher temperature than the sodium boiling point, the melt-sodium contact is expected to lead to solidification and fragmentation with very small particles. In the case of metal fuel, which melts below the boiling point of sodium, the melt-sodium contact may result in incomplete fragmentation or fragmentation with very small particles. Such fragmentation may improve the lateral spreading once the materials have moved into more open areas, but the melt will still have a high rate of decay-heat generation.

For the case with small diameter channels, where freezing and plugging is likely to obstruct the coolant channels, the melt will probably emerge from the assembly by melt-through of the duct wall. In that case the flow will continue within the spaces between the ducts, with the fuel melt displacing sodium as it flows downward and may eventually melt through another neighboring duct wall. Considering the difficult downward path for the melt and the low temperatures in the regions below the core, it will take a relatively long time for the fuel to reach the core support plate and the decay heat level will be considerably lower at that time. The melt will enter the sodium gradually, leading to complete fragmentation with larger particles, but the degree of lateral spreading may be small. The melt will contain considerably more steel than in the previous case.

Once the core materials have moved down and out of the bottom of the assemblies, a debris bed would form in the inlet plenum. The debris bed coolability calculations require assumptions about the debris spread, bed porosity, and particle size and sphericity. Experiments performed at ANL with metal fuel and sodium [2-9] showed the formation of high porosity debris beds (0.76-0.95), but the porosity of oxide fuel beds is considerably lower. The heat generated in the porous bed is removed by conduction, convection without boiling, and eventually coolant boiling if the first two heat transfer mechanisms are not sufficient to cool the debris.

Another scenario to be considered is the possibility that the melt entering the sodium will not fragment or will fragment but not quench. This would result in the accumulation of a molten pool in the inlet plenum on the core support structure. It is also possible that a particulate bed could form initially and later form a molten pool. The study of the coolability of the molten pool requires tracking the time evolution of the crust formation at the pool interfaces with the core support structure at the bottom and the liquid sodium at the top. These calculations must also include the reduction of the plate thickness due to melting, as melt-through or failure of the lower plate in the inlet plenum would cause the melt to relocate to the bottom of the reactor vessel or to the core-catcher if the reactor is equipped with one.

2.3.3.3 Vessel Failure Mechanisms

Three potential vessel failure mechanisms have been identified: a) pressure loading due to an energetic power excursion, b) melting due to direct contact with high temperature molten fuel/steel mixture, and c) creep rupture at elevated temperature over an extended period of time.

a. Pressure Loading

Initiating phase transient analyses and/or phenomenological considerations for both oxide-fueled and metal-fueled reactor concepts [2-1, 2-5] indicate that the CDAs considered do not produce energetic severe accident sequences that challenge the containment during this phase. The initiating phase encompasses the sequence of events that occur after the accident initiation, while the fuel assembly walls retain their integrity and only intra-assembly material relocation occurs. For such a loading to develop, it is necessary to heat the fuel and/or sodium coolant to a level such that rapid vaporization occurs, producing pressures sufficiently high (hundreds of atmospheres) to cause a rapid expansion of the vapor and movement of the liquid sodium above the core and/or the remaining structures in the upper core region towards the top of the reactor vessel. The impact on the top of the reactor would have to be large enough to break the reactor vessel. However, the development of fuel vapor, if it occurs, leads to rapid fuel dispersal and rapidly decreases the reactivity and power of the reactor. In addition, the heat capacity ratio of the coolant and structures to the fuel is such that the fuel will quench following such a power excursion without raising the non-fuel materials to temperatures where their vapor pressures become significant. Concerning the potential for sodium vaporization and sodium explosions due to direct contact with the fuel it is noted that such events require spontaneous nucleation on contact. The contact temperature for the oxide-fuel and sodium system is well below the spontaneous nucleation for liquid sodium [2-1], and the same applies for the metal-fuel systems, where the fuel melting temperature is considerably lower, ruling out the potential for sodium vapor explosions.

Recognition of the dispersive nature of the fuel motion immediately or soon after the cladding failure and the absence of energetic fuel-coolant interactions during the early stages of the accident shifts the emphasis to the Post Initiating or Transition Phase. The Transition Phase is characterized by inter-assembly material relocation and begins when the integrity of the fuel assembly walls is lost, Power excursions during the Transition Phase might result if the fuel relocation were to add positive reactivity, either in-core or ex-core. As discussed above, fuel/steel freezing and channel plugging can trap relocating fuel and raise the possibility of re-criticality and a power excursion. For oxide fuel, it appears likely that once the Transition Phase started, the accident will progress until most or all of the assemblies are melted, and the accident will only terminate with a power excursion that disperses the fuel or by removing fuel from the molten pool by draining into the lower parts of the core and beyond. The Transition Phase may be averted altogether by providing a path for the molten fuel to escape to the inlet plenum and beyond prior to the breaching of the fuel assembly walls. The concept of Controlled Material Relocation, which involves changes in the design of the fuel assembly, is being considered by some SFR designers with the intention of providing paths for molten fuel to exit the core region prior to breaching of the assembly walls that could eventually lead to the formation of a molten pool. This concept, intended to ensure the axial fuel relocation to the upper and lower plenum, is described in Section 3.

For metallic fuel, experimental results show that progression to a Transition Phase characterized by inter-assembly material relocation may not occur due to the more favorable fuel dispersal characteristics in the coolant channels of intact pin geometry. Recriticalities may be avoided entirely and the accident terminated by early fuel dispersal that permanently removes fuel from the core region. Several TOP experiments in TREAT have demonstrated the metal fuel upward relocation with only limited fuel freezing and partial blockage formation [2-15]. For a LOF, scoping analyses indicated that in a metal-fueled reactor the blockage formation below the core and thus the obstruction of the downward fuel relocation can be avoided by designing the coolant channels below the core with a larger hydraulic diameter.

Once fuel reaches the lower plenum, where a large fraction of the fuel is expected to relocate eventually in a LOF scenario, it is necessary to ensure that no re-criticality occurs and the debris bed remains coolable. The radial spreading of the fuel debris and the debris-bed characteristics are important factors in determining the potential for re-criticality. Neutronic analyses indicate that the subcritical reactivity can be maintained even if the full core relocates to the inlet plenum, provided that there is sufficient horizontal spreading of the molten and solid relocated core materials.

b. Melting Due to Direct Contact

Direct contact of core fuel or fuel/cladding mixture with the reactor vessel steel walls can result in liquefaction of the steel if the fuel temperature is high enough. The normal melting point of the steel is around 1700 K. In an oxide-fueled reactor the fuel melting temperature is around 3000 K, considerably higher that the steel melting temperature. In a metal fueled reactor the fuel melting temperature is around 1400 K, lower than the steel melting temperature, but chemical interactions between fuel and the steel components are likely to lower the effective melting temperature of the steel. The rapid eutectic penetration temperature is approximately 1400 K. Melt-through of the reactor vessel might occur if direct contact with fuel was established and the vessel cooling was insufficient to maintain the local wall temperature below the effective melting temperature. If the core materials are coolable in the inlet plenum, either as a particulate bed or a molten pool, then direct contact between the core fuel and the reactor vessel will be precluded and core materials can be contained in the inlet plenum in a coolable state. It is noted that the ALMR reactor provides a backup core support plate to retain molten core material that would penetrate the lower core support plate that forms the floor of the inlet plenum. The ability of the inlet plenum and backup support plate to contain the relocated core materials in a coolable configuration is an important safety issue that was mentioned in the PRISM PSER Final Report [2-14], and should be evaluated more closely in future studies. The addition of an in-vessel core-catcher that would play a similar role is a CDA mitigation feature that is further discussed in Section 3 of this report.

c. Creep Rupture at Elevated Temperature

Creep rupture of the reactor vessel subject to elevated temperatures for extended periods of time is the third candidate mechanism for vessel failure. A scoping analysis for a metal-fueled ALMR estimated the static tensile stress for accident loads at two locations, neglecting transient thermal stresses. The two locations considered were the vessel wall exposed to the hottest sodium (near the top just below the head), and the lower core support plate. The analysis indicated that at the normal sodium boiling temperature the creep rupture time at the top of the vessel would be greater than 10 years. However, the lower support plate was found to fail due to creep after about 100 hours if the fuel/steel mixture is at 1071 C but remain intact indefinitely at the fuel/steel mixture has the temperature 721 C. These results indicate that the creep rupture failure of the primary system structures must be considered as part of the overall in-vessel retention analysis.

2.3.4 Radiological Consequences

The end product of the CDA consequence assessment is an estimate of the radioactivity release outside the containment building and the associated exposure to the public. The analysis of radiological consequences involves several phases including the source-term evaluation, secondary-containment evaluation, radioactivity transport and release, and atmospheric dispersal.

The source-term evaluation, which specifies the quantity of radioactive material released from the primary reactor system, represents the greatest uncertainty of these phases.

In the absence of hydrodynamic disassembly of the reactor and/or energetic fuel-coolant interactions, the occurrence of a benign core melt-down including fuel melting and fragmentation and debris-bed formation on available surfaces must be considered in the evaluation of the source term. If there are enough surfaces that remain coolable by conduction, convection, or sodium boiling within the beds, to cool the relocated fuel/steel mixture, the core materials will remain cooled indefinitely within the reactor vessel resulting in an essentially negligible source term for radioactive release to the secondary containment.

The processes associated with a melt-through of the core material out of the reactor vessel must also be considered. These include the chemical reaction of the sodium with concrete and the resultant release of water vapor from the concrete, the generation of hydrogen as a result of the sodium-water or sodium-concrete reactions, and the interaction of molten fuel with concrete. The principal concern with respect to off-site radiological consequences is the integrity of the containment envelope due to possible buildup of internal pressure, temperature, or explosive gases. Studies have indicated that in the absence of an engineered out-of-vessel core catcher, fuel in contact with concrete will cause concrete melting and limited penetration, due to the mixing of the molten fuel with concrete and resulting dilution and heat flux decrease. There is substantial evidence that the inherent response of the reactor structures and mitigating features, such as containment atmosphere venting and cleanup systems, would keep the radioactivity release within tolerable limits. System concepts involving confinement combined with a filtered system may offer a far more efficient protection than the pressure-tight containment [2-10].

Containment building temperature and pressure transients are determined primarily by the chemical reactions of the sodium expelled from the reactor system and by the decay heat of released fuel and its chemical interactions with containment materials. The release of the resultant noble gases and aerosols depends on the mixing with the containment atmosphere and the leakage paths present. The halogens and solids are subject to removal mechanisms within the containment, primarily dependent on aerosol mechanics. Aerosol agglomeration, settling, and plateout can remove particulate radioactivity with a removal time constant of the order of hours. Removal of aerosols in the leak paths can also play a significant role in reducing the release of radioactivity from the containment building.

The atmospheric dispersal of radioactivity after release from containment and the corresponding potential exposure to individuals are calculated by conventional methods used in LWR safety analysis.

3. DESIGN MEASURES FOR SEVERE ACCIDENT PREVENTION AND MITIGATION

This section describes specific design features used in the past and proposed for the future, with the goal of preventing the occurrence of irreversible core geometry or structural changes due to an initiating event that could otherwise lead to a CDA, or, in the case that a CDA would occur, mitigating its consequences.

3.1 Prevention of Irreversible Core Geometry or Structural Changes

3.1.1 Self-Sctuated Shutdown Devices (SASS)

3.1.1.1 SASS using Magnetic or Fuse Devices [3-13]

In order to achieve high reliability of the Plant Protection System (PPS), two separate shutdown systems have in general been included in the design of US and overseas SFRs. If the primary system fails, a very low probability event, the secondary system would shut down the plant if necessary. In general, the set-points for the actuation of the secondary system are somewhat higher than those of the primary system, so that the secondary system will not be actuated if the primary system functions correctly. To further reduce the probability of the shutdown system in SFRs, a third self-actuated shutdown system (SASS) has been studied extensively in the US and other countries [3-15, 3-16]. The SASS would have actuation set-points higher than both the primary and secondary system, and would normally not be activated if either the primary or secondary system functions correctly. It should have the following main characteristics: a) outside dimensions identical to those of a normal fuel assembly, and fully contained within these dimensions, b) insertable into the core in any chosen subassembly location, and easily replaceable, c) capable of introducing negative reactivity with very high reliability by either removing fuel from, or adding neutron poison to the active core region, d) actuated inherently by either coolant temperature or power level, e) capable of preventing core disruption for all anticipated low probability initiating faults by shutting down the reactor prior to loss of coolable core geometry (the system is not required to prevent fuel damage under all circumstances), f) insensitive to core deformation and seismic events, g) having a small impact on core performance as regards both neutronics and thermal-hydraulics, h) having a system lifetime equal to or longer than that of the core, i) testable both ex-core and in-core and re-settable.

In any SASS the following essential functions can be distinguished: 1) the sensing function, 2) triggering logic function, and 3) lock-release or actuation function. For the sensing-triggering function, devices based on three different principles were identified in [3-13] as being promising: 1) melting-point-operated devices, including fuses, 2) ferromagnetic Curie-point-operated devices, and 3) thermal-expansion-operated devices.

Conventional fuse-type devices have the merit of simple designs. They have, however, two important disadvantages, namely: 1) the fuse is destroyed in the process of its actuation, precluding the possibility of testing the device, and 2) in order to obtain a short time response, the fuse device has to be operated close to its melting point, which limits the force that can be applied without incurring excessive creep. The most promising device is the Curie-point-operated device which has been described in [3-13, 3-16] and is further discussed below. This device is also the basis of the SASS adopted for Japanese SFR described later in Section 4. Thermal expansion devices were examined but were considered the least promising, and little effort was spent on their development.

Table 1 provides an overview of some important SASS design options, and the choices selected in [3-13] are indicated by asterisks. These choices were used to define the SASS characteristics and examine the interaction between the SASS and its host reactor system. The conceptual design analyzed, shown in Figure 2, had the following main characteristics:

a) The absorber material in pin bundle configuration is held above the core, partially within the upper blanket;
b) The absorber is held by a ferromagnetic Curie-point-operated device, which combines the sensing, triggering, and lock-release functions;
c) The heating of the ferromagnetic yoke for protection against LOF accidents is provided by two peripheral rows of fuel pins which heat up the sodium;
d) The heating of the ferromagnetic yoke for protection against TOP accidents is provided by fissile material located in close contact with the yoke.

The physics evaluation of the SASS was performed with respect to the CRBR plant and included the determination of the optimum absorber enrichment and position, and an assessment of the impact on core parameters. The calculations were performed for first and equilibrium CRBR cores. It was found that the overall worth of a SASS assembly increases with the enrichment of the Boron (B-10) used, but the worth increase becomes marginal at about 75% B-10 enrichment. With this enrichment, the highest worth positions for the CRBR inner core were located in the 5th fuel ring, with worth 3.37 dollars. The central flat positions of the 5th ring are most desirable since they have a high reactivity and minimum interaction with existing control rod assemblies.

The replacement of two CRBR inner core fuel assemblies by SASS assemblies results in two basic reactor parameter changes: a) the first is the removal of a small amount of fuel from a high worth region, b) the second is the placement of a substantial amount of absorber material in the upper axial blanket. Removing a small amount of high worth fuel reduces the excess reactivity which in turn shortens the operating cycle length. Two SASS assemblies would reduce the normal cycle length by 20 days. This can be compensated by a small increase in the core enrichment of 1.5%, which would keep the initial cycle length the same. This in turn would lower the reactor breeding ratio by 2.8%. The addition of SASS control poison in the axial blanket degrades the neutron economy and depresses the flux. Two SASS assemblies positioned at central flats of the 5th ring result in a 5.5% reduction in axial blanket breeding and a 0.6% reduction in overall reactor breeding.

Accident analyses using the SAS3A code were carried out [3-17] to evaluate the performance of a SASS for an LMFBR based on the CRBR design. For all runs the insertion of negative reactivity was initiated 0.5 sec after reaching the set point and was assumed to be linear, at a rate of -8 $/s up to a value of -4.0 $. From the analyses performed it was concluded that core damage can be averted by using a SASS for the postulated transients.

Table 1 Primary Design Characteristics of SASS
(tentatively-preferred options marked by asterisks)

Characteristics	Design Options
Concept Type	(1) Combine Sensing, triggering, and lock-release functions*, or (2) combine sensing and triggering functions, keep separate lock-release function
Sensor-trigger type	(1) Melting-point-operated device, (2) ferromagnetic Curie-point-operated device* or (3) thermal-expansion-operated device
Insertion force	(1) Gravity*, (2) spring, (3) pneumatic, (4) hydraulic, or (5) explosive
Reactivity control	(1) Absorber insertion*, or (2) fuel removal
Form of reactivity control material	(1) Solid block with perforations, (2) Rod bundle*, (3) articulated (chain) configuration*, (4) balls, or (5) liquid
Type of reactivity control material	(1) Fuel, (2) boron carbide*. (3) tantalum, or (4) europrium oxide

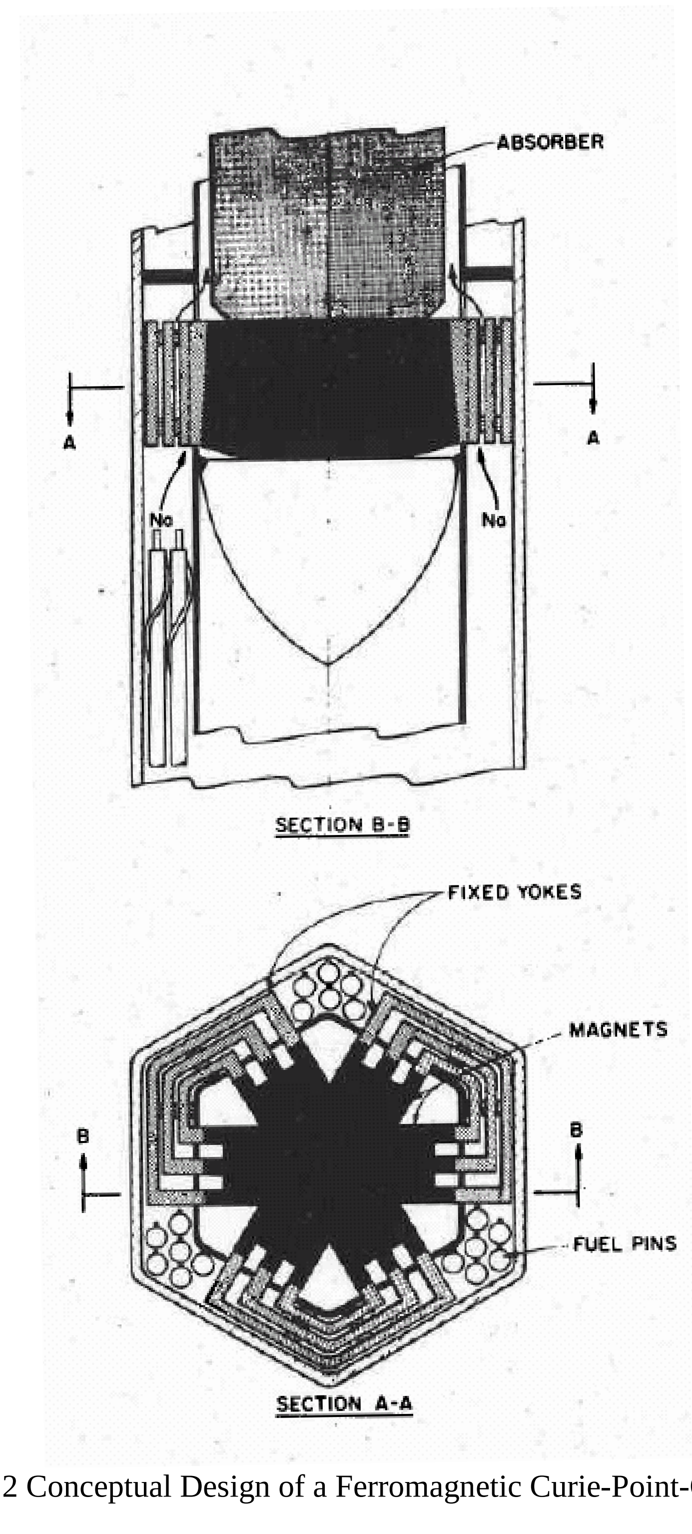

Figure 2 Conceptual Design of a Ferromagnetic Curie-Point-Operated Device

3.1.1.2 SASS using Hydraulically Supported Absorber Balls [3-14]

A SASS which relies on hydraulically supported absorber balls has been studied at the Atomics International (AI) division of Rockwell Corp. This system, referred to as Inherent Shutdown System (ISS) is both completely independent and diverse from the regular safety rod systems, and thus provides the core protection even in the extremely unlikely event of a common-mode failure of the rod-type systems. A number of columns of small tantalum balls (~0.25 in. diameter) are held above the active core region by the reactor coolant flow. These columns automatically fall into the core and shut the reactor down if the coolant flow of the reactor is lost during a LOF event. A thermally-actuated valve within each ISS assembly interrupts the flow and causes the system to shut down the reactor during a TOP event.

The reference ISS concept consists of small tantalum balls which are hydraulically raised by the reactor's coolant and stacked above the active core inside a hexagonal assembly housing. The sodium pumps are slowly ramped to the reactor's full flow rate to achieve complete stacking. Once complete stacking is achieved the reactor flow can be reduced as a flow as low as 40% nominal without any balls falling back into the active core. A Curie-point temperature-operated magnetic device containing a fissile material trigger is located below the core. In the event of a TOP one portion of the magnetic device will overheat and release a plug that closes the coolant orifice causing the balls to fall into the core. A schematic of an LMFBR with an ISS is shown in Figure 3, and the ISS subassembly is illustrated in Figure 4.

The main components of the ISS are the absorber balls and the flow shutoff valve. The length of the absorber ball column section is twice the length of the active core. The absorber balls are usually stacked above the active core to a height equal to the length of the active core, and when released into the active core they completely fill its height. The absorber balls are prevented from going out at the top of the assembly by a grate that contains coolant passages with a diameter smaller than the absorber ball diameter. A similar grate at the bottom prevents the balls from dropping below the active core. A cylindrical tube is used in the center of the ISS subassembly allowing part of the coolant to bypass the column of absorber balls. This central bypass tube contains perforations along the upper section corresponding to the length of the absorber column. The length of this perforated section determines the dropout characteristics of the system in response to a decrease of the coolant flow rate. The absorber balls used in the reference design are made of tantalum since it has the highest density of the common neutron absorber materials and would have a shorter drop time. Their size (0.25 in.) is a trade-off between the drop time of the ball column and the sodium flow rate required to levitate the balls.

The other main component of the ISS is the flow shutoff valve which consists of a thermally-actuated plug that is pushed upward by spring when the temperature of a section of the device is raised above the Curie point of its material, shutting off the coolant flow through the ISS subassembly and causing the absorber ball column to fall into the active core. By placing a fissile material in one part of the magnetic circuit the temperature of that section increases rapidly when the neutron flux increases during a TOP event.

The ISS was designed with a control worth of 1$/assembly, and a total of 5 to 7 ISS subassemblies was found to be needed to meet the reactivity control requirements for the CRBR-type reactor considered. Analyses of both LOF and TOP events with failure of the normal reactor scram indicated that the ISS can adequately shut down the reactor and protect the public during these events. The ISS subassembly is very simple and has no connection with the top shield. Therefore, during a seismic event, no parts of the control mechanism need to move through the top shield area, or between that area and the active core, in order to scram the reactor. Even if the core is severely distorted and the normal control rods cannot enter the core, the absorber balls would still drop into the active core since they are very small and can move through extremely distorted channels.

Since the ISS subassemblies can operate with the reactor power varied from 40% to 100%, no penalty is associated with the normal plant operation, If 7 ISS subassemblies are added to a large LMFBR, the fissile inventory of the reactor would have to be increased by 0.18%. The sodium flow rate through each of these assemblies is about two-thirds of the flow rate of a normal fuel assembly. Since the sodium is not significantly heated as it passes through the ISS assembly, the sodium outlet temperature from the core would be decreased by about 3 F by the addition of 7 ISS assemblies.

The ability to maintain some of the reactor components and repair them may be affected by the use of this system since some corrosion products from the tantalum absorber balls will be transported through the primary system by the coolant and deposited at various places. Since the tantalum will be partially transmuted to radioactive Tantalum 182, this material will be carried around the primary loop and cause it to become somewhat radioactive. Calculations have indicated that if pure tantalum is used, its corrosion rate could cause the IHX to become too radioactive for hands-on maintenance. However, limited experimental data indicates that a tantalum alloy, T111, which is 90% tantalum, 8% tungsten, and 2% hafnium, would have substantially lower corrosion rates. An alternative may be to use absorber balls that are vapor deposited with chromium, which would lead to very little radioactivity transferred to the primary loop.

Major uncertainties associated with this system include: ball jamming possibilities, absorber ball self welding, absorber ball corrosion rates, and a reliable flow shutoff valve design. Another key uncertainty of this device is the actual absorber insertion rate during realistic flow coast-down rates when the reactor pumps are tripped. A number of additional water, sodium, and irradiation experiments are recommended in [3-14] to further study the characteristics of this system.

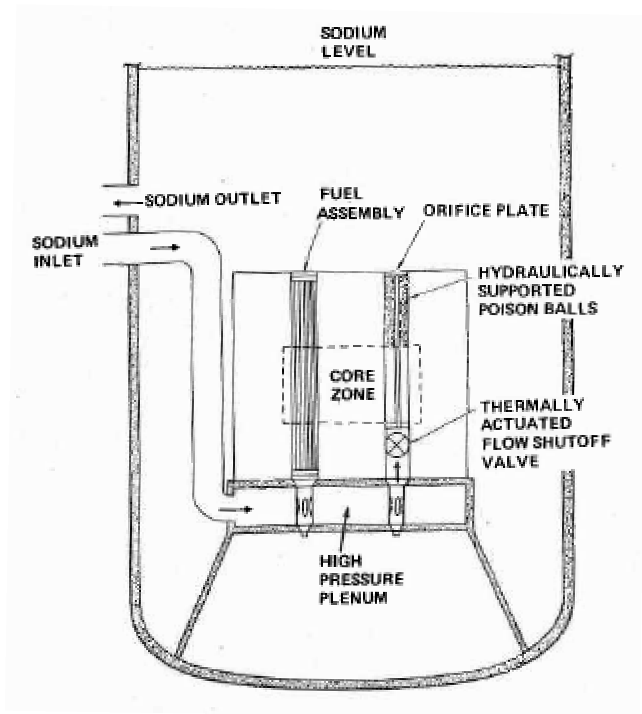

Figure 3 Hydraulically Supported Absorber Ball Assembly in a SFR

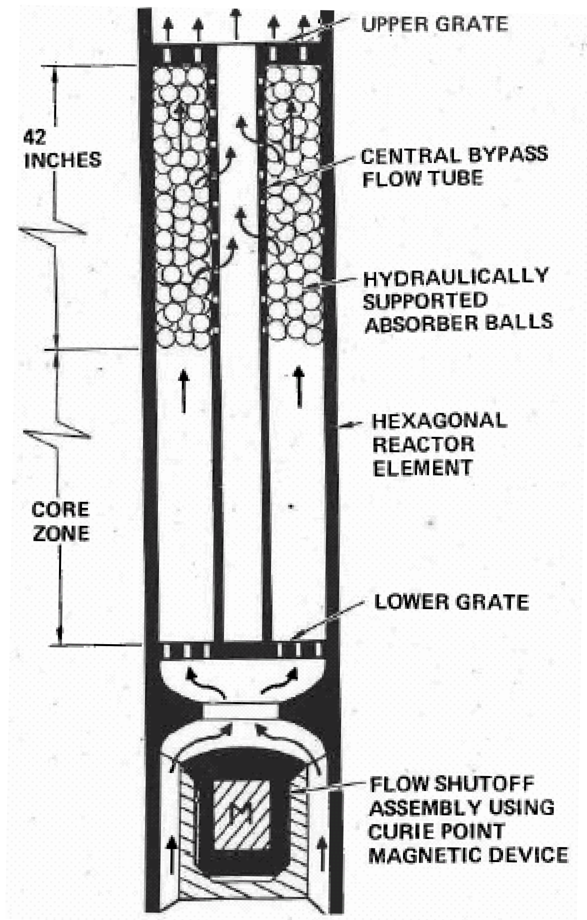

Figure 4 Absorber Ball Assembly Conceptual Design

3.1.2 GEMS [3-1, 3-2]

The concept of the Gas Expansion Module (GEM) device was introduced to increase the margin to boiling in the core by introducing negative reactivity, and thus help lower the power level in the core, if the main coolant pumps malfunction or stop. The negative reactivity introduced by GEMS is used to overcome the positive Doppler reactivity in the case of an unprotected LOF in oxide-fueled cores. Lower operational fuel temperatures in metal fueled cores yield a smaller positive Doppler feedback in an LOF, so the GEM is not needed. The GEM is essentially an empty assembly duct at the periphery of the core, filled with an inert gas, sealed at the top and open at the bottom, which is connected to the core high-pressure inlet coolant plenum. Sodium voiding at the periphery of the core causes a negative reactivity feedback due to the decreased neutron reflection and resulting increased neutron leakage from the core. When the pumps are operating, sodium is pumped into the GEM at the bottom, and the trapped gas is compressed into a region which is above the active core. Sodium then occupies the portion of the GEM adjacent to the fueled region of the core. If the pressure in the core inlet plenum drops, as would occur during a flow coastdown, the gas region expands downward into the core region and increases the voiding in that region. Since the GEM assemblies are placed at the periphery of the core, typically in the first row of reflectors, this effect adds negative reactivity and contributes to inherent reduction of the core power thus increasing the margin to coolant boiling. Since the reactivity worth of the GEM assembly depends on the change in neutron leakage, this device is dependent upon core size, with generally lower GEM worth per assembly as the core size increases.

In order to demonstrate the effectiveness of the GEM, Westinghouse Hanford Company (WHC) carried out a series of tests at the Fast Flux Test Facility Reactor (FFTF) in the 1980s. This was a series of unprotected loss of flow tests (LOF/WOS) from power levels up to 50% (200MWt). It was part of the Inherent Safety Test (IST) program at FFTF. There was also a pump start test to demonstrate that the reactivity transient is benign for an inadvertent pump start-up event. A number of GEM devices were built and six were loaded into the reactor, equally spaced azimuthally at the core periphery. After preliminary tests for feasibility, the LOF/WOS series was carried out. The first set of tests was with flow coast-down to primary pump pony motor flow level and then the tests were repeated with coast-down to natural circulation level. Tests were run from fractions of full power up to 50% power. The maximum coolant core outlet temperature for the two series was < 950F. The tests were limited by temperature limits which prevented further tests starting at initial power greater than 50% of nominal full power. The results show the margins that can be achieved. Subsequent modeling of these tests, which included the reactivity feedback from the response of the GEM devices along with the other reactivity feedback phenomena such as radial core expansion and control rod driveline expansion, was used to provide initial validation for computer codes such as SASSYS/SAS4A [3-36].

GEMs were also proposed in the GE PRISM design, part of the DOE advanced liquid-metal reactor (ALMR) project, for use if an oxide fuel core was used. During the interactions with the NRC, issues were raised about the possibility of adding to the overall risk by the introduction of GEMs and the qualification of the GEMs. Inadvertent start-up of a primary pump from near critical conditions could result in positive reactivity insertion, and leakage of the GEM gas with migration of the gas bubbles into central regions of the core could add to the overall risk.

3.1.3 Natural Circulation Decay Heat Removal

Both pool-type and loop-type SFRs will require robust, redundant auxiliary cooling systems (ACS) or decay heat removal systems (DHRS) to remove decay heat generated in the core and transferred to the primary sodium coolant during routine shutdown and emergency conditions when the primary heat removal system has failed. The ACS must be designed with enough heat removal capability and redundancy to ensure that for an anticipated transient event, such as a loss of flow (LOF) or loss of heat sink (LOHS), either with or without scram, system temperatures would remain low enough so that a significant margin to core damage due to overheating the fuel or cladding would exist. Depending on the event, the core may still be expected to remain in an operable condition with no fuel deformation or melting and no cladding failures.

The capacity of the ACS is a design choice, and will depend on other aspects of the reactor system design and the anticipated transients that it will be expected to accommodate. The ABTR Preconceptual Design Report [3-27] states as one of its base requirements that "…the design shall have a passive means of negative reactivity insertion and decay heat removal sufficient to place the reactor system in a safe stable state for specified anticipated transient without scram (ATWS) events without significant damage to the core or reactor system structure." The term 'passive' means a system that does not require an external action, i.e., a system which would be 'active', to have the system function. What qualifies as "significant damage" is not defined, although it is likely that it is expected to mean that assembly and fuel pin geometry was still maintained, so that there is no permanent damage to the reactor. This would allow for some fuel damage or deformation, such as would occur for overheated fuel pins, and would render the core inoperable until all of the damaged fuel was replaced.

In order for the reactor to survive an ATWS event without cladding damage, the ACS would be required to be designed and function with adequate heat removal from the primary coolant and subsequent primary coolant natural circulation through the core. As the core and primary coolant continued to increase in temperature during the transient, sufficient inherent negative reactivity feedback effects would be required to shut down the reactor. The heat removal, natural circulation, and reactivity feedback must be sufficient to maintain the reactor core temperatures below the failure limits of the cladding, and the primary coolant below the local and bulk boiling temperatures.

Reactivity feedback from fuel Doppler, coolant density, axial expansion, radial expansion, and control rod driveline expansion would be considered in determining the overall inherent reactivity effect at the given reactor conditions. The net reactivity change from the combination of the individual effects is not expected to be large [3-27] and must therefore be well understood and defendable. Allowing for coolant local or bulk boiling would cause voiding in the core region and the addition of potentially significant amounts of positive reactivity (>$1.00), which have unacceptable consequences.   Negative reactivity addition for metallic fuel extrusion, allowing for more neutron leakage, could be a significant driver for using metallic fuels. Other passive features such as GEMS or SASS would need to be evaluated from a nuclear safety related equipment basis point of view. The role that these types of additional features would play in evaluating the transient behavior and credit for accident progression would need to be further investigated.

3.1.3.1 Direct Reactor Auxiliary Cooling System (DRACS)

The Direct Reactor Auxiliary Cooling System (DRACS) was patented in 1982 by Westinghouse Electric Corporation [3-31]. The DRACS is simply a liquid metal to liquid metal heat exchanger placed in the cold plenum downcomer portion of a liquid metal fast reactor. An air cooled to liquid metal secondary heat exchanger, located outside of the reactor vessel, removes the heat to the environment via an exhaust stack. By locating the primary heat exchanger in the cold plenum, the heat exchanger acts to maintain the cold plenum temperature lower than the core and hot outlet plenum coolant temperature. The lower temperature, higher density coolant in the cold plenum establishes a pressure head for natural circulation flow through the reactor core. Although forced flow can be designed into the DRACS coolant and air coolant flow paths, a more passive approach is to incorporate natural convection flow for both flow paths. Electromagnetically latched flow dampers limit the air flow through the secondary heat exchanger during normal operating conditions, limiting heat loss through the DRACS, but allowing for natural circulation flow to be maintained during normal reactor operations, ensuring the availability of the system when needed. A loss of power or scram would de-energize the flow damper latches to allow for full operating conditions to be established. A manual actuation of the flow damper can also be incorporated into the design.

DRACS can be used in both pool-type and loop-type liquid metal reactors. Four DRACS are proposed in the pool-type ABTR concept [3-27] with three being required for adequate heat removal capability and one spare for redundancy. The Japan sodium fast reactor (JSFR) loop-type concept [3-28, 3-29] proposes to use one DRACS and two primary reactor auxiliary cooling systems (PRACS), one in each leg of the dual-loop concept. The PRACS units are very similar to the DRACS. The difference is that the PRACS are designed to fit within the upper plenum of the intermediate heat exchanger (IHX). Natural circulation flow is established though the cold leg inlet piping to the inlet plenum of the reactor core. The DRACS operation is similar to that described above, with natural convection flow developed within the reactor vessel. A similar concept was used in the EBR-II reactor, which included two shutdown heat removal loops, with the sodium/sodium heat exchanger placed in the sodium pool, as illustrated in Figure 5. Other SFR concepts [3-30] include DRACS units in their design for backup decay heat removal, when the primary heat removal system is inoperable.

The positive aspects of the DRACS are that, since it is located within the primary coolant and can be made passive, it is both efficient at removing heat and reliable and it can be scaled and used in parallel with other units. The DRACS can be designed with essentially no moving parts and natural flow convection through all flow paths. The negative aspects are that: a) additional penetrations must be made through the primary vessel; b) the primary heat exchangers must be accommodated at the appropriate locations within the reactor vessel, and c) the power losses due to continuous passive operation are larger than possible alternatives described in below in Section 3.1.3.2.

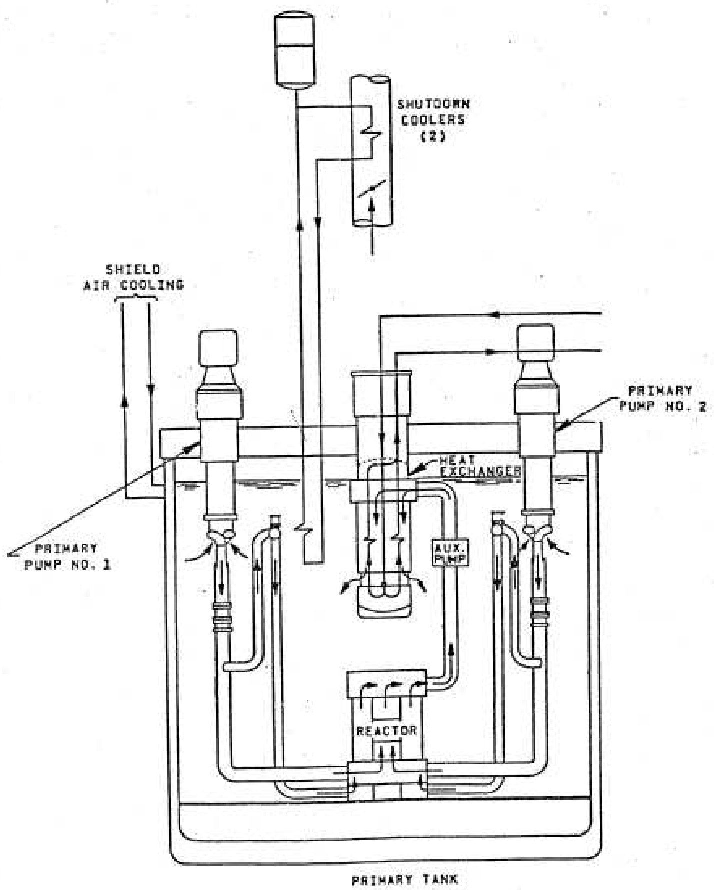

Figure 5 EBR-II Primary System [3-35]

The Advanced Burner Test Reactor (ABTR) heat removal concept is shown in Figure 6. Four DRACS are proposed in the ABTR concept [3-27]. With three of the four operating, 0.5% of the total reactor power can be removed from the primary coolant. The ABTR concept uses NaK as the secondary coolant, which has a melting temperature of -13°C versus 98°C for Na. Using NaK ensures that that the DRACS coolant will not solidify at nominal atmospheric conditions. The pressure in the NaK flow path is higher than the Na primary coolant to ensure that a leak in the primary heat exchanger will not allow primary coolant into the DRACS. Each of the four DRACS units is rated to 0.625 MW or 0.25% of the full power rating of the ABTR (250 MW). At normal operating conditions, the units run at ~1% of their full cooling capacity (~6 kW each). This allows for the natural circulation flow to be established in the NaK secondary coolant and in the air tertiary coolant, with the system ready for immediate full operating conditions while limiting heat losses from the system.

Accident analyses using the SAS4A/SASSYS-1 computer code have predicted that the DRACS units are capable of removing sufficient heat such that the LOF event with scram and without scram will not result in any fuel or cladding damage in the core.

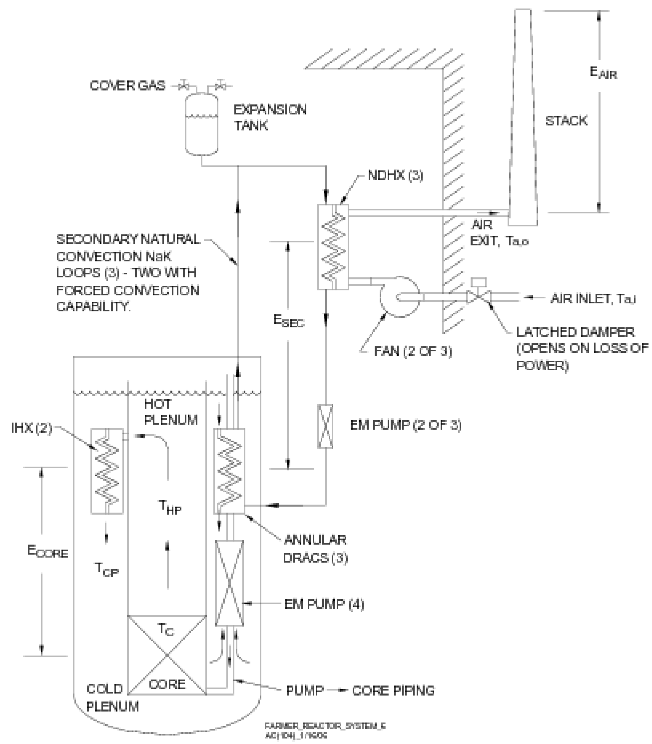

Figure 6 ABTR Pool-Type Heat Transport System Configuration [3-27]

The JSFR heat removal concept is shown in Figure 7. As stated previously, the JSFR concept uses one DRACS and two PRACS. The systems work similarly to the ABTR concept with the major difference being that Na is used as the coolant instead of NaK. Accident analysis results predict that the heat removal of the DRACS and PRACS is sufficient to prevent fuel or cladding damage for both protected (where the scram system operates) and unprotected (scram system fails to operate) anticipated transient conditions. For the unprotected conditions, the additional SASS feature decouples the control rods allowing them to be inserted into the core and shut down the reactor.

In conclusion, the DRACS concept appears to be a robust and reliable approach for auxiliary cooling of the SFR in both the pool-type and loop-type configuration. Since the concept uses tube and shell type heat exchangers and natural circulation, the approach is scalable to larger systems. Multiple units can be employed for redundancy and the passive nature of the concept allows for high reliability. Accident analysis has shown that for ATWS events, the heat removal, combined with the natural circulation though the core and negative reactivity feedback, can effectively prevent accident progression to core damage.

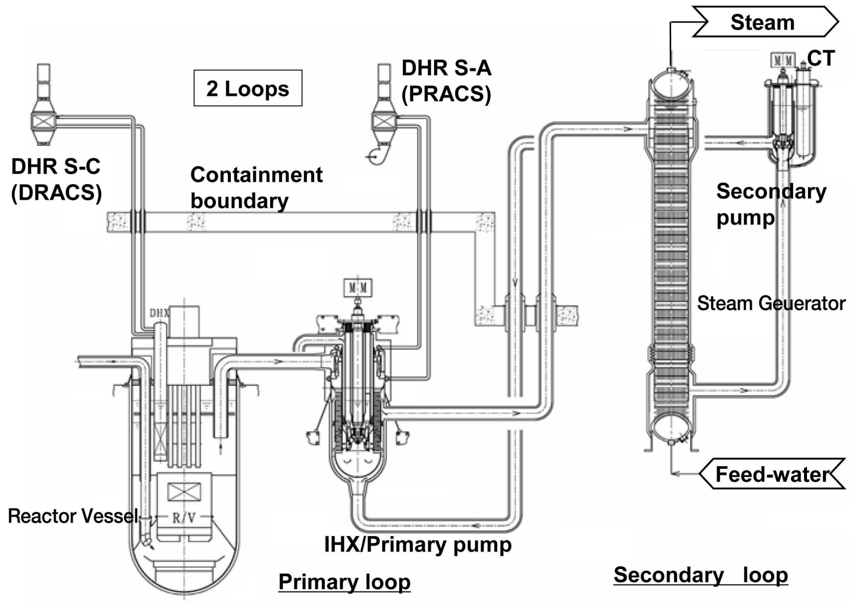

Figure 7 JSFR Loop-Type Heat Transport System Configuration [3-28]

3.1.3.2 Reactor Vessel Decay Heat Removal System (RVDHRS) [3-3, 3-4]

The reactor vessel decay heat removal system (RVDHRS) is a system which removes heat passively from the reactor containment (guard) vessel by the natural convection of air flowing up past the outside of the vessel. ALMR designs utilized the RVDHRS as the safety-grade system for removing residual heat from the reactor core. Core heat is conducted through the reactor vessel wall from the sodium coolant and transferred across the argon gas gap between the reactor vessel and the containment vessel to the reactor containment vessel. This heat is passively removed by the air flow around the outside of the containment vessel with natural convection of the heated air to the ultimate heat sink which is the atmosphere above the below-grade silo. To complete the passive air flow loop, cooler air from the atmosphere is concurrently drawn downward to the bottom of the silo, where it is turned inward and upward to flow around the outer surface of the containment vessel. The RVDHRS is therefore completely passive and always in operation.

In the case of the ALMR design, the design predictions showed that the RVDHRS is able to keep the reactor vessel temperatures below the ASME Service Level C limits for the RVDHRS design-basis event. This event involves a reactor scram with only RVDHRS cooling. The RVDHRS handles the design basis event and brings the reactor to hot standby but in considerably longer time than 36 hours. The temperatures are above the ASME Service Level B limit so the potential for damage to internal components during these events was noted by the NRC, but the design duty cycle anticipated only one such postulated event during the 60 year life of the plant .The temperatures are also elevated in the silo, so in the view of the NRC, the materials utilized in the silo will require substantial justification. Blockage of the RVDHRS passages was postulated as the most credible failure mode of the RVDHRS.

To resolve the postulated issues and to confirm the design performance of the RVDHRS concept, Argonne National Laboratory (ANL) carried out a series of RVDHRS tests in the Natural Convection Shutdown Heat Removal Test Facility (NSTF) when the ALMR program was terminated The NSTF is a large-scale test facility located on the ANL site. The facility was originally developed to provide confirmatory data for the ALMR RVDHRS design. The NSTF has a full-scale vertical segment of the air-side of the ALMR RVDHRS system, with electrical heaters in the lower part to simulate heating from the reactor vessel. The principal components of the facility are the structural module, electric heaters, instrumentation, insulation, and a computerized data acquisition and control system. Figure 8 is a schematic overview. The key features of the structural module are an inlet section, a heated zone that mocks up the exterior of the reactor guard vessel and the collector (duct) wall (surrounding the guard vessel) of the RVDHRS, and an unheated stack or chimney. All sections, with the exception of the inlet section, are thermally insulated to minimize parasitic heat losses to the environment.

The objectives of the RVDHRS tests at NSTF were to assess: (I) the air-side thermal-hydraulic performance of the system; (II) its performance under degraded system operation (partial to complete blockage of the system inlet); and its performance under different wind conditions including high wind gusts. The Phase I tests were performed with smooth and finned wall surfaces in the rectangular heated test section. A total of 71 tests (runs) were performed with the smooth configuration, and 41 tests with the finned configuration. The data from the 71 tests for the smooth wall surfaces were used to develop a heat transfer correlation of the Dittus-Boelter type. Due to ALMR program termination, the Phase II tests were not carried out, but some blocked air inlet experiments were performed. As of program termination, it appeared that the RVDHRS system could be a viable approach for an ACS.

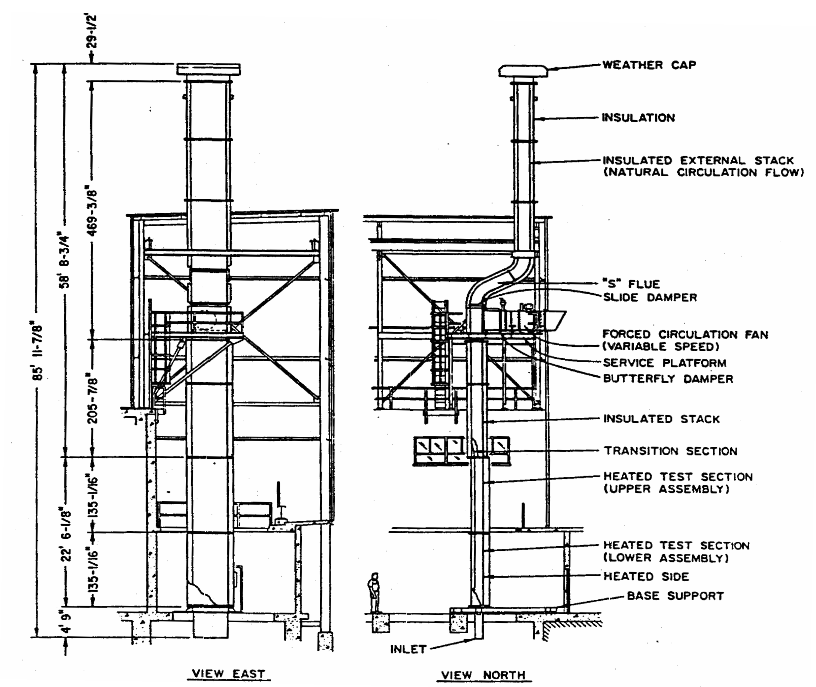

Figure 8 Schematic Overview of the NSTF

3.1.4 Core Restraint System Design for Negative Power/Flow Coefficient Reactivity Feedback

The radial expansion of the core is caused by the radial temperature distributions present in the core, both within each assembly and from assembly to assembly. Changes in the core temperatures cause changes in the dimensions of the core structures, which lead to changes in the radial core dimensions and an associated reactivity feedback. The mechanism that underlies the radial expansion is the behavior of a fuel subassembly in the presence of temperature differences between the six sides of the assembly hexcan. A new fuel subassembly will be nominally straight when inserted in the reactor. As power is increased, the subassembly will experience both axial and radial temperature gradients through the core region. The radial and axial temperature gradients arise from the corresponding variations in power generation, both between subassemblies and within any given subassembly. These temperature gradients will cause a deflection of the subassembly from the vertical, because each hexcan wall will potentially have a different axial temperature distribution and consequently each wall could have a different length. The tendency of some hexcan walls to have a greater length than the opposite wall results in creating stresses within the hexcan walls and deflection of the subassembly from vertical. The direction of the subassembly deflection depends on the direction of the net radial temperature gradient considering all six walls of the hexcan, with an unrestrained subassembly deflecting from the vertical towards the regions of lower temperatures. The net radial temperature gradient is zero in the lower region of the fuel subassembly where no power (heat) is being generated, increases approximately linearly through the core region due to the power generation, and is assumed to then remain approximately constant in the upper region of the assembly, retaining the temperature differences between each wall of the hexcan that exist at the top of the core region, again since no power is generated in that region to modify the hexcan wall temperatures. Any changes in reactor power will cause the radial and axial temperature gradients to change in proportion to the reactor power, and will thus tend to increase the radial deflection of the fuel subassemblies, especially in the upper regions of the subassembly.

The radial deflection of the fuel subassemblies can be affected by the core restraint system, depending on how the core restraint system is designed. The goal of the core restraint system is to provide a confined geometric boundary for the core subassemblies to limit or eliminate reactivity fluctuations associated with radial subassembly movement during operation and to facilitate loading and refueling by assisting in positioning the assemblies. The design of the core restraint system will also determine how it interacts with the core assemblies to define the core geometry during operation and is essential in determining the reactivity feedback that will be generated from the changing radial dimension of the core as a result of changes in power or core temperatures, including conditions that occur during off-normal or accident events. For simplicity, the core restraint system can be visualized as a ring, or multiple rings, which surround the core at one or more elevations, and which limit the radial expansion of the core subassemblies at those elevations. The driver, blanket, reflector, shield, and control subassemblies may have ‘load pads’ (i.e., thicker sections of the hexcan duct) at the elevations of the restraint rings and at other elevations as needed, depending on the design of the core restraint system.

The core restraint system must be designed to facilitate insertion and removal of the subassemblies by having sufficient net clearance between the core subassembly load pads and the restraint ring(s), which is maintained at hot standby conditions used for refueling operations. The clearances at hot standby are made large enough to ensure that the subassemblies can be removed and inserted within the limit of force allowed by the design of the subassembly and the refueling machine, even in the presence of irradiation effects which permanently deform the subassemblies, reducing or eliminating the intended clearances at hot standby and making them more difficult to remove from the core during refueling. The clearances should also be chosen to be small enough so that the subassemblies will be able to bend in response of the radial temperature gradients, as described above, until the clearances are closed, the subassemblies are tightly held, and as a result the core geometry is fixed and not subject to random variations, i.e., core geometry is ‘well-defined’. If the core restraint system is properly designed, as power is increased beyond the point where all the available clearances have been closed, which typically occurs at about 75-85% of nominal full power, the subassembly begins to take a different shape, providing a reliable increase in average core diameter in response to increases in core temperature, which decreases core reactivity, that is, negative reactivity feedback. Such changes in the core geometry and the associated reactivity decreases were shown in analyses of the FFTF reactor and verified by the experimental data. The ability of a SFR to incorporate a core restraint system design that introduces a negative reactivity feedback in response to power-to-flow ratios or temperatures above nominal was also shown to be a potentially significant contributor in achieving a safe inherent reactor response to accident conditions, resulting in benign termination of such postulated unprotected accidents.

In general, the number and location of the subassembly load pads and core restraint rings is a design option and has varied from one design to the next, as did the number of core restraints. As described, certain arrangements will result in a more favorable reactivity feedback due to radial core expansion, enhancing the 'inherent safety' characteristics of the reactor.

3.1.5 Control Rod Driveline Design for Negative Reactivity Feedback

The control rod drivelines connect the control rod drives located outside of the reactor with the control rods in the reactor core. The control rod drives are typically located above the reactor vessel, either on the reactor vessel closure for a pool-type plant or on the reactor vessel head for a loop-type plant. In both cases, the drivelines are inside the reactor vessel, and pass through the outlet sodium plenum directly above the subassemblies. As a consequence, the drivelines are heated by the sodium leaving the core. At steady-state conditions, the drivelines are heated to the nominal temperature of the outlet plenum, which determines their length. The control rod drives position the control rods in the core by moving the control rod drivelines, so that a given control rod position in the core will be associated with a length of the control rod driveline and the position of the control rod drive itself. If the temperature of the sodium exiting the core subassemblies changes, the temperature of the control rod driveline will also change, which will in turn change the length of the control rod driveline, and as a result the position of the control rod in the core and the reactor core reactivity. Thus, an increase in the outlet plenum sodium temperature will effectively insert the control rods further into the core to provide a negative reactivity feedback. Since an increase in sodium temperature will occur in many accident situations, the thermal expansion of the control rod drivelines has the potential to be a significant contributor to the negative reactivity feedback that is needed to allow the reactor to inherently respond to such accidents leading to a benign termination. There is also a relation between the control rod mechanism expansion and the axial fuel expansion discussed above, as noted earlier in Section 2.2.1. As the fuel expands upward due to increased fuel and cladding temperatures, this effectively causes an insertion of the control rod adding negative reactivity.

However, if the control rod drives are supported on the vessel head, while the core is supported by the vessel walls, the relative position of the control rod drives and the reactor core is determined by the reactor vessel itself, and the heating of the reactor vessel walls will also change the relative position of the core and control rods. For example, if the reactor vessel temperature increases, the core will be lowered away from the control rods, which is equivalent to withdrawing the control rods from the core, and will lead to a positive reactivity addition. As a result, the net reactivity effect of the inherent movement of the control rod drive line by the entire reactor vessel and control rod drive system in response to temperature changes is determined by the sum of these two reactivity components. The design of the control rod driveline provides an opportunity to increase negative reactivity feedback in response to core temperature changes. In all cases, though, it is important to note that the heating of the control rod driveline can occur very rapidly, as in the early stages of an accident, while the heating of the reactor vessel occurs over an extended time. The reactivity feedback from relative motion between the control rods and the core is often seen to be significantly negative shortly after the initiation of an accident, providing substantial negative reactivity feedback to reduce core power, but eventually can become less negative, or even slightly positive at times well after the initiation of the accident if there is a large increase in the temperature of the reactor vessel. Conditions in the reactor are changing very slowly by that time.

3.1.6 The Impact of Sodium 'Void' Worth

As discussed in an earlier section, the effect on reactor core reactivity due to sodium boiling, which would replace the liquid sodium coolant with sodium vapor ('voiding') can be very large depending on the reactor design. For a compact, 'neutronically-efficient', fast reactor core, where the ratio of the surface area of the envelope of the active core to the volume of the active core is relatively small, the total sodium void worth can be as high as 6-8$ of positive reactivity, and can be a function of design choices such as the type of fuel (oxide, metal, carbide, nitride). The void worth varies considerably with location in the core, from significantly positive near the center, to much lower values or even negative values at the top and bottom of the core.

The possibility of large-scale voiding as a result of extremely unlikely or unforeseen events, e.g. multiple ruptures of inlet coolant pipes, cannot be entirely dismissed. Such voiding has the potential to introduce substantial positive reactivity ($4-$6) in current SFR designs and could present an obstacle to their licensing because of the potential for severe reactor damage and of the resulting risk to the public. For this reason, there remains a strong incentive to minimize the reactivity that can be added if sodium boiling is initiated, and thus to minimize the consequences of voiding in the extremely unlikely event that it occurs [3-32]. Since the sodium void worth is affected by the design of the reactor core, a large number of studies [3-33, 3-34] have evaluated the sodium void worth characteristics of different reactor concepts using different core geometries and compositions to reduce the sodium void reactivity. For core geometries that have a larger surface area of the envelope of the active core relative to the core volume, the sodium void worth can be lower, or even negative. This is typically accomplished by shortening the height of the active fuel region of the core while increasing the diameter so that the core volume is the same. The difficulty with measures to reduce the sodium void worth is that they also affect other aspects of fast reactor performance. A reactor with a lower sodium void worth typically has:

1. a larger change in reactivity during an irradiation cycle, leading to the need for more control rods, possibly of higher individual reactivity worth;
2. a shorter irradiation time between refueling, leading to a lower capacity factor;
3. a physically-larger reactor for the same power generation.

The design changes for reduced void worth tend to penalize the overall reactor safety performance. For example, the larger reactivity change increases reliance on reactivity control, and a short core increases the sensitivity to control rod motion. These trends are unfavorable to safety performance in several classes of accidents, such as TOPs and seismic events.

The question is one of determining whether a reactor core with a positive sodium void worth, even as high as 6-8$, represents a significant difference in safety performance from a reactor core with a much smaller or negative sodium void worth. In making such an evaluation, it is important to remember that the sodium void reactivity is only one component of overall reactivity feedback, and it is the total reactivity feedback that will determine accident progression and consequences. In the PRISM PSER Final report [2-14] the NRC staff has concluded that the consequences of events that could lead to core damage as a result of the positive void coefficient should be analyzed, and the results of these analyses considered in the context of the overall risk of each reactor design.

The reason that the sodium void worth is a significant safety concern is that once sodium boiling is initiated, as may occur only in very low probability accidents, the increase in core reactivity due to sodium voiding will result in a rapid increase in reactor core power. At the same time the fuel is no longer adequately cooled since sodium vapor is much less effective at removing heat than liquid sodium and, in such accidents as the unprotected LOF, the coolant flow rate decreases. Depending on the detailed evolution of the accident sequence, the fuel pin will tend to overheat leading to fuel and cladding melting and relocation. Because the fuel relocation in the coolant channels, following cladding failure, is generally dispersive introducing large amount of negative reactivity, the importance of fuel relocation reactivity effects tends to quickly overcome the importance of the sodium voiding effects. However, during a very short period following the cladding failure (a few milliseconds) the in-pin fuel relocation effects are of comparable magnitude with and may even dominate the negative reactivity effect of the still accelerating dispersive fuel motion in the coolant channel. If the cladding failure is located near the core center line, the in-pin fuel relocation may introduce positive reactivity, with the net result being that for a very short period following the cladding failure the net fuel reactivity contribution decreases very slowly or is slightly positive. These reactivity effects are discussed in more detail in the subsequent sections of this report. However, the rate of insertion of sodium void reactivity plays an important role in determining the net reactivity and power level at which the fuel melting and relocation is initiated, which in turn determines the amount of energy produced before the negative fuel reactivity will shut down the reactor. If the sodium voiding reactivity is such that the fuel pin failure occurs when the core reactivity is near one, the very early fuel positive reactivity contribution, combined with the continued sodium voiding, can lead to a short lived power excursion which is terminated by the negative reactivity due to the fuel dispersal in the coolant channels, which quickly becomes dominant. On the other hand, if the pin failure occurs at a reactivity level which is well below prompt critical the initial fuel relocation reactivity contribution and continued sodium voiding are not sufficient to increase the reactivity above prompt critical before the negative reactivity due to fuel dispersal in the fuel channels becomes dominant, and a power excursion does not occur.

The amount of sodium voiding reactivity that may be introduced during an accident if sodium boiling were to occur depends on both the spatial location and extent of the boiling and on its timing. First, the sodium void worth has a strong spatial dependence in the reactor core. The central region of the core typically has a high positive sodium void worth, while the outer regions have a negative sodium void worth, even for a core that has an overall significantly positive sodium void worth. This spatial variation in sodium void worth is extremely important in understanding the impact of sodium voiding on accident progression. Second, by definition, sodium voiding during an accident will occur first where the local sodium temperature reaches the local sodium boiling point. Since the boiling point is a function of pressure, the sodium boiling point is lowest at the top of the core and highest at the bottom of the core even under static conditions. When full coolant flow is still available, the pressure drop across the core increases this difference in boiling point between the top and the bottom of the core. Third, due to the coolant flow upwards through the core, the sodium temperature increases as the sodium moves from the bottom of the core to the top of the core, with the result that the highest sodium temperature is at the top of the core under normal conditions. All of these contribute to a predisposition for coolant boiling to begin at or near the top of the core, depending on the accident conditions.

Details of the accident sequence must also be examined. First, we consider the accidents where coolant flow is maintained, such as the unprotected TOP where there is an uncontrolled withdrawal of a control rod and a failure to scram, and the unprotected LOHS where the reactor is isolated from the balance-of-plant and there is a failure to scram. As core temperatures increase in response to such events, if the coolant boiling point is exceeded, it will occur first at or near the top of the core, where the sodium void worth is lower, except possibly in the case of very high reactivity addition where the power increase is much faster than the time required for the sodium coolant to move through the core. It should be noted that accidents such as the unprotected TOP and LOHS do not result in such serious consequences if the reactor is designed to make best use of the favorable reactivity feedbacks. More serious accidents are typically required to cause sodium boiling, such as the uncontrolled withdrawal of all control rods from the core with a failure to scram, or loss of all heat removal capability from the reactor, including all backup auxiliary and decay heat removal systems, with failure to scram.

The differences between subassemblies must also be considered. Not all subassemblies have the same power, flow rate, and temperatures due to the range of relative power and flow that occur in a fast reactor core. Higher coolant flow rate is provided for subassemblies that have higher power, with the goal of keeping the core fuel and cladding temperatures within limits while producing the highest amount of power. This typically results in a range of power-to-flow ratio for driver fuel subassemblies, and also for internal and radial blanket subassemblies if they are used. The effect of this range is that there are a few subassemblies that will have the highest temperatures at nominal steady-state conditions, and will experience the fastest rise in temperature during an accident. The result is that when sodium boiling first occurs, it only occurs in a few assemblies, not the entire reactor core. The initial sodium voiding reactivity feedback is that generated only by these few assemblies, and is only a small fraction of the sodium void worth for the entire core.

At this point, it becomes necessary to examine the accident progression following such limited sodium boiling. This can be affected by a number of design choices, including the type of fuel (oxide, metal, nitride, carbide, etc.), since sodium boiling inevitably will lead to some amount of fuel melting, cladding failure and fuel relocation. As noted above, the fuel relocation quickly becomes dispersive and introduces large amounts of negative reactivity, dominating the sodium voiding reactivity effects. However, depending on the cladding failure location, as described in sections below, the early fuel motion may introduce only small amounts of negative reactivity or may even introduce a limited amount of positive reactivity. In the case of unprotected accident where the coolant flow is maintained, the pin failure location tends to be located in the upper part of the pin, above the core mid-plane, leading to an early negative reactivity contribution due to fuel relocation. For a reactor design where the fuel and cladding properties favor continued longer term dispersive fuel relocation and the time interval between boiling initiation in the highest power-to-flow ratio assemblies and the next group of assemblies approaching boiling is long enough to allow the fuel relocation reactivity to become strongly negative, boiling in the remaining assemblies may be reduced or inhibited, thus limiting the amount of sodium void worth introduced and leading to a benign shutdown.

A different situation exists for accidents where coolant flow is lost, such as in an unprotected LOF where there is a loss of offsite power that is typically used to run the coolant pumps along with a failure to scram the reactor. In this case, as a result of the flow coastdown, core temperatures increase and reactor power automatically reduces as a result of the reactivity feedback. It is possible to have sodium boiling if the power reduction is not rapid enough with respect to the flow coastdown, which is the reason that some reactor designs have extended flow coastdown mechanisms such as increased pump inertia or by electronically controlling the pumps to best use the available inertia and avoid coolant boiling for this accident initiator. For modern designs, more severe accident initiating conditions are required to lead to coolant boiling, such as instantaneous failure of all of the coolant pumps with no coastdown and failure to scram the reactor, as might occur in an extremely large earthquake for example. With the very rapid reduction in flow rate, the axial coolant temperature profile through the core begins to change rapidly, with the result that sodium boiling may begin much below the top of the core region. In this case, although only a few assemblies will boil at first, the reactivity addition may be larger than for cases where boiling begins at the top of the assembly, partly because the boiling begins in a region of higher sodium void worth, and partly because there is little flow remaining to prevent the boiling region from extending rapidly downward to the core midplane and below.

Similar to the unprotected TOP and LOHS, there will be subsequent fuel and cladding melting, fuel pin failure, and fuel and cladding relocation. The dispersive fuel relocation in the coolant channels that introduces large amount of negative reactivity soon becomes dominant, but the early fuel reactivity contribution depends on the axial failure location and may be slightly positive for a short time in the case of midplane failures. If the fuel properties and reactor design favor continued longer term dispersive fuel relocation the core heterogeneity may allow enough time for the fuel dispersal to become dominant in the lead assemblies and reduce the reactivity and power level before the occurrence of extensive sodium boiling in other fuel assemblies. The further progression of the accident will depend on the axial pin failure locations and on the core reactivity level at the time of failure, which depend on the choice of fuel type and other design details.

Thus, to answer the question on the safety significance of the sodium void worth, it is necessary to perform detailed transient analyses of reactor response for the extremely severe accidents of very low probability that may result in sodium voiding and subsequent fuel relocation. It is essential to have accurate models for the phenomena that may occur in such events, and there has been significant effort in the past to simulate such conditions and to obtain the experimental data needed to support development of such models, as described below in Section 5.

Typical sequences of events for an unprotected LOF accident have been outlined for both oxide-fueled and metal-fueled cores in Section 2. In developing the reactor model, it is important to include all of the features that directly affect the molten fuel relocation, such as the presence of axial blankets and/or other structures that can promote fuel freezing and blockage formation, and special fuel assembly features designed to facilitate the axial fuel relocation such as the FAIDUS concept discussed in Section 3.2.2 below.

The safety significance of the sodium void worth depends on the reactor design and design choices. With proper design and suitable choices, it is possible to relegate the significance of the sodium void worth to accidents of extremely low probability, below those typically considered for reactor licensing and possibly relevant only for affecting the generation of a potential source term for the containment.

3.2 Mitigation of Accident Consequences After the Onset of Irreversible Core Geometry Changes

3.2.1 In-Pin Fuel Relocation Prior to Cladding Failure in Transients without Scram

The in-pin molten fuel relocation, which rapidly introduces negative reactivity during very low probability accidents that are progressing towards significant core damage, can play an important role in mitigating the accident consequences when it occurs prior to cladding failure and fuel ejection into the coolant channel. The sequence of events is outlined below [3-21].

During both the LOF and TOP postulated accidents, the mismatch between energy generated in the fuel pin and the energy removed by the coolant can lead to overheating of the fuel pin. During the early period, limited fuel relocation occurs due to the axial expansion of the fuel pin, which introduces limited negative reactivity. As the accident proceeds, the inside of the fuel pin begins to melt at the centerline of the pin at an axial location where fuel temperatures are highest, leading to the formation of an internal cavity as shown in Figure 9. This cavity is filled with a mixture of molten fuel and fission gas and grows continuously, both radially and axially, due to continued fuel melting. The fuel-gas mixture in the molten cavity is pressurized due to the presence of fission gas released from the fuel and increasing temperatures and can move under the influence of local pressure gradients. During this period fuel relocation occurs due to both axial expansion of the solid fuel pin and the in-pin hydrodynamic relocation of the molten fuel. As long as the cavity maintains a bottled-up configuration the hydrodynamic fuel relocation is due only to the compressibility of the molten fuel and fission gas mixture. This fuel relocation is limited and tends to introduce an amount of negative reactivity comparable in magnitude to the negative reactivity introduced by the axial expansion of the solid fuel. As the cavity walls continue to melt there is a competition between two effects illustrated in Figure 10:

a. The axial extension of the cavity, which can cause the cavity to reach the top of the fuel pin. When this happens, the pressurized molten fuel in the cavity is connected to the lower pressure in the upper pin plenum and can relocate suddenly, leading to a large insertion of negative reactivity;

b. The radial expansion of the cavity and cladding pressurization and melting which can cause cladding failure. When cladding failure occurs, the inner cavity is connected to the coolant channel which is at a significantly lower pressure and the molten fuel inside the pin cavity is accelerated rapidly toward the cladding failure location. This initial in-pin fuel relocation can have either a negative or a positive reactivity contribution, depending on the location and extent of the failure and on the axial failure propagation. The molten fuel is ejected into the coolant channel where it is dispersed axially. This axial fuel dispersal leads to a large insertion of negative reactivity and quickly becomes the dominant reactivity effect. If the continued fuel dispersal is not stopped at a later time by fuel freezing and channel blockage formation, the associated negative reactivity insertion leads to neutronic shutdown of the reactor.

It is noted that during a brief time after cladding failure the reactivity effect due to in-pin fuel relocation, which may introduce positive reactivity if the failure location is near the core centerline, can dominate the reactivity effect due to the fuel dispersal in the coolant channel. This can lead to a net positive reactivity insertion due to fuel relocation if the failure location is near the core centerline. If the cladding failure occurs when the reactor is near prompt critical, as tend to be the case in some rapid unprotected LOF events, the post-failure fuel relocation can lead to a short prompt critical power burst, which is terminated by the rapid fuel dispersal in the coolant channel. If the in-pin fuel relocation is initiated prior to cladding failure, however, it acts as a "fuse" introducing rapidly a significant amount of negative reactivity. Even if the cladding failure occurs later, the reactivity is significantly below prompt-critical at the time of the failure, and the early post-failure fuel relocation limited reactivity effect is not sufficient to lead to a critical condition and the associated power burst.

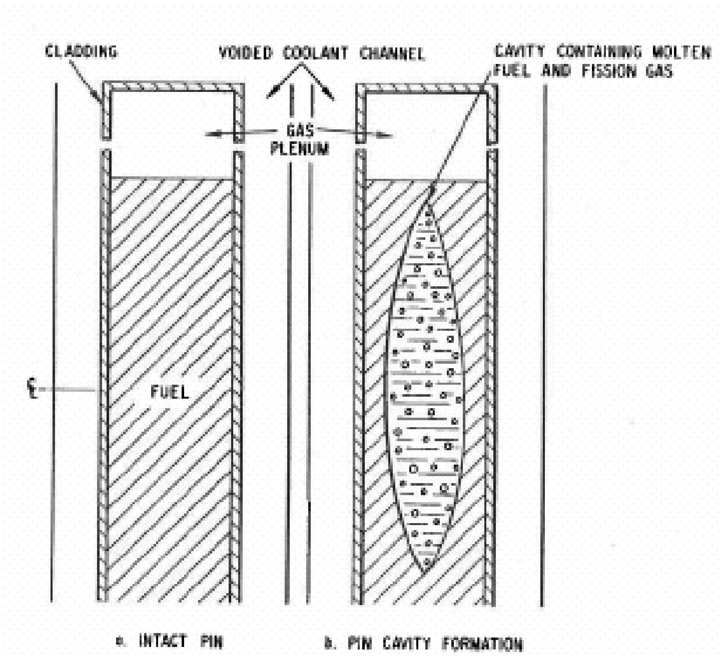

Figure 9 Cavity Formation During the Initial Accident Phase

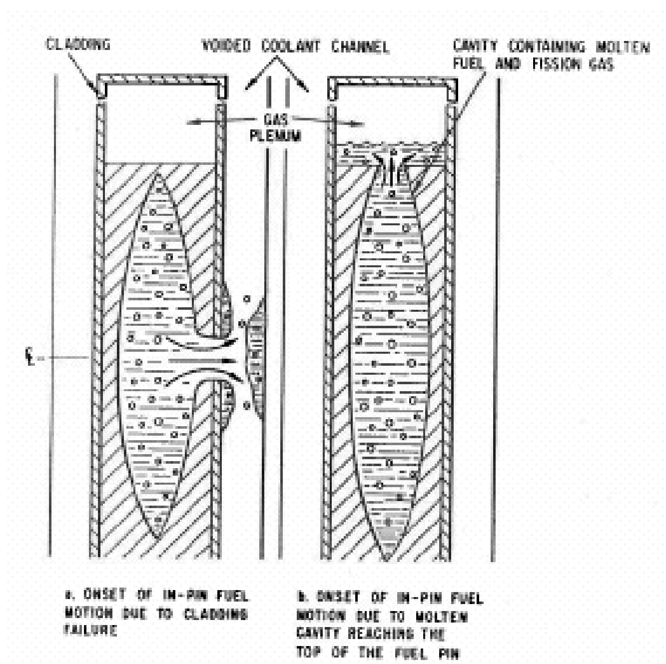

Figure 10 Molten Fuel Relocation Modes

In metal-fueled cores, due to the metal fuel high conductivity, the molten region that develops inside the fuel pin during postulated severe accidents tends to be biased towards the upper part of the fuel pin much more than the corresponding molten cavity in oxide fuel pins. This is due to the higher thermal conductivity of the metal fuel, which causes the temperature in the metal fuel pin to more closely follow the axial coolant temperature profile while the temperature in the oxide fuel pin more closely follows the axial power shape, which peaks at or near the core midplane. This favors the possibility that during many severe accident sequences the molten fuel region will extend to the top of the fuel column prior to cladding failure, leading to the onset of rapid in-pin fuel relocation that acts as a self-limiting accident mitigation feature [3-22]. This phenomenon is less likely to occur in oxide fuel cores, where the molten fuel region is generally closer to the core mid-plane and cladding failure tends to occur before the onset of rapid in-pin fuel relocation. Research has been conducted to evaluate the extent in-pin molten fuel relocation within annular oxide-fuel pins [3-23]. Experiments performed with annular fuel pins have shown that the effectiveness of in-pin fuel motion may be limited by fuel freezing in the colder regions of the fuel pin leading to blockage of the internal channel, and the potential early escape of the fission gas which, for regular fuel pins, is retained in the molten cavity.

In addition to the fuel type, the amount of fuel that can relocate upwards from the molten cavity depends of the fuel pin design. Thus, in a fuel element with an upper blanket and dimples that limit the blanket movement, the amount of fuel ejected from the molten fuel cavity in the driver fuel could be small, with the in-pin fuel relocation limited by the upper blanket material. In a fuel pin without an upper blanket the early rapid in-pin fuel relocation is un-constrained and controlled by the pressure difference between the molten cavity and plenum. The fuel pins used in transmutation reactor designs will not have axial blankets, thus facilitating the rapid in-pin fuel relocation and the associated negative reactivity feedback.

3.2.2 Modified Fuel Assembly Design for Enhanced Fuel Relocation

The concept of a modified fuel assembly for enhanced fuel relocation has been developed at JAEA in conjunction with oxide-fueled cores, and the discussion in this section is focused on analyses and results obtained for such cores. Extensive analyses of the Initiated Phase events in oxide-fueled cores have shown that the potential energy deposition during this phase is limited due to the dispersive nature of the fuel motion soon after the fuel cladding failure and the absence of energetic fuel-coolant interactions (FCI). However, in the later stages of the Initiating Phase, which is characterized by intact fuel assembly walls and a lack of inter-assembly fuel relocation, the dispersal of the fuel can be reduced and even stopped due to the molten fuel and/or cladding freezing and channel blockage formation in the colder regions of the assembly. The difficulty of accident termination in the Initiating Phase is due to the high heat capacity provided by the upper and lower axial blanket regions, which can result in temporary freezing of the molten cladding and/or fuel and blockage formation. This has shifted the emphasis from Initiating Phase energetics to concerns related to the potential energetic re-criticality events in the Transition Phase, which include the subassembly disruption, onset of inter-assembly material relocation, pool formation, and fuel escape to the lower and upper plenum. In a best-estimate scenario for an oxide-fueled core [3-17], where molten fuel and cladding re-freezing occurred in the colder regions of the assembly, the resulting bottling up of the fuel in the active core region gave rise to rapid radial attack of the subassembly hexcan walls, formation of liquid fuel pools and potential energetic re-criticality events. The accident terminated in the Transition Phase due primarily to the downward fuel relocation through the Control Rod Guide Tubes (CRGT) following melt-through of the hexcan walls, as illustrated in Figure 11a. It is noted that although the core may include fuel pins without blankets, reducing the potential for fuel freezing and blockage formation, the potential for terminating such an accident prior to the Transition Phase will have to be evaluated for the specific design proposed.

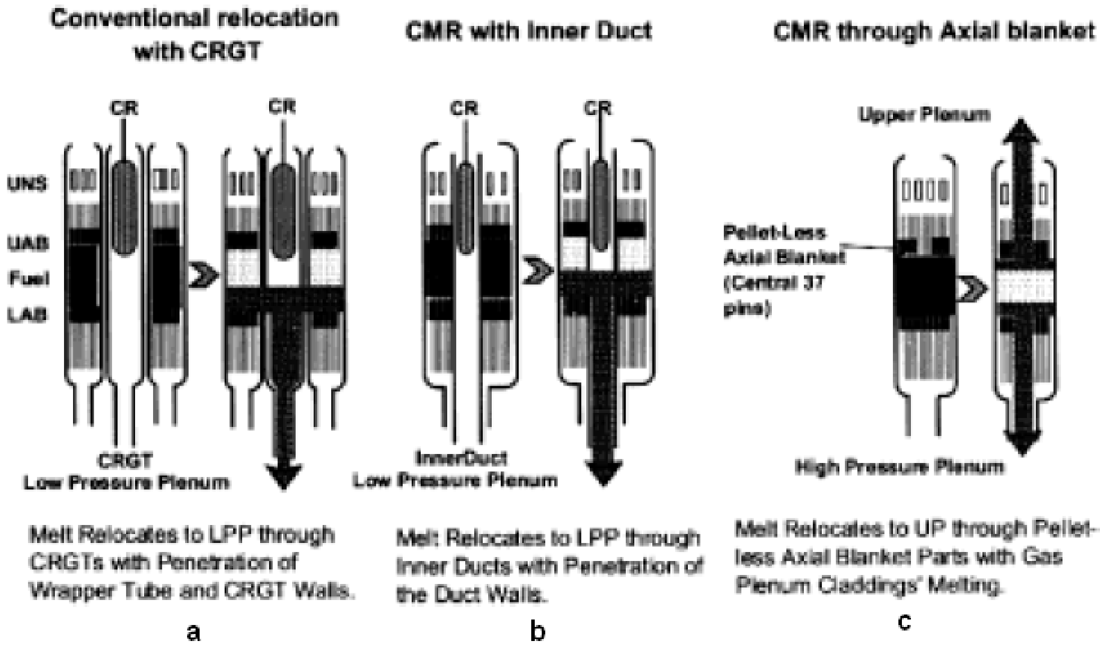

Figure 11 Post-Accident Material Relocation with CMR

The results of these severe accident assessments of fast reactor designs using oxide fuel have led JSFR designers to propose design changes offering Controlled Material Relocation (CMR) [3-18, 3-19] with the goal of achieving an early benign accident termination in the Initiating Phase, avoiding the potential of the bottled-up core scenario. CDA termination in the Initiating Phase means that the subassembly hexcan wall have not been breached and the absence of inter-assembly material relocation, and eliminates the potential for significant fuel pool formation and associated concerns related to energetic "sloshing" pool re-criticality events. Furthermore, by introducing CMR it becomes possible to enhance the potential for inherent in-vessel debris coolability. Several CMR approaches have been studied, including: a) Fuel Assembly Inner Duct System (FAIDUS) concept, and b) Limited Blanket Removal Concept, which are described below.

One version of the FAIDUS concept is illustrated in Figure 11b. In contrast to the traditional subassembly design, the CRGTs are placed inside the driver subassemblies. Each driver subassembly contains an inner duct (with or without control rods), resulting in driver fuel assemblies with less fuel than traditional driver fuel assemblies. The balance of the beneficial and detrimental effects of such a significant change in the fuel subassembly must be considered in the reactor core design. The principal success criterion for accident termination in the Initiating Phase is the early failure of the thinner inner duct wall followed by downward fuel relocation in the extended Inner Duct, allowing fuel to escape from the core and relocate below the active core region. This fuel relocation introduces substantial negative reactivity and causes a rapid reduction in core power and temperature, thereby preventing the failure of the outer subassembly hexcan wall. Thus, it is necessary for the Inner Duct to fail first, before the outer subassembly hexcan wall would be breached, and with sufficient time provided for the downward molten fuel relocation to reduce the power level and avoid the hexcan wall breaching.

The complex physical phenomena that compete to determine the effectiveness of the CMR and the accident sequence must be well understood. An extensive experimental program, the EAGLE project, has been undertaken in Japan to confirm the capability of the fuel removal using the FAIDUS concept.

The Limited Blanket Removal (LBR) concept, illustrated in Figure 11c, relies on the removal of the axial blanket regions from a small fraction of the fuel pins in each subassembly, which reduces the heat capacity of the pins in that region and thus facilitates extended fuel relocation. The analysis presented in [3-17] estimates that removal of the upper and lower axial blanket fuel in the 37 centrally located fuel pins within the subassembly will ensure the extended fuel relocation that provides sufficient negative reactivity feedback to lower reactor power and prevent significant radial fuel attack on the subassembly hexcan structure, thereby assuring the termination of the Initiating Phase without Transition Phase initiation.

Post-accident material distributions in connection with the Subassembly Inner Duct and Limited Blanket Removal concepts differ primarily in terms of the potential extended fuel dispersal to the upper sodium plenum. The following Post-Accident Material Relocation (PAMR) distributions are suggested in [3-17] following a LOF event:

- Formation of a complete temporary blockage in the lower axial blanket, but leaving the 37-pin region without blanket pellets open in the LBR case, results in ~15% downward fuel removal based on a 1 m active core. This applies to both concepts.
- A similar fuel fraction (~15%) will move into the upper axial blanket resulting in freezing in the fuel pin region still containing the blanket material. In the case of the LBR concept the region containing the 37 pins without blanket pellets will remain open
- Due to the fuel crust formation on still intact hexcan walls following the fuel pin disruption about 20-30% of the fissile material will remain in the active core region. This scenario is applicable to both concepts.
- The rest of the fissile material (~45%) will be dispersed upward by fission gas and steel vaporization through the 37-empty-pin region in the LBR case, or relocated downward following the failure of the subassembly inner duct wall in the FAIDUS case. Some of the fuel dispersed upward will be plated out as fuel crust as it travels through the upper core region (~10%) with about 35% of the fuel entering the upper sodium plenum. In the absence of reflux cooling, about 50% of the active fuel is expected to relocate eventually to the lower sodium plenum in the case of the FAIDUS concept.

The analyses described above apply only to oxide fuel. Experimental results for metal fuel have demonstrated a different behavior. Due to the thermo-physical properties of the metal fuel and the formation of relatively low melting point alloys with cladding material during the accident sequence, the dispersal of fuel-containing materials in coolant channels is enhanced, and the tendency of forming blockages is reduced. For a reactor using metal fuel, the CMR approaches described above may not be necessary, although uncertainties about fuel relocation with metal fuel still exist, and similar approaches may need to be considered, depending on the outcome of further experiments.

3.2.3 Core Catchers

Core catchers have been studied for both advanced LWRs and SFRs. The name refers to devices intended to "catch" reactor debris in the event of a core meltdown as may occur in the case of a severe accident, containing and cooling the molten core materials. Several in-vessel and ex-vessel core catcher concepts proposed in the literature are reviewed in [3-20]. Some of the key approaches associated with these concepts include:

• *A geometry to contain relocated corium (i.e. the mixture of oxide fuel and core structural materials) that enhances cooling and freezing by lower plenum coolant*. Many designs "catch" relocated corium in a structure that can contain and cool the corium (through fins and narrow gap cooling).

• *High-temperature materials to reduce corium decay power density*. Some designs use materials that form a lower temperature eutectic with uranium dioxide in the corium and dilutes the decay heat power density in relocated materials.

• *Materials to chemically absorb fission products and associated decay heat*. Some designs use a glass material that catches and absorbs heat from relocating corium until it becomes a molten mass containing dissolved uranium and fission products.

• *Multiple structures to reduce corium decay heat and retain corium materials*. Some multiple structure designs incorporate a sacrificial layer above a container that can structurally support the cooled corium. As the corium forms a eutectic with the sacrificial layer, the power density of the corium material is lowered (reducing the heat load to the underlying container).

It is noted that these designs pertain to the oxide-fueled reactors. Scoping analyses with metal fuel have also been performed, but due to the different nature of the fuel relocation and the ultimate distribution of core materials following extremely severe accidents, the need for core catchers in reactors that use metal fuel has not been established at this time.

3.2.3.1 In-Vessel Core Catcher

If the molten fuel/steel mixture relocated from the core cannot be contained in a coolable configuration in the lower plenum, it could eventually make its way to the lower head of the reactor vessel, potentially causing it to fail due to melting or creep (if the molten fuel/steel mixture is not coolable in this location), and release the molten core material outside the primary system. The role of the in-vessel core catcher is to ensure that the relocated core material does not come into direct contact with the reactor vessel, but is retained in a sub-critical, coolable configuration inside the reactor vessel. This enhanced safety margin has the potential to improve the plant economics due to reduced regulatory requirements, and increase public acceptance due to reduced plant risk.

A core-catcher concept must meet several challenges: it must resist, for instance, high thermal loads, elevated temperatures (estimated at up to ~2500°C), and mechanical loads. The concept should be able to operate under passive conditions and should provide enough surfaces to ensure that the core material can retained in a sub-critical, coolable configuration. It must also cope with unexpected events, since the description of the core degradation and relocation to the lower regions of the reactor vessel is still subjected to many uncertainties. Last but not least, the implementation cost should be as low as possible. A multi-layered in-vessel core catcher considered for the JSFR is illustrated in Figure 12. A core catcher with neutron absorbers, illustrated in Figure 13, is being considered in France. The efficiency of the core catcher depends on multiple phenomena, all of which must be analyzed, including: a) debris bed formation and cooling, b) melt pool formation frozen crust formation at the lower boundary fuel/plate interface) and upper boundary (fuel/sodium interface), c) convection heat transfer in the molten pool and heat transfer to the frozen crust, d) neutronics and power generation in the debris bed and/or molten pool, e) external cooling of the lower head, f) mechanical behavior of the vessel.

The design goals of the in-vessel design catcher include [3-20]:

- The core catcher shall prevent re-criticality of relocated material
- The core catcher shall fit within the reactor vessel
- The core catcher shall reduce the decay heat power density of relocating corium
- The core catcher shall contain relocated material (size, structurally, thermal shock resistance)
- The core catcher shall be as inexpensive as possible
- The core catcher shall facilitate long-term coolability using passive means
- Interactions between core catcher materials and relocated debris shall not result in exothermic reactions or generation of combustible gases
- The core catcher shall not present a seismic hazard
- The core catcher shall be stable for the lifetime of the reactor
- The core catcher shall be easily installed and maintained
- The core catcher shall not adversely affect reactor performance or coolant circulation

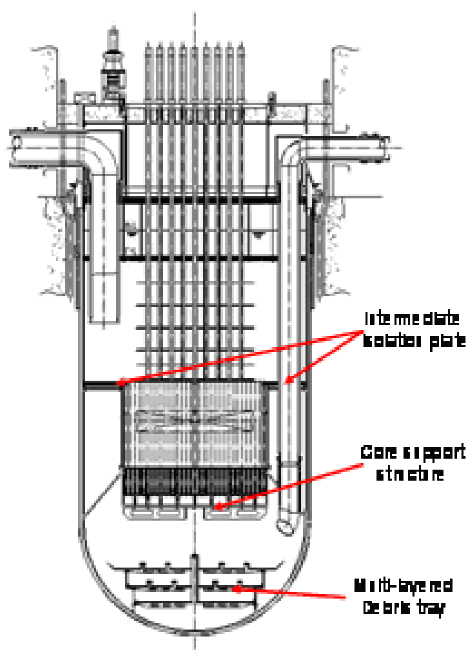

Figure 12 Schematic View of JSFR with Multi-Layered Debris Tray

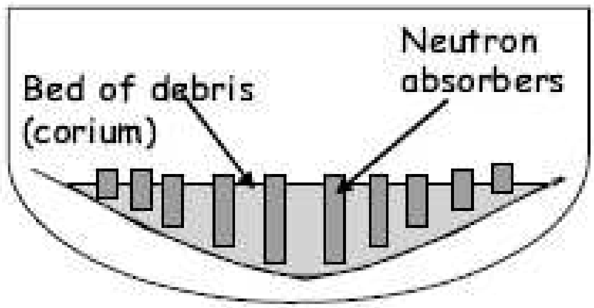

Figure 13 Core Catcher with Neutron Absorbers

3.2.3.2 Ex-Vessel Core Catcher

Ex-Vessel Core Catchers are designed to mitigate the accident consequences by containing and cooling the core melt that would leave the reactor vessel in case of a hypothetical CDA leading to vessel melt-through. The role of an ex-vessel core-catcher is to prevent the foundation erosion and to stabilize and control the corium within the containment. The objective is to preserve the integrity of the containment as the main barrier of fission product release to the environment. The conditions existing shortly after a postulated failure of the reactor vessel are highly uncertain. It is possible that a stream of debris falling onto the reactor cell floor may locally fail the steel liner. This would expose the concrete to chemical attack by core debris and sodium. The expected chemical reactions are exothermic, generating both gases and aerosols and consuming primary system sodium. Calculations indicate that the conductive heat transfer into the concrete wall of the reactor cell is not sufficient to prevent continuous sodium boiling [3-24]. Complete sodium boil-off from the reactor cell would occur within approximately 100 hours, and may take only 28 hours in the case of complete liner failure. The core debris is expected to spread into a thin layer at the bottom of the reactor cell. No penetration into the concrete structure would be expected if a particle layer forms. A molten layer will cause limited melting attack. The transient containment response after reactor vessel failure has been studied in detail. The areas of main interest include concrete dehydration, concrete interactions with molten fuel, steel, and sodium [3-25], containment pressure and temperature time histories, and radiological consequences. Parametric studies have revealed a high sensitivity to the amount of water present at the accident site. This water is contained in the concrete structures in the form of free water, adsorbed water, interlayer water, and bound water. Interactions in the reactor cell could be minimized by insulating the concrete structure with a layer of refractory oxides such as Al_2O_3, which would also reduce the water release into the reactor cell following liner failure.

Much of the current research is focused on advanced water reactors, with the Corium Spreading and Coolability (CSC) project [3-26] as an example. The two generic processes studied by the CSC project include: a) spreading of the corium under various conditions, and b) cooling of the melt. The cooling of the molten corium using direct water contact, studied for advanced LWRs, is not applicable to SFR ex-vessel core-catchers, since there would not be any cooling water in this region, but liquid or gaseous sodium instead. However, the information on corium spreading and freezing and the analytical tools developed by this project will be valuable in the evaluation of SFR ex-vessel corium behavior.

3.2.4 Inerted Head Cavity Isolation Domes [3-2, 3-5]

The capability to seal the reactor head compartment was one of the design fallback features which was considered and evaluated during the FFTF regulatory review. For additional safety margin it was thought that this fallback design feature could be implemented for the purpose of imposing an additional barrier to the release of sodium and fission products from the reactor to the containment building should the need for an additional barrier be identified .The evaluation was based on assessments of HCDA energetics, mechanical consequences of sodium impact and predicted head lift, corresponding sodium expulsion from the reactor and offsite radiological consequences for a range of sodium and radioisotope release assumptions. There is the potential for inerting this sealed cavity to mitigate the possibility of sodium combustion. The FFTF project conclusion was that the sealed compartment head would not provide a substantial increase in containment performance under HCDA conditions but would have serious adverse effect on real operational safety. No further technology development work was therefore done on this concept.

Subsequently, ALMR introduced the concept of a containment boundary consisting of a containment vessel surrounding the reactor vessel connected to a low-leakage pressure retaining containment dome above the reactor vessel head. This provides a low-pressure/low volume controlled leakage barrier around the primary system. The containment dome is fitted with ports to allow maintenance access. Refueling and the removal and replacement of small equipment could be carried out. During normal operation, the design provided for the use of mechanically secured seal plugs to seal the access ports. Preliminary design assessments were performed which showed that the design should withstand the effects of extreme events. A large primary breach with release of fission products to the containment dome accompanied by a sodium fire, which would continue until all the containment volume oxygen is consumed, would have margin for containment pressure and temperature conditions and the releases are within the Protective Action Guideline limits.

With the termination of the ALMR project no further work was done beyond the preliminary design assessments.

3.2.5 Filtered/Vented Containment; Leak Detection and Fire Suppression Systems [3-6, 3-7 through 3-12]

There has been considerable design, construction and operational experience with LMR containment systems, both international and US, but it has been concentrated on low-leakage/ pressure retaining containments. The focus has not been on the high leakage filtered/vented containment concept. As such, the safety design bases for the containment design have included the criteria that the containment system should withstand the pressure and temperature resulting from the consequences of the maximum postulated sodium leakage. However, there exists a considerable body of work on the consequences of beyond-design-basis events which involved hypothetical core disruptive accidents for fast reactors using oxide fuel. In this extreme limit, analyses results can be found that show the need for venting to prevent excessive pressurization of the containment after meeting the NRC requirement on an initial period of containment integrity with low leakage. Recommendations were made for filtering, cavity cooling and sacrificial beds. Some of the venting/filtering concepts were incorporated into designs. There is a database on phenomenon such as sodium-concrete interaction, decomposition of concrete, generation of hydrogen, aerosol production, pool scrubbing and containment over pressurization, and mitigating systems such as containment vents, purge systems, containment cleanup, re-combiner, and re-venting.

Given that a sodium leak has occurred in connection with a CDA that eventually resulted in vessel failure, then mitigation of the consequences of such a leak is governed by the early detection of that leak and the prompt application of the fire suppression systems. This section focuses on the detection and mitigation of liquid sodium metal leaks into a gas (inert or air) environment external to guard piping and vessels. The detection and mitigation of liquid sodium metal leaks into water is a separate subject and is not discussed here.

Technology measures for the detection of sodium coolant leaks into a gas atmosphere encompass a variety of sensor and instrumentation depending upon the size of the leak. Small leaks (below 100 gm/hr) are detected through the use of sodium aerosol detectors and radiation monitors. The category of aerosol detectors includes sodium ionization detectors and plugging filter aerosol detectors. The detection time ranges from 10 hours to 100 hours. Chemical analysis for atmosphere monitoring in selected cells can also be used. Continuity detector of the contact (spark-plug) and cable types, which depend upon liquid sodium causing electrical shorts between electrodes, do not work reliably in this range. Continuity detectors, radiation monitors, and aerosol detectors can all be used to detect large leaks (20 kg/min to 100 kg/min). Large leaks would also result in measurable changes in cell pressures and temperatures in minutes. Temperature and pressure sensors can used to detect this range of leaks. Depending upon the particular sodium system which is leaking, changes in the system sodium level process instrumentation can be also be used. Intermediate leaks (20 kg/min to 100 gm/hr), as can be expected, are detected by a combination of the measures for large leak and small leak detection. In an air atmosphere, smoke detectors can also be utilized.

Upon detection of a leak there are operator procedures which call for manually initiated quick-draining of the affected system depending upon event conditions but there other design measure which suppress the consequences resulting from the chemical activity of sodium in the presence of air, water or concrete. These measures can be categorized as passive or active depending upon the need for operator action. Basic to all design is the compartmentalization of the areas with sodium systems into cells. These cells are heavy concrete structures lined with steel plate. In this way, the partitioning of the floor area limits the spreading of spilled sodium and the concrete is protected from interactions with the liquid sodium. The cells with the radioactive primary sodium system have an inert gas atmosphere while the non-radioactive secondary sodium systems are in cells with an air atmosphere. The primary system cells therefore have an added level of assurance against sodium combustion in the case of primary coolant leaks. In addition to these design measures, the sodium fire protection system also includes catch pan systems and fire extinguishers. Portable sodium carbonate fire extinguishers have been proposed. Designs have also included secondary sodium inert atmosphere blanketing systems.

The catch pan system is designed to mitigate the consequences of a sodium spill in an air-filled cell. It consists of steel catch pans and steel fire suppression decks. The catch pan consists of carbon steel plate assemblies which cover the cell floor and extend vertically up the wall to a height of ~feet above the maximum sodium level in the catch pan to prevent liquid flow over the edge of the pan. The fire suppression deck is essentially an airtight cover for the catch pan. Drain pipes are welded to the deck and extend downward to a point approximately ½ inch above the catch pan plate. Liquid metal spillage drains from the deck into the catch plan and as the drain pipe is partially filled, the effective burning surface is reduced to the cross section of the drain pipe. When the pipes are blocked with combustion products, air is prevented from reaching the liquid metal surface.

All this technology is commercial-off-the-shelf (COTS), readily available and has been implemented in a number of plants which were built .There is an R&D base of sodium combustion experiments. There is also operating experience as a number of sodium spill incidents have occurred. The failure of the thermo-couple well tube on the intermediate sodium transport loop at the Monju reactor in Japan is probably the event which has been most widely-reported and evaluated. It provided a full scale test of the sodium leak detection and fire suppression technology. The failure of the well tube, while the reactor was being brought up to 40% power for a plant trip test, led to sodium leaking into the Sodium Heat Transport System (SHTS) Loop C piping room and consequently, a sodium fire incident. The sodium high temperature alarm, a smoke detector alarm and a sodium leak detector alarm all activated pointing to a sodium leak in the piping room of SHTS Loop C. The presence of smoke was confirmed when the piping door room was opened, and the plant was taken into shut down mode with initiation of draining of loop C. A semicircular mound (~300kg) of sodium compounds was found on the steel floor liner but no damage was observed except to a ventilation duct and an access walkway. Post–incident examination estimated that the liner thickness had been reduced by 0.5 mm to 1.5mm. The concrete wall was dehydrated locally but the structural strength was not affected. Sodium compounds were found over the entire floor of the steam generator room connected to the piping room. It was recognized by the investigating working groups and task forces after the incident that there were deficiencies in operator timely response in diagnosing and monitoring the event. These were attributed to deficiencies in the operator procedures but the follow-on Monju improvement program also included enhancements to the sodium leak detection and fire detection systems. Improvement in mitigation measures also included the ventilation system. A nitrogen gas injection system was to be introduced in order to extinguish sodium fires more rapidly and thermal insulation was to be added to the concrete walls for protection.

4. REVIEW OF APPROACHES TO SFR SEVERE ACCIDENTS

This section provides an overview of the approaches to SFR severe accidents in Japan, France, and the USA. Both Japan and France have focused their attention on the oxide-fuelled SFR as the primary target for a future commercial fast reactor. The US, after studying the oxide-fuelled SFR for many years, has focused the research effort on the metal-fueled SFR as a promising target in the search for an inherently safe fast reactor.

4.1 Review of the Japanese SFR Safety Approach - JSFR

Japan Atomic Energy Agency (JAEA) initiated the Feasibility Study (FS) on commercial Fast Reactor (FR) Cycle Systems in 1999. The goal of the project, which includes the participation of all concerned parties in Japan, is to establish by 2015 the fast reactor cycle technology that enables the fast reactor system to become a future primary energy source with highly competitive characteristics [4-1].

A large-scale sodium-cooled fast reactor named JAEA Sodium-cooled FR (JSFR) has been developed through the FS that achieves all the development targets. JSFR is a sodium-cooled, MOX fueled fast reactor, with advanced loop type design that evolves from Japanese fast reactor technologies and experience. It is envisioned to have JSFR available for deployment by the year 2015.

4.1.1 Safety Targets and Design Principles

The sodium-cooled fast reactor (SFR) with oxide fuel is being developed in the Fast Reactor Cycle Technology Development (FACT) Project by a Japanese team that includes Japan Atomic Energy Agency (JAEA) and Japan Atomic Power Company (JAPC), aiming at its commercialization by 2050. The conventional safety approach to the severe accident issue is: 1) to minimize the occurrence probability of CDA by utilizing, for example, an additional passive self-actuated reactor shutdown system and 2) to assess the mechanical energy release due to hypothetical CDA events and confirm the intactness of the reactor vessel and containment. In addition to these efforts JAEA and JAPC have pursued the development of a recriticality-free core concept, with the aim of eliminating the recriticality event issue during postulated CDAs for oxide-fueled fast reactors.

The Japanese SFR safety targets were set aiming at world wide acceptance. The SFR design must ensure:

(1) A comparable or superior safety level to that of same-generation LWRs.
(2) A risk much lower than the risks encountered in daily activities, without taking into account the need for offsite emergency responses

These targets are consistent with the safety-related goals or user requirements in the Generation IV project and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO/IAEA).

The safety design principle is Defense-in-depth, with the following defense levels:

1. Prevention of abnormal occurrences
2. Control abnormal operations
3. Control accidents
4. Manage severe accidents
5. Offsite emergency response

While levels 2 and 3 above refer to Design Basis Events (DBE), the severe accidents referred to in level 4 are not DBEs but are considered Design Extension Conditions (DEC). A deterministic approach for both DBEs and DECs is taken into account for the system design, although appropriate design margins are provided by adopting conservative design evaluations for DBEs and by using best estimate design evaluations for DECs. The deterministic approach according to the Defense in Depth is adopted to specify safety functions, such as reactor shutdown system (RSS) and decay heat removal system (DHRS) for prevention of core damage. The defense-in-depth approach for JSFR [4-2] is illustrated in Fig. 14, while a schematic view of JSFR is shown in Fig. 15.

The defense levels 2 and 3 (DBE) rely on two independent Reactivity Shutdowns Systems (RSS), the primary RSS and the secondary RSS. The Decay Heat Removal System (DHRS) for these events includes redundant systems with passive operation, such as the current design with one Direct Reactor Cooling System (DRACS) and two Primary Reactor Auxiliary Cooling Systems (PRACS), illustrated in Figure 16. An earlier design with one DRACS and two Intermediate Reactor Auxiliary Cooling Systems (IRACS) is illustrated in Figure 17. An evaluation of the two DHRS has shown that the system with one DRACS and two PRACS provides an improved stability of the sodium natural circulation [4-1]. Additional safety features of the JSFR primary cooling system include a reactor vessel without pipe penetrations and double wall pipe design.

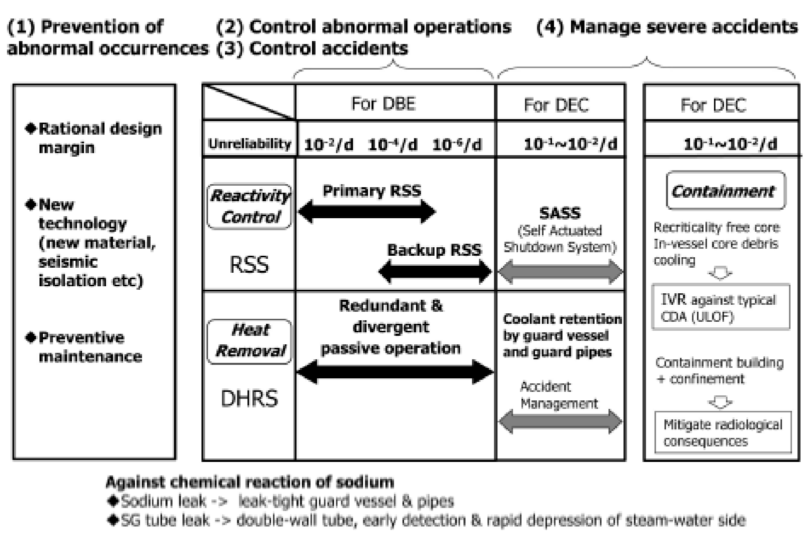

Figure 14 Safety Assurance Strategy for JSFR [4-2]

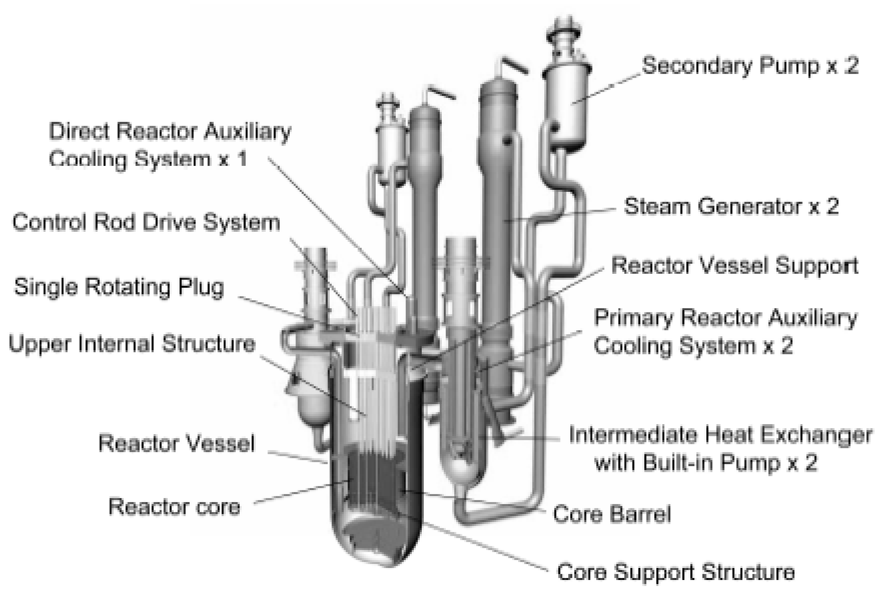

Figure 15 JSFR Plant View [4-2]

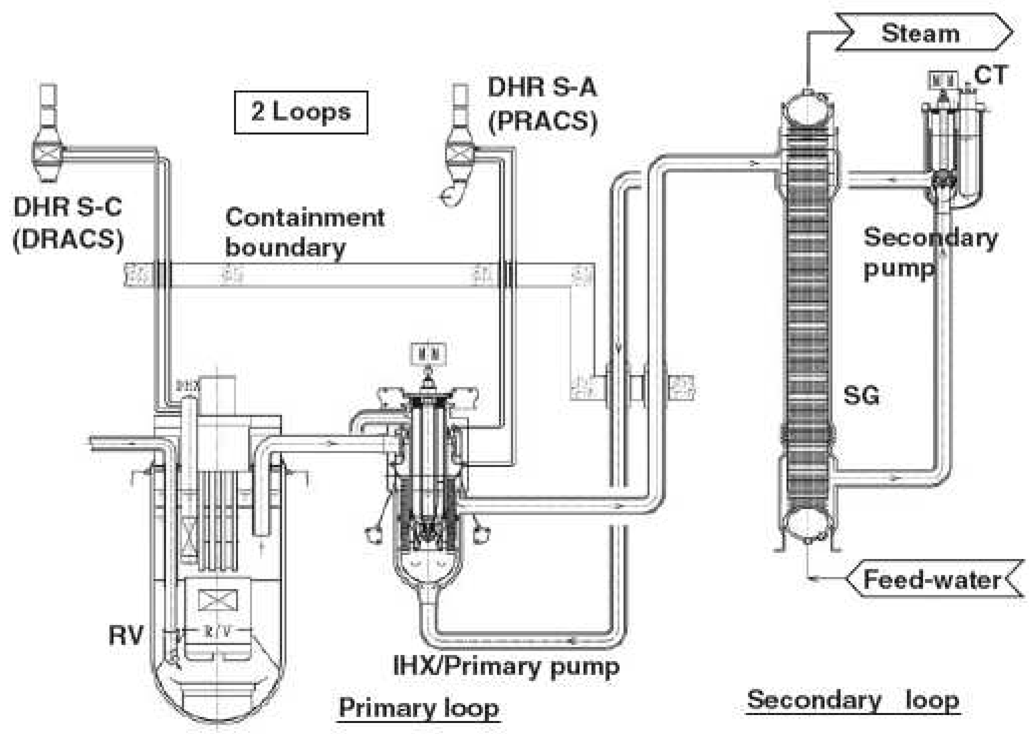

Figure 16 Schematic View of the Modified JSFR Heat Transport System [4-1]

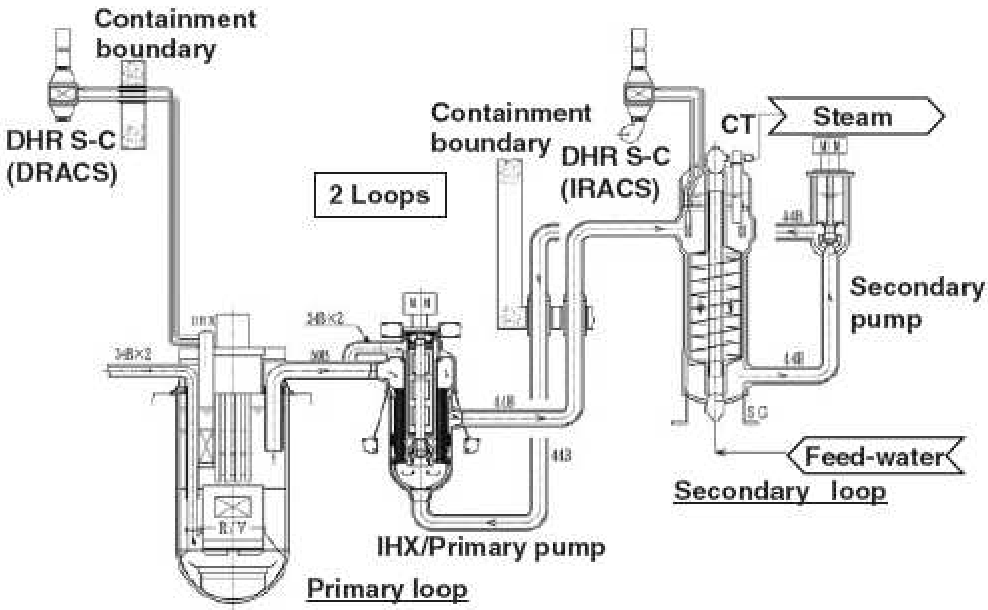

Figure 17 Schematic View of the Original JSFR Heat Transport System [4-1]

The defense-in-depth strategy is complemented by a risk-informed approach, with the goal of an estimated large offsite release frequency lower than 10^{-6} /site-year. Other risk related goals include: a) core damage frequency (CDF) less than 10^{-6} /reactor-year (/reactor-year), and b) unreliability of containment capability sufficiently small under representative Core Disruptive Accident (CDA) conditions. A preliminary PSA based on existing reliability data base (including data from JOYO, MONJU, EBR-II, FFTF, and Japanese LWRs) resulted in CDF << 10^{-6} /ry: a) ATWS: 10^{-8} /ry to $3x10^{-8}$ /ry, b) LORL: $4x10^{-9}$ /ry, and c) PLOHS: $2x10^{-8}$ /ry

4.1.2 The Role of Severe Accidents

The prevention and mitigation of Severe Accidents relies on several elements:

1. An additional passive self-actuated shutdown system (SASS)
2. Conditions for the elimination of severe re-criticality:
    - The sodium void worth is less than 6 dollars.
    - The core height is less than or around 1m.
    - Enhanced molten fuel discharge from the core region.
3. Long term cooling of fuel debris (IVR)

The prevention of re-criticality is a central issue in the consideration of severe accidents. The fast reactor core is not in its maximum reactivity geometry. The minimum critical mass is less than several hundred kg, or fuel in several sub-assemblies (SA) (MOX, Pu-fissile 15%, sphere, no reflector). During a severe accident that leads to fuel melting in several SAs providing an early molten fuel escape from the core can avoid re-criticality and a severe power burst. This target scenario provides the motivation for modified SA designs that enhance the early molten fuel escape from the core.

The JSFR design includes a self-actuated shutdown system (SASS), illustrated in Figure 18, which relies on a passive de-latch mechanism utilizing the magnetic properties change of the sensing alloy at the Curie-point temperature. The combination of two independent RSSs and SASS (2 strong + 1 weak) is sufficient to reduce the occurrence frequency of ATWS to very small value (around 10^{-8}/ry). However:

- Rapid accident progression of ATWS could result in early large release (Cliff-edge effect).
- PSA for advanced reactors has less operating experiences available: 11,000RY for LWRs, but only 300RY for SFRs.
- Many Gen-IV plants will be constructed and be used for long term period in various societies.
- The re-criticality issue is an important issue in fast reactor safety and regulatory and public concern is rather high.

Therefore, JAEA considers that the re-criticality issue should be resolved, and thus it should be considered in design to avoid a large mechanical energy release, but the cost should be minimized. Improvement of counter measures is essential in order to minimize the impact on core performance and fabrication cost.

4.1.2.1 Controlled Material Relocation Concept

An important counter measure against re-criticality in the JSFR is a fuel assembly designed to facilitate the early molten fuel escape from the core, also referred to as Controlled Material Relocation (CMR) which was described in Section 3. Various CDA analyses have shown that re-criticality events can be avoided if about 20% to 30% of the initial core fuel inventory is discharged from the active core region. This portion depends on the fissile enrichment of the fuel. In a smaller core with higher fissile enrichment fuel discharge of ~30% of initial inventory is necessary to reach a deep sub-critical condition, while 20% is enough for a larger core with lower fissile enrichment. This has led to the fuel assembly concept with an internal duct structure for controlled fuel relocation as illustrated conceptually in Figure 19.

A sequence of FA designs have evolved to reduce the impact on core performance and fabrication cost. The initial FAIDUS (Fuel Assembly with Inner Duct System) design had a centrally located channel for fuel escape as shown in Figure 20a [4-8]. The internal central duct has an orifice at the top, and thus is filled with sodium flowing very slowly under normal operating conditions. If an accident such as ULOF occurs sodium in the pin bundle region will boil off, fuel will melt and the internal duct wall will be melted through, opening a path for the molten fuel to relocate. The molten fuel will be discharged through the duct to the inlet plenum by the gravity force, the sodium vapor pressure caused by the fuel-coolant interaction and fission gas plenum released from the molten fuel. The fuel discharge capability of the initial FAIDUS was investigated using the SIMMER-III code [4-9]. The analysis indicated that more than 60% of the initial fuel mass in the FSA is discharged from the core region within 3 seconds mainly due to gravity, sodium vapor pressure, and fission gas pressure. The fuel discharge through the

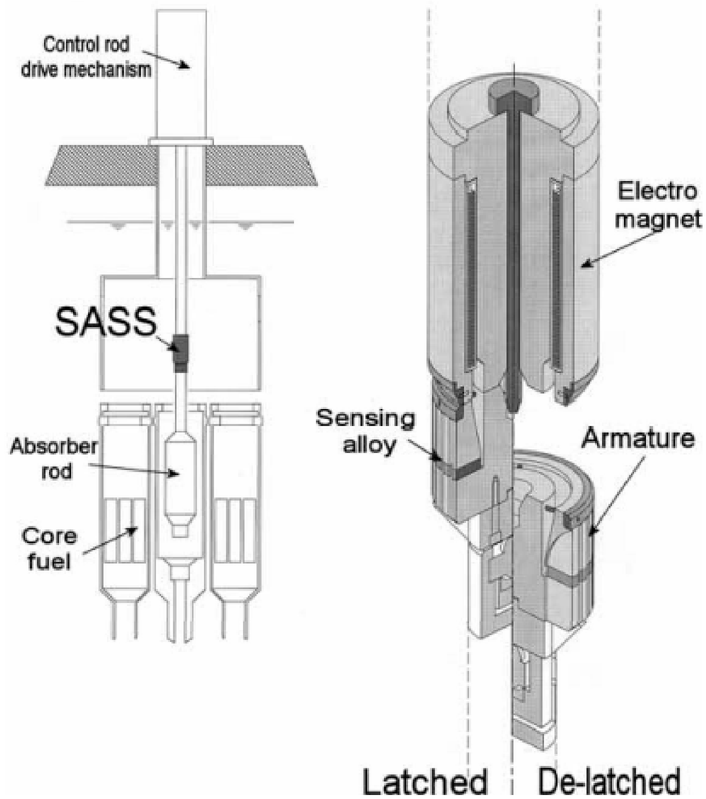

Figure 18 Self-Actuated Shutdown System [4-2]

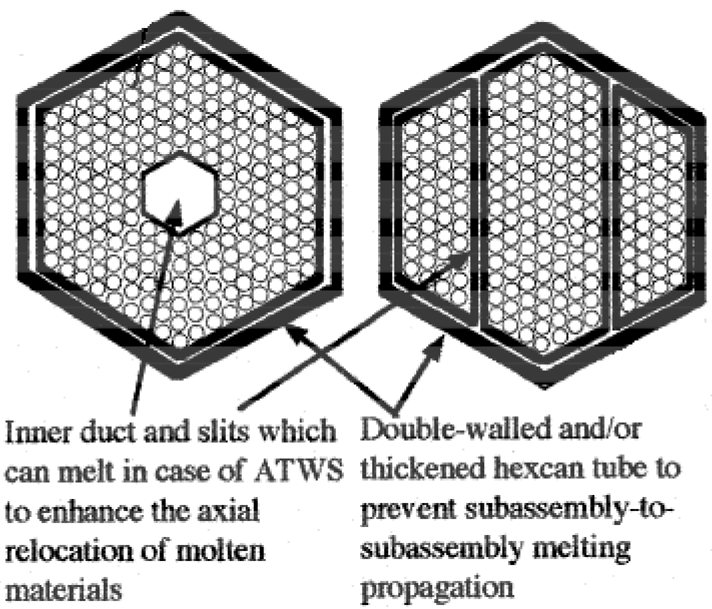

Figure 19 Concept Fuel Assemblies for Enhanced Molten Fuel Escape from the Core [4-4]

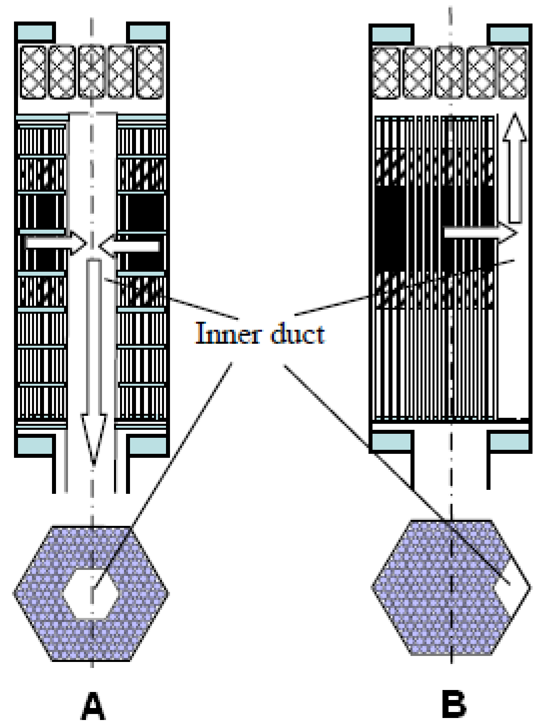

Figure 20 FAIDUS Concepts: A) Original FAIDUS and B) Modified FAIDUS

inner duct in FAIDUS has been confirmed by a series of experiments performed as part of the EAGLE project [4-10]. This concept has a disadvantage from the viewpoint of core performance, because the fuel mass in the FAIDUS subassembly with a central fuel relocation channel is about 10% less than that of a normal design FSA. In addition, a special grid-type spacer is needed and the fabrication of the FSA with the internal duct at the center of the assembly is more difficult.

These disadvantages have been addressed in the current reference design, the Modified-FAIDUS, which is shown in Figures 20b. A more detailed representation of the M-FAIDUS fuels assembly is shown in Figure 21. The Modified-FAIDUS uses a corner channel for fuel escape, with the fuel expected to relocate upwards to the upper plenum. The fuel loss is smaller than in the original FAIDUS design because the cross section of the duct could be reduced, due to the shorter length of the duct, while maintaining the same ratio length/diameter. In the Modified-FAIDUS conventional wire-spacer can be used, and the fabrication of the assembly is easier. An important aspect of the Modified-FAIDUS design, that requires evaluation, is the efficiency of the upward fuel motion against gravity. To evaluate the performance of the Modified-FAIDUS design a typical ULOF accident in a SFR was analyzed using the SIMMER-III code. The results indicated that most of the molten fuel is discharged upward via internal duct in a high power FSA. The total discharged fuel fraction exceeds 20% of the initial inventory, which implies the elimination of the possibility of a re-criticality event. The demonstration of upward fuel discharge has been included in the second series of the EAGLE experiments, EAGLE-2. The current R&D effort studies the use of a slim CRGT (Control Rod Guide Tube) to provide the paths for molten fuel escape. This approach would allow a normal fuel SA fabrication and require no fuel loss. The effect of the early fuel relocation due to the use of a

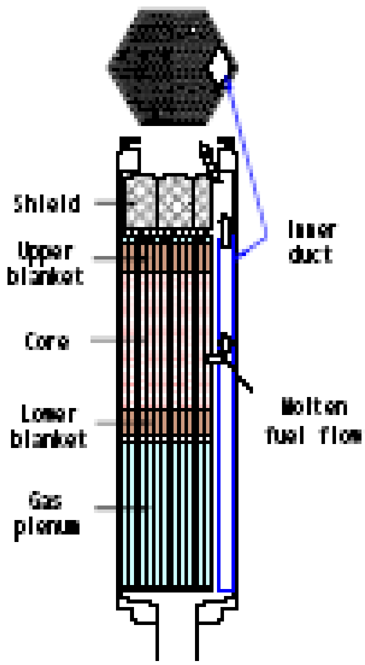

Figure 21 Modified-FAIDUS - Fuel Assembly Design [4-3]

FAIDUS design is illustrated in Figure 22 [4-11], which shows that the early fuel relocation prevents the formation of the large molten fuel pool in the active core region, and leads directly to the Post-Accident Heat Removal phase, eliminating any potential energetic events.

The Post Accident Heat Removal (PAHR) for JSFR is designed to prevent the debris bed from reaching the limit conditions for a coolable debris bed:

- Critical thickness : >30cm
- Cooling limit (bed dry out condition with porosity 0.5)
  - 10 cm for bed formation just after shutdown
  - 15 cm for bed formation 1000 second after shutdown

The basic idea that underlies the PAHR design is to broaden the fuel debris bed as much as possible inside the reactor vessel:

- Upward relocation and in-place cooling inside the core will help to reduce the amount of the molten fuel which might reach the bottom of the reactor vessel (RV). For the fuel relocating upward, the intermediate isolation plate (Figure 23) can retain a debris bed with up to 40% of the core fuel within the limit conditions.
- For the downward relocating fuel, the core support structure and the multi-layered debris tray (Figure 23) can retain 100% of the fuel within the limit conditions. The core support structure is designed to protect against direct molten fuel jet attack, while multi-layered debris tray at the bottom of the RV is designed for debris retention the debris bed height limits of cooling and sub-critical state.

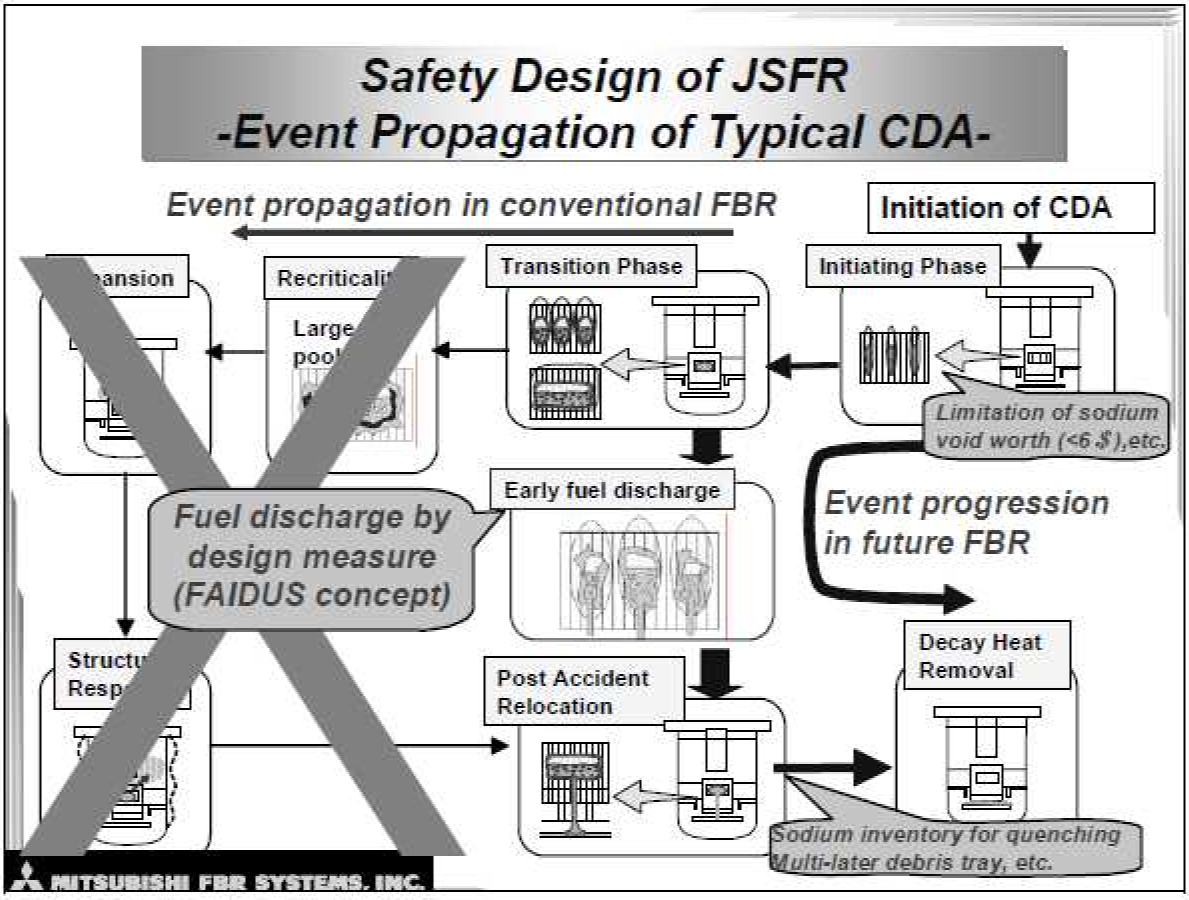

Figure 22 The Role of Early Fuel Discharge During the Transition Phase [4-11]

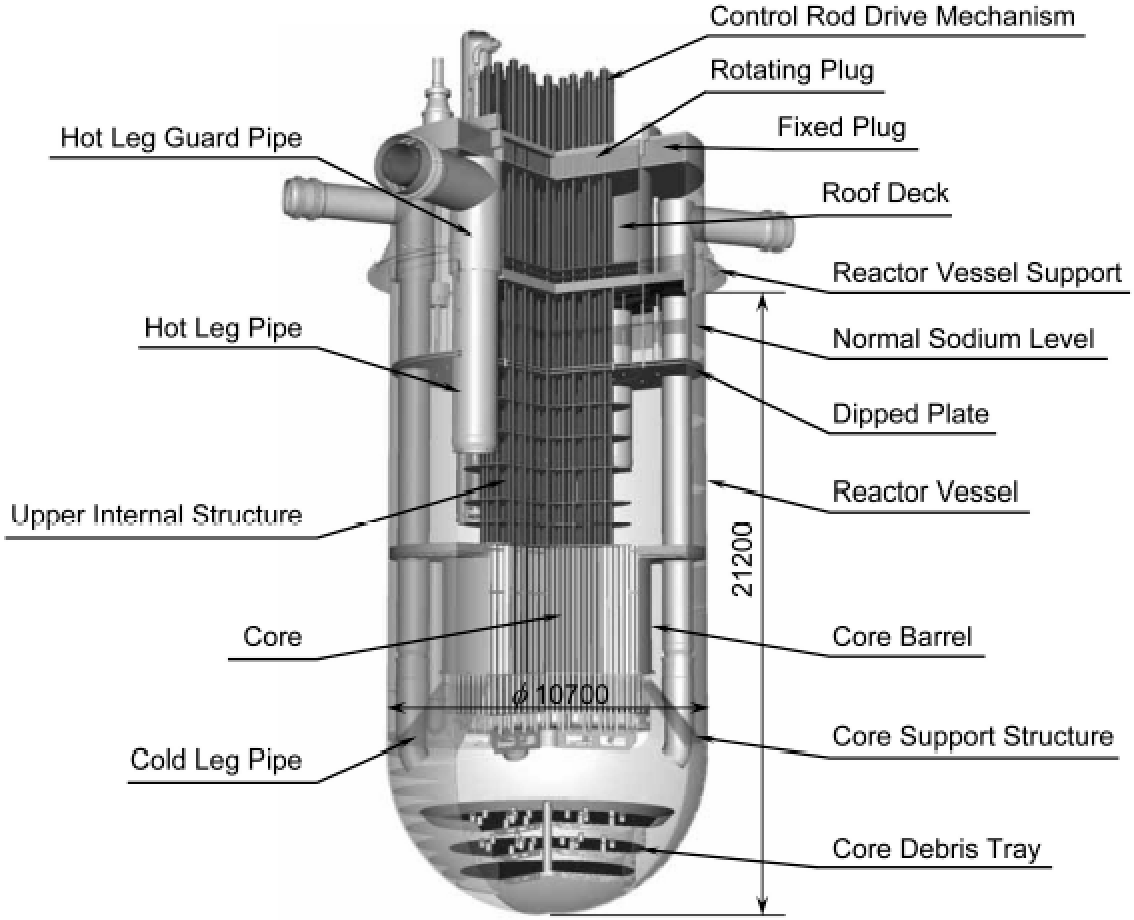

Figure 23 JSFR Reactor Vessel and Internal Structures [4-2]

4.1.3 Core Disruptive Accident Analysis

CDAs are not DBA but BDBE. They are assessed in the licensing and the assessment must be conducted with "realistic" (best estimate) analysis codes. Selected BDBEs, including ULOF, UTOP, LOPI, and Local Fault are evaluated in order to confirm the safety margin, or no cliff-edge effects. ULOF was recognized as an envelope of core damage effects and the whole ULOF scenario and consequences were evaluated for DFBR, Monju, and JSFR. A ULOF analysis was completed for JSFR with modified FAIDUS assemblies [4-11]. The initiating phase was calculated with the SAS4A code and showed limited fuel melting just after the initial power transient due to fuel pin failure. Continuation of the SAS4A calculations for another 3 s showed a gradual increase of the fuel melting with a relatively small sub-criticality. The core conditions at this time were then used as initial conditions for the SIMMER III calculations. The fuel relocation after the failure of the inner tube was calculated using the SIMMER code. After the failure of the inner duct, the fuel escape was driven by the pressure difference between the core and upper sodium plenum. The modified-FAIDUS expelled 90 % of molten materials in core.

The fuel escape behavior in several typical sub-assemblies with various power levels was integrated to estimate the whole core behavior. The results support the feasibility of the avoidance of severe re-criticality by fuel escape from the core in the early phase of CDA.

The fuel relocation results from SAS4A and SIMMER calculations were used to evaluate the longer term post-accident heat removal (PAHR). In these calculations, which considered the decay heat just after shutdown, 40% of the fuel was located on the intermediate isolation plate and 60% remained inside the core. The PAHR simulation used 3D fluid dynamics inside the RV and 1D net-work flow for the primary circuit with DRACS x 1 and PRACS x2. The outermost row of the core, radial blanket and radial shield were available for cooling paths, while the rest of the core region was assumed to be totally blocked. The results indicated that the decay heat balanced the removed heat around 30 minutes after the start of the transient event and the maximum temperature of the debris support structure remained below 700°C. Thus, the structure integrity could be preserved.

4.1.4 Summary

CDAs are evaluated in the category of Beyond Design Base Events (BDBE) or Design Extension Conditions (DEC). ATWS/ULOF is one of the major concerns for the severe accident consideration. Both CDA prevention and mitigation features are included in the JSFR design:

- Prevention: SASS
- Mitigation: Elimination of severe re-criticality CMR and IVR

R & D works under way for the development, enhancement, and evaluation of these CDA prevention and mitigation capabilities. The SFR designers in Japan indicate that severe core damage can be ruled out from the design and accident management considerations by achieving sufficiently low occurrence probability.

Improvement of the M-FAIDUS fuel assembly is considered essential to reducing the design impact. The current R&D target is the Slim CRGT design, which will be analytically and experimentally studied in detail. A SIMMER-III/IV collaboration is considered desirable by JAEA for this purpose.

4.2 Review of the French (AREVA/CEA) SFR Safety Approach

The European Fast Reactor (EFR) project, which was launched in 1988, provides the basis for the development of commercial fast reactors. The goal of this project is to develop a design that takes advantage of the experience gained from fast reactors in France (PHENIX, SUPERPHENIX) as well as around the world [4-7]. The EFR includes advanced features that have been shown by extensive R&D to yield benefits in terms of safety, reliability, and economy.

4.2.1 Safety Approach

The European Sodium Fast Reactor is based on a pool-type design illustrated in Figure 24.

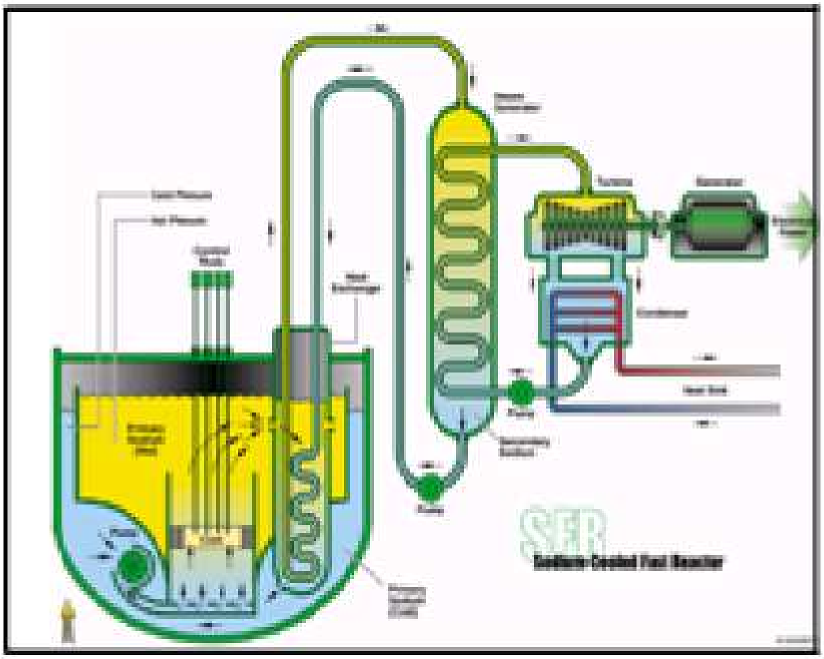

Figure 24 Schematic View of the European Sodium Fast Reactor

The EFR safety approach is based on risk minimization. It includes:

- Comprehensive and balanced safety concept by harnessing the favorable features of LMFBR
- Avoiding weak points and cliff edge effects in the beyond design basis area
- Prevention of accidents and minimization of their consequences if they occur
- ALARP principle

CDA might occur in case of:

- Failure of the shutdown function
- Failure of the decay heat removal (DHR) function
- "Exotic" initiators: e.g., fast structural failure or insertion of large gas amount into the core

Improvement of CDA prevention requires improvement of the shutdown and heat removal functions.

Improvement of the shutdown function is obtained by adding a third shutdown system, which operates both passively and actively in case of postulated failure of the two basic shutdown systems. The two basic shutdown systems are designed for efficient operation in case of serious imbalances between produced and removed power:

- Failure of primary pumps
- Loss of main heat sink
- Withdrawal of absorbers

The objective of the third shutdown system is to relegate unprotected transients such as ULOF, ULOHS, and UTOP outside the realm of technical imagination.

Improvement of the DHR function is obtained through complementary DHR measures, including the implementation of a debris tray for long term retention and coolability of molten core materials.

4.2.2 The Role of Severe Accidents

For Superphenix, a 1200 MWe liquid-metal reactor prototype, the design-basis accident was a loss of electric power with the very conservative assumption of a simultaneous failure of the reactor shutdown system. The accident was characterized by an extended pump coastdown, eventually followed by sodium boiling which would trigger a power excursion and subsequent core disruption. To mitigate the consequences of this scenario, significant construction improvements were needed, including the ensuring the ability of the main vessel and the cover slab to withstand the mechanical energy released during this scenario, the installation inside the main vessel, below the core, of a core catcher capable of retaining the molten core debris, and a dome over the cover slab to retain any leaks of radioactive materials. Through its consequences, this accident scenario envelops many less-severe potential accident situations, which therefore did not require special consideration. For the reactor projects that were studied afterward, the occurrence probability of this postulated extreme accident, already very low for Superphenix, was further decreased through improvements in the reliability of the shutdown systems. The study of the whole-core disruptive accidents continued, but only in the framework of residual risk analyses.

The post-EFR generation of European commercial LMFBR is designed with consideration of CDAs. After improvement of safety functions, no CDA scenario is credible. The frequency of a combination of CDA initiators with postulated failure of the first and second shutdown systems and the third shutdown level is significantly below 10^{-7} per year. CDAs are considered Beyond Design Conditions. CDA analyses, traditionally based on ULOF, are performed to assure that there are no cliff-edge effects. They also provide appreciation of the relative importance of core characteristics in order to obtain a good balance between design and beyond design requirements.

Reasonable containment measures are provided for the mitigation of radiological releases in beyond design conditions. They include:

- Improvement of the primary containment for increased resistance against mechanical energy release from CDA
- Implementation of a debris tray for long term retention and coolability of molten core
- Definition of beyond design basis plant states for demonstrating the effectiveness of secondary containment

4.2.3 Core Disruptive Accident Analysis

CDA analyses are performed to identify the core characteristics which minimize the consequences of CDAs. The LMFBR characteristics lead to the consideration of the ULOF scenario for assessing the core behavior. The codes used for the initiating phase analysis include SAS4A, FRAX, and PHYSURAC. For the analysis of the transition phase, research activities are focused on the development of the SIMMER code.

4.2.4 Summary

In France, CDAs are evaluated in the category of Beyond Design Basis Events (BDBE) or Design Extension Conditions (DEC). ATWS/ULOF is one of the major concerns for the severe accident consideration. Both CDA prevention and mitigation features are included in the EFR design:

- Prevention: A third shutdown system, both passively and actively activated
- Mitigation: Enhanced in-vessel coolability and retention

R & D work is under way for the development, enhancement, and evaluation of these CDA prevention and mitigation capabilities. The position of the French SFR designers is that severe core damage can be ruled out from the design and accident management considerations by achieving sufficiently low occurrence probability.

4.3 Review of the US SFR Safety Approach - IFR

Inherent safety was a primary goal in the development of the Integral Fast Reactor (IFR) [4-12] in the US. The IFR metallic fuel design is an advanced concept developed as a result of experience with metallic fuels in EBR-II [4-13] and other reactors. The IFR design has demonstrated the potential for design features to mitigate the accident consequences and in some cases to render them benign.

4.3.1 Safety Approach

In the IFR concept, a pool-type primary system arrangement is combined with an advanced metallic fuel design and an on-site fuel-cycle facility. The pool-type primary system together with the low-pressure, liquid-metal coolant, provide substantial margins to coolant boiling and fuel melting during both normal and off-normal events. All the primary system sodium is contained within the reactor vessel, along with the core, the primary pumps, and the intermediate heat exchangers. Natural circulation cooling of the fuel is assured for both normal shutdown heat removal and for abnormal events. There is no external piping for the primary coolant system, eliminating pipe break accidents as a source of coolant loss. Intrinsic protection against accidents caused by pipe rupture is therefore provided by the pool-type primary system. In addition, the high heat capacity of the pool concept provides long time margins for corrective action in the event of a heat-sink loss. In the IFR fuel design the fuel is cast as a uranium-plutonium-zirconium alloy. The physical properties of the IFR metallic fuel are compared with the corresponding properties of a typical oxide fuel in Table 2. The metallic fuel is denser than the oxide, with a thermal conductivity higher by an order of magnitude, and a lower specific heat. The thermal expansion coefficient of metallic fuel is higher than oxide, and the melting point is much lower. Since the U-Pu-Zr alloy is chemically compatible with sodium, the fuel rod is submerged in liquid sodium inside the cladding. The bond-gap sodium, together with the high

Table 2 Oxide and Metallic Fuel Thermo-Physical Characteristics [4-12]

Nominal Composition	UO_2-20% PuO_2	U-15% Pu-10% Zr
Density, g/cc	10.6	15.8
Thermal Conductivity, W/cm-^oC	0.023	0.22
Specific Heat, J/g-^oC	0.38	0.20
Thermal Expansion Coefficient, 1/^oC	$1.2x10^{-5}$	$2.0x10^{-5}$
Melting Point, ^oC	2750	1160
Fuel Pin Thermal Time Constant, sec.	~3	~0.3

thermal conductivity of the metal, give the metallic fuel pin an order-of-magnitude faster thermal response time compared to the oxide fuel which has a lower conductivity and is gas-bonded. The high thermal conductivity of the bond-gap sodium lowers the fuel surface temperature of the metallic fuel when compared to the oxide fuel. In addition, due to its higher thermal conductivity, metallic fuel exhibits relatively small radial temperature gradients. Metallic fuel therefore operates at much lower temperatures than oxide fuel, and the amount of energy stored in the fuel under normal operating conditions is reduced correspondingly.

4.3.2 The Role of Severe Accidents

The prevention of core disruption in unprotected (i.e. without scram) overpower and under-cooling accidents relies on diverse and redundant scram and shutdown systems to ensure that accidents that may have the potential to cause serious damage to the reactor core are of extremely low probability, and on the provision in the design for inherent, passive mechanisms which respond to the accident conditions and act to restore between the reactor energy production and the system cooling energy removal. Core disruption refers to any irreversible re-arrangement of the reactor core. To assure a self-limiting response to accident conditions, specific features are included in the system design. In the unprotected loss-of-flow and overpower accidents, the upset condition leads to an increase in the coolant temperature rise through the core. Negative reactivity feedbacks keyed to this coolant temperature increase can be effective in limiting accident consequences. Two such mechanisms are provided by: a) radial core expansion driven by subassembly duct bending and above-core load pad thermal expansion, and b) differential thermal expansion of control rod drives and the core support structure, leading to a net insertion of the control rods.

During an unprotected LOF event, radial core expansion and control rod drive elongation provide the overall negative reactivity feedback to lower the reactor power. Other reactivity effects that must be considered are the fuel Doppler feedback which introduces a positive feedback as the power decreases, the positive coolant density feedback due to the coolant temperature increase, and the positive fuel density feedback due to the fuel temperature decrease. Due to the low operating temperature of metal fuel, there is less positive reactivity feedback created as the reactor power is reduced as compared to the oxide fuel case.

In an unprotected TOP event, fuel overheating will lead to a prompt negative Doppler reactivity feedback. With continued fuel heating, the fuel temperature in the interior of the pin reaches a level at which the fuel strength is reduced and eventually melts. This is especially relevant for the metallic fuel, which loses strength and melts at temperatures lower than those for the steel cladding. If unreleased fission gas bubbles are present, they can pressurize the low strength or molten fuel, which can then be extruded upwards producing a strong negative reactivity feedback. Researchers investigating the oxide fuel behavior [4-13] have concluded that for annular oxide fuel pins, significant fuel relocation would occur in high ramp rate (greater than 3 $/s) TOPs. For lower ramp rates typical of unprotected TOP accidents internal fuel relocation was found to be inhibited by freezing and plugging in the colder upper pin regions. For solid oxide fuel pins the in-pin fuel relocation will also be prevented until the molten fuel cavity reaches the top of the fuel pin. These constraints do not apply in the case of metal fuel pins, since the axial temperature profile peaks at the top of the core, causing the molten cavity to reach the top of the fuel pin at an early time in the TOP transient and the relocating molten fuel to encounter hot temperature conditions. Therefore, pre-failure in-pin fuel motion can occur in metal fuel pins even at lower overpower ramp rates associated with uncontrolled rod withdrawal, providing a significant negative reactivity feedback that can mitigate the TOP consequences.

The main thrust in the US ALMR program in the 1990s was to develop a modular reactor plant concept, PRISM (Power Reactor - Inherently Safe Module) illustrated in Figure 25, which can be built up, in modular fashion, to larger plant ratings. In an earlier design, each power block is comprised of three 471-MWth reactor modules connected to a single 465-MWe turbine-generator [4-16, 4-17]. More recently the PRISM plant has been up-rated to improve economics and reduce excessive conservatism [4-15]. Each reactor produces 1000 MWt, with two reactors combined into a power block that supplies steam to a single turbine-generator. The PRISM reactor system design takes advantage of passive and inherent mechanisms to accommodate postulated unprotected accidents that could potentially lead to CDAs otherwise. Notable examples of the plant's innovative characteristics include:

a) The use of ternary metal fuel
b) Inherent reactor shutdown and stabilization by thermal and reactivity response characteristics of the reactor, even under extremely unlikely accident conditions
c) Passive decay heat removal systems
d) Reactor seismic isolation
e) Containment consisting of a guard vessel around the reactor vessel and a reactor enclosure with seal-welded penetrations
f) Elimination of power-dependent auxiliary cooling systems.

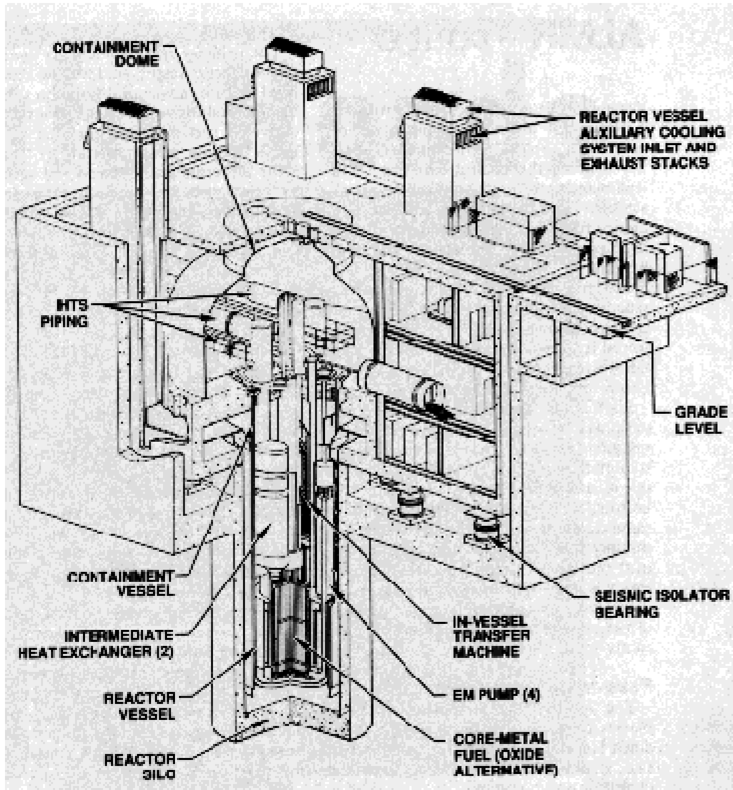

Figure 25 PRISM Reactor Module [4-18]

High shutdown heat removal reliability in PRISM is accomplished by a combination of the normal heat removal path through the Intermediate Heat Transport System and an inherent air natural circulation heat removal system referred to as RVACS (Reactor Vessel Auxiliary Cooling System). The RVACS is designed to provide shutdown heat removal for all conditions associated with loss of heat removal through the normal heat transport system. A typical RVACS configuration is shown in Figure 26. The natural air flow is maintained at all operating conditions even at normal operating power. At transient conditions, with higher sodium and vessel wall temperatures, the heat removal rate increases rapidly, approximately with the third power of the vessel temperature.

The PRISM reactor is equipped with six GEM assemblies distributed symmetrically around the core as shown in Figure 27.The GEMS consist of a hollow pressure tube that is capped at the top and filled with Helium. Sodium coolant is allowed to enter at the bottom and fill the pressure tube when the coolant pumps are operating. The GEM's are designed to lower the power level in the core if the main coolant EM pumps malfunction or stop. In this event, the helium pressure will cause the sodium level in the pins to drop and thus allow more neutrons to leak out of the core. Since the neutron flux of this core is very sensitive, the lower number of neutrons causes the reactivity and power level to drop. The GEMS were retained for the oxide backup core, and are likely to be removed from the PRISM design if a metal fuel core is pursued, as noted earlier in Section 3.1.2.

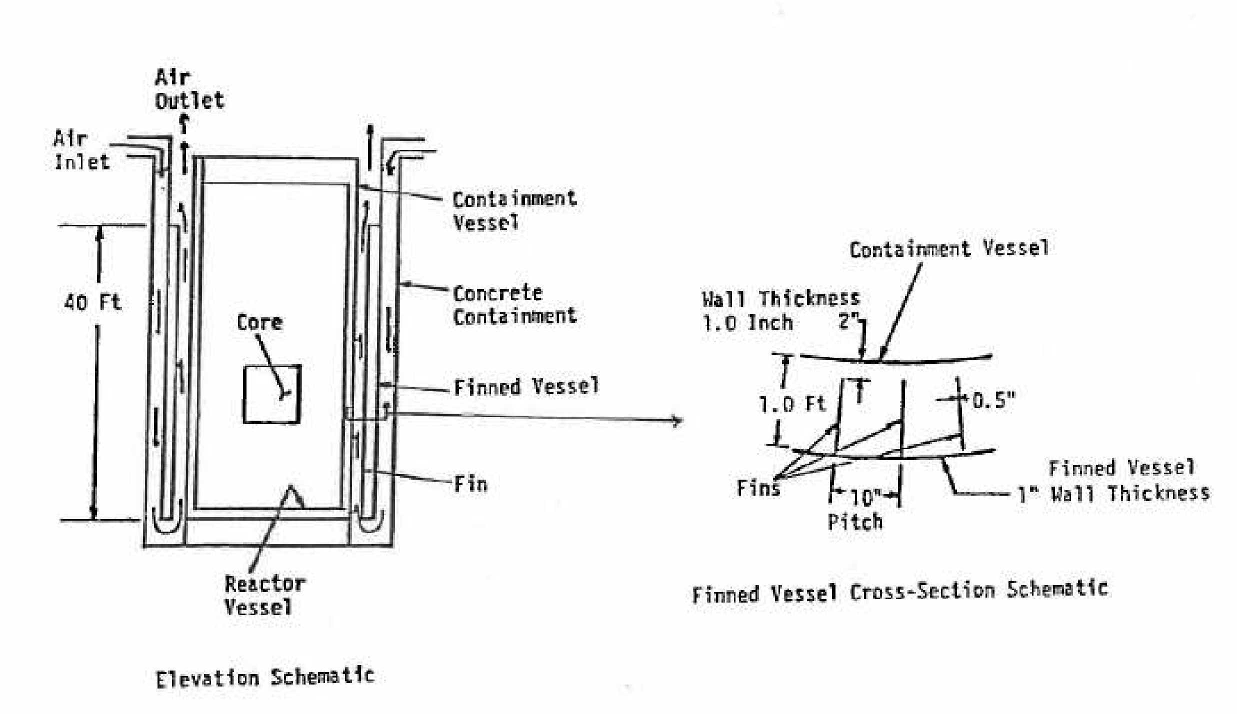

Figure 26 Schematic Representation of the RVACS

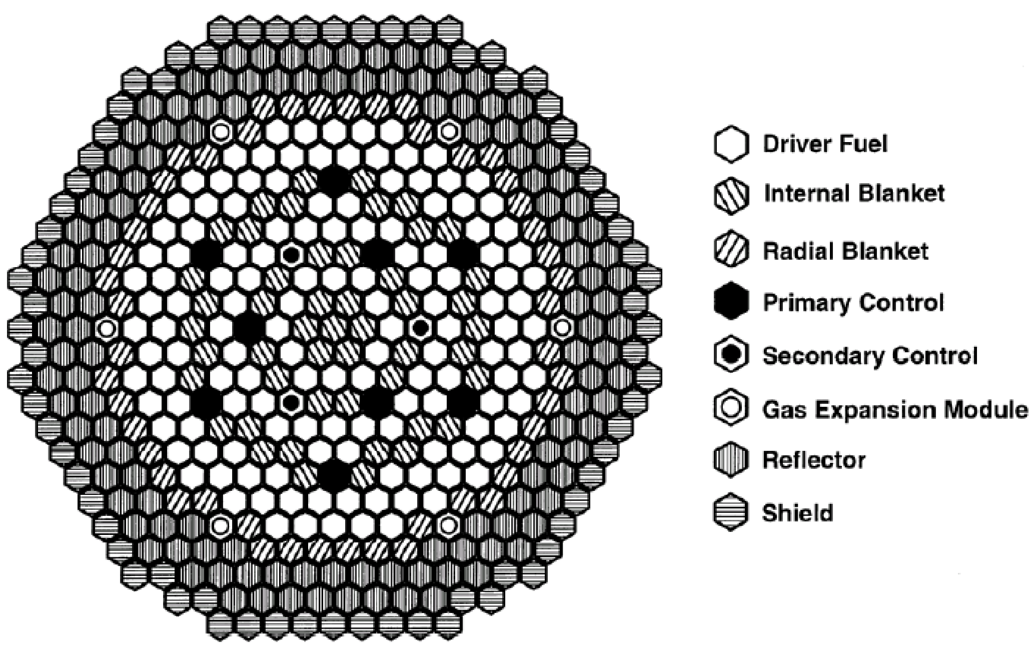

Figure 27 GEM Assembly Locations in the PRISM Core [4-16]

The PRISM design described in the PSER Final Report [2-14] includes an Ultimate Shutdown System (USS) assembly, illustrated in Figure 28. Should the control rods fail to insert into the core, this central assembly can shut down the reactor "single-handedly". This assembly is in the center of the core and consists of a hollow assembly tube with a thin membrane covering the top of the assembly. Connected to the top of the assembly tube is a wedge shaped plunger and container filled with thousands of boron-carbide neutron-absorber balls. When activated, the plunger breaks through the membrane seal and allows the boron balls to fall into the active core region. This sudden negative reactivity insertion guarantees that the core will achieve shutdown. The more recent USS design was modified to use segmented absorber materials instead of the of the boron balls.

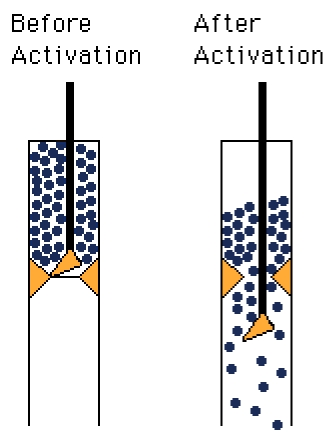

Figure 28 Ultimate Shutdown System Assembly

4.3.3 Unprotected Accident Analysis

The SAS4A code system, coupled with the SASSYS code [4-14], was used to analyze the reactor behavior during postulated unprotected accidents for both oxide-fueled reactors such as CRBR and for the metal-fueled reactors represented by the IFR concept, and other reactor design concepts based on the IFR. The results of these analyses provide an assessment of the outcomes of various accident sequences and provide guidance for future experimental needs and computational model development. SAS4A allows the modeling of reactivity feedbacks and phenomena of specific interest for both oxide-fueled reactors and metal-fueled reactors in analyses of postulated unprotected accident sequences. The SAS4A code includes models describing the radial core expansion and the control rod drive expansion that occur during an unprotected accident and the associated negative reactivity feedbacks. The SAS4A oxide-fuel version of SAS3A has detailed models that describe oxide-fuel phenomena during irradiation and during postulated unprotected accident conditions, including the fuel pin mechanics model DEFORM and the material relocation models PLUTO2 and LEVITATE. The SAS4A metal-fuel version has been extended to include models that describe the specific phenomena associated with the metal fuel pins, including the addition of the pre-failure in-pin fuel relocation model PINACLE, and significant model additions in the DEFORM fuel pin mechanics model and in the LEVITATE post-failure material relocation model. These models are described in more detail in Section 5.1.1 below.

4.3.4 Summary

The US approach to SFR severe accidents has focused on the prevention of core disruption and mitigation of consequences of unprotected overpower and under-cooling accidents. The prevention of core disruption is achieved through the provision in the design for diverse and redundant shutdown systems, complemented by the use of inherent, passive mechanisms which respond to the accident conditions and act to restore or maintain the balance between the reactor energy production and the system cooling energy removal. This approach can be used for both oxide and metal fueled designs, although it is considered easier to achieve benign termination of severe unprotected accidents with the use of metal fuel [4-19]. The core disruption prevention features include the radial core expansion, control rod driveline expansion, and the use of the metal fuel which has a low operating temperature. Features that help mitigate the consequences of core disruptive accidents include the low melting temperature of the metal fuel and its compatibility with sodium, the pre-failure in-pin fuel relocation due to metal fuel characteristics, and a conventional containment.

5. EVALUATION OF THE IMPACT OF DESIGN MEASURES ON SEVERE ACCIDENT PREVENTION AND MITIGATION

The evaluation of severe accidents has played a prominent role in the safety analysis of SFRs. During the licensing evaluations of FFTF and CRBR extensive experimental and analytical work was performed in the US to evaluate the phenomena, accident path, and consequences of postulated accident initiators that could lead to core disruptive accidents. During the US CRBR project in the 70s and 80s, US researchers played a lead role in the study of severe accidents for SFRs with oxide fuel pins. The research included in-pile and out-of-pile experiments for the study of severe accidents phenomena as well as an extensive code development effort for the quantitative evaluation of the postulated severe accident consequences. Specialized modules of the SAS4A code were developed to describe the material relocation during UTOP events (PLUTO) and ULOF transients (LEVITATE). These modules allowed SAS4A to perform detailed whole core analyses of the initiating phase of postulated severe accident such as ULOF, UTOP, and LOF driven TOP. Through international agreements, SAS4A was shared with research organizations in Japan, France, Germany, and UK and became the worldwide standard for the study of the initiating phase of postulated severe accidents in oxide fueled SFRs.

With the re-focusing of the US SFR program on the metal fuel SFR during the 1980's and 1990's, the US experimental and analytical activities related to severe accidents shifted to metal fuel phenomenology. Existing modules of SAS4A such as DEFORM were modified to allow the modeling of metal fuel pins, and a new module was developed to describe the pre-clad-failure in-pin fuel relocation (PINACLE). A series of metal fuel experiments was performed to study the severe accident phenomena typical for metal fuel SFRs, and these experiments were analyzed with the SAS4A-M (metal fuel) code.

In the meantime, the development and validation of the SAS4A oxide fuel code was continued by an active collaboration involving Japan, Germany, France, UK, and the EU. The US researchers continued to participate in some of the meeting of this group and provided technical advice as needed. The SAS4A code continues to be used actively in Japan, France, and Germany for the study of the initiating phase of postulated SFR severe accidents. In France the codes used for the initiating phase analysis also include FRAX, and PHYSURAC. The SAS4A code, including the LOF material relocation module LEVITATE, was used for the ULOF analyses performed for severe accident analyses during the re-licensing of the Monju reactor in Japan.

Current research activities in Japan and France are focused on the development of the SIMMER-III and SIMMER-IV codes for the analysis of the transition phase for oxide-fueled reactors. The integrated analysis of the CDA sequence of events is a central goal in both Japan and France, with the events ranging from normal operation to the initiating phase intra-subassembly material relocation described by the SAS4A code, and the subsequent events starting with the Transition Phase modeled by the SIMMER code. As described below in Section 5.2, the initial conditions for the SIMMER in analyses performed in Japan are provided by previous SAS4A analyses. The change from one code to the other occurs at the time of the failure of the inner tube, when radial molten fuel relocation can become significant. Thus the interfacing between SIMMER and SAS4A becomes an important area of the severe accident analysis where US researchers can provide valuable expertise. Multi-dimensional (2-D and 3-D) material relocation that occurs within a fuel assembly prior to the breaching of the assembly wall can aslo play an important role in determining the timing of and physical conditions present at the initiation of inter-assembly material relocation. In the US a study and model development of the multi-dimensional material relocation in fuel assemblies was initiated during the US New Production Reactor (NPR) project, when the development of a 2-D version of the LEVITATE code (DIANA) was undertaken.

5.1 Major Codes for Evaluation of CDA Energetics

This section reviews the features and capabilities of two major codes used in the US, Japan, France, and other countries for the evaluation of CDA energetics: a) the SAS4A code used for the analysis of the initiating phase events, and b) the SIMMER-III code used for the analysis of the transition phase events.

5.1.1 The SAS4A Code System

The SAS system of accident analysis codes, developed in the US at Argonne National Laboratory (ANL), has played an important role in the assessment of energetics potential for postulated SFR severe accidents. The SAS4A code system [5-1], the latest generation of this code system, models the events talking place in a LMFBR core during the initiating phase of postulated accidents such as TOP and LOF. Developed initially for the analysis of oxide-fuel SFRs, the code models have been later expanded to allow the simulation of postulated accident sequences in metal-fueled SFRs. The reactor core is subdivided into channels, each channel containing a number of assemblies. All the fuel assemblies in a channel are assumed to behave identically. The flow within each assembly is assumed to be one-dimensional. The end of the initiating phase treated by SAS4A is thus defined by the breach of the assembly wall, when two-dimensional and three-dimensional flow effects between adjacent subassemblies become significant and the one-dimensional flow assumption becomes invalid. SAS4A has a modular structure, with various modules describing specific phenomena relevant to the SFR accident scenario. Specific models describe the coolant flow and boiling, fuel pin heat transfer, the mechanical response of the fuel pin, cladding melting and relocation prior to fuel pin failure, fuel pin melting and in-pin fuel relocation, and multiple material relocation and interaction after the cladding failure and fuel pin disruption. Various combinations of modules are active at any given time in each channel, depending on the local conditions, and providing a highly flexible framework for the mechanistic modeling of the complex phenomena that occur during a postulated CDA.

The Loss of Flow (LOF) accident has been studied extensively in the US, Japan, and Europe as the accident that provides an envelope for the CDA conditions in an oxide-fueled reactor and can lead to the Transition Phase. The SAS4A module that describes the sequence of events during a postulated LOF accident, from the initiation of fuel motion to the end of the Initiating Phase (IP) is the LEVITATE model [5.2]. LEVITATE is likely to be active in a SAS4A channel at the end of the IP, when the SIMMER-III code, described in the following section, is activated to describe the Transition Phase (TP) events. Understanding the similarities and differences between the models used in LEVITATE and SIMMER-III is thus important to ensuring a code transition that preserves the important physical characteristics of the core at the transition time.

The LEVITATE code is a multi-phase multi-component model that describes the fuel assembly in a one-dimensional geometry, assuming that all the pins in the subassembly behave coherently. The moving components include the fuel and cladding, which can exist as solid particles, liquid, or vapor, the coolant which can exist as liquid or/and vapor, and the fission gas which can be either released fission gas or fission gas still retained in the moving solid fuel. Each of these components is associated with one of the three velocity fields available: liquid velocity, solid particle velocity, and vapor velocity. A typical LEVITATE configuration is presented in Figure 29. Three basic thermal-hydraulic models are used for to describe the LOF events in each subassembly: 1) the hydrodynamic model describing the material relocation inside the fuel pin cavities, which contain molten fuel and fission gas, 2) the hydrodynamic model describing the material relocation in the coolant channel bounded by the outside cladding surface and the subassembly hexcan wall, and 3) the stationary structure model describing the heat transfer and melting/freezing processes associated with the solid fuel pin stubs and the hexcan wall. LEVITATE describes a wide spectrum of physical phenomena, including pin-disruption modes, multiple fuel/steel flow regimes, fuel/steel freezing and plug formation, and a tight coupling with the sodium slug dynamics [5-3]. LEVITATE has been initially developed for the analysis of LOF postulated accidents in oxide-fueled SFRs, and has been successfully validated in the US in analyses of multiple TREAT experiments [5-2, 5-4, 5-5]. The oxide-fuel version has been later validated in analyses of the CABRI experiments through an international collaboration that included Japan, France, and Germany, as described below in Section 5.2.1. In the US, as the focus of SFR research shifted to metal-fueled reactors, LEVITATE has been expanded to model metal-fuel specific phenomena. An important capability added during this phase was the coupling of LEVITATE with the newly developed pre-failure in-pin fuel relocation module PINACLE [5-6].

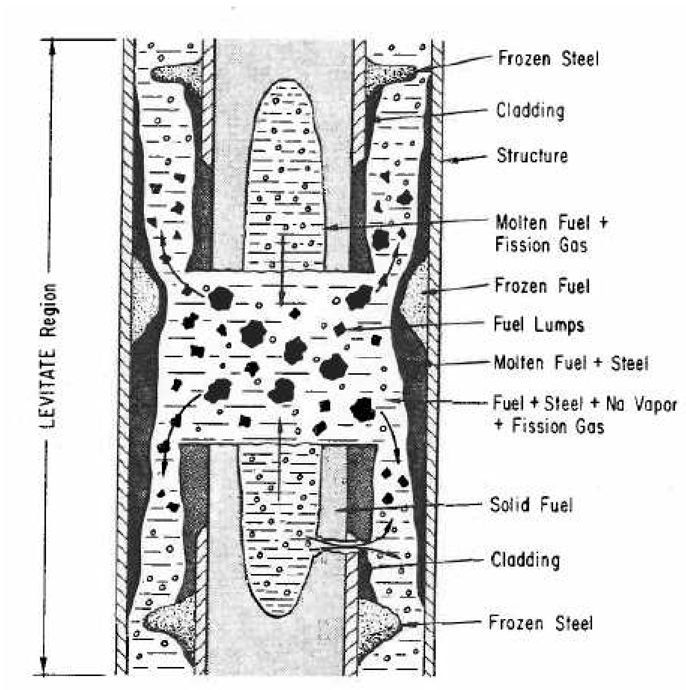

Figure 29 Typical LEVITATE Configuration

5.1.1.1 The SA4A Metal Fuel Version

The SAS4A code was initially developed for the analysis of unprotected hypothetical accidents in oxide-fueled SFRs. With a change in focus in the US towards the metal-fueled SFRs, a new version of the code has been developed. This section describes the main SAS4A phenomenological modules affected by the introduction of metal fuel pins [5-16]. These modules include: a) the in-pin molten fuel relocation module PINACLE, b) the cladding failure module DEFORM-5, and c) the post-failure material relocation module LEVITATE.

a) The PINACLE module describes the relocation of molten fuel inside the pin cladding, prior to cladding failure. As the accident proceeds, the mismatch between the heat generated in the fuel and the heat removed by the coolant leads to the fuel heat-up and the formation of a molten fuel cavity. The molten fuel inside the cavity can relocate under the influence of the pressure gradients, causing potentially significant negative reactivity changes. As long as the cavity maintains a bottled-up configuration, only limited fuel relocation can occur. However, if the molten fuel cavity reaches the top of the active fuel column, the molten fuel can be ejected in the space above the active fuel. This situation, illustrated in Figure 30, is typical for metal fuel pins, where the axial temperature profile peaks near the top of the active fuel. The amount of fuel ejected above the active fuel column depends on the pressure difference driving the molten fuel and on the specific pin design. One of the models incorporated in PINACLE calculates the breach of the fuel column top, which controls the onset of rapid in-pin fuel relocation. PINACLE also models the freezing of the in-pin molten fuel as the power decreases, and the associated reduction, both axial and radial, of the molten cavity. This allows an accurate modeling of the decreasing-power part of the transient overpower sequence.

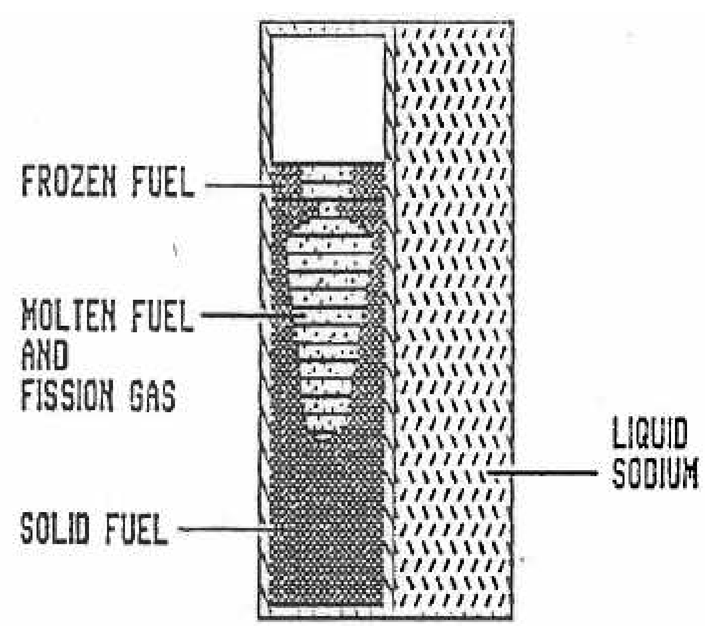

Figure 30 PINACLE Configuration

b) DEFORM-5 is the SAS4A module designed to provide the metal fuel pin and monitor the pin failure margin. It has been coupled with the PINACLE module to provide an integrated model of the pre-failure pin behavior. Because the cladding is considerably stronger than the fuel, the initial model development has concentrated on the cladding transient response. Because of the large interconnected porosity that develops with burnup, the pressure in the gas plenum and the pressure in the fuel pin are assumed to remain equal up to the time when a significant molten region develops and PINACLE is initiated. After the initiation of PINACLE the internal pressure acting on the cladding is calculated by PINACLE. The pressure equilibration between the molten fuel cavity and the plenum occurs as a result of the in-pin fuel relocation modeled by PINACLE. The cladding plastic strain and failure margin are determined using the forces acting on the cladding and the cladding temperatures. Penetration of the cladding due to the formation of the uranium-iron eutectic is also considered. Eutectic penetration of the cladding reduces its load bearing capabilities. The DEFORM-5 module determines the time of cladding failure and triggers the initiation of the LEVITATE model.

c) LEVITATE describes the physical processes that occur in a fuel assembly following the cladding failure. In the metal-fuel version of SAS4A LEVITATE is initiated at the time of cladding failure in both voided and unvoided coolant channels. This is different from the oxide-fuel analyses, where the early fuel fragmentation and relocation for unvoided channels is described by the PLUTO2 model of SAS4A. The use of LEVITATE in unvoided coolant channels in the analysis of metal fuel behavior is due to the metal fuel tendency to foam rather than fragment when ejected in unvoided coolant channels, and to the need for modeling of early cladding melting and relocation due to eutectic penetration. LEVITATE is interfaced with the PINACLE module, and continues the in-pin fuel relocation after the pin failure using the PINACLE results as initial conditions. The LEVITATE cladding failure model is compatible with the DEFORM-5 cladding failure model, and LEVITATE continues to monitor the cladding failure axial extension after the initial cladding failure.

5.1.2 The SIMMER-III Code

SIMMER-III is a general two-dimensional, multi-phase, multi-component fluid dynamics code coupled with a space-time energy-dependent neutron transport model, developed for the mechanistic analysis of the transition phase of CDAs [5-12]. The code has been initially developed in the US at Los Alamos National Laboratory (LANL), and earlier versions of the code have played a significant role in the advancement of the mechanistic simulation of CDAs, focusing on the transition phase events. Modeling limitations identified during these early analyses have led to the development of a new generation code, SIMMER-III in a joint effort of Japan Nuclear Cycle Development Institute (JNC), the French Commission for Nuclear Energy (CEA), and the German Research Center Karlsruhe (FZK). In the early stages of the development effort the UK Atomic Energy Authority and in the US LANL also participated in this effort.

The conceptual framework of SIMMER-III is shown in Figures 31 and 32. The SIMMER-III code models the basic SFR materials: fuel, steel, sodium, control rod, and fission gas. Some of these materials can exist in different physical sates, e.g. the fuel can exist as stationary fuel pin,

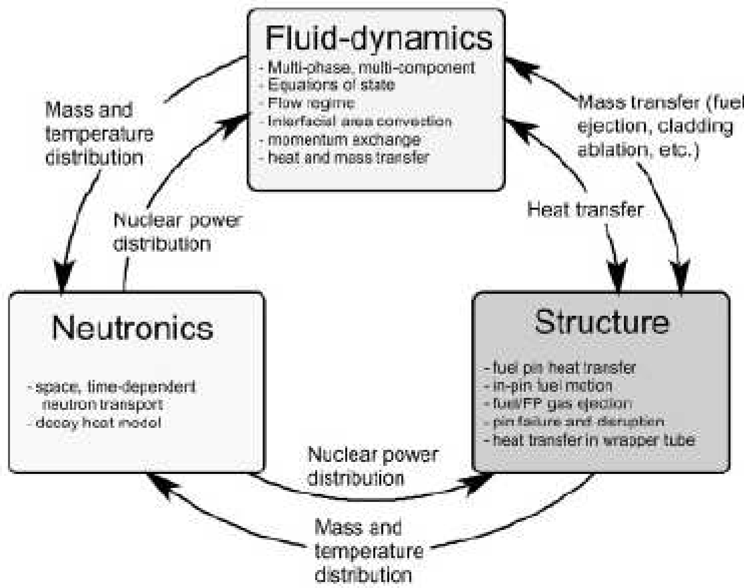

Figure 31 Overall Framework of the SIMMER-III Code

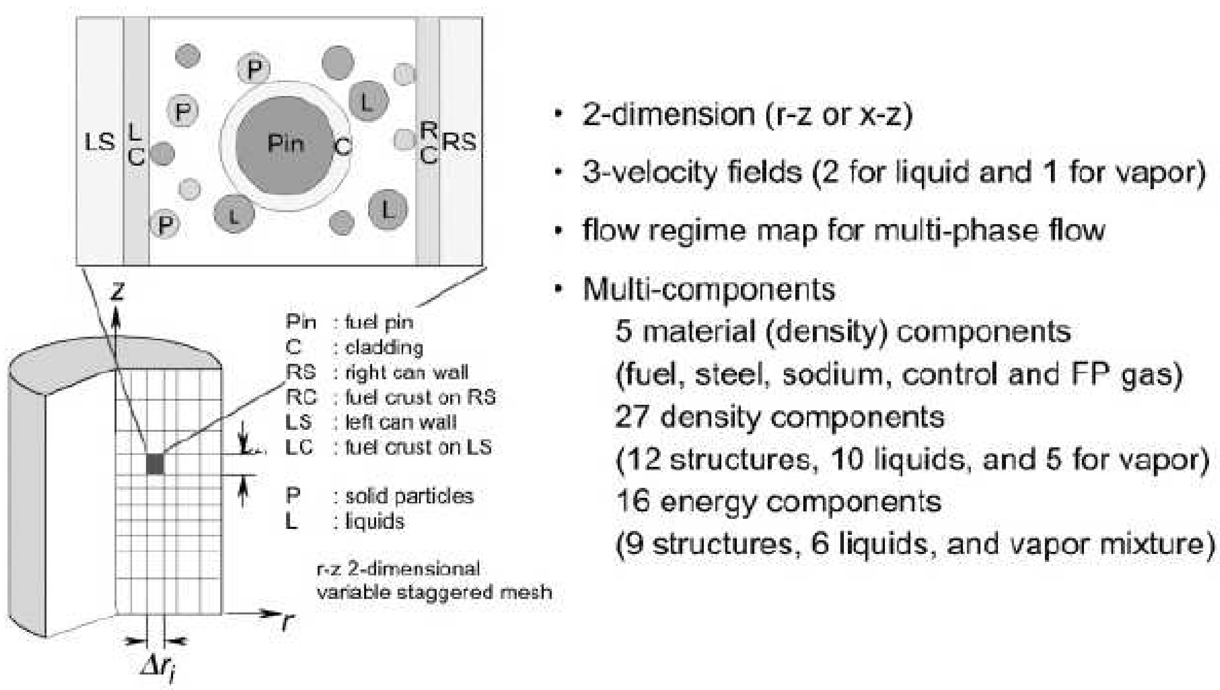

Figure 32 Multiphase, Multi-Component Fluid Dynamics Model in SIMMER-III

moving liquid fuel, stationary crust refrozen on the structure, moving solid particles and vapor. Thus the material mass distribution is described by 27 component mass distributions in the current version of the code. Some of these material components share the same temperature, which allows a reduced number of energy conservation equations, currently 16. The mobile components which include the liquids, vapors, and solid fuel particles, are assigned to one of three velocity fields. Two of these velocity fields are used for liquids and particles, and one is used for vapor components, allowing the relative motion of different moving components to be modeled. A fuel pin model is also provided, with the fuel pin pellet represented by several radial cells.

The fluid dynamics model uses a semi-implicit procedure to solve the inter-cell convection on an Eulerian staggered mesh. The material convection is treated using an explicit scheme, while the pressure propagation uses an implicit scheme. The intra-cell mass, momentum, and energy sources and inter-component transfer terms are determined separately from the inter-cell convection, an approach similar to the approach used in the LEVITATE/SAS4A code. The structure model describes the time-dependent configuration of the fuel pins and subassembly can walls. Two can walls can be modeled at the left and right of mesh domain. The presence of a can wall at the cell boundary prevents radial fuel convection and provides a surface on which the molten fuel can freeze and the vapor can condense. The breakup of the structure components is based on thermal conditions and the wall-thickness-dependent threshold for mechanical breakup. The fuel pin model includes the treatment of conduction in the solid fuel represented by several radial cells with individual temperatures, models of the molten fuel cavity and the fission gas plenum.

The space-time-dependent neutron kinetics model in SIMMER-III is based on an improved quasi-static method with a diffusion acceleration technique where the flux shape is calculated by a standard Sn neutron transport theory. Since the changes in the material number densities and temperatures play an important role, a cross-section model is included in the code to perform self-shielding calculations in order to determine the effective macroscopic cross-sections whenever the reactivity is updated.

5.2 Evaluation of CDA Scenarios in Japan

The evaluation of the mechanical consequences of a postulated core disruptive accident has been one of the major efforts in the safety analysis of oxide-fueled LMFBRs. The efforts to accumulate the experimental knowledge and to improve the computer codes by adding new models that describe oxide-fuel phenomenology and reflect the experimental results have continued steadily. During the 1970s and 1980s new experimental knowledge gained from TREAT, ACRR, CABRI, and SCARABEE test reactors was used to enhance and validate the SAS3D and SIMMER-II codes. These codes were used as the safety assessment tools in several oxide-fueled reactor safety analyses, with SAS3D being used to analyze the Initiating Phase (IP) events, and SIMMER-II describing the Transition Phase (TP) events. In the 1990s the existing and new experimental data from the CABRI and SCARABEE in-pile tests have been used for the further development and validation of a new generation of mechanistic computer codes, SAS4A and SIMMER-III, which have further advanced the LMFBR safety analysis. The progression of a postulated unprotected LOF accident in the prototype LMFBR with oxide fuel has been evaluated with these codes reflecting the latest experimental and analytical knowledge on CDAs [5-7]. Using the latest safety research knowledge based on experimental information in the context of the advanced accident analysis codes the CDA energetics was assessed to be more benign than predicted in previous analyses. The important new experimental data and the corresponding code models implemented in the SAS4A and SIMMER-III codes for the analysis of oxide-fueled core accident sequences are outlined below.

5.2.1 Initiating Phase Analysis

The initiating phase accident progression is analyzed with the SAS4A code. The oxide-fuel version, originally developed at Argone, was later modified in Japan to reflect the more recent oxide fuel experimental data obtained in the CABRI test series. The initiating phase events are driven initially by the insertion of sodium void reactivity. Subsequent reactivity changes due to fuel expansion, fuel and cladding relocation, and the Doppler effect play a role in determining the total reactivity changes and thus the energetics during this phase.

5.2.1.1 Initiating Phase Model Enhancements

The reactivity changes that occur during the initiating phase are directly related with various thermo-hydraulic phenomena such as coolant boiling, the transient response of the fuel pin, and the fuel breakup and relocation. Several model improvements based on the experimental data obtained from the CABRI experiments allow a more accurate description of these phenomena and their combined effects lead to significantly lower predicted energetics during the LOF sequence than previously calculated. These new models or model improvements describe the following phenomena:

a) The axial fuel expansion of the fuel was neglected in previous LOF analyses performed in Japan. However, based on the extensive experimental data obtained from the CABRI tests, this mechanism is now take into account under weak cladding restraint conditions [5-7]. The fuel stack above the uppermost fuel-cladding contact position freely expands axially, whereas the fuel column below this location is assumed to be completely stuck without contribution of the axial cladding expansion and without partial slip of the constrained fuel. This model was found to be appropriate in the analysis of the CABRI results for LOF-TOP conditions [5-7], although the simple treatment of the stuck fuel conditions somewhat underestimates the single-pin fuel expansion CABRI data;

b) The release of the plenum fission gas into the coolant channel upon cladding rupture was often observed in the LOF CABRI tests. The plenum gas pressure can be a driving force for the fuel stub motion described below and the blowdown of plenum fission gas has a favorable effect in suppressing the stub motion. The plenum fission gas released in the coolant channel also contributes to the fuel dispersal and because the coolant voiding has already expanded to the lower end of the fissile region at the time of the plenum gas release the additional void reactivity is limited;

c) The fission gas released from the molten or disrupted fuel, together with the sodium vapor, plays an important role in the early fuel dispersal. A significant experimental data base has been accumulated on irradiated fuel disruption under LOF conditions, showing that the amount of retained fission gas is 40-60% of the fission gas produced during irradiation. These experimentally based values have been used in the SAS4A fission gas models in the LOF sequence analysis;

d) The intact fuel pellets inside the cladding near the upper and lower core boundaries may move toward the core midplane after the occurrence of the fuel pin breakup. This phenomenon, referred to as "stub motion", is driven by the plenum gas pressure when the cladding restraint is lost due to the temperature rise. In the CABRI tests in which the transient power pulse was applied before the boiling extension to the plenum region, the stub motion of a few cm was observed at approximately 100 ms after the pin disruption. On the other hand, the stub motion did not occur if the power pulse was applied after the boiling extension to the plenum region. The modeling of the stub motion was not included in the original LEVITATE model, and understanding the details of the new models will require discussions with the Japanese developers;

e) The location of the initial fuel pin failure can significantly affect the energetics of the LOF events. The CABRI tests have indicated that a burst-type fuel pin failure occurred when the fission gas released from the fuel pellets increased the fuel pin internal pressure so that the cladding stress exceeded its temperature dependent strength. This temperature-dependent cladding failure mechanism implies that the pin failure always occurs in narrow range of the axial height that is well above the core mid-plane, about 65% of the core height. The ejection of the molten fuel through the cladding rip decreases the pressure in the molten fuel cavity and thus the subsequent axial rip extension is limited. The mechanistic of the burst-type fuel pin failure, reflecting the experimental knowledge obtained in the CABRI experiments, has been implemented in SAS4A and was used in the LOF accident analyses;

f) The uncertainties in the reactivity coefficients were reduced by the consistent evaluation of critical assembly experiments [5-8]. The resultant uncertainties were 20% for the void reactivity and 14% for the Doppler coefficient.

5.2.1.2 Initiating Phase Analysis Results

Simulations of a postulated LOF accident in the prototype LMFBR with oxide fuel were performed in Japan using the enhanced models described above for a core model with 34 channels and most probable conditions [4-9]. In this nominal case the stub motion was not taken into account because it occurs 100 ms after the fuel breakup and has no effect on the initial reactivity and power peak which occurs at 10 ms after the breakup. Figure 33 shows the power and reactivity histories for the nominal case. The individual reactivity components are shown in Figure 34. The maximum power is Pmax = 26.4 Pnominal and the maximum reactivity is 0.93 \$. Before the boiling onset the total reactivity is negative as a result of the balance of fuel axial expansion, coolant density, and Doppler reactivity changes. After the boiling onset the coolant voiding and cladding relocation insert positive reactivity and lead to power increase. This accelerates the fuel melting and disruption, leading to fuel dispersal which rapidly decreases the reactivity. The results of this nominal case indicate that the most probable outcome of the initiating phase is a non-energetic entry into the transition phase. A conservative case was also analyzed, in which several conservative assumptions were added simultaneously to the nominal case: the void reactivity was increased by 30%, the negative Doppler coefficient was decreased by 14%, the amount of fission gas which contributes to fuel dispersal was decreased by 50%, the cladding breakup stress was decreased by 30% to increase the coherency of coolant boiling and the stub motion was taken into account. In the conservative case the stub motion following the

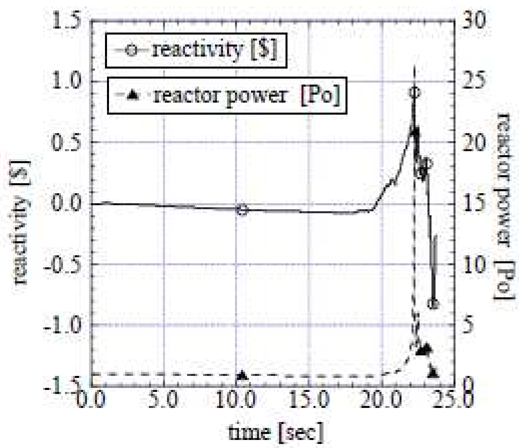

Figure 33 Total IP Reactivity and Power History for the Nominal LOF Case

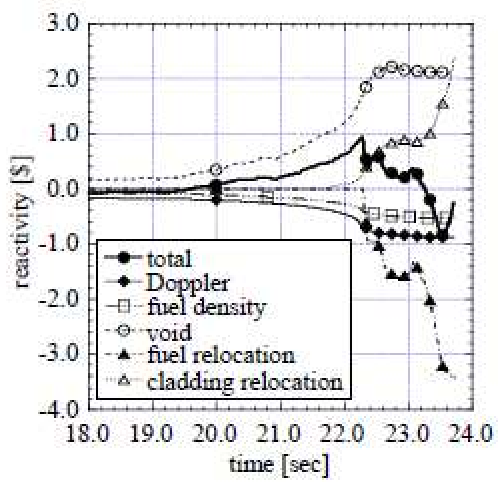

Figure 34 Reactivity Components during the IP Nominal LOF Case

fuel disruption inserted reactivity after the first power peak compensating the reactivity decrease due to the early fuel dispersal. Thus a second power peak was due mainly to the void reactivity and cladding reactivity, which melts the fuel particles and causes rapid fuel dispersal and reactivity decrease. The maximum fuel temperature is 3000 K and the maximum pressure is 6 atm. No appreciable mechanical energy would be generated at this enthalpy level. The energy level calculated in the conservative case is substantially lower than in former evaluations, due to the improvements of the models used in SAS4A, which reflect the knowledge obtained from the CABRI experiments and the reduction of the uncertainty of the reactivity coefficients.

5.2.2 Transition Phase Analysis

The objective of the Transition Phase (TP) analysis is to evaluate the sequence and energetics of the events that occur as the molten fuel region expands radially beyond the fuel subassembly boundaries and spreads from a subassembly scale pool to a core scale pool. The event progression in this phase is dominated by the increased fuel mobility resulting from the enlargement of the molten region and the associated thermo-hydraulic events that determine the fuel reactivity changes. Another important phenomenon is the escape of the fuel from the core through the open paths which become accessible to the molten fuel when the subassembly walls begin to melt. The TP analysis is performed using the SIMMER-III code, which was developed in a common effort by the Japan Nuclear Cycle Development Institute (JNC), the French Commission for Nuclear Energy (CEA), and Research Center Karlsruhe (FZK).

5.2.2.1 Transition Phase Model Enhancements

The Transition Phase analysis for the reference LMFBR with oxide fuel reported in [4-9] was performed with SIMMER-III reflecting the improved understanding of relevant phenomena gained from experimental studies and improved SIMMER-III models. The more model improvement and validation efforts relevant for this analysis include:

a) The progress in understanding the molten fuel escape through the Control Rod Guide Tube (CRGT), which was recognized as one of the dominating phenomena in the TP analysis, was taken into account. The breakup of the structure walls of the CRGT and the radial blanket, which are adjacent to the molten fuel pool, provides an efficient fuel escape path through these structures. The experimental results from SCARABEE [5-9] and CAMEL [5-10] experiments indicated that the wall melt-through occurred in a few seconds and that the fuel escaped efficiently through these structures and SIMMER-III analyses have shown similar results;

b) The size of the fuel particles plays an important role in determining the fuel motion since it determines the drag force between the particles and the surrounding fluid. It was shown that a slow power transient produced fuel particles of a fuel pellet size, while a fast power transient produced finer fuel particles with a diameter less than 1.0 mm due to fission gas driven dispersion.[5-11];

c) The axial blanket region is a direct path of fuel escape from the core within the fuel assembly which is available at the beginning of the TP. Fuel freezing in the blanket region can obstruct this fuel escape path and plays an important role in the TP. Experiments that have simulated the melting and relocation of cladding and subsequent freezing with solid fuel pellets [5-10] have been analyzed with SIMMER-III. The code was shown to predict conservatively short penetration lengths;

d) The boiling behavior of the molten steel and fuel mixture plays a key role determining the TP fuel motion and associated reactivity changes and has been studied in several experiments. The BF2 experiment in SCARABEE, which focused on a boiling fuel pool under in-pile nuclear heating was analyzed with the SIMMER-III code, showing that the SIMMER-III models can simulate the fuel boiling behavior reasonably well.

5.2.2.2 Transition from SAS4A IP Analysis to SIMMER-III TP Analysis

The initial conditions for the SIMMER-III TP analysis are based on the results of the IP analysis performed with SAS4A. The data mapping between the two codes is performed by the SAME-II code developed at JNC as an automated interfacing tool. Because the core representation and physical models used in SAS4A and SIMMER-III are not the same, the data mapping can have a significant effect on the subsequent TP calculations. A description of the SAME-II code does not appear to be available in the open literature. Direct discussions with the Japanese developers of the code will be needed in order to understand the impact the transition from the SAS4A analysis to SIMMER-III analysis may have on the LOF accident analysis. The timing of the transition from SAS4A to SIMMER-III is an important decision in the integrated LOF analysis. The SAS4A code is designed to treat the one-dimensional fuel relocation in a fuel assembly before the melting of the assembly wall, while SIMMER-III treats the two-dimensional fuel relocation that occurs after the melting of the assembly wall. Thus the transition from the SAS4A IP analysis to the SIMMER-III TP analysis must occur just before the first occurrence of an assembly wall melting. The LOF analysis described in [4-9] this transition was performed at a time when the reactor power was P= 0.9 Pnominal and the reactivity was about -1.0 $. Almost all the sodium was expelled from the core and the fuel was mostly disrupted except for the outermost core regions. Most of the disrupted fuel was still in the active core region and only 4% of the core fuel had penetrated into the axial blanket regions.

5.2.2.3 Transition Phase Analysis Results

In the nominal case the fuel penetration into the axial blanket and the fuel escape through the CRGT were both considered. However, it was assumed that the fuel freezes and forms blockages in the lower and upper regions of the CRGT where thick steel structures exist. The history of the power and reactivity calculated for the nominal case are shown in Figure 35. In the early stages of the TP analysis the reactivity increased due to gravity-driven downward fuel relocation causing the reactor power to increase to P= 2.0 Pnominal. The fission gas release due to fuel melting dispersed the fuel again, causing the reactivity and power to decrease. The failure of the CRGTs occurred between 2.4 s and 3.5 s into the TP, allowing the fuel to escape and further decrease the reactivity and power. At the same time the sodium vapor caused by the fuel contact with the coolant further accelerated the fuel penetration into the upper blanket. The fuel escape from the core decreased the reactivity continuously and the accident terminated without recriticality. The amount of fuel that left the core at the end of the calculation was 28% of the fuel inventory.

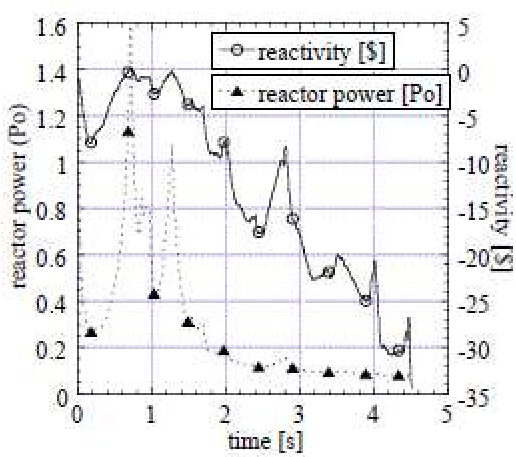

Figure 35 Power and Reactivity Histories for the TP Nominal Case

The geometry used in the SIMMER-III analysis of CRGT fuel removal is illustrated in Figure 36, where a single CRGT with six surrounding fuel subassemblies is modeled in two dimensions. SIMMER-III predicted that more than 50% of the active fuel is discharged into a CRGT channel within a few seconds after the subassembly wall melt-through.

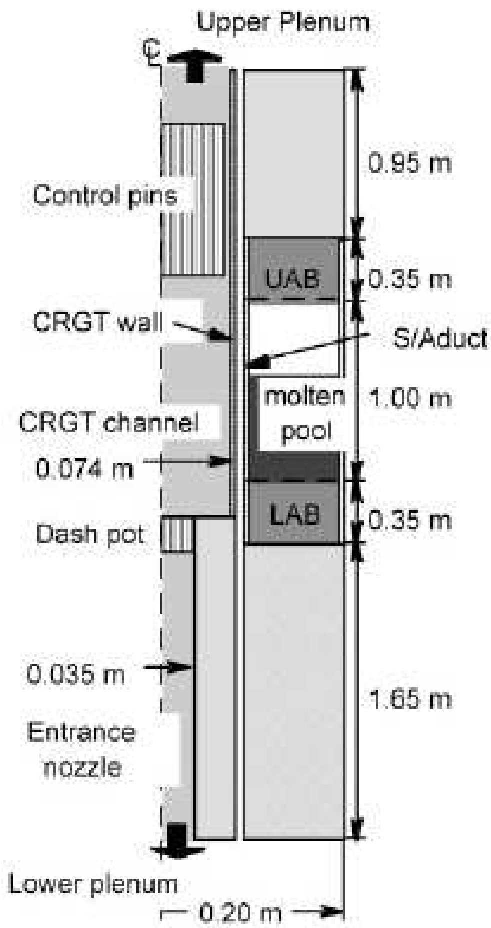

Figure 36 Geometry of Simmer-III Analysis of CRGT Fuel Removal

5.2.3 Integrated LOF Analysis Summary

The integrated LOF analysis using the SAS4A code for the initiating phase analysis coupled with the SIMMER-III code used the transition phase analysis is a significant step towards a consistent mechanistic assessment of the CDA energetics. The knowledge about CDA phenomenology gained from tests in the CABRI and SCARABEE reactors and from separate effects out-of-pile experiments was incorporated in the enhanced models of the SAS4A and SIMMER-III codes. The LOF analyses using these models indicated a significant reduction in the postulated CDA energetics due to the presence of multiple inherent physical mechanisms and design features which mitigate the consequence of the accident. Additional experimental data and corresponding model improvement and validation will further improve our understanding of the CDA progression.

5.3 Evaluation of CDA Scenarios in France

The study of whole-core disruptive accidents, initiated by a loss of electric power supply and simultaneous failure of the shutdown system, was initially considered to envelope many smaller less-severe potential accident situations, which then would not require special consideration. The concern remained however as to whether other less severe but more likely initiators, such as coolant channel blockage in a fuel assembly, could play a more important role in the overall risk and lead to hazardous core melting scenarios. Because of the difficulty in defining the size of the coolant channel blockage, it was decided to consider the most severe possible blockage, a Total Instantaneous Blockage (TIB) at the entrance of a subassembly at full power. In the study of this hypothetical accident the control systems are considered to remain in operation, but because the flow in the subassembly is assumed to stop instantaneously the detection systems do not react, at least for some time, and the accident evolves at full power. To ensure that this accident will not be an initiator for a whole-core meltdown, the licensing authorities have required a demonstration that the meltdown of core materials will not propagate beyond the six neighboring assemblies.

The analysis of the TIB accident was performed using the SIMMER-III code [5-13]. The TIB scenario has been investigated in the SCARABEE experimental program [5-14], and knowledge gained from these experiments has been used to enhance the SIMMER-III models used in the TIB analysis. The SIMMER-III calculation describes the faulty subassembly surrounded by six neighboring subassemblies separated from the faulty assembly by the inter-assembly gap. The location of the faulty assembly in the core is shown in Figure 37. The transient phase is induced by a TIB that occurs in the selected fuel assembly at nominal operating power. The geometry modeled by SIMMER-III is illustrated in Figure 38, which shows that the fissile pin bundle with 217 fuel pins is represented by nine concentric regions. The six neighboring subassemblies are represented in a second concentric pin region.

The steady state conditions are obtained with a SIMMER-III quasi-transient calculation that increases the power to the nominal level and is followed by a constant power period. The SIMMER-III neutron transport model is activated during the transient phase of the simulation which describes the TIB events and begins by reducing the inlet the sodium inlet flow rate to zero. The sequence of events calculated by SIMMER-III includes: a) sodium boiling and clad dryout, b) cladding melting and draining, leading to the formation of a steel plug in the colder subassembly zone above the inlet, c) liquid steel pool formation above the frozen steel plug, d) fuel melting and collapsing into the liquid steel pool, leading to boiling of the steel-fuel mixture, e) melt-through of the hex-can wall and molten material entering the subassembly gap and freezing on the surrounding hexagonal wrappers, and causing sodium boiling in the first pin region of the surrounding subassemblies.

During the TIB sequence the material relocation induces small power variations in the faulty fuel assembly illustrated in Figure 39: a) a power increase at the time of the fuel pin disruption due to

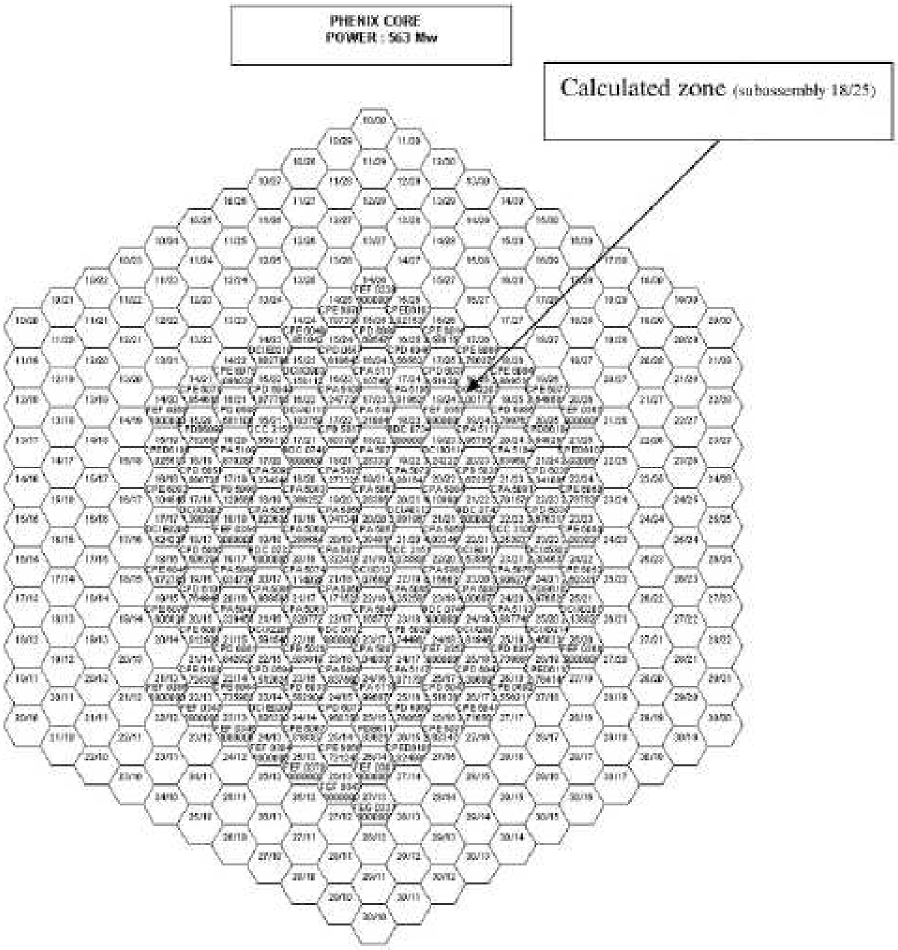

Figure 37 Phoenix Core Configuration and Location of Postulated Blockage

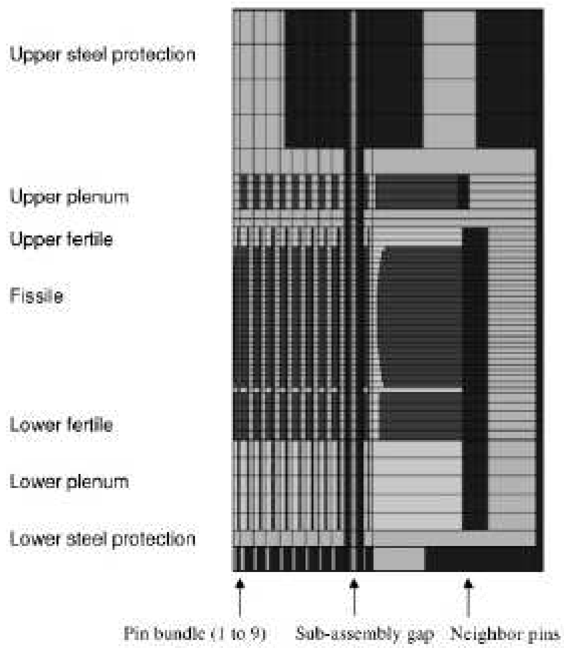

Figure 38 SIMMER-III Geometry of the Assemblies Considered in TIB Analysis

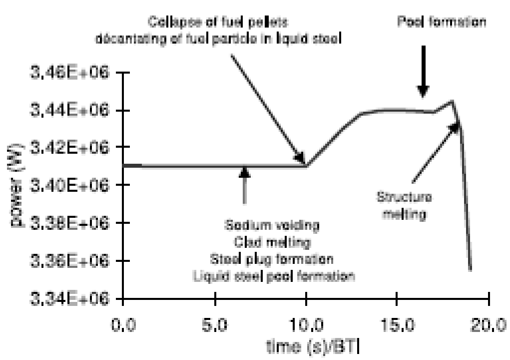

Figure 39 Power Generated in the Blocked Assembly during the Postulated TIB Accident

fuel compaction, and b) a power decrease at a later time due to the fuel-steel mixture boiling and then relocation to the neighboring assemblies. These small power variations are associated with core reactivity changes which parametric studies have shown can be detected with sufficient margins earlier by a positive variation in core reactivity and/or later by a negative reactivity variation for any core position of the blocked subassembly. Thus, the reactor scram can be triggered sufficiently early to limit the molten zone to the blocked assembly and the neighboring assemblies, taking into account the decay power and cooling after scram.

5.4 Evaluation of Severe Accident Scenarios in US

The metal-fuel versions of the SAS4A and SASSYS code systems are used to analyze the core behavior under beyond-design-base transient conditions for various advanced liquid-metal reactor designs. The results of theses analyses provide an assessment the outcomes of various accident sequences. Due to the emphasis on inherent safety for LMRs, in addition to diverse and redundant systems for reactor protection, models that represent the various reactivity feedback mechanisms such as the radial core expansion, control rod drive line expansion, and in-pin fuel relocation have been included in these codes, allowing the study of the response of various LMR designs during postulated unprotected accidents.

In order to quantify the inherent safety margins available in metallic and oxide-fueled reactors, unprotected LOF and TOP accidents in two representative US reactor designs have been analyzed [4-12] using the SASSYS computer code, including consideration of the inherent safety mechanisms described above. One design was a 1000 MWe plant and the second was a 365 MWe plant. Both plants have a pool-type primary system layout. An oxide-fueled core and a metal-fuelled core were designed for each plant size. A heterogeneous core layout, with internal blanket breeder subassemblies arranged within the core, was used for both cores. Results from a study of an unprotected LOF accident in the large reactor design are shown in Figure 40. As the accident develops, the coolant temperature rises, causing radial core expansion and control rod drive elongation. No in-pin fuel extrusion was observed due to the fuel chilling in all cases. The major positive reactivity feedback is provided by the Doppler effect in the cooling fuel. Because the temperature drop is smaller in the high conductivity metal fuel, the metallic core experiences a much milder reactivity and temperature transient. After 1000 s following the accident initiation, the power level in the metallic core is approaching decay heat removal levels and the coolant temperature has risen less than 100 C. In the oxide core the power remains significantly above decay heat removal levels, and the peak coolant temperature is approaching the boiling point.

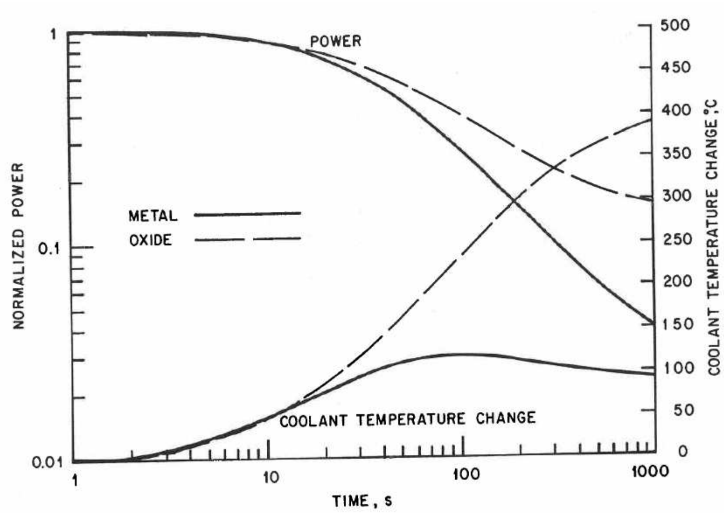

Figure 40 Reactor Power and Peak Coolant Temperature Change Histories
for the Unprotected LOF Accident

Analyses of very severe unprotected accidents that could lead to core disruption have also been performed with the SAS4A code, in order to quantify the energy deposition during such events. A SAS4A analysis of the thermal-hydraulic and neutronic events that occur in a low void worth metal fuel core during a very rapid unprotected LOF accident with a flow decay half-time of 0.3 s is described in [5-15]. This LOF was selected in order to evaluate the consequences of extremely unlikely accident initiators that would lead to fuel pin failure and subsequent fuel relocation. The only mechanistic initiator that can lead to such a rapid LOF is, possibly, a severe earthquake. For slower LOFs, pin failure and fuel relocation do not occur, as negative reactivity from other core feedback effects has enough time to counteract the positive reactivity introduced by the early sodium boiling.

As the coolant flow decreases, the fuel and coolant temperatures begin to increase. Eventually, boiling begins in some of the channels, leading to an increase in reactivity and power. These events are described by the pre-boiling and boiling modules of SAS4A. At the same time, fuel melting begins, and in-pin molten fuel relocation may occur prior to cladding failure. These events are described by the PINACLE model which interacts closely with the DEFORM5 pin mechanics model. DEFORM5 monitors the cladding conditions and predicts the location and timing of cladding failure. After cladding failure, fuel relocation occurs both in the coolant channel and inside the molten pin cavities. Rapid pin disruption can occur due to fuel melting and cladding eutectic penetration. The axial fuel dispersal is driven by the fission gas pressure and the pressure of the remaining sodium vapor. The post-failure events are described by the LEVITATE model. All reactivity effects are integrated by the SAS4A neutronics module, which provides the feedback for the reactor power calculations.

During the accident sequence, various assemblies in the core behave differently because of their exposure to various power and flow conditions. To account for this, the SAS4A code models the core as a number of distinct channels. Each channel contains a number of assemblies that are assumed to be identical. In the LOF calculation described a 10-channel core model was used and all channels are assumed to contain irradiated fuel with 10 atom % burnup. The results of the SAS4A calculations are illustrated in Figure 41, which shows the normalized power and reactivity histories. Because of the decrease in sodium flow, the fuel and sodium temperatures increase gradually, leading eventually to sodium boiling and fuel melting. The positive reactivity due to coolant boiling causes a gradual increase in power, which reaches $P= 4.1\ P_0$ at t=4.1 s when the first fuel pin failure occurs in channel with the axial failure location at $H=0.83\ H_{core}$ (83% of the core height). The axial failure location plays an important part in determining the reactivity contribution due to the early fuel relocation. Although the reactivity effect of the fuel dispersal in the coolant channel is generally negative, the in-pin fuel relocation can have a negative or positive reactivity effect depending on the failure axial location. Because the early fuel relocation effect is dominated by the in-pin fuel relocation, a failure location near the core mid-plane can lead to a temporary positive fuel reactivity contribution. However, when the axial failure location is far enough from the core midplane, as is the case in this LOF analysis, the effect of the fuel relocation is negative from the beginning. Axial failure locations located in the upper part of the fuel pin are favored in metal fuel cores because of the axial fuel temperature profile.

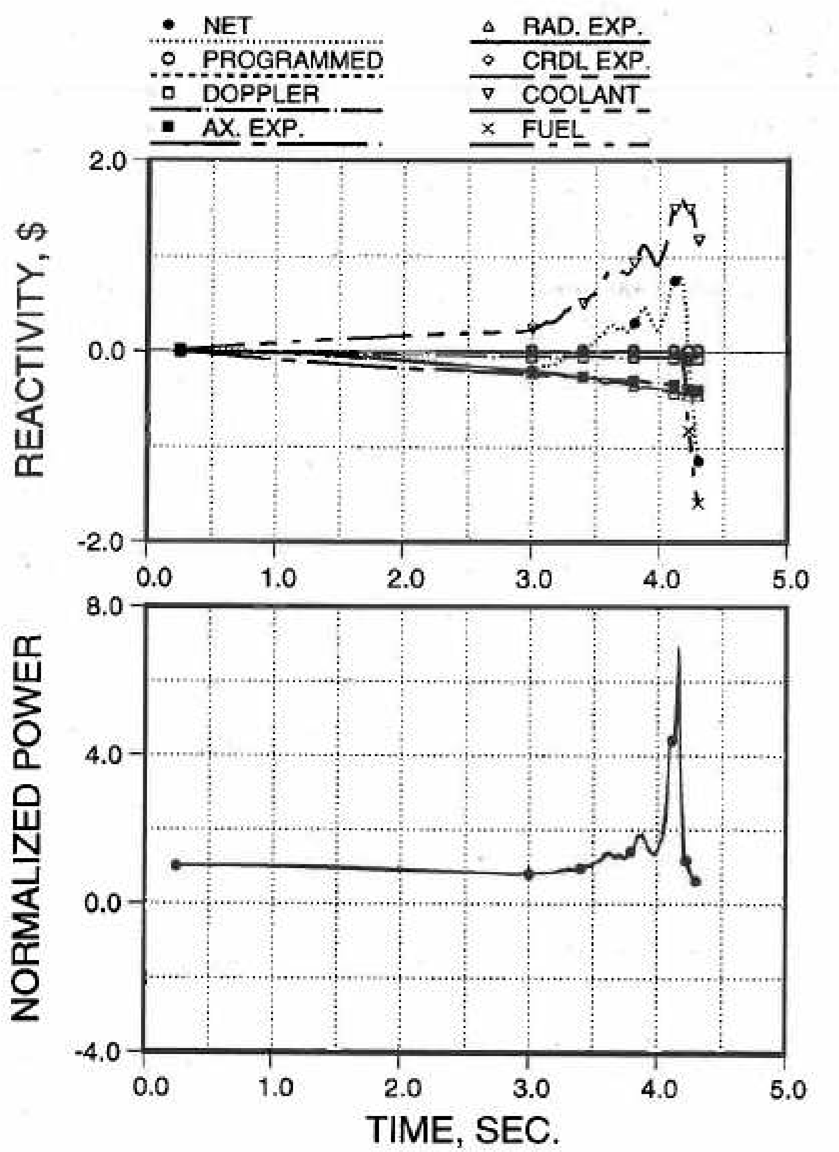

Figure 41 Power and Reactivity Histories for a Rapid Unprotected LOF

During the first 60 ms after the first pin failure the positive boiling contribution remains larger than the fuel negative reactivity contribution and reactivity continues to increase, reaching a maximum value of 0.82 $ and a maximum power $P = 6.5\ P_0$. Afterward, the negative fuel reactivity addition becomes dominant and the reactivity and power decrease rapidly. Fuel pin failures occur in channels 2 and 3, in both channels at H = 0.83 Hcore, causing the fuel relocation to have a negative reactivity contribution from the beginning. At 4.3 s when the calculation was terminated, the net reactivity was -1.14 $ and the power $P = 0.6\ P_0$. The continued fuel relocation was introducing negative reactivity at a rate of -9.6 $/s. The results of this LOF analysis show that even in the extremely unlikely event of a nearly instantaneous LOF without scram that leads to a CDA in a metal fuel core, the inherent safety characteristics of the metal fuel core ensure a benign initiating phase sequence of events.

6. CONCLUSIONS

This report presents a review of the role of severe accidents in the safety analysis of SFRs, design features developed or considered for the prevention and/or mitigation of SFR severe accidents, and current SFR safety approaches in the US, Japan, and France.

Despite substantial differences in the reactor and plant design there are many similarities between the safety approaches to severe accidents in Japan and France. In both cases the CDAs are considered in the plant design. The goal is to improve both the prevention and mitigation functions. The plant design and safety analysis efforts are focused on oxide-fueled SFRs.

The central element in the prevention approach is the addition of a third, passively activated, shutdown system. While the other two redundant reactivity shutdown systems are considered for Design Basis Events, the third shutdown system is considered for severe accidents. When the third shutdown system is considered, the combined probability occurrence of CDA initiators and failure of all three shutdown systems is significantly below 10^{-7} per year. Thus the CDAs are excluded from the Design Basis Events, and are considered in the safety analysis as Design Extension Conditions (DEC) or Beyond Design Basis Events (BDBE).

The mitigation approach is focused on preventing re-criticality and assuring in-vessel retention of core materials if a CDA does occur. Both JSFR and EFR include a debris tray, designed to ensure the retention of the relocated core materials, to protect the reactor vessel from direct contact with the molten fuel, and to maintain a debris bed that is both sub-critical and coolable. A feature unique to the JSFR design is the modified fuel assembly, M-FAIDUS, designed to allow an early escape from the core of the molten fuel and thus prevent a severe power burst during the transition phase.

CDA analyses, traditionally based on ULOF, are performed for JSFR and EFR to assure that there are no cliff-edge effects and to provide an understanding of the relative importance of core characteristics for the design and beyond design requirements. In Japan these analyses have used the SAS4A code for the initiating phase and on the SIMMER code for the transition phase. A modified version SIMMER named PAMR is being developed for the analysis of long-term post accident events. In France the codes used for the initiating phase analysis include SAS4A, FRAX, and PHYSURAC, while SIMMER is used for the transition phase analysis.

The US approach to SFR severe accidents has focused on the prevention of core disruption in unprotected overpower and under-cooling accidents through the provision in the design for diverse and redundant shutdown systems, complemented by inherent, passive mechanisms which respond to the accident conditions and act to restore the balance between the reactor energy production and the coolant system energy removal. The plant design and safety analysis efforts are focused on metal-fueled SFRs. Metal-fueled reactor designs provide significant advantages in the prevention of postulated CDAs and mitigation of their consequences.

To assure a self-limiting response to accident conditions, specific features are included in the system design, which respond to the increase in the coolant temperature rise through the core associated with the unprotected loss-of-flow and overpower accidents. Negative reactivity feedbacks keyed to this coolant temperature increase can be effective in limiting accident consequences. Two such mechanisms are provided by: a) radial core expansion driven by subassembly duct and above-core load pad thermal expansion, and b) differential thermal expansion of control rod drives and the core support structure, leading to a net insertion of the control rods. An additional inherent negative reactivity feedback mechanism provided by the metal fuel pins is the pre-failure in-pin fuel relocation, which can play an important role in the mitigation of unprotected severe accidents consequences.

US researchers have developed significant safety design advancements through the Advanced Liquid Metal Reactor (ALMR) and Integral Fast Reactor (IFR) programs. These advancements constitute a progressive approach to prevention of severe accident consequences, even in the event of accident initiators that would in the past have been assumed to lead to coolant boiling, cladding failures, and fuel melting. This approach, exemplified by the EBR-II Shutdown Heat Removal Test series and the FFTF Inherent Safety Test series, utilizes the unique performance properties of sodium-cooled fast reactors to provide self protection through fission power reduction to shutdown and natural coolant circulation for shutdown heat removal. This level of safety performance can be assured through selection of materials and arrangement of components, and furthermore it can be verified by full scale testing. Detailed deterministic analyses using modern simulation techniques can be employed to address phenomenological uncertainties and to extend testing and verification results to new reactor and plant designs. Risk-informed probabilistic analyses can be employed to assess the likelihood of accident sequences that might exceed the enhanced safety margins provided by the ALMR/IFR safety design approach. This safety approach focused on the prevention of unprotected accident consequences through selection of materials and arrangement of components can make a significant contribution to enhancing the inherent safety characteristics of the Advanced Sodium Fast Reactor prototype.

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