GA-A16138

THE SAFETY PROGRAM PLAN FOR THE GAS-COOLED FAST REACTOR

by
A. TORRI* and B. E. BOYACK**
General Atomic Company
and
A. P. KELLEY, JR.*
Helium Breeder Associates

Work supported in part under
Contract DE-AT03-76SF71023
for the San Francisco Operations Office
Department of Energy
and under Interim 1978 GCFR Program Agreement for
Helium Breeder Associates

*Present Address: Pickard, Lowe and Garrick, Inc. Solana Beach, California
**Present Address: Los Alamos Scientific Laboratory, Los Alamos, New Mexico

GENERAL ATOMIC PROJECT 6214
DATE PUBLISHED: DECEMBER 1980

GENERAL ATOMIC COMPANY

ABSTRACT

The Gas Cooled Fast Breeder Reactor (GCFR) Safety Program has developed and adopted a probabilistic risk framework. This report defines a risk goal and quantifies the defense-in-depth concept by defining distinct lines of protection (LOPs). Each LOP has a quantitative frequency and consequence goal which is consistent with the overall risk goal. LOPs -1 through -3 are dedicated to preventing accidents. The normal operating systems constitute LOP-1. LOP-2 comprises the dedicated safety systems, and LOP-3 includes a new LOP made up of inherent safety features to prevent unusual events from progressing into severe accidents. LOPs -1 and -2 are design features normally in current nuclear power plants.

LOPs -4 through -6 are dedicated to mitigating the consequences of accidents if LOPs -1 through -3 fail and lead to core melting. LOP-4 exploits the prestressed concrete reactor vessel (PCRV) as a barrier to contain accidents and to mitigate the release of activity. LOP-5 assures that the containment serves as an effective barrier against releasing activity to the environment if the LOP-4 barrier is postulated to fail. LOP-6 quantitatively reduces the consequences due to site and environmental effects, including emergency planning.

This plan is intended to comprehensively state the GCFR safety approach. This plan will be implemented in detail through the implementation plan for accident prevention (i.e., the GCFR Plant Specification for reliability) and the implementation plan for accident mitigation, respectively.

CONTENTS

FIGURES

TABLES

1. INTRODUCTION

1.1. PURPOSE

This document is meant to plan for a timely and orderly execution of the Gas-Cooled Fast Breeder Reactor (GCFR) Safety Program. This plan establishes a logical framework for identifying the technology to demonstrate that the GCFR can achieve the requisite degree of public risk safety and still compete economically wth alternate power generation technologies.

This plan is intended to identify all potential areas of investigation within a comprehensive framework. This plan is not intended to assess state-of-the-art knowledge nor to prioritize and detail safety program tasks and schedules; the Safety Program Implementation Plan will include these activities as the next logical step in developing the program.

1.2. SCOPE

Nuclear electricity generation has two categories of public health and safety considerations: (1) risks associated with operating the nuclear plant and (2) risks associated with nuclear fuel supply and disposal outside the plant. This plan deals only with risks associated with operating the GCFR nuclear plant.

Risks associated with operating the nuclear plant may be further categorized in terms of the location and magnitude of plant radioactivity. As long as adequate cooling is provided, most radioactivity is located in the fuel pins within the reactor core. Other smaller plant radioactivity sites include the spent fuel and waste gases, liquids, and solids. This plan deals with risks from all these plant radioactivity sites; however, because by far the largest radioactivity inventory is within the reactor core, this plan emphasizes risks associated with reactor core faults.

For large amounts of radioactivity to be released from the core fuel, the core must severely overheat and essentially melt to present any potential hazard to the public. Risk measures the plant hazard (i.e., the probability of a given radioactivity release to the environment). Thus, all levels of plant operation should be studied to ensure that an appropriate or acceptable level of risk is not exceeded; this study should reduce accident probabilities and/or consequences. This plan therefore addresses the probability of fuel melting accidents and the ability of the plant to mitigate fuel melting consequences.

This plan identifies tasks to accomplish the following:

1. Define safety goals and criteria.
2. Develop analytical models.
3. Conduct analyses to
    a. Assess criteria compliance.
    b. Establish test requirements for confirming models or design performance.
    c. Identify necessary design improvements.

This plan does not identify milestones, funding, priorities, nor schedules; a separate periodic implementation plan will include this information.

1.3. BACKGROUND

1.3.1. Safety Design Philosophy

Federal regulations* in applying for a nuclear power plant construction license require that the preliminary safety analysis report assess the risk to public health and safety resulting from facility operation, determine the safety margins during all stages of plant operation, and determine the adequacy of safety related structures, systems, and components. This risk assessment has traditionally been made within the context of "multiple levels of safety design" by deterministically evaluating conservative plant conditions ranging from anticipated operating modes to accident conditions of exceedingly low probability. The Nuclear Regulatory Commission (NRC) made the final judgment that no undue risk would result from plant operation on the basis of this spectrum of evaluations. The limited number of serious accidents and the absence of harm to any member of the public even though 70 light water reactors (LWRs) are operating verifies the judgment.

1.3.2. Design Basis Events

To quantitatively assess the safety assurance levels of a particular reactor design, standard practice defines an enveloping set of events, or design basis, which that design must accommodate with little or no damage. The NRC requires the license applicant to demonstrate that the plant can survive all events within the design basis without substantial core damage and in accordance with requirements in applicable sections of the Code of Federal Regulations.

1.3.3. Core Disruptive Accident Considerations

For fast reactors, a class of more severe accidents can be postulated which could lead to partial or complete core disruption. These extremely

---

*Title 10 of the Code of Federal Regulations (10CFR) (Refs. 1-1 through 1-3).

low probability accidents, commonly referred to as core disruptive accidents (CDAs), have postulated consequences which range up to those which substantially impact public health and safety. These large consequences arise because the fuel in a fast reactor core is not in its most reactive configuration. Under CDA conditions, material motions in a partially disrupted or molten core could theoretically release enough energy to disrupt the integrity of the primary system boundary and the reactor containment building.

Because large consequences are possible, hypothetical core disruptive accidents (HCDAs) significantly influenced fast reactor licensing and safety approval to date. However, they have been treated as accidents beyond the design basis, and they have not been accommodated under the same rigorous guidelines as events within the design basis. In the recent Fast Flux Test Facility (FFTF) and Clinch River Breeder Reactor (CRBR) safety evaluations, HCDAs were extensively analyzed to demonstrate that substantial structural and thermal margins existed in the plant to accommodate the nominally predicted consequences of a range of postulated CDAs. Clearly, however, CDAs have been treated as accidents beyond the design basis only on the premise that they have been or could be shown to be sufficiently improbable.

1.3.4. Probabilistic Analyses and Risk Assessment

Over the past 20 years, fast reactor safety analysts have developed more detailed mechanistic models of the various accident-related phenomena. The ever-larger accident analysis computer codes containing these models have predicted the various scenarios of interest with increasing detail.

While detailed mechanistic analyses help safety analysts better appreciate the range of consequences which should be associated with each scenario, they have not addressed two important aspects of safety analysis:

1. Low-probability system failures which initiate an accident sequence.
2. The probability distribution of predicted consequences.

Available accident analysis tools can predict a range of consequences for each accident of concern. This range of consequences is due to the remaining phenomena uncertainties and lack of sufficiently detailed models.

To address the important issues of initiator probability and consequence distribution, reactor safety analysis uses probabilistic analysis methods including event trees, fault trees, cause-consequence analysis, failure modes and effects analysis, and sensitivity analysis. Consequence and probability information can estimate risk.

In the past few years, risk assessments have been associated with a wide variety of endeavors. The Reactor Safety Study (RSS) (Ref. 1-4) conducted on LWRs by the NRC established the usefulness of risk assessment for reactor safety analysis. More recently, similar studies were completed for liquid metal fast breeder reactors (LMFBRs) and high temperature gas-cooled reactors (HTGRs) (Refs. 1-5 and 1-6). These risk assessments quantitatively assessed accident risks and placed these risks in perspective with other societal risks; this objective evidence for judging acceptability has been made available to the public and regulatory agencies responsible for licensing. Risk assessment methodology is expected to grow in fast reactor safety analysis because it can meaningfully treat the consequences of low-probability events.

1.3.5. GCFR Safety Characteristics

The GCFR has safety-related characteristics intrinsic to using a helium coolant:

1. Chemically inert. Helium is noncorrosive and will not chemically react with other substances.

2. Radioactively stable. Helium is not activated in-core, leading to low circulating activity levels and eased maintenance conditions.

3. Single phase. Helium does not undergo phase changes detrimental to heat transfer upon pressure loss or overtemperature.

4. Neutronically transparent. Change in helium density has minimal effect on core reactivity.

5. Optically transparent. Helium transparency permits remote visual inspection of primary coolant system components, facilitating fuel handling and other maintenance and surveillance.

The chief intrinsic safety drawback of helium coolant is that it has a relatively low volumetric heat capacity. Other design and inherent features compensate for this problem.

Initial concept design work on the GCFR began in the U.S. in the early 1960s; since the late 1960s, GCFR design work has proceeded with considerable international cooperation. On the industrial side, cooperation between General Atomic Company (GA) and the Kraftwerk Union Aktiengesellschaft (KWU) of the Federal Republic of Germany (FRG) has steadily increased in design, research and development (R&D), and safety studies directed toward joint participation in a demonstration plant program. Major GCFR R&D programs are under way at national laboratories in the U.S. and the FRG. GCFR development work is also being performed by experimental core heat transfer studies in Switzerland and Germany and in-pile loop testing in the Belgium Reactor-2 (BR-2) at Mol.

Being developed is a 350-MW(e) GCFR demonstration plant design employing three main cooling loops. The entire primary coolant system is enclosed within a massive prestressed concrete reactor vessel (PCRV), as shown in Fig. 1-1. The reactor core and its associated structural support and shielding components are located in the central PCRV cavity. The three main cooling loops, each with a steam generator and a helium circulator, are

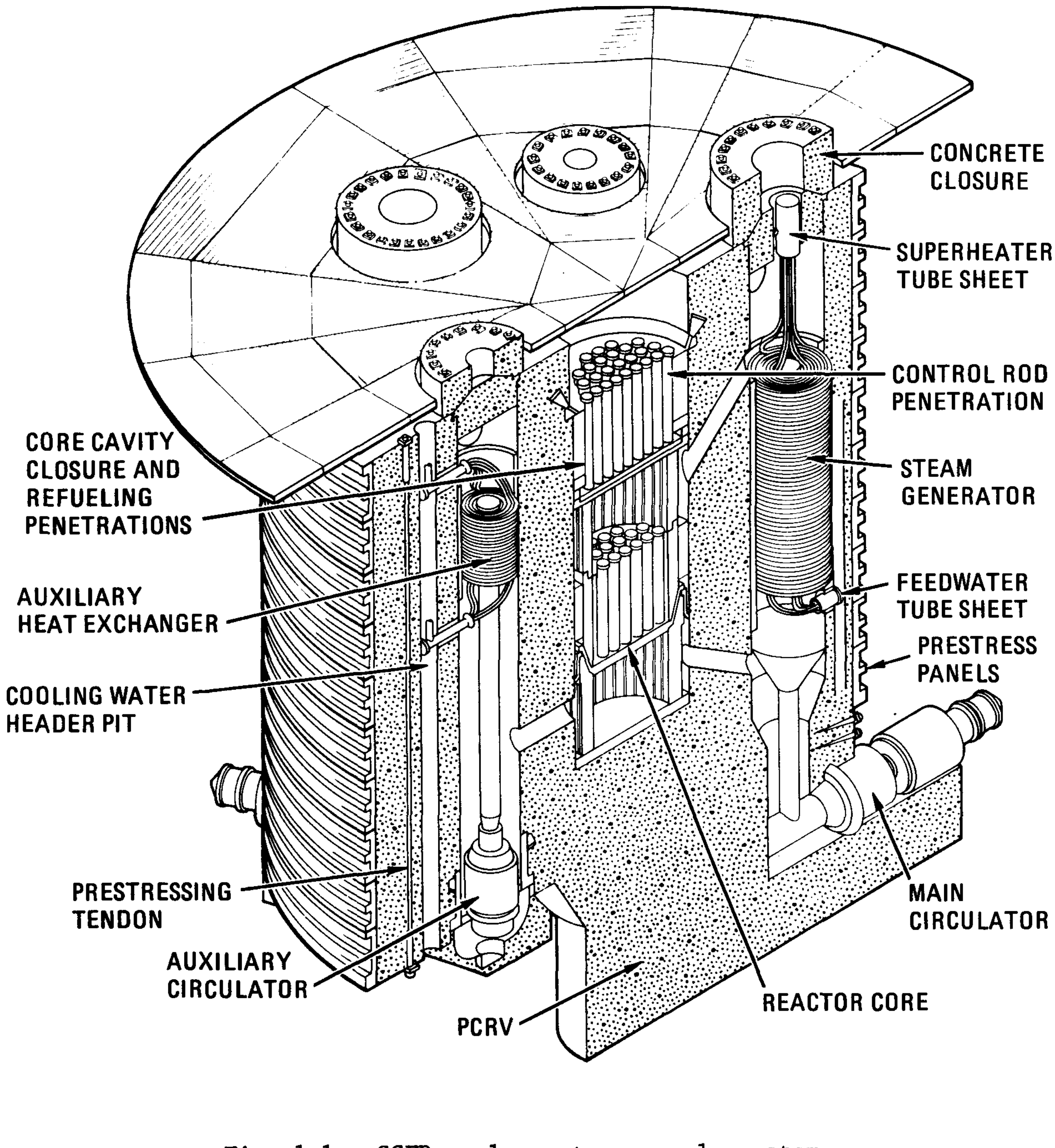

Fig. 1-1. GCFR nuclear steam supply system

located in three peripheral cavities in the vessel walls. Three auxiliary loops are contained in smaller cavities located in the PCRV walls between the steam generator cavities. The balance of plant is very similar to conventional LWR systems.

The GCFR licensing process began in the U.S. in 1971 when GA submitted a Preliminary Safety Information Document (PSID) (Ref. 1-7) to the then Atomic Energy Commission (AEC) Directorate of Licensing (DOL). In August 1974, the DOL issued a Preapplication Safety Evaluation Report (SER) (Ref. 1-8), and in November 1974, the Advisory Committee on Reactor Safeguards (ACRS) (Ref. 1-9) issued an interim letter to the AEC chairman. The DOL SER identified several areas of concern but concluded that the GCFR demonstration plant can potentially be designed and constructed to operate without undue risk to the public health and safety. The ACRS interim letter recognized certain GCFR safety advantages and identified several areas which it felt deserved further evaluation.

1.4. LINES OF PROTECTION

The GCFR plant and other nuclear plants provide primary physical barriers to protect the public from exposure to the core radioactivity. These barriers include the steel clad, which encloses the core fuel; the reactor vessel, which houses the core and coolant; the containment building, which houses the nuclear steam supply system (NSSS); and the site itself, which places distance between the public and the plant. Maintenance of the first barrier has rightfully received the traditional first priority in the plant design. Three independent and separate lines of protection (LOPs) maintain this first barrier:

1. The normal operating systems which provide the normal electrical power generation and protect the fuel and clad from becoming overheated.

2. The dedicated safety systems which protect the core only when the normal operating systems fail.

3. Inherent features which ensure that clad damage would be limited even if the above systems fail.

The goals of the GCFR Safety Program Plan will primarily be met by developing six separate and independent LOPs. The first three LOPs (operating systems, dedicated safety systems, and inherent features) maintain gross cladding integrity, while the remaining three LOPs (primary vessel, secondary containment, and the site) mitigate the consequences of accidents resulting in the release of radioactivity from the core. Each LOP provides a sequential and quantifiable risk barrier between the public and the radiological hazards associated with postulated GCFR accidents, as illustrated in Fig. 1-2. The six LOPs and their functions are described more specifically below:

1. Operating systems reliability. LOP-1 minimizes the incidents requiring plant shutdown and provides a first means to reliably shut down and cool down the reactor core following all residual occurrences which require shutdown. To accomplish this safety function, LOP-1 employs the GCFR operational and design features that provide normal electrical power generation: reactor core, reactor vessel, reactor internals, plant control and instrumentation, main loop cooling system, control rod system, and related balance of plant systems.

2. Dedicated safety systems. LOP-2 provides automatic reliable core shutdown and cooldown in the event that the LOP-1 operating systems fail. LOP-2 includes systems dedicated to providing this safety function that are independent of normal electrical power generation systems: core auxiliary cooling system, plant protection system, and related balance of plant systems.

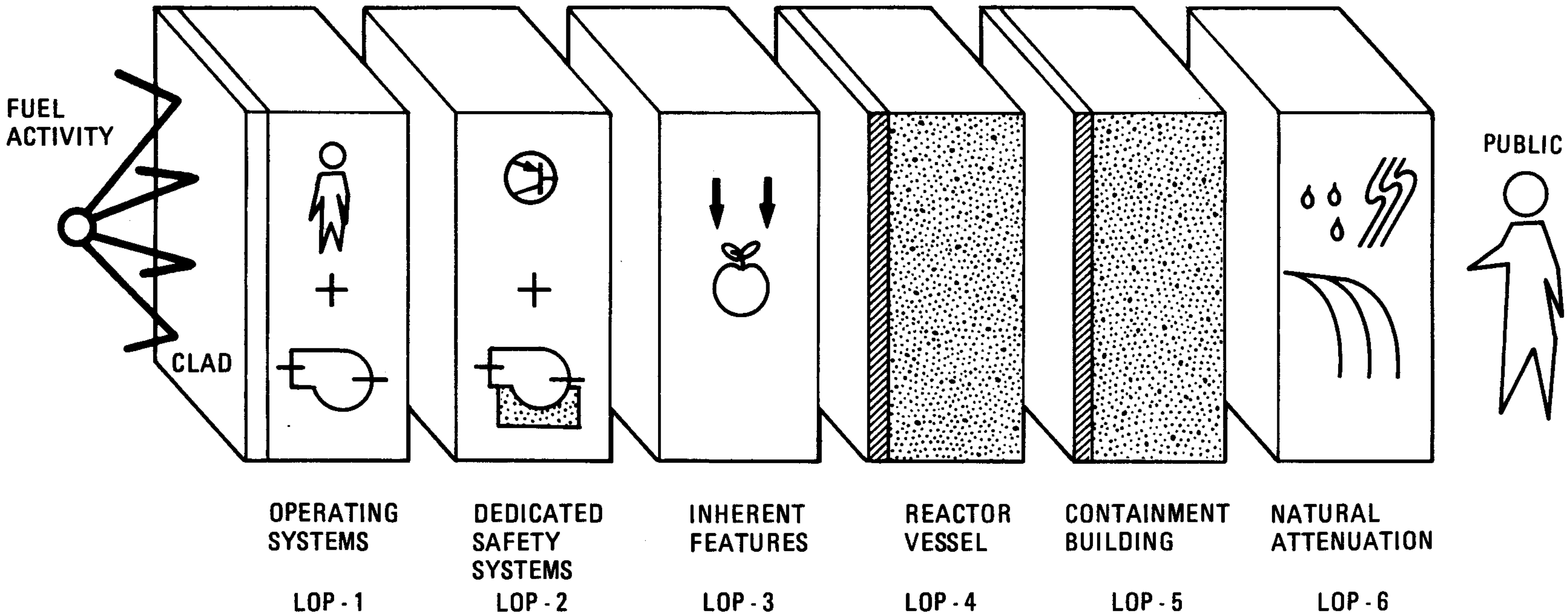

Fig. 1-2. Safety program barriers

3. Inherent accident prevention. LOP-3 demonstrates that the inherent reactor system response will limit core damage even if the active systems in LOP-1 and -2 fail. LOP-3 provides this function with inherent features, free from human intervention, providing an additional level of protection against common cause failure mechanisms. LOP-3 includes the following features: natural convection core cooling, inherent reactor shutdown mechanisms, and inherent local fault accommodation.

4. In-vessel accident containment. LOP-4 demonstrates that the PCRV structure and associated systems inherently protect the containment against consequential failure in the event of whole core disruption resulting from the failure of LOPs -1 through -3. LOP-4 deals with two threats to containment integrity: energetics and core debris. Successful core debris containment requires the PCRV liner cooling system to function.

5. Containment integrity. LOP-5 demonstrates that the containment building structure and associated systems can delay, control, and reduce the release of radioactivity to the environment even in the event of LOP-4 failure. LOP-5 deals with missiles, containment pressure buildup, flammable gas, and heat load.

6. Radiological attenuation. LOP-6 demonstrates that naturally occurring attenuation mechanisms limit radioactivity transported in the environment to significantly affect public health even if the preceding LOPs fail. LOP-6 deals with aerosol depletion, weather and siting, and emergency procedures.

The LOPs defined above separate the core disruptive accident sequence into its major components. Each LOP independently reduces the probability and consequence (risk) of a given accident initiator. The failure of each successive LOP defines the initial conditions to examine the response of each succeeding LOP.

LOPs-1 and -2 deal with design basis safety features, while LOPs -3 through -6 address the capability of the GCFR to accommodate and mitigate events traditionally considered beyond the design basis. The LOP approach therefore extends the traditional defense-in-depth concept to consider accommodating accidents much more severe than design basis. Additionally, LOPs -1 through -3 render an extremely low probability to any accident which could potentially release significant radioactivity to the environment, and LOPs -4 through -6 mitigate the consequences of these low probability accidents in the unlikely event that they should occur.

1.5. OVERALL SAFETY GOALS

The quantitative risk approach to identifying technology requirements for each LOP entails two problems:

1. Identify the overall risk acceptance criteria for the plant.

2. Allocate goals to each LOP consistent with the overall acceptance criteria.

Problem No. 1 is beyond the scope of this safety plan and must be determined at a national or even international policy level. However, this plan addresses Problem No. 2.

The problem of allocating goals to each LOP does not have a unique solution. Innumerable combinations of weightings might be assigned to each LOP which would be consistent with the overall acceptance criteria. The optimal allocation of LOP goals minimizes plant operating, design, or research costs. If goals are quantified before complete information is available, design or research cost objectives that are not optimal might be selected. However, since the alternative of having an unfocused program is considered much less desirable, early identification and numerical quantification of program goals are considered necessary.

In general, generic risk acceptance criteria have not been established for U.S. nuclear power plants. However, the NRC has provided some guidance by setting risk goals for the LMFBR in Ref. 1-10:

1. The design should assure the capability to minimize the risks associated with core meltdown events to an extent comparable to LWR designs.
2. There will be no greater than one chance in one million per year (i.e., 10^{-6}/reactor yr) for potential consequences greater than 10CFR100 (Ref. 1-2) guidelines for an individual plant.

Until risk acceptance criteria are established for nuclear power plants, the above guidance will be assumed to present an acceptable risk objective for design development and prelicensing NRC reviews for the GCFR, leading to a GCFR demonstration plant program.

Objective No. 2 above provides a single point on a probability versus consequence plot to establish a risk envelope. Objective No. 1 above provides additional points on such an envelope at probabilities below 10^{-6}/yr if the associated LWR risks can be quantified, as was done most extensively by the Reactor Safety Study group (Ref. 1-4). Additional points at probabilities above 10^{-6}/yr can be established by maintaining the objective of not exceeding the Code of Federal Regulations (CFR) LWR consequence (Refs. 1-1 through 1-3).

A resultant risk envelope can be established on a probability versus consequence plot, as shown in Fig. 1-3. The requisite degree of safety is achieved if risks are shown to lie to the left of the curve in the shaded region. Probability is measured in terms of frequency per reactor year, and

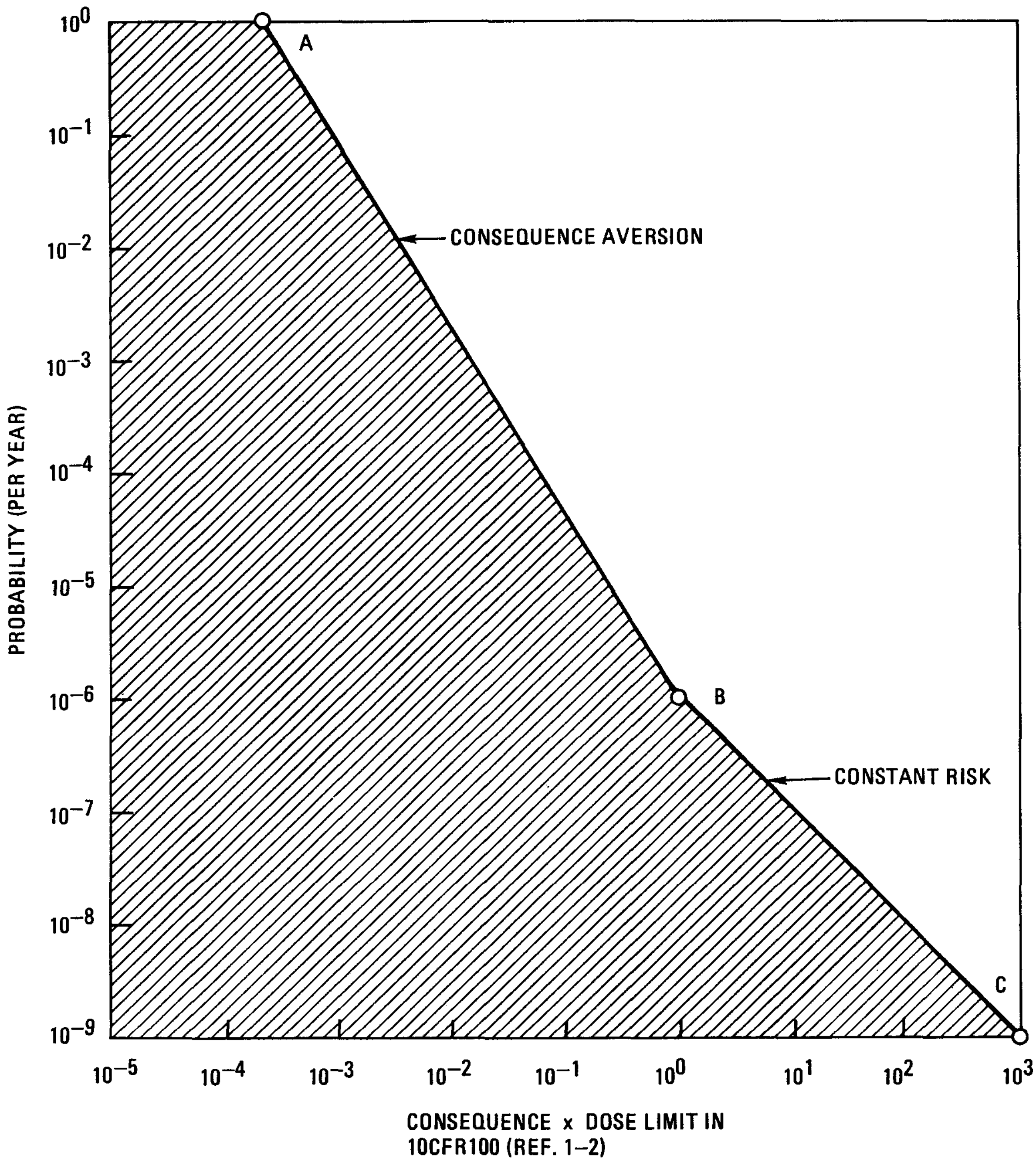

Fig. 1-3. Risk goals

consequence in terms of multiples of the dose allowed in 10CFR100 (Ref. 1-2). The envelope is defined by the points described below:

Point A. 10CFR50, Appendix I (Ref. 1-1) limits to be met during normal plant operation. Under unusual operating conditions, public exposure is still limited to a small fraction of doses from natural background.

Point B. 10CFR100 (Ref. 1-2) limits to not be exceeded at a frequency greater than 10^{-6}/reactor yr.

Point C. Reactor Safety Study (Ref. 1-4) risk associated with core meltdown not to be exceeded.

The resultant curve between points B and C has a slope of constant risk (probability decreasing in proportion to consequence increase). Between points A and B, the curve has a slope of consequence aversion (probability decreasing faster than consequence increase). In general, the envelope defined in Fig. 1-3 assures that those situations occurring frequently shall yield little or no consequence and those extreme situations having potential to the greatest public harm shall have an extremely low probability of occurrence.

The curve in Fig. 1-3 is not unique. Other possible interpretations of regulatory requirements result in slightly different risk envelopes. The curve in Fig. 1-3, however, forms a consistent basis for detailing numerical goals at lower levels of the program plan.

Goals may now be allocated to the six LOPs. Figure 1-3 shows that a total probability decrement of 10^{-9} and consequence of 10^7 (from 10^{-4} for Point A to 10^3 for Point C) must be divided. This allocation should assign realistic and demonstrable probability goals to each LOP. The optimal goals should minimize design or research costs. Unfortunately, this conceptual stage of the GCFR does not have available information for making tradeoffs to optimize these costs. Lacking such information, goals may be allocated on a basis equivalent to commercial LWRs. Since this approach can also best apply relevant LWR operating experience, commercial LWR failure data are considered below.

Tables 1-1 and 1-2 provide data on the range of failure rates which appear to be achieved by current LWR systems. These data consider common cause failures. Table 1-1 provides experience data on common cause failures in terms of a beta factor. Beta essentially represents the fraction of failures of a given component which commonly occur in the other redundant components in the system. Table 1-1 shows that 6% to 53% of the component failures in the sample are common cause, although most common cause failures occur in the range of 10% to 20%. Considering that the failure probability for such active components lies generally in the range of 10^{-2} to 10^{-4}/yr their common cause failure probability would be in the range of 10^{-3} to 10^{-5}/yr. Considering that a typical system consists of many such components, the achieved system failure probability rate is in the range of 10^{-2} to 10^{-4}/yr. Notably, the last entry in Table 1-1 shows that, even for a mature system like commercial jet aircraft, the common cause failure fraction is not notably improved.

Table 1-2 provides the system unavailabilities calculated for the LWR systems in the Reactor Safety Study (Ref. 1-4), again considering common cause failures. Fifty percent of the LWR systems had calculated unavailabilities of $\geq 10^{-3}$/yr; 75% had unavailabilities of $\geq 10^{-4}$/yr.

The above indicates that the achieved LWR systems failure probability is typically in the range of 10^{-2} to 10^{-4}/yr. Considering that several systems must respond for each LOP, an LOP failure probabilities goal in the range of 10^{-1} to 10^{-3}/yr appears realistic, based upon current industry experience. Maintaining the LOP target failure probabilities within this range helps ensure that work packages can be defined with technically achievable probability goals, in spite of common cause failures.

TABLE 1-1
PERCENTAGE OF COMMON CAUSE (β) FAILURES

Component Type	Failure Mode	No. of Common-Cause Failures	Total No. of Component Failures	β Factor
Diesel generation	Fail to start Fail to run	7 4	50 30	0.14 0.13
Trip system sensor channel	Fail to trip	14	153	0.09
Valve	Fail to open (close)	30	132	0.23
Pump	Fail to start Fail to run	2 0	14 12	0.14 0.06[a]
Pressure, level, flow switch	Fail to trip	41	77	0.53
Aircraft jet engine	Fail to run	136	1702	0.08

[a]Obtain from the binominal distribution at 50% confidence.

TABLE 1-2
REACTOR SAFETY STUDY CALCULATED SYSTEM UNAVAILABILITIES[a]

Unavailability Range	Number of Systems			Cumulative (%)
	PWR[b]	BWR[c]	Total	
10^{-6} to 10^{-5}	0	1	1	100
10^{-5} to 10^{-4}	5	4	9	98
10^{-4} to 10^{-3}	4	7	11	75
10^{-3} to 10^{-2}	10	3	13	50
10^{-2} to 10^{-1}	3	3	6	15
TOTAL	22	18	40	

[a] Ref. 1-4.
[b] Pressurized water reactor.
[c] Boiling water reactor.

Considering the failure probability goal above and the goal of maintaining some equivalence with LWR systems, Fig. 1-4 partitions the risk envelope into individual probability and consequence targets for each LOP. The partitioning in Fig. 1-4 notably places a maximum reliance of 10^{-3}/yr in probability and 10^{-2}/yr in consequence for each LOP. The combined goal of LOPs-1 and -2 (the systems traditionally provided to meet the design basis) is 10^{-4}/yr. This target is consistent with the mean core melt frequency calculated in the LWR Reactor Safety Study (Ref. 1-4). The barriers provided in addition to LOPs -1 and -2 accommodate accidents traditionally beyond the design basis.

Further, the consequence aversion portion of the risk envelope is to be achieved by the LOPs with the highest achievable reliability (namely, LOPs -1, through -3, which include systems and features which prevent loss of coolable core geometry). Less stringent probability targets are assigned to LOPs -4 through -6 where the extreme complexity of core disassembly phoneomena must be quantified.

Table 1-3 describes and expands upon the resulting success criteria for each LOP. Table 1-3 interprets the public consequence criteria for each LOP into success criteria for plant inherent and design features. At the higher frequency of events dealt with by LOPs -1 and -2, economic criteria are expected to be more limiting than the public consequence criteria; hence, the plant success criterion is concerned with limiting damage to plant equipment. Therefore, in LOPs -1 and -2, the safety program will emphasize reliability goals. In LOPs -3 through -6, the public consequences criteria become limiting; therefore, the safety program must emphasize both reliability and consequence goals. Notably, if any of the first five barriers succeeds, significant harm to the public health and safety is prevented.

The success criteria defined here for each LOP should not be considered unchangeable. The safety program will continue to optimize the allocation of risk criteria to the six LOPs.

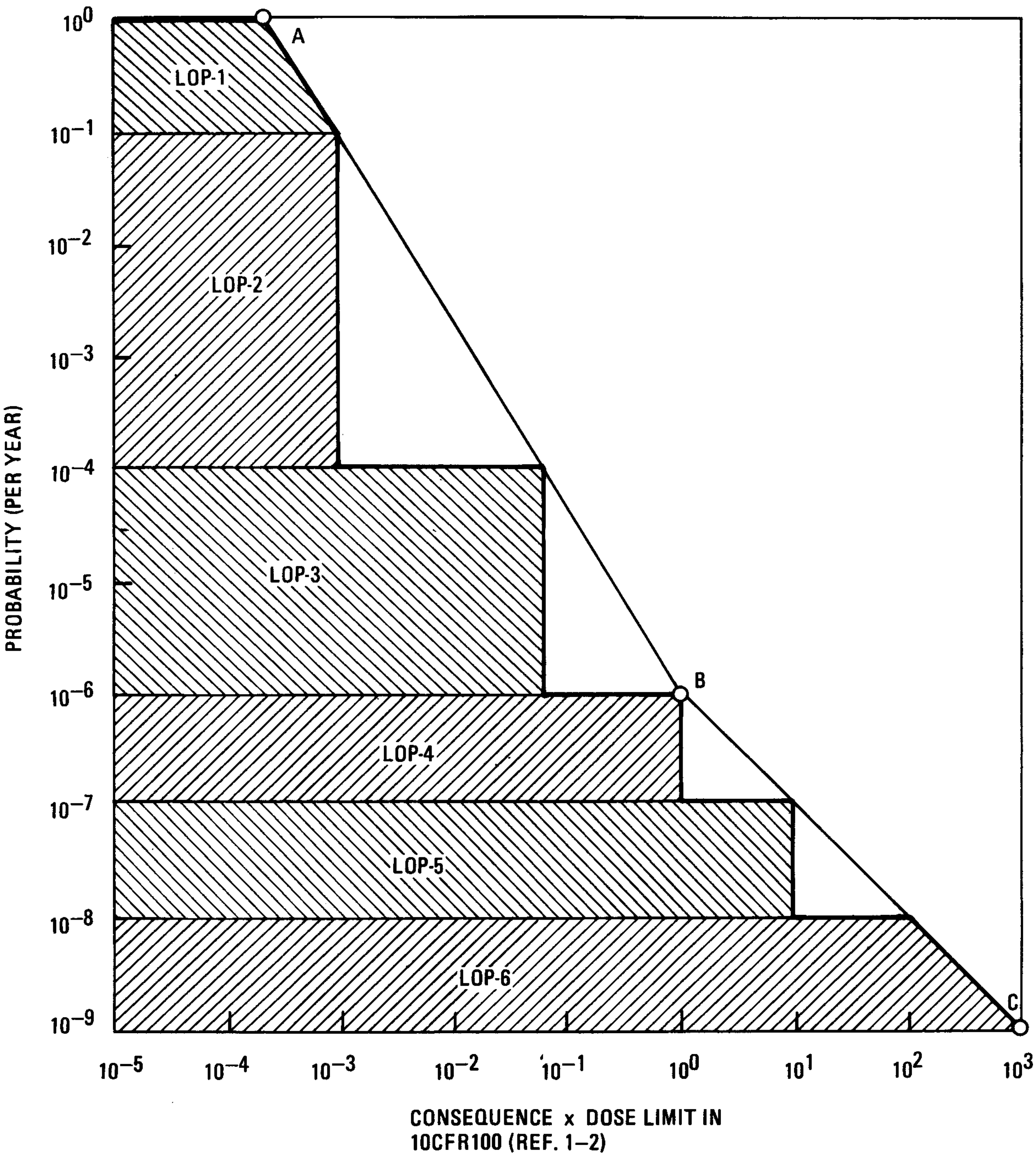

Fig. 1-4. LOP goal allocation

TABLE 1-3
LOP DEFINITIONS AND SUCCESS CRITERIA

LOP Barrier	Function	Probability (Times/Yr)	Plant Consequence	Public Consequence
1. Operating systems	Shutdown/cooldown core following anticipated operational occurrences.	$<10^{-1}$	Reoperable without extensive repair.	Plant contributes less than 1% to background exposure (Ref. 1-1).
2. Dedicated safety systems	Shutdown/cooldown core in the event the operating systems in LOP-1 fail.	$<10^{-4}$	No lifetime reduction to permanent components.	Exposure does not exceed a small fraction of natural background.
3. Inherent features	Shutdown/cooldown core in the event the active systems in LOP-2 fail.	$<10^{-6}$	No loss of core cooling geometry.	Annual radiation worker exposure limit (Ref. 1-3) not exceeded in any member of public.
4. PCRV vessel	Contain debris/energy release following core meltdown from failure of LOPs -1 through -3.	$<10^{-7}$	No loss of liner or penetration integrity of vessel which could consequentially cause loss of containment integrity.	No acute health effects (Ref. 1-2); no significant latent effects.
5. Containment	Delay/control the release of activity from LOP-4 failure.	$<10^{-8}$	No unacceptable loss of containment leaktight integrity.	No acute fatalities.
6. Natural attenuation	Attenuate radiological consequences resulting from LOP-5 failure.	$<10^{-9}$	No criteria for plant, possible site criteria.	Maximum LWR consequences not exceeded.

1.6. PLAN STRUCTURE

To achieve the LOP goals defined in the previous sections, the GCFR Safety Program Plan has a work breakdown structure, that is, a hierarchial tree of products to accomplish program objectives. Figure 1-5 shows the top level of this structure.

The three top level products are as follows:

1. Safety program integration. This task area manages the program and project support functions, conducts studies to define and guide the overall GCFR Safety Program, establishes necessary administrative safeguards to meet technical goals, performs the integrated analyses as required to determine whether the overall program goals are being met, and provides the integrated program test requirements and test plans.

2. Core accident accommodation. This task area develops the technology base to support the six independent LOPs to protect public health and safety from GCFR accidents.

3. Noncore activity release accommodation. This task area establishes design criteria for the nonreactor aspects of the GCFR plant to ensure that they do not excessively risk the public health and safety. This task area provides containment for ex-reactor fuel pressure systems, radwaste, and circulating activity.

The following sections of this document define the products below the three top level products.

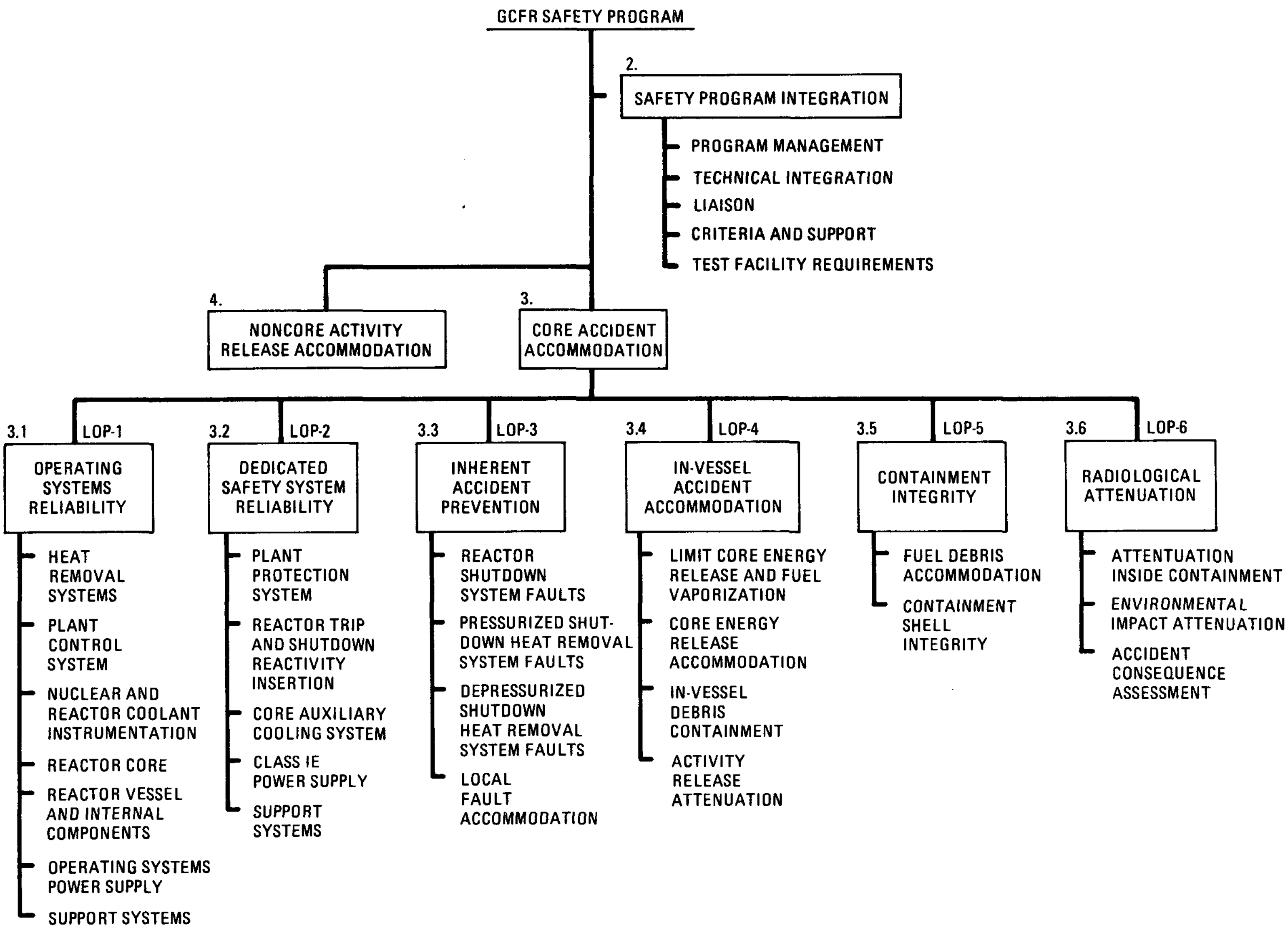

Fig. 1-5. GCFR safety program structure

1.7. UNRESOLVED ISSUES

This plan represents the first attempt to order the GCFR program defense-in-depth safety philosophy into a number of distinct, separable, and quantifiable barriers to prevent abnormal occurrences from progressing into severe accidents. While substantial progress is believed to have been made, further work must optimize the goals and the composition of the three LOPs dedicated to preventing accidents (LOPs -1 through -3). These three LOPs, while logically defined, evidently cannot be made fully independent of each other. The following examples indicate this interdependence:

1. The passive elements of the core auxiliary cooling system (CACS), such as the heat exchangers, pipes, helium valves, PCRV, dampers, cooling towers, and pressure relief valves, serve both LOP-2 in the forced circulation mode and LOP-3 in the natural circulation mode.

2. The shutdown cooling system (SCS) is dedicated to LOP-1, because it shares some major components with the main loop cooling system (MLCS). On the other hand, it is a safety class system, its operation is initiated and terminated by the plant protection system (PPS) (an LOP-2 system), and it may be served by its dedicated power supply or even by the LOP-2 power supply.

3. The plant control system (PCS) is an LOP-1 function, while the reactor trip function is a LOP-2 function. Both insert the control rods for negative reactivity.

Further revisions are most likely in the the following two areas:

1. As described previously, the risk envelope adopted for this program has evolved over time and is based on interpreting existing NRC requirements. These requirements include a letter to the CRBR project from the NRC (Ref. 1-10). This letter does not

have the same status as a Regulatory Guide. This GCFR risk envelope was adopted only to demonstrate to the NRC that a GCFR can be designed to meet all the NRC requirements and proposed goals. The risk envelope is only a means to bring these requirements to a common denominator. Adopting this risk envelope should in no way be interpreted as indicating that the GCFR program believes this to be an appropriate risk envelope for nuclear power plants. In fact, very likely the national effort in progress to establish a quantitative safety goal for nuclear power plants will be less stringent than the risk envelope adopted here. Particularly Point B in Fig. 1-3 is expected to be relaxed significantly.

2. Preliminary assessments of the ability of the GCFR conceptual design to meet the LOP-1 and -2 reliability goals indicate the following potential difficulties:

   a. The LOP-1 reactor shutdown goal may be difficult to meet. The current LOP-1 definition includes the PCS for normal plant shutdown but not PPS-initiated reactor trips. The frequency of PPS-initiated reactor trips probably cannot be reduced to less than 0.1/yr.

   b. The LOP-2 goal may be difficult to meet with respect to residual heat removal by the CACS in the forced circulation mode only, particularly since the CACS in the natural circulation mode also has to span the LOP-3 goal.

One potential approach to resolve these difficulties may be to increase the LOP-1 goal to 10^{-2}/yr and to include the reactor trip function in the LOP-1 definition. This would increase the demand on the LOP-1 cooling systems while easing the demand on the CACS. Shutdown and cooling systems would also achieve greater consistency, since the safety-related functions that share components with the normal operating systems are consistently treated as LOP-1 functions. For example, the SCS shares the main steam generator and circulators with the MLCS, and the reactor trip system shares the control rods and drive lines with the reactivity control system. The backup shutdown system would then take on a role parallel to the CACS (namely, a PPS-actuated, active operating mode in LOP-2 and a self-actuated, inherent operating mode in LOP-3).

This report has not attempted to resolve these issues, because in its current form, this plan is consistent with the GCFR Plant Specification for Reliability, (Ref. 1-12), and both documents would have to be revised to maintain consistency.

REFERENCES

1-1. "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water Cooled Nuclear Power Reactor Effluents," in _Code of Federal Regulations_, Title 10, Part 50, Appendix I, U.S. Government Printing Office, Washington, D.C., issued May 5, 1975, subsequently amended.

1-2. "Reactor Site Criteria," in _Code of Federal Regulations_, Title 10, Part 100, U.S. Government Printing Office, Washington, D.C., issued April 12, 1962, subsequently amended.

1-3. "Standards for Protection Against Radiation," in _Code of Federal Regulations_, Title 10, Part 20, U.S. Government Printing Office, Washington, D.C., issued November 17, 1960, subsequently amended.

1-4. "Reactor Safety Study," U.S. Nuclear Regulatory Commission Report WASH-1400, October 1975 (NUREG-75-014).

1-5. "CRBRP Risk Assessment Report," Clinch River Breeder Reactor Plant Report CRBRP-1, March 1977.

1-6. "HTGR Accident Initiation and Progression Analysis Status Report, Phase II Risk Assessment," DOE Report GA-A15000, General Atomic Company, April 1978.

1-7. "Gas-Cooled Fast Breeder Reactor - Preliminary Safety Information Document," Gulf Energy and Environmental Systems Report GA-A10298, February 1971.

1-8. "Preapplication Safety Evaluation of the Gas Cooled Fast Breeder Reactor," USAEC-DOL Report, August 1974.

1-9. Stratton, W. R., "Conceptual Design for Prototype Gas-Cooled Fast Breeder Reactor (GCFBR)," letter from Chairman of Advisory Committee for Reactor Safeguards to Honorable Dixy Lee Ray, Chairman of U.S. Atomic Energy Commission, November 8, 1974.

1-10. Denice, R. P., letter from Assistant Director for Special Projects, Nuclear Regulatory Commission to L. W. Chaffey, Clinch River Breeder Reactor Project Office, May 6, 1976.

1-11. Fleming, K. N., and P. H. Raabe, "A Comparison of Three Methods for the Quantitative Analysis of Common Cause Failures," General Atomic Report GA-A14568, May 1978.

1-12. "GCFR Plant Specification for Reliability," General Atomic unpublished data, September 9, 1980.

2. SAFETY PROGRAM INTEGRATION

A successful safety R&D program requires harmony among the program tasks and with other programs. This task area provides internal and external integration tasks to guide the overall safety program. As such, this task assures that program goals and work packages are properly balanced to do the following:

1. Ensure that all relevant GCFR safety characteristics are properly understood.

2. Ensure that the necessary support is provided from and to the safety program to confirm this understanding.

3. Factor this understanding of safety characteristics into the design, licensing, construction, and operation of GCFR plants.

Figure 2-1 shows the task breakdown.

2.1. PROGRAM MANAGEMENT

2.1.1. Introduction

To ensure a cost effective program, the most important R&D activities must be identified, assigned criteria for successful resolution, prioritized, costed, and scheduled with assigned organizational responsibilities. The U.S. Department of Energy (DOE) and Helium Breeder Associates (HBA), which both provide funding for GCFR safety research, are ultimately responsible for this function. This task provides the tools by which the DOE and HBA can be assured that their programs will be accomplished in a technically competent manner.

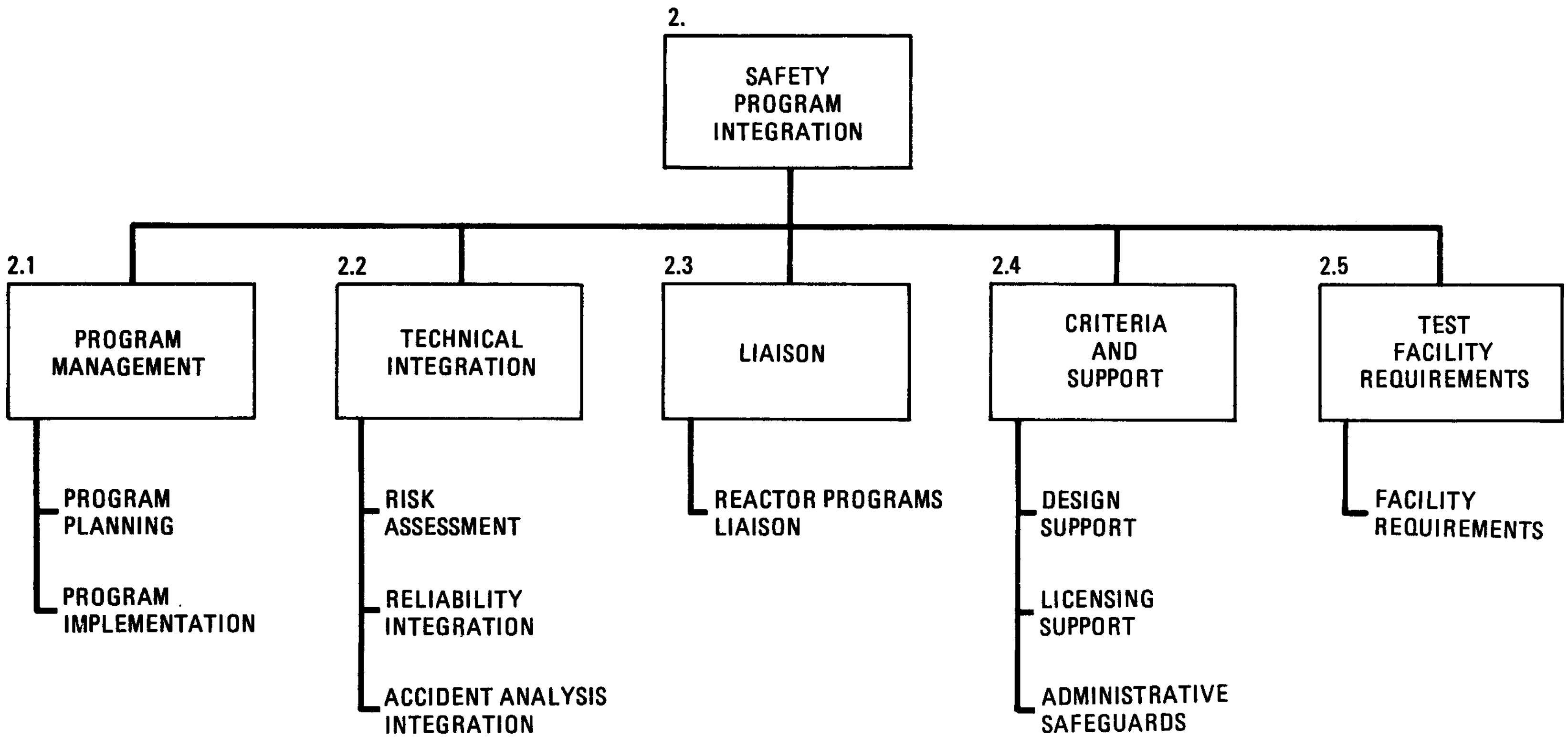

Fig. 2-1. Safety program integration work breakdown structure

2.1.2. Objective

This task defines technical objectives and tasks for the safety program down to level 4 of the work breakdown structure. It does not develop administrative methods related to program controls (i.e., cost and performance measuring systems). Figure 2-1 shows the task breakdown.

2.1.3. Work Packages, Program Planning

This task is intended to establish safety program task responsibilities, schedules, funding requirements, and priorities. The following work package meets this objective: prepare and maintain the GCFR Safety Program Implementation Plan.

2.2. TECHNICAL INTEGRATION

2.2.1. Introduction

This task integrates technical methods and associated data bases which cross the LOPs and which must be established to support task management decision making. This task includes the overall risk assessment, reliability methods and data integration, and accident analysis integration.

2.2.2. Objective

This task is intended to integrate technical methods, data, and analysis to assess GCFR accident risk and program priorities.

2.2.3. Scope

This task develops integrated methods and data which cross the boundary of the six LOPs and which must establish relative GCFR accident risks. Methods and data specific to one LOP alone will be covered under that LOP in the work breakdown structure. Figure 2-1 shows the task breakdown.

2.2.4. Work Packages

The following tasks and associated work packages will achieve the overall task area objective.

2.2.4.1. Risk Assessment. This task is intended to quantitatively assess GCFR accident risk by integrating the technical accomplishments of each LOP. This task should provide a balanced safety perspective and ensure the relative independence of each LOP. The tools and data developed under Sections 2.2.4.2 and 2.2.4.3 will assess GCFR accident risk (where risk is measured by accident frequency and consequence). This task output will be directly compared with the safety goals identified in Section 1.5 to identify where the greatest safety improvements can be made or where safety research should be most optimally directed.

The following work packages will meet this objective:

1. Conduct qualitative and quantitative risk trade-off studies of safety program alternatives to select R&D priorities.

2. Perform quantitative risk assessments of the GCFR plant designs.

2.2.4.2. Reliability Integration. This task is intended to integrate methods and the data base to evaluate the reliability performance of each LOP and to conduct the integrated reliability analysis required in Section 2.2.4.1. This task will acquire methods to integrate probabilistic techniques into the GCFR engineering effort. The GCFR Plant Specification for Reliability (Ref. 2-1) is the controlling document for this activity.

The following work packages will meet this objective:

1. Establish reliability goals for the major GCFR systems.

2. Establish analytical methods for evaluating the reliability of GCFR systems and components.

3. Provide a GCFR reliability data base.

2.2.4.3. Accident Analysis Integration. This task is intended integrate methods and the data base to evaluate the consequences of GCFR accident sequences. This task will acquire integrated methods and data for analyzing GCFR system dynamic performance, determining core behavior during postulated accident scenarios, establishing PCRV and containment behavior, and calculating on- and off-site radiological transport and consequences.

The following work packages will meet this objective:

1. Establish analytical methods for evaluating the dynamic performance of GCFR systems following postulated accidents.

2. Establish analytical methods for evaluating reactor core behavior under postulated accidents.

3. Establish analytical methods for calculating radionuclide tansport within the PCRV and containment/confinement and off site.

4. Establish analytical methods for determining on- and off-site doses and environmental impacts resulting from accident radionuclide transport.

5. Provide a GCFR accident analysis data base to support analysis with the above tools.

2.3. LIAISON

2.3.1. Introduction

The GCFR safety R&D program must be closely coordinated with other related design and research programs, particularly safety programs for other reactors. This liaison would ensure maximum benefit to the incremental GCFR program.

2.3.2. Objective

This task is intended to interface with groups establishing the direction of LMFBR, LWR, HTGR, and related foreign safety research programs to ensure maximum benefit to the GCFR program.

2.3.3. Scope

To the extent possible, this task will ensure consistency between GCFR and other program objectives. It will attempt to ensure GCFR representation and participation on safety-related policy and working committees and will follow activities of DOE-formed safety technical management centers. Figure 2-1 shows the task breakdown.

2.3.4. Work Packages

The following work packages will meet this objective:

1. Establish and maintain liaison and technical exchange activities with other reactor concept safety research programs.

2. Support the development of generic code standards and research programs for nuclear safety that apply to GCFRs.

2.4. CRITERIA AND SUPPORT

2.4.1. Introduction

The GCFR safety R&D program must directly support other GCFR program activities, particularly the design, licensing, and operation of the GCFR demonstration plant.

2.4.2. Objective

This task is intended to provide direct safety-related support to the design, licensing, and operation of the GCFR demonstration plant.

2.4.3. Scope

Except for the specific deliverables identified below, this task provides support on an as-required basis. Figure 2-1 shows the task breakdown.

2.4.4. Work Packages

A program comprising the following tasks and associated work packages will meet this overall objective.

2.4.4.1. Design Support. This task is intended to provide the overall safety-related performance criteria for the GCFR design and to conduct design reviews to ensure compliance as necessary. This task will prepare top-level criteria documents, such as the GCFR Plant Specification for Nuclear Safety (Ref. 2-2), the GCFR Plant Specification for Reliability (Ref. 2-1), and the plant transient specification. The following work packages will meet this objective:

1. Prepare and maintain the top-level safety-related design criteria documents.

2. Conduct design criteria and design reviews as necessary to ensure compliance with the above specifications.

2.4.4.2. Licensing Support. This task is intended to support the licensing or safety approval process of GCFR projects. It will provide appropriate analysis documentation for GCFR licensing topical reports (LTRs), preliminary safety analysis reports (PSARs), final safety analysis reports (FSARs), and environmental reports. The following work package will meet this objective: provide consultation, documentation, and general assistance to parties in the process of licensing GCFR projects.

2.4.4.3. Administrative Safeguards. This task is intended to help establish quality assurance, maintenance, operating, and related procedures for GCFR projects. This task will ensure that such administrative procedures enforce and support the degree of reliability required by the safety-related design criteria. The following work package will meet this objective: provide consultation, documentation, and general assistance to parties in the process of establishing administrative safeguards for GCFR projects.

2.5. TEST FACILITY REQUIREMENTS

2.5.1. Introduction

Test facilities must provide experimental data to verify GCFR safety methods. Both dedicated safety facilities and nondedicated facilities gather data. This task provides safety-related test requirements for these supporting experiment programs.

2.5.2. Objective

This task is intended to prepare the functional and test requirements for safety-related GCFR experiment projects. This task includes developing safety test requirements for projects such as the Core Flow Test Loop (CFTL), Gas-Reactor In-Pile Safety Test (GRIST-2), and Low Power Safety Experiment (LPSE) and for the preoperational tests of GCFR plants.

2.5.3. Scope

This task does not cover design, construction, and operating charges for facilities. This task is limited to supplying the interfacing functional and test requirements. Figure 2-1 shows the task breakdown.

2.5.4. Work Packages

The following work package will meet the task objective: establish test requirements for safety-related GCFR experiment projects.

REFERENCES

2-1. "GCFR Plant Specification for Reliability," General Atomic unpublished data, September 9, 1980.

2-2. "GCFR Plant Specification for Nuclear Safety," General Atomic unpublished data, May 30, 1980.

3. CORE ACCIDENT ACCOMMODATION

The six LOPs defend against core accidents. LOPs -1 through -3 are dedicated to accident prevention, while LOPs -4 through -6 are dedicated to accident mitigation. Accidents are successfully prevented as long as core cooling geometry is maintained. To maintain core cooling geometry, gross cladding integrity must be maintained. Therefore, successful accident prevention is characterized by small releases of core activity, probably dominated by local fuel pin failures. Radiological consequences to demonstrate that the consequence limits of LOPs -1 through -3 are met, therefore, are not expected to require a safety R&D effort. The accident prevention portion of the GCFR Safety Program thus reduces to a reliability integration and assurance program.

The GCFR Plant Specification for Reliability (Ref. 3-1) constitutes the implementation plan for the accident prevention program. It defines the reliability target allocation to systems, subsystems, and components; it defines the responsibility of the reliability and design organizations for meeting accident prevention objectives; it defines the data base and analysis methods; and it defines the acceptance criteria by which a design is judged to meet its assigned reliability goal. Since Ref. 3-1 is the working document for accident prevention, the GCFR Safety Program Plan (particularly for LOPs -1 and -2, the traditional active systems) is only intended to define the Ref. 3-1 objectives and general activities.

Additional safety-related R&D is anticipated before LOP-3, inherent design safety features, can be optimally used. Therefore, Section 3.3 on LOP-3 more comprehensively discusses inherent design feature options and the R&D required to solve the feasibility and/or desirability of these options.

The accident mitigation portion of the plan (LOPs -4 through -6) emphasizes consequence-related R&D for two reasons:

1. Substantial uncertainties remain in the consequences of core melt accidents both with respect to the physical phenomena and the magnitude of physical effects.

2. A firm quantitative basis for the probabilistic achievements of LOPs -4 through -6 is unlikely to be established, regardless of the magnitude of effort devoted to it.

Thus, the probabilistic aspects of LOPs -4 through -6 are deemphasized except for a few selected and specific reliability assessments, such as the reliability of the liner cooling system for molten fuel containment. The confidence that the probabilistic targets of LOPs -4 through -6 are indeed met is derived from studying a sufficiently wide range of core melt accident initiators and sequences and by performing consequence sensitivity assessments of major assumptions and parameters. On this basis, the unquantifiable residual probability that higher consequence sequences have been omitted can be rationally judged to be less than 10%.

A future task will accomplish a detailed implementation plan for the accident mitigation program.

3.1. LOP-1: OPERATING SYSTEMS RELIABILITY

LOP-1 uses the designed operability, reliability, and safety enhancement features of normal operating systems. Since these systems constitute the first barrier to be challenged during normal operation, they should be designed such that the plant can achieve or exceed the allocated reliability goal while maintaining economic power generation.

3.1.1. Introduction

The operating systems are an important contributor to overall GCFR safety. These systems provide a barrier against a too-frequent demand for the dedicated safety systems. LOP-1 protects against the higher frequency events, designated as anticipated operational occurrences. The public consequence limits are not expected to limit the design of LOP-1 systems. Therefore, the LOP-1 safety program will emphasize attaining the LOP-1 reliability goal.

Figure 3-1 shows the work breakdown structure for LOP-1 operating systems reliability. The work breakdown structure is organized to highlight the components and systems which comprise the operating systems: the primary heat removal systems, the PCS, instrumentation systems, the reactor core, the PCRV, and internal components and support systems.

3.1.2. Objectives

The primary safety objective of the operating systems is to shut down and cool down the reactor core following anticipated operational occurrences listed in the GCFR Plant Specification for Nuclear Safety (Ref. 3-2). The success in providing this function is measured against the following criteria:

1. The failure of the LOP-1 barrier will be less frequent than one time in 10 reactor years of operation.

2. All event sequences that are successfully terminated by LOP-1 features will require outage and repair times within the outage times identified in the GCFR Plant Specification for Availability (to be developed) (Ref. 3-3).

3. The plant will contribute less than 1% of the radiation exposure encountered by the public due to natural background radiation.

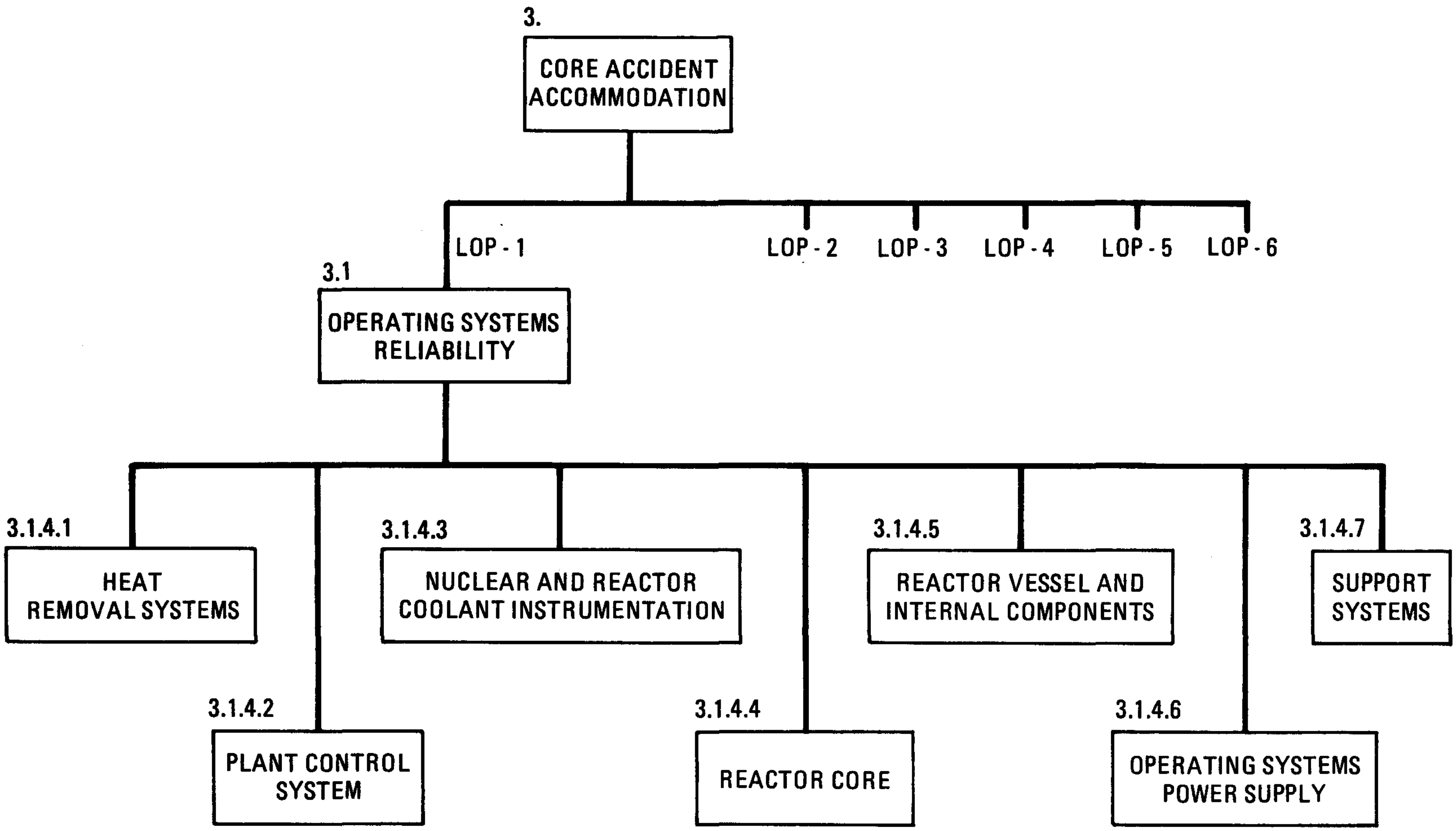

Fig. 3-1. LOP-1 task breakdown

3.1.3. Scope

The operating systems satisfy a dual role: (1) economic electrical power generation and (2) reactor shutdown/cooldown following higher frequency events. The design function designs, develops, and tests the operating systems and their components. Development plans cited in the following subsections outline these activities. However, those operating systems functions related to system reliability are included in the GCFR Safety Program Plan, because they directly affect achieving overall reliability goals which are, in turn, closely related to plant safety.

To ensure that this LOP satisfies the stated success criteria, the following activities are required:

1. Develop reliability targets and criteria.
2. Assess system reliability.
3. Assess the design against the reliability objectives at the completion of major design phases.

The operating systems share some hardware with the dedicated safety systems discussed in Section 3.2. However, the functions of LOP-1 operating systems are clearly delineated:

1. Provide normal electrical power generation with safe, reliable operating systems. These systems will accommodate expected internal and external disturbances that can potentially impact continued plant operation and will minimize plant shutdowns.

2. Provide the initial means of reactor core shutdown and cooldown.

3.1.4. LOP-1 Task Breakdown

Figure 3-1 shows the LOP-1 task breakdown. Seven task areas have been identified. Each task area corresponds to either a single hardware system or a grouping of hardware systems. Since individual development plans have been or are being prepared by the design activity, the GCFR Safety Program Plan focuses on those activities which are primarily of safety significance. The generic work packages for each task area include the following:

1. Criteria.
2. Reliability assessment.
3. Dynamic performance (if appropriate).
4. Design assessment.

3.1.4.1. <u>LOP-1 Heat Removal Systems.</u> The MLCS and the SCS are heat removal systems dedicated to support the LOP-1 function. The MLCS performs two principal functions:

1. In the normal power generating mode, the MLCS with the power conversion system is dedicated to reliable power production. In this function, the primary safety-related objective is to minimize the frequency of faults which require reactor shutdown, minimizing the frequency of demand for residual heat removal.

2. In the event of a fault occurrence which requires reactor shutdown, the MLCS with the turbine bypass system provides the normal means of residual heat removal.

In the residual heat removal mode, the GCFR MLCS employs three heat transfer fluid systems: (1) the primary coolant system using high pressure helium, (2) the turbine bypass steam and feedwater system using high purity water, and (3) the plant heat rejection system using an evaporative cooling tower with a circulating water loop. Reference 3-4 is the development plan for the primary coolant system prepared by the design function.

The SCS is a safety-class system designed to perform the residual heat removal function for all conditions except the more rapid depressurization accidents. It is the first backup for the MLCS with which it shares the most reliable components [i.e., the steam generator, the main circulator (with a dedicated pony motor), and the main loop isolation valve]. The SCS provides an independent, redundant and diverse means for residual heat rejection from the steam generator to the ultimate heat sink. Each of the three independent secondary water circuits includes a motor driven pump and a multitube steam/water-to-water heat exchanger in a large water drum condenser. The condenser has sufficient heat capacity to operate for ~30 min without water makeup. The principal motivation for including the SCS in the LOP-1 function is two-fold:

1. The LOP-1 goal is difficult to meet with the MLCS alone, because of the limited reliability provided by the condensate, feedwater, and circulating water system.

2. The reliabilities of the SCS and the MLCS are difficult to separate, particularly in assigning a separate reliability target, because of the shared equipment between the two systems.

This task is intended to assure that the LOP-1 heat removal systems can perform residual heat removal for all initiating events within the LOP-1 scope and that the combined reliability of the MLCS and the SCS is adequate to support the LOP-1 reliability goal stated in Section 3.1. The following four work packages will meet this objective:

1. Criteria. Develop a reliability target allocation for the combined reliability of the MLCS and the SCS which will support the reliability goal for LOP-1. Develop success criteria for the LOP-1 heat removal systems needed to meet the reliability target.

2. Reliability assessment. Evaluate the reliability of the LOP-1 heat removal systems. Review proposed design modifications for

their impact on the previously assessed reliability. Prepare recommendations on those design options which meet the reliability target. Develop reliability demonstration requirements for LOP-1 heat removal system components for which the data base is inadequate.

3. Dynamic response. Evaluate the dynamic response of the LOP-1 heat removal systems to key initiators. Identify special requirements for verifying safety-related dynamic response and methods development.

4. Design assessment. Perform a confirmatory design review of the LOP-1 heat removal systems at the completion of conceptual and preliminary designs to assure that the criteria in item 1 are satisfied. Submit a failure modes and effects analysis and reliability assessment to the NRC.

3.1.4.2. Plant Control System (PCS). The control system developed for the GCFR demonstration plant will provide stable manual or automatic control over the 25% to 100% range for normal electrical power generation on base load. The PCS will provide the following functions:

1. Maintain constant main steam temperature and pressure.
2. Regulate reactor power to sustain plant output.
3. Balance outlet steam temperatures from the steam generators.

The control system is based on a reactor-follow-turbine load scheme. Reference 3-5 is a development plan for the PCS developed by the design function.

This task is intended to assure that PCS reliability supports the overall reliability goal presented in Section 3.1. The following four work packages will meet this objective.

1. Criteria. Develop a target allocation for the PCS which will support the overall reliability allocation for the operating systems.

2. Reliability assessment. Evaluate the reliability of the PCS. Review proposed design modifications for their impact on the previously assessed reliability. Prepare recommendations on those design options which meet the reliability target. Develop reliability demonstration requirements for control system components if the reuqired data base is inadequate.

3. Dynamic response. Evaluate the dynamic response of the PCS to key accident initiators. Identify special requirements to verify safety-related dynamic response and methods development.

4. Design assessment. Perform an independent design review at the completion of conceptual and preliminary designs to assure that the criteria in item 1 are satisfied.

3.1.4.3. Nuclear and Reactor Coolant Instrumentation. The instrumentation systems provide the plant operator with the required information to operate the plant in a safe and efficient manner. They provide signals for the PCS and PPS. The systems covered include the following:

1. The nuclear instrumentation measures the reactor neutron flux level, rate of change, and gross spatial distribution from shutdown to above full-design power operation and during initial startup. The instrumentation generates appropriate control, alarm, and trip signals with a high degree of reliability over the life of the plant.

2. The core-element temperature instrumentation monitors individual core elements to help the operator select the size of the flow control orifices to assure proper flow distributions.

3. The reactor coolant instrumentation measures temperature, pressure, core flow rate, gross gamma activity, and delayed neutorn activity at locations within the reactor coolant system.

4. The water ingress monitor system detects moisture in the primary coolant that would result from sources such as leakage of water or steam from a steam generator.

5. The PCRV structural instrumentation system monitors the concrete temperature, PCRV prestress, and the tendon strains.

6. The analytical instrumentation measures and monitors the chemical containments in the primary coolant.

This report does not explicitly describe the other control and protection signal instrumentation that is typical of steam power plant practice.

This task is intended to assure that the reliability of the nuclear and reactor coolant instrumentation supports the overall reliability goal presented in Section 3.1. The following three work packages will meet this objective:

1. Criteria. Develop a target allocation for the nuclear and reactor coolant instrumentation which will support the overall reliability for the operating systems.

2. Reliability assessment. Evaluate the reliability of the nuclear and reactor coolant instrumentation. Review proposed design modifications for their impact on the previously assessed reliability. Prepare recommendations on those design options which meet the reliability target. Develop reliability demonstration requirements for instrumentation if the required data base is inadequate.

3. Design assessment. Perform an independent design review at the completion of conceptual and preliminary designs to assure that the criteria in item 1 are satisfied.

3.1.4.4. Reactor Core. The GCFR reactor core is comprised of fuel assemblies, blanket assemblies, control and shutdown assemblies, and radial shield assemblies. The reactor core is primarily intended to provide fission-produced energy at a rate and in a spatial distribution which allows the heat to be removed safely and reliably. Reference 3-6 is a development plan for the reactor core system developed by the design function.

This task is intended to assure that the reactor core reliability system supports the overall reliability goal presented in Section 3.1. The following three work packages will meet this objective:

1. Criteria. Develop a target allocation for the reactor core system which will support the overall reliability allocation for the operating systems.

2. Reliability assessment. Evaluate the reliability of the reactor core system. Review proposed design modifications for their impact on the previously assessed reliability. Prepare recommendations on those design options which meet the reliability target. Develop reliability demonstration requirements for instrumentation if the required data base is inadequate.

3. Design assessment. Perform an independent design review at the completion of conceptual and preliminry designs to assure that the criteria in item 1 are satisfied.

3.1.4.5. Reactor Vessel and Internal Components. The PCRV encloses the entire primary coolant system, providing a leak-tight containment for the reactor coolant. All cavities and ducts within the PCRV are lined with a steel liner to provide leak tightness. The liner is protected from the high helium temperatures by an insulation system on the coolant side and by cooling water circulating in embedded piping on the concrete side. Closure plugs or caps are provided for PCRV penetrations. A pressure relief system ensures that the PCRV is not overpressurized. The components of the reactor internals systems include the core support structure, radial shield assembly, and the upper and lower plenum shield assemblies. References 3-7 and 3-8 are development plans for the reactor vessel system and reactor internals system.

This task is intended to assure that the reliability of the reactor vessel and internal components systems supports the overall reliability goal presented in Section 3.1. The following three work packages will meet this objective:

1. Criteria. Develop a target allocation for the reactor vessel and internal components which will support the overall reliability allocation for the operating systems.

2. Reliability assessment. Evaluate the reliability of the reactor vessel and internal components. Review proposed design modifications for their impact on the previously assessed reliability. Prepare recommendations on those design options which meet the reliability target. Develop reliability demonstration requirements for instrumentation if the required data base is inadequate.

3. Design assessment. Perform an independent design review at the completion of conceptual and preliminary designs to assure that the criteria in item 1 are satisfied.

3.1.4.6. Operating Systems Power Supply. The operating systems power supply, normally called the preferred power supply, in the GCFR consists of (1) the off-site power supply and (2) the on-site main generator supply. The main generator supply is an important element of the preferred power supply, because the GCFR program requires that the turbine generator have a run-back capability to house load without turbine trip and reactor scram in the event of a loss of off-site power. In addition, the operating systems power supply includes any dedicated ac power supplies for the SCS if required or provided to meet the LOP-1 relability target.

This task is intended to assure that the reliability of these power supplies supports the overall LOP-1 reliability objective defined in Section 3.1. The following three work packages will meet this objective:

1. Criteria. Develop a reliability target allocation for the operating systems power supply which will support the overall LOP-1 reliability goal. Define success criteria necessary for the operating systems power supply which are required to meet the reliability target.

2. Reliability assessment. Evaluate the reliability of the operating system power supply. Identify any improvements required to meet the reliability target and assess proposed modifications for their impact on the operating system power supply reliability. Prepare recommendations for those design options which meet the reliability target. Develop reliability testing requirements for components which have an inadequate reliability data base.

3. Design assessment. Perform an independent design review at the completion of conceptual and preliminary designs to assure that the criteria in item 1 are satisfied.

3.1.4.7. Support Systems. The support systems for the operating systems include (1) the instrument and service air system, (2) the auxiliary steam supply system, and (3) the component cooling water systems. A previous study (Ref. 3-9) has shown the important role of the support system in attaining overall reliability objectives for the operating systems.

This task is intended to assure that the reliability of the operating system support systems supports the overall reliability goal presented in Section 3.1. The following three work packages will meet this objective:

1. Criteria. Develop a target allocation for the support systems which will support the overall reliability allocation for the operating systems.

2. Reliability assessment. Evaluate the reliability of the support systems. Review proposed design modifications for their impact on the previously assessed reliability. Prepare recommendations on those design options which meet the reliability target. Develop reliability demonstration requirements for instrumentation if the required data base is inadequate.

3. Design assessment. Perform an independent design review at the completion of conceptual and preliminary designs to assure that the criteria in item 1 are satisfied.

3.2. LOP-2: DEDICATED SAFETY SYSTEM RELIABILITY

LOP-2 is based on the reliability and safety features of the dedicated safety systems. These systems are required in the event the operating systems in LOP-1 fail. The successful operation of the dedicated safety systems ensures that the public is not at risk and that the plant will not sustain extensive damage.

3.2.1. Introduction

The dedicated safety systems have historically provided a major barrier to the progression of accident sequences. These systems have provided for reactor shutdown and cooldown with a high degree of reliability. Therefore, the systems which constitute LOP-2 are assigned a major role in responding to and terminating accident sequences. Public consequence criteria are not expected to limit the design of LOP-2 systems; therefore, the LOP-2 safety program will emphasize meeting the LOP-2 reliability goal.

Figure 3-2 presents the work breakdown structure for LOP-2, dedicated safety systems reliability. The work breakdown structure is organized to highlight the systems which comprise the dedicated safety systems. These include the reactor scram systems, the PPS, the Class IE electrical power, the CACS, and the support systems.

3.2.2. Objectives

The primary objective of the dedicated safety systems is to shut down and cool down the reactor core in the event that the operating systems in LOP-1 fail. In addition to accommodating failure of the operating systems to successfully terminate event sequences within their design envelope, the dedicated safety systems also are designed to terminate the sequences resulting from the initiators listed in the GCFR Plant Specification for Nuclear Safety (Ref. 3-2). The performance of the dedicated safety systems is measured against the following success criteria:

1. The failure of the LOP-2 barrier will be less frequent than once in 10,000 reactor years of operation.

2. The plant will not expose the public to radiation greater than a small fraction of the natural background radiation. This fraction is currently not defined.

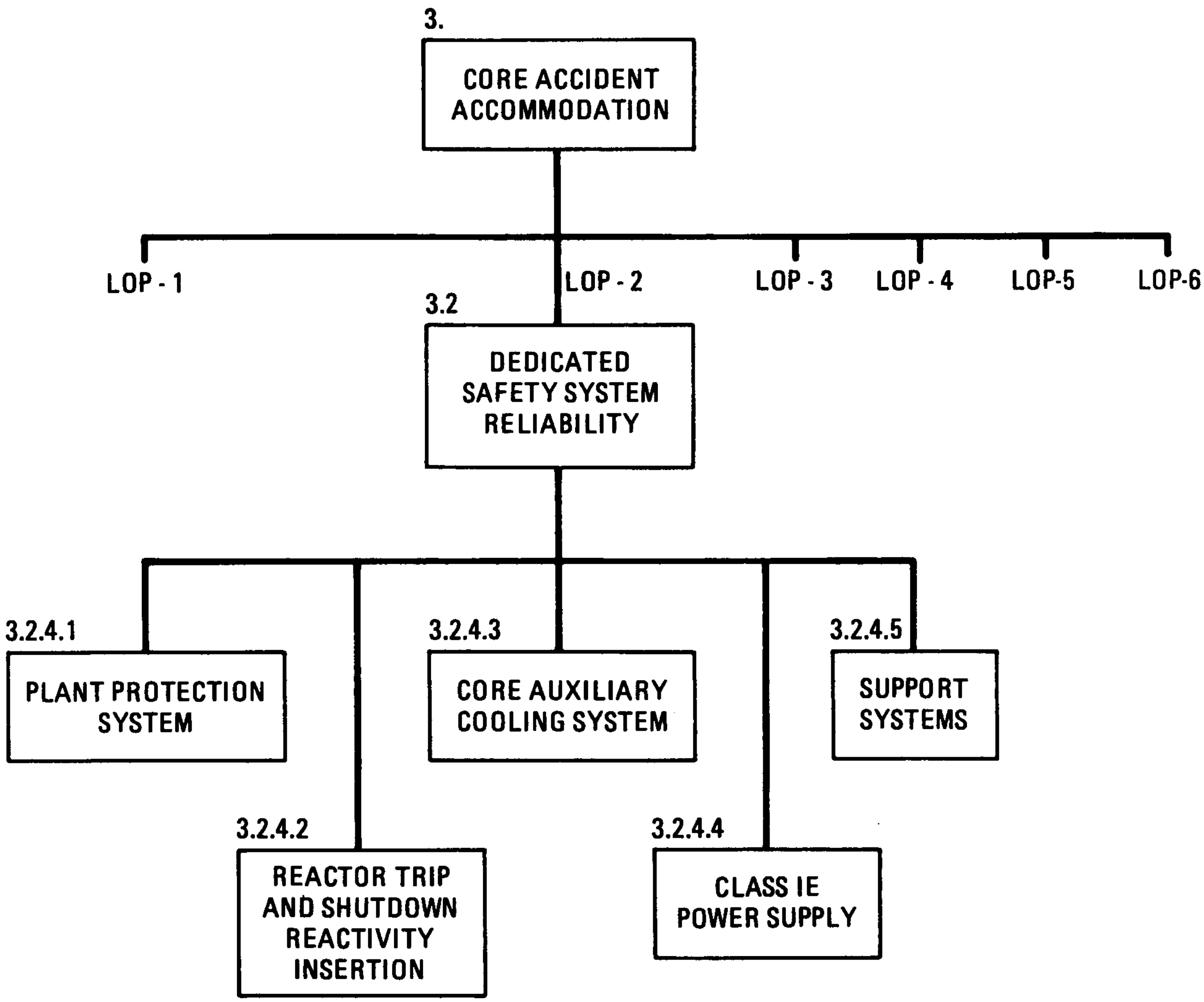

Fig. 3-2. LOP-2 task breakdown

3.2.3. Scope

In contrast to the LOP-1 operating systems, the dedicated safety systems are primarily intended to safely shut down the plant and to provide long-term residual heat removal. However, this GCFR Safety Program Plan does not list all activities in the dedicated safety systems. It details only those activities to (1) develop safety-related criteria, reliability targets, and reliability criteria; (2) assess system reliability; (3) assess dynamic performance of the dedicated safety systems as they relate to overall dynamic response of the core cooling systems; and (4) assess the design relative to the reliability objectives. This latter assessment is performed after conceptual and preliminary designs are completed.

The design function designs, develops, and tests the dedicated safety systems and their components. Development plans sited in the following subsections outline these activities.

This report clearly delineates the functions provided by the LOP-2 operating dedicated safety systems. The LOP-2 systems provide automatic, reliable shutdown and cooldown of the core in the event of the following:

1. The operating systems of LOP-1 fail to provide their designated functions of core shutdown and cooldown for anticipated operational occurrences.

2. Core shutdown and cooldown is required for events outside the design envelope for LOP-1 systems but within the design basis envelope for the total plant.

3.2.4. LOP-2 Task Breakdown

Figure 3-2 shows the LOP-2 task breakdown. Each of the five identified task areas corresponds to a hardware system. The generic work packages for each task area are outlined as follows:

1. Criteria.
2. Reliability assessment.
3. Dynamic performance.
4. Design assessment.

3.2.4.1. Plant Protection System (PPS). The reactor PPS includes all the sensors, logic, and actuators (e.g., trip actuators, valve actuators, etc.) to generate and process protective function signals. These signals (1) actuate reactor trip, (2) actuate backup shutdown rod insertion, and (3) initiate the safety-related residual heat removal functions of the core cooling systems for all conditions that result in a reactor trip. Reference 3-10 is a development plan for the PPS prepared by the design element.

This task is intended to assure that the PPS provides the required safety function of detecting off-normal performance and initiating reactor shutdown and cooldown. In addition, this function will have sufficient reliability to support the overall reliability goal presented in Section 3.2. The following four work packages will meet this objective:

1. Criteria. Develop criteria identifying the top-level safety requirements which must be satisfied by the PPS. Develop a target reliability allocation for the PPS which will support the overall reliability allocation for the GCFR.

2. Reliability assessment. Evaluate the reliability of the PPS. Review proposed design modifications for their impact on the assessed base case reliability. Prepare recommendations for those

design options which meet the relability target. Develop reliability demonstration requirements for PPS components for which the data base is inadequate.

3. Dynamic performance. Evaluate the dynamic performance of the PPS. Focus on those operational characteristics that are directly related to the overall plant shutdown and cooldown characteristics.

4. Design assessment. Peform an independent design review at the completion of conceptual and preliminary designs to assure that the criteria in item 1 are satisfied.

3.2.4.2. Reactor Trip and Shutdown Reactivity Insertion. The PPS (Section 3.2.4.1) assures reliable actuation of the release mechanism for the reactor trip system and for the backup shutdown system if a PPS setpoint is exceeded. The reactor trip and backup shutdown reactivity insertion assures that the reliability of physical insertion of the control and the shutdown rods, given a PPS-actuated release, is commensurate with the LOP-2 reliability goal.

The PPS-actuated release of the control rod holding magnet trips the reactor. The reactor trip reactivity insertion thus encompasses the gravity fall of the control rods into the control assembly guide channels and the deceleration of the rods upon full insertion.

The PPS-actuated backup shutdown is not fully resolved. A self-actuated release on the backup shutdown rods recently incorporated as an inherent LOP-3 feature may deviate the need for a PPS actuated backup shutdown. However, until resolved, a PPS-actuated insertion of the backup shutdown rods is maintained and, to the extent practical, is independent, redundant, and diverse from the PPS control rod actuation. This function has considered magnet release mechanisms and motor driven insertion.

This task is intended to assure that the PPS-actuated reactivity insertion by the control rods (reactor trip) and by the backup shutdown rods (if required) can shut down the reactor with a reliability commensurate with the overall LOP-2 reliability goal defined in Section 3.2. The following four work packages will meet this objective:

1. Criteria. Develop a target reliability allocation for reactivity insertion by the reactor trip and the backup shutdown systems following a PPS-actuated release. Define requirements for PPS-actuated reactivity insertion to meet the reliability target.

2. Reliability assessment. Evaluate, to the extent practical, the reliability of reactivity insertion by the control rods (reactor trip) and by the backup shutdown rods, given a PPS-actuated release. Review proposed design modifications for their impact on the assessed base case reliability. Prepare recommendations for those design options which meet the reliability target and assess the need for a PPS-actuated backup shutdown rod release. Develop reliability demonstration requirements for components with an inadequate data base.

3. Dynamic performance. Evaluate the dynamic performance of PPS-actuated reactivity insertion by the control rods and the backup shutdown rods.

4. Design assessment. Perform an independent design review at the completion of conceptual and preliminary designs to assure that the criteria in item 1 are satisfied.

3.2.4.3. Core Auxiliary Cooling System (CACS). The CACS is one of two safety-class residual heat removal systems in the GCFR. Section 3.1.4.1 discussed the SCS. The CACS must provide adequate cooling and prevent the temperatures of the fuel, cladding, and reactor internals from exceeding prescribed limits for all accident sequences within the GCFR design basis envelope. The CACS is designed to operate in a forced circulation mode. In this mode, active systems are required to operate continuously to provide the desired cooling. The CACS is also designed to provide adequate core cooling by natural circulation (passive means), provided the primary coolant is pressurized to at least 0.1 MPa (10 atm). This inherent operating mode is not within the scope of LOP-2, but is required in LOP-3, as discussed in Section 3.3.

The CACS is comprised of three independent loops. Each loop consists of a core auxiliary heat exchanger, an auxiliary circulator driven by a variable speed electric motor, and a core auxiliary cooling loop. The water loop consists of a pressurized surge tank, circulating water pumps, and an auxiliary loop cooler, which is an air-blast heat exchanger. Reference 3-11 is a development plan for the CACS prepared by the design element.

This task is intended to assure that the reactor shutdown system provides the required core cooling functions with sufficient reliability to support the overall relability goal presented in Section 3.2. The following four work packages will meet this objective:

1. Criteria. Develop criteria identifying the top-level safety requirements for the CACS. Develop a target reliability allocation for the CACS which will support the overall GCFR reliability allocation.

2. Reliability assessment. Evaluate the reliability of the CACS. Review proposed design modifications for their impact on the assessed base case reliability. Prepare recommendations for these design options which meet the reliability target. Develop reliability demonstration requirements for CACS components for which the data base is inadequate.

3. Dynamic performance. Evaluate the dynamic performance of the CACS. Focus on those operational characteristics that are directly related to the overall plant shutdown and cooldown characteristics.

4. Design assessment. Perform an independent design review at the completion of conceptual and preliminary designs to assure that the criteria in item 1 are satisfied.

3.2.4.4. Class IE Power Supply. The dedicated safety systems are normally powered from the preferred power supply. The Class IE power supply constitutes the safety-grade backup power supply for the dedicated safety systems. The station diesel generators constitute the source of Class IE ac power, which is distributed to the end uses through the IE power buses. The CACS and its dedicated support systems are the principal systems receiving Class IE power; however, the SCS may also be supplied by the Class IE power supply to meet the safety classification requirement. A separate and diverse source of Class IE power for the SCS has been considered, but the reliability need has not been fully resolved (see Section 3.1.6).

Three Class IE storage batteries provide the uninterruptible dc power supply for switchgear control annunciators, indicating lights, emergency lighting, and the uninterruptible power systems.

This task is intended to assure that the Class IE power supply supports the dedicated safety systems with sufficient reliability to meet the LOP-2 reliability goal. The following three work packages will meet this objective:

1. Criteria. Develop a target reliability allocation for the Class IE power supply which will support the overall LOP-2 reliability goal. Develop requirements for the Class IE power supply to meet the reliability allocation.

2. Reliability assessment. Evaluate the reliability of the Class IE power supply. Review proposed design modifications for their impact on the assessed base case reliability. Prepare recommendations for the design options which meet the reliability target. Develop reliability demonstration requirements for components with an inadequate data base.

3. Design assessment. Perform an independent design review at the completion of conceptual and preliminary designs to assure that the criteria established in item 1 are met.

3.2.4.5. Support Systems. The reactor plant cooling water system is the principal support system required by the dedicated safety systems; other support systems are not essential. Component specific support systems, such as the auxiliary circulator support system, are treated as part of the component to which they are dedicated (i.e., the auxiliary circulator).

Dependence of both the LOP-1 heat removal systems and the CACS on common support systems has been recognized early as a fundamental limitation in the achievable residual heat removal reliability. Therefore, all support systems for the dedicated safety systems must be fully independent of the support systems serving the LOP-1 systems.

This task is intended to assure that support system independence has been achieved and that the support systems for the dedicated safety systems perform their functions with a reliability commensurate with the overall LOP-2 reliability target. The following three work packages will meet this objective:

1. Criteria. Develop a reliability target for support systems dedicated to LOP-2 which will support the overall LOP-2 reliability goal. Develop support system requirements to meet the reliability target.

2. Reliability assessment. Evaluate the reliability of the LOP-2 dedicated support systems. Review proposed design modifications for their impact on the assessed base case reliability. Prepare recommendations for design options which meet the reliability target. Develop reliability demonstration requirements for components with an inadequate data base.

3. Design assessment. Perform an independent design review at the completion of conceptual and preliminary designs to assure that the criteria in item 1 are met.

3.3. LOP-3: INHERENT ACCIDENT PREVENTION

3.3.1. Introduction

LOPs -1 and -2 assure that the operating and the active safety systems reduce the frequency with which LOP-3 features are required to prevent an accident to less than 10^{-4}/reactor yr. Beyond these normally provided safety features, essentially passive design characteristics should further reduce the probability of core damage. To clarify the meaning of inherent accident prevention, three component categories are defined as follows:

1. Active components. These require external power to perform their designated function. A system of sensors, actuators, and controllers usually governs operation of an active component. Active components are extensively used in LOPs -1 and -2. Active components should be avoided in accomplishing the LOP-3 objectives for inherent accident prevention.

2. Reactive components. These require moving parts but are either self-actuated or powered from a stored energy source (i.e., spring, battery pressure bottle). The process that the component controls should inherently actuate a reactive component. Check valves and self-actuated control rod releases are examples of reactive components. Reactive components can be used to meet an LOP-3 inherent accident prevention objective.

3. Passive components. These perform their design function without moving parts and constitute essentially all the structural components.

An accident-prevention feature is defined as inherent if the safety-related function does not depend on active components, the PPS, nor the PCS.

3.3.2. Objectives

Accomplishing the LOP-1 and -2 objectives assures that all design basis events traditionally considered in nuclear power plant licensing are met. Additional LOP-3 features assure significant protection beyond these normally considered events before an event sequence can progress into a core-damage accident. While LOPs -1 and -2 provide all the traditional safety features, in all probabilistic risk assessments to date, these systems cannot be expected to prevent core damage at a probability as low as 10^{-6}/yr. Rather, they limit the core damage frequency to the range of 10^{-4} to 10^{-5}/yr. Furthermore, LOP-1 and -2 active systems are believed to be exploited to the extent practical, and adding an additional active safety system barrier probably would not accomplish the desired confidence of very low probability core melt. Therefore, LOP-3 emphasizes inherent reactor shutdown and residual heat removal features, which are as independent from human intervention, power supplies, and PPS actions as is practically achievable. With such an approach, the core melt probability is believed possible to reduce to a level as low as 10^{-6}/reactor yr.

To accomplish these objectives, three specific success criteria are established for LOP-3 as follows:

1. Demonstrate that inherent safety design features incorporated into the GCFR design reduce the cumulative frequency for loss of core cooling geometry to less than 10^{-6}/reactor yr (i.e., to a frequency which is a factor of 100 lower than that required for LOPs -1 and -2 alone).

2. Demonstrate that local faults within an assembly can be accommodated without loss of assembly coolability nor assembly duct integrity.

3. The dose consequences to any individual in the public from any event sequence which is accommodated by LOP-3 features is not permitted to exceed the annual radiation exposure limit established for radiation exposed workers. This limit is approximately equivalent to 10% of the 10CFR100 (Ref. 3-12) dose limits and is to be evaluated using realistic consequence models and assumptions.

The consequence limit in criterion 3, above, is expected to be satisfied as long as core cooling geometry is maintained. The criterion is maintained for completeness and because the radiological consequences of local faults in the core are not well understood. However, no R&D tasks for consequence mitigation are identified in LOP-3.

3.3.3. LOP-3 Task Breakdown

Figure 3-3 shows the LOP-3 task breakdown. Four LOP-3 barrier task areas are identified: (1) reactor shutdown system faults, (2) pressurized shutdown heat removal system faults, (3) depressurized shutdown heat removal system faults, and (4) local faults. Pressurized shutdown heat removal system faults are basically accommodated by natural circulation, while depressurized shutdown heat removal system faults do not have natural circulation available in the primary coolant. The generic work packages for each task area are outlined as follows:

1. Criteria.
2. Design option development.
3. Reliability assessmet.
4. Dynamic response.
5. Instrument requirements.
6. Test requirements.
7. Design selection/design review.

3.3.3.1. Reactor Shutdown System Faults. These prevent negative reactivity from being inserted into the core to terminate the fission chain reaction due to failures in the PCS, the PPS, and/or their respective insertion or absorber release mechanisms.

Objective. Demonstrate that if all active reactivity insertion is postulated to fail inherent GCFR reactivity insertion can insert sufficient negative reactivity in a short enough time to maintain core cooling geometry until the reactor is permanently shut down. The inherent reactor shutdown features and active shutdown systems should reduce the overall failure

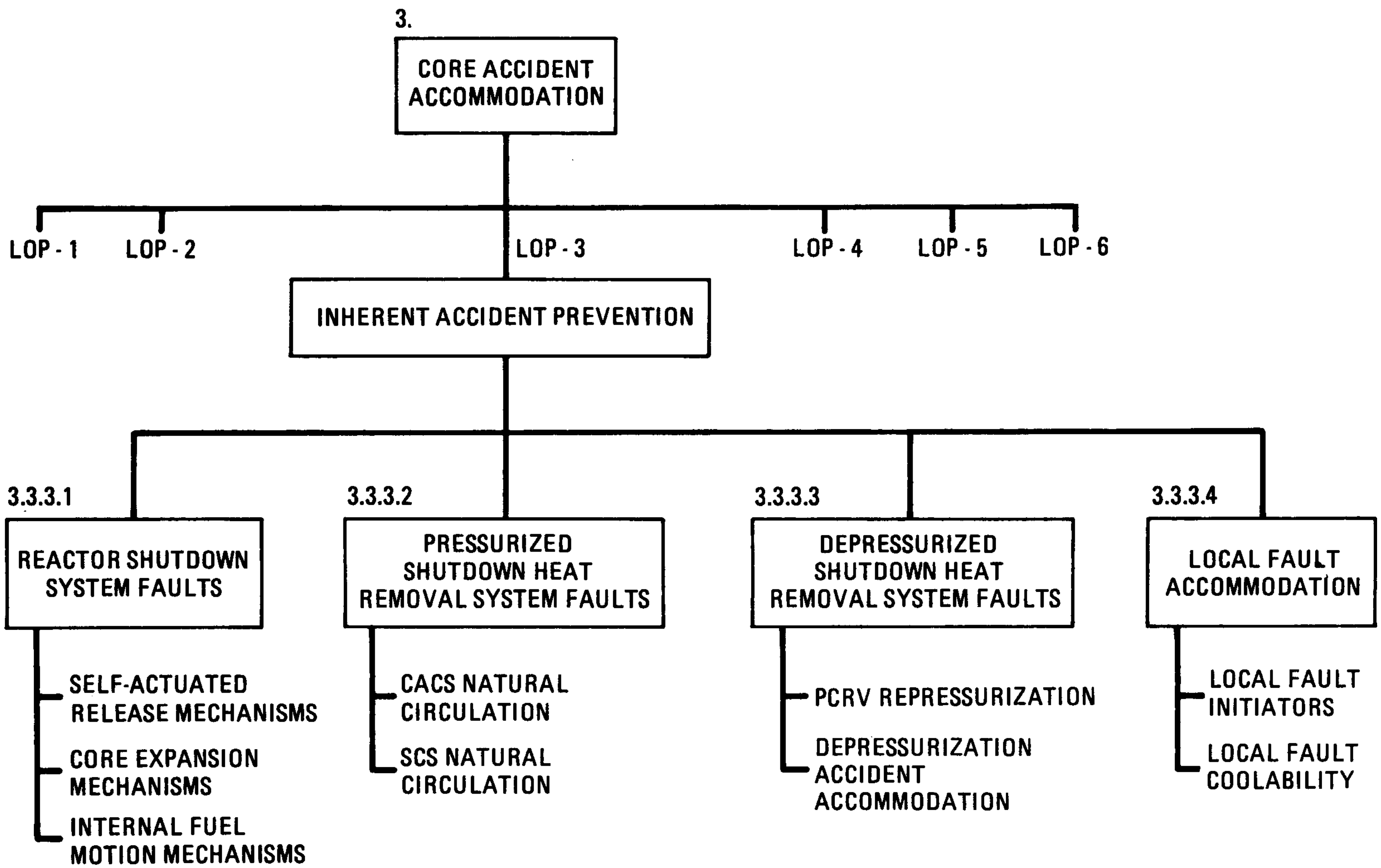

Fig. 3-3. LOP-3 task breakdown

probability to terminate the fission chain reaction to a freqency sufficiently below 10^{-6}/yr to meet the overall LOP-3 probabilistic safety goal.

Three potential mechanisms for inherent reactivity insertion will be investigated to accomplish this second level product.

3.3.3.1.1. Self-Actuated Release Mechanisms.

Objective. Investigate the feasibility of self-actuated release for absorber insertion to meet the objectives of task 3.3.3.1.

Work Packages. The following seven work packages will meet this objective:

1. Criteria. Develop necessary and sufficient criteria for self-actuated release to meet the task objective. Criteria will specifically address initiator, performance, and reliability.

2. Design option development. (Design program task. This work may be executed under the GCFR design program rather than the safety program.) Develop feasible GCFR design options for self-actuated release. This task will be based on the LMFBR program design options. Self-actuated shutdown may consider the following options:

    a. Self-actuated release incorporated into the secondary shutdown system.

    b. A secondary shutdown system with self-actuated release only.

    c. A third shutdown system with self-actuated release only.

    Rank options according to design preference.

3. Reliability assessment. Help develop design options and select the reference concept by evaluating for reliability potential and limitations. Develop requirements to demonstrate reliability for the options considered. Rank options according to reliability.

4. Dynamic response. Help develop design options by analyzing transients for the release mechanisms against the criteria identified in item 1. Identify requirements for developing dynamic methods for the options considered. Rank options according to dynamic response preference.

5. Instrumentation requirements. Identify instrumentation requirements, if any, for the options considered.

6. Test requirements. Collect requirements for the options considered to test design development, dynamic response, and reliability. Develop test facility reqirements for the options considered. Rank options according to test facility requirements.

7. Design option evaluation. (Potential design program task.) Evaluate the design alternatives. Recommend a preferred option and associated GCFR development requirements. These requirements should be considered in conjunction with other preferred options under task 3.3.3.1.

3.3.3.1.2. Core Expansion Mechanisms.

Objective. Investigate the feasibility of utilizing inherent core expansion mechanisms to perform inherent reactor shutdown and to meet the objectives of task 3.3.3.1.

Work Packages. The following six work packages will meet this objective:

1. Criteria. Develop necessary and sufficient criteria for core expansion mechanisms to perform inherent shutdown function.

2. Design option development. (Design program task.) Investigate core design options to enhance negative reactivity insertion due to core thermal expansion in response to loss of flow and positive reactivity insertion initiators. Develop experiment requirements to develop the design of each option. Rank options according to design preference.

3. Reliability assessment. If possible, help evaluate the design option by considering reliability. Rank options according to reliability.

4. Dynamic response. Perform core transient response analyses for loss of flow and reactivity insertion initiators for the design options identified in item 2. Develop dynamic test requirements. Rank options according to dynamic response preference.

5. Test requirements. Collect test requirements for each option. Develop test facility requirements. Rank options according to test facility requirements.

6. Design option evaluation. (Potential design program task.) Evaluate the available design options. Recommend the preferred option and associated development program.

3.3.3.1.3. Internal Fuel Motion Mechanisms.

Objective. Investigate the feasibility of negative reactivity effects due to fuel motion within the fuel pins to meet the objectives for task 3.3.1.

Work Packages. Internal fuel motion can potentially result in negative reactivity effects during overpower conditions which may be sufficient to meet objectives of task 3.3.3.1 for overpower initiators. The following two work packages will meet this objective:

1. Criteria. Define necessary and sufficient criteria to meet the objective.

2. Design option development. Review the ongoing LMFBR work on internal fuel motion and determine its applicability to the GCFR. On the basis of this assessment, identify any GCFR-specific work packages to establish the feasibility of this option for the GCFR.

3.3.3.2. Pressurized Shutdown Heat Removal System Faults. Pressurized shutdown heat removal system faults constitute those conditions where the LOP-1 and -2 active core cooling systems are reduced to below the minimum capacity required to maintain core cooling geometry in the shutdown reactor, but primary coolant system pressure remains above the minimum pressure required for natural circulation core cooling.

Objective. Demonstrate that inherent GCFR shutdown heat removal features maintain core cooling geometry following a postulated total failure of all active components in the LOP-1 and -2 shutdown heat removal systems. The inherent shutdown heat removal reliability combined with the active LOP-1 and -2 shutdown heat removal systems should reduce the overall probability of loss of decay heat removal to a frequency sufficiently below 10^{-6}/reactor yr to meet the overall probabilistic LOP-3 safety goal.

The CACS and SCS both remove heat from the core to the ultimate heat sink by natural circulation to accomplish this second-level product. Task 2.2.4.2 defines the extent to which this capability is required. Two third-level tasks support these task objectives.

3.3.3.2.1. CACS Natural Circulation.

Objective.

1. Demonstrate that shutdown heat can be removed from the core to the ultimate heat sink by CACS inherent features when (a) all active CACS components are postulated to fail and (b) the primary coolant is above the minimum pressure required for natural circulation.

2. Demonstrate the extent to which the reliability objective for task 3.3.3.2 can be met by considering failures in the reactive and passive components of the CACS.

Work Packages. The following seven work packages will meet these objectives:

1. Criteria. Define and document criteria for CACS natural circulation shutdown heat removal, including initiator requirements, natural circulation performance criteria, minimum helium pressure requirements, etc.

2. Design option development. (Design program task.) Develop design options for CACS natural circulation shutdown heat removal and assess the technical feasibility of each option. Rank options by design preference. Develop a detailed design for the option selected under item 7 and issue a design development plan if required.

3. Reliability. Evaluate the shutdown heat removal reliability improvement for each feasible option. Identify special reliability demonstration requirements for each option and rank options by reliability improvement. Perform a detailed reliability assessment of the option selected under item 7 and develop reliability demonstration requirements if required.

4. Dynamic response. Evaluate each option with respect to the dynamic response to key initiators. Identify special requirements for dynamic response verification and methods development. Rank each option according to dynamic response preference. Perform a detailed dynamic analysis of the option selected under item 7 and issue a development plan if required.

5. Instrument requirements. Identify instrumentation requirements for each option and rank each option according to instrumentation simplicity.

6. Test requirements. Collect test requirements to verify design development, reliability, and dynamic response; establish test facility requirements and test program scope. Develop detailed test facility requirements and a test program plan for the option selected under item 7.

7. Design selection. Evaluate the design option assessment under items 2 through 6. Recommend an overall preferred option for the GCFR program to implement. Perform an independent design review after completing the conceptual and preliminary designs to assure that the criteria in item 1 are met.

3.3.3.2.2. SCS Natural Circulation. The requirements for SCS natural circulation are dependent upon the conclusions of task 3.3.3.2.1 and on integrating the results of that task into the integrated reliability analysis for shutdown heat removal which is performed under task 2.2.4.2.

To the extent that such requirements for SCS natural circulation are identified, the following objectives apply to this task.

Objective. Demonstrate that design features for SCS natural circulation are incorporated into the SCS system design to meet the requirements identified for SCS natural circulation. Demonstrate that the combined capability for CACS and SCS natural circulation shutdown heat removal in conjunction with the LOP-1 and -2 active shutdown and heat removal systems meet the reliability objective for task 3.3.3.2.

Work Packages. Work packages identical to task 3.3.3.2.1 will meet these objectives.

3.3.3.3. Depressurized Shutdown Heat Removal System Faults. Depressurized shutdown heat removal system faults reduce the primary coolant system pressure either accidentally or intentionally to below the minimum pressure required for natural circulation, then lose all LOP-1 and -2 active cooling systems. Intentional primary coolant depressurization may occur for several reasons: refueling, repair of components inside the PCRV, in-service inspection (ISI), etc. Depressurization accidents involve structural failures at the primary coolant system boundary. Combined with the loss of LOP-1 and -2 active systems, this category of fault conditions may be of such low probability that depressurization accident cooling faults may not have to be accommodated with an inherent feature to meet the probabilistic safety goal for LOP-3.

Objective. Demonstrate that inherent shutdown heat removal features for depressurized conditions are available or can be restored to maintain core cooling geometry following a postulated loss of all active components in LOP-1 and -2 shutdown heat removal systems. Low probability structural failures combined with LOP-1 and -2 active cooling system failures need not be accommodated by an inherent barrier if the combined probability for progression into LOP-4 of all exempted event sequences is less than 10^{-7}/reactor yr. Inherent shutdown heat removal accommodation for depressurized conditions combined with active system failures in LOPs -1 and -2 should reduce the probability of accident progression into LOP-4 to sufficiently below 10^{-6}/reactor yr to meet the overall LOP-3 probabilistic safety goal.

Two tasks support this second level product:

3.3.3.3.1. PCRV Repressurization. For intentionally depressurized conditions, the status of primary coolant boundary seals is known, and a sealed condition can be restored. PCRV represurization can then restore natural circulation as the inherent LOP-3 core cooling feature. Ideally, repressurization would be accomplished by only reactive and passive components to qualify as an inherent feature. A program requirement for PCRV repressurization has been established.

Objective. Demonstrate that the PCRV repressurization system qualifies as an inherent feature and that, in combination with natural circulation and LOP-1 and -2 active systems, it reduces the probability of accident progression into LOP-4 with the PCRV depressurized to sufficiently below 10^{-6}/reactor yr to meet the reliability objective for task 3.3.3.2.

Work Packages. Seven work packages will meet this objective:

1. Criteria. Develop criteria for PCRV repressurization, including initiator requirements, time to repressurize, minimum pressure, cooling requirements during repressurization, etc.

2. Design option development. (Design program task.) Develop design options for PCRV repressurization and rank options according to design preference.

3. Reliability assessment. Evaluate reliability improvement for depressurized shutdown heat removal for each option and rank options by reliability.

4. Dynamic response. Evaluate core cooling and transition to natural circulation for each option and rank options by dynamic response preference.

5. Instrument requirements. Develop instrumentation and actuation requirements for each option and rank options according to instrumentation and actuation simplicity.

6. Test requirements. Collect test requirements from items 2 through 5 and develop test facility requirements. Rank options according to test requirements.

7. Design review. Evaluate design options for PCRV repressurization and recommend an overall preferred option to GCFR management for implementation. Perform an independent design review after completing conceptual and preliminary designs to assure that the criteria in item 1 are met.

3.3.3.3.2. Depressurization Accident Accommodation. This task is only identified for completeness. A task plan and work packages will be developed if a requirement for inherent accommodation of depressurization accidents is established under task 2.2.4.2.

3.3.3.4. Local Fault Accommodation. Preventing local faults in the core fuel, such as local fuel failures or flow blockages, is established as a specific LOP-1 objective. This task considers means to accommodate such local faults if postulated to occur in spite of LOP-1 local fault prevention.

Objective. Demonstrate that local faults, which may develop in spite of LOP-1 prevention of local faults, can be accommodated in LOP-3 without failure or excessive distortion of the subassembly wall. Demonstrate that the probability of local fault propagation to subassembly wall damage is consistent with the frequency goal for LOP-4 initiators.

Two third level tasks support the objectives of this task.

3.3.3.4.1. Local Fault Initiators.

Objective. Define local fault initiators to be considered for accommodation within LOP-3 limits and provide probabilistic justification for excluding more severe local faults from LOP-3 accommodation.

Work Packages. Five work packages will meet this objective:

1. Criteria. Define criteria for local fault initiator selection, fault propagation limits, and success criteria for local fault accommodation. Establish local fault detection and diagnostic requirements.

2. Initiator selection. Define local fault initiators for LOP-3 accommodation consistent with criteria in item 1.

3. Design option development. Develop core assembly design options, if necessary, to meet the LOP-3 objectives for local fault accommodation.

4. Reliability assessment. Support the selection of local fault initiators appropriate for consideration within the LOP-3 reliability objectives. Develop probabilistic evidence to support exclusion of more severe local faults from the LOP-3 accommodation requirement.

5. Test requirements. Identify test requirements to support the local fault initiator selection and to support exclusion of more severe local faults from LOP-3 consideration.

6. Local fault detection. Develop local fault detection and diagnostic capability consistent with the criteria developed in item 1. The fuel rod pressure equilization system (PES) is expected to satisfy most of these requirements.

3.3.3.4.2. Local Fault Coolability.

Objective. Demonstrate that local faults identified in task 3.3.3.4.1 are coolable without exceeding the limits for fault propagation. Demonstrate that the cumulative combination of fault initiator probability and the probability of successful fault cooling is sufficiently below 10^{-6}/reactor yr to meet the overall probabilistic objective for LOP-3.

Work Packages. Four work packages will meet this objective:

1. Blockage formation. Characterize the formation of flow blockages within a core assembly and define the blockage geometry for coolability analysis for the fault initiators identified in task 3.3.3.4.1.

2. Blockage coolability. Demonstrate that the flow blockages and blockage geometries identified in item 1 are coolable without exceeding the criteria for fault propagation limits and for the success criteria for local fault accommodation identified in task 3.3.4.1. Define minimum cooling requirements for successful blockage cooling.

3. Reliability assessment. Determine the reliability of blockage cooling from the cooling requirements developed in item 2.

Determine the cumulative probability of local fault development combined with unsuccessful blockage cooling.

4. Test requirements. Develop test requirements to support the demonstration that blockage development and blockage coolability meet the criteria developed in task 3.3.3.4.1

3.4. IN-VESSEL ACCIDENT ACCOMMODATION

3.4.1. Introduction

LOP-4 evaluates the PCRV as a successful barrier to contain accidents which proceed to loss of core cooling geometry by failure of LOPs -1 through -3. To accomplish this function, the PCRV must contain energy releases that may occur in the sequence of core melting and the molten core debris. This capability can be assigned to the GCFR reactor vessel, because the PCRV is very massive and structurally redundant with an inherently very large capability to contain energy release and because the normal cooling system to cool the PCRV liner can also remove decay heat from molten core debris inside the PCRV.

3.4.2. Objectives

The success of the LOP-4 barrier is measured by the following specific objectives:

1. Establish the limiting energy release, fuel vaporization fraction, and fission product release from the core for identifiable accident sequences with a probability greater than 10^{-7}/reactor yr by mechanistic accident analyses and supporting experiments.

2. Demonstrate that the boundary integrity of the PCRV is maintained for the bounding energy releases established by analysis.

3. Demonstrate that for identifiable accident sequences with a probability greater than 10^{-7}/reactor yr the core debris can be contained within the PCRV.

4. Demonstrate that for identifiable accident sequences with a frequency greater than 10^{-7}/reactor yr (i.e., LOP-4 terminated sequences) the activity release to the containment is limited such that the expected exposure to the public will not exceed the limits in 10CFR100 (Ref. 3-12), augmented by dose limits from plutonium of 75 rem to the lung and 150 rem to the bone.

5. Mechanistic accident analyses to demonstrate that objectives 2 through 4 are met should include variations in analysis parameters and accident paths. The unquantifiable residual probability that higher consequence sequences have been omitted should be expected to be less than 10%.

3.4.3. Scope

The LOP-4 program is intended to define specific work packages necessary to meet the objectives of this task. To ensure that the task objectives are met, the program will develop specific success criteria, realistic analysis methods, mechanistic accident analyses, experiment requirements, experiment plans, safety-related design or functional requirements (to be met by the design development program), and a design review (to assure meeting the safety-related requirements).

3.4.4. LOP-4 Task Breakdown

Figure 3-4 shows the LOP-4 task breakdown. LOP-4 is broken into four third level tasks. Each task addresses a specific LOP-4 objective:

1. Investigate core melt and core disruption accident sequences. Identify the accident category and the specific accident sequences which yield the limiting core energy release, fuel vaporization, fission product release, and core melt debris volume.

2. Quantify the PCRV response to the limiting core energy release. Identify specific functional requirements to meet LOP-4 objectives.

3. Investigate the PCRV capability to contain molten fuel in the lower central PCRV cavity. Establish the functional requirements for in-vessel molten fuel containment.

4. Quantify the attenuation of activity releases from the PCRV to the containment. Establish functional requirements necessary to accomplish the LOP-4 exposure limits.

3.4.4.1. Limit Core Energy Release and Fuel Vaporization. This task is intended to perform the core accident analyses necessary to accomplish objectives 1 and 5 for LOP-4. Each accident sequence investigated under this task involves core damage due to a loss of normal core cooling geometry. This is because, by definition, the accident prevention measures in LOPs -1 through -3 have failed. Indeed, some accident sequences require core disruption to attain neutronic shutdown and a stable subcritical core configuration. Core disruption releases fission products from the fuel and may cause partial fuel vaporization and mechanical energy release. Analysis of these accident sequences will establish the range of core mechanical

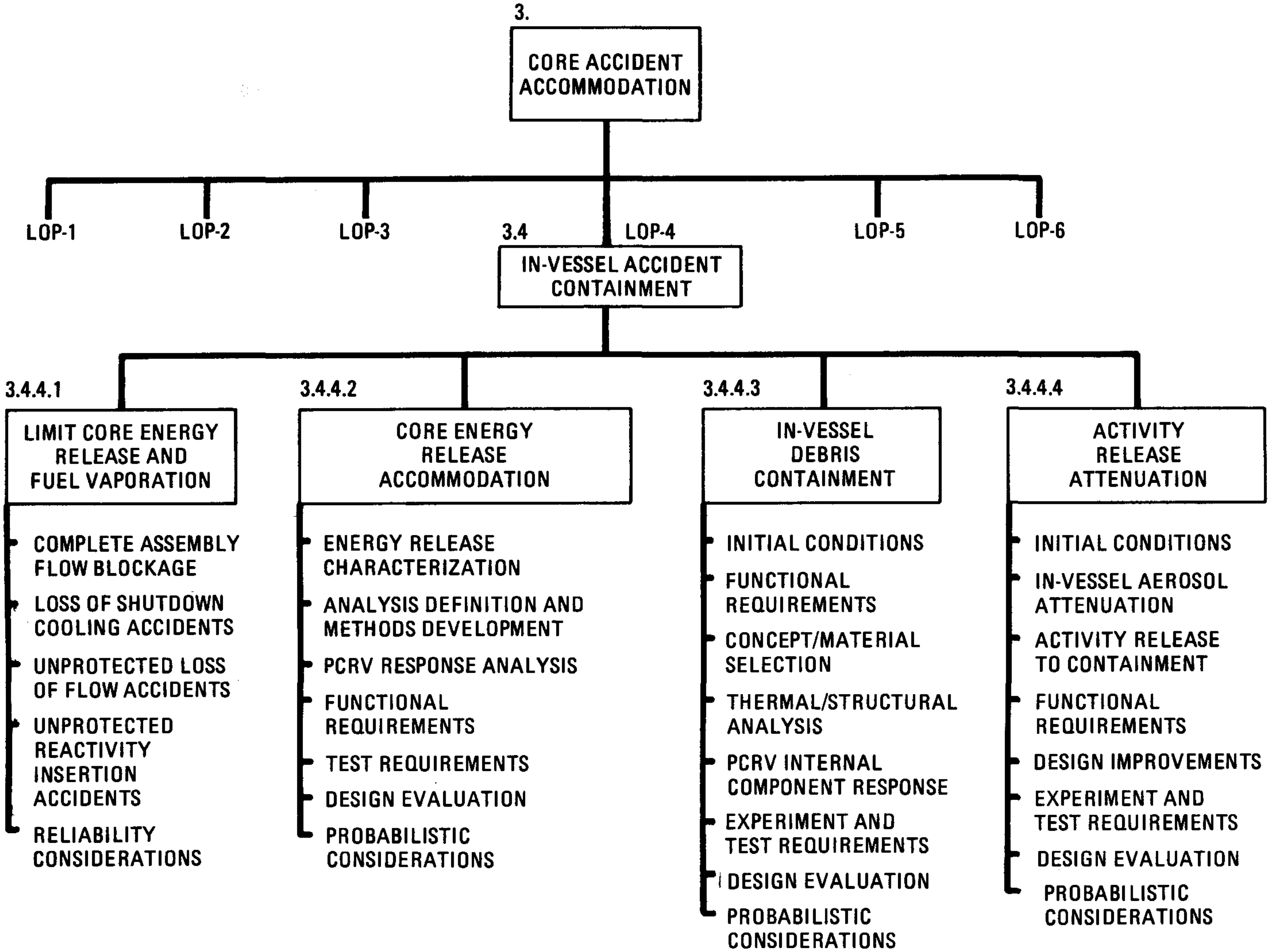

Fig. 3-4. LOP-4 task breakdown

energy release; the range of core fuel vapor and fission product release; and the rate, timing, and condition of molten fuel relocation to the bottom of the central PCRV cavity. These latter variables help demonstrate that the PCRV structural integrity is retained as an LOP-4 barrier. Four basic accident categories are established, spanning the range of core melt and core disruptive accidents in the GCFR. Each accident category exhibits a distinct pattern of core behavior and physical phenomena; however, variations in assumptions and parameters within a given category are similar. Therefore, each accident category can be treated generically to identify phenomenological R&D requirements, methods development, and integral test requirements. This report identifies the objectives and work packages for each accident category. The GCFR Safety Program Implementation Plan will discuss the state of knowledge and the means for completing each work package.

3.4.4.1.1. Complete Assembly Flow Blockage. Local flow blockages which are too large to maintain design cooling geometry need be considered for two reasons:

1. At the very low accident probabilities for which LOP-4 provides consequence mitigation, all mechanisms which could potentially cause complete assembly flow blockages cannot be eliminated in spite of the design provisions which prevent complete flow blockages by any single piece of debris. Obvious reasons are the accumulation of multiple debris at the assembly inlet or smaller debris at the inlet manifold or the grid spacers.

2. Melting and draining of cladding as a result of flow blockage can cause steel blockages to form near the core-lower axial blanket interface. The subsequent accumulation of molten fuel upon this steel blockage may propagate blockage to neighboring assemblies by lateral melt-through of the assembly duct walls. Therefore, damage propagation to neighboring assemblies must be bounded to

assure that local flow blockages do not control LOP-4 frequency and consequence.

Objective. Complete assembly flow blockages will be investigated to demonstrate that the consequences of damage propagation to neighboring assemblies are less than other accident categories in terms of energy release, fuel vaporization, fission product release, and molten debris generation.

Work Packages. The following five work packages will meet this objective:

1. Damage propagation mechanisms. Identify mechanisms for damage propagation from flow blocked assemblies to neighboring assemblies. Define analysis methods required to determine the extent of damage propagation.

2. Methods development. Develop or adapt analysis methods with the capability identified in item 1.

3. Analysis. Using the methods developed under item 2, analyze assembly flow blockage and damage propagation accident sequences. Perform damage propagation sensitivity analyses to demonstrate that the objective for this task is met.

4. Test requirements. Where warranted by uncertainties, define experiment requirements to substantiate the physical models in the analysis methods and data. Define test requirements, if necessary, to verify the integrated analyses prediction of damage propagation. Define experiment/test facility requirements to support the experiment/test needs.

5. Design improvements. Identify improvements in core design or plant operating procedures, where necessary, to meet the task objective.

3.4.4.1.2. Loss of Shutdown Cooling Accidents (LOSC). The LOSC accident category includes events and multiple failures which shut down a reactor with inadequate core heat removal, such that core cooling geometry is lost due to decay heat alone. While such accident sequences are considered to be of adequately low probability, they tend to be of somewhat higher probability for current GCFR designs than accident sequences where reactor shutdown fails. The safety program acknowledges this tendency by (1) allocating most LOP-1 through -3 failure probability to LOSC sequences and (2) emphasizing the investigation of core melt sequences in the shutdown reactor.

Objective. Investigate LOSC accidents to bound the core consequences in terms of energy release, fuel vaporization, fission product release, and molten debris generation. Mechanistic analyses and sensitivity studies will bound consequences in support of LOP-4 objectives 1 and 5. Experiments will supplement where necessary.

Work Packages. Relocating of molten fuel during the LOSC accident sequence may cause recriticality. This causes major phenomenological uncertainties in regard to (1) the physical conditions for recriticality, (2) the ability of the GCFR core to avoid recriticality during molten fuel relocation, and (3) the recriticality consequences of core energy release and fuel vaporization. Due to these uncertainties, LOSC accidents rank high in the GCFR Safety Program Plan. Figure 3-5 shows the current understanding of the LOSC accident sequence.

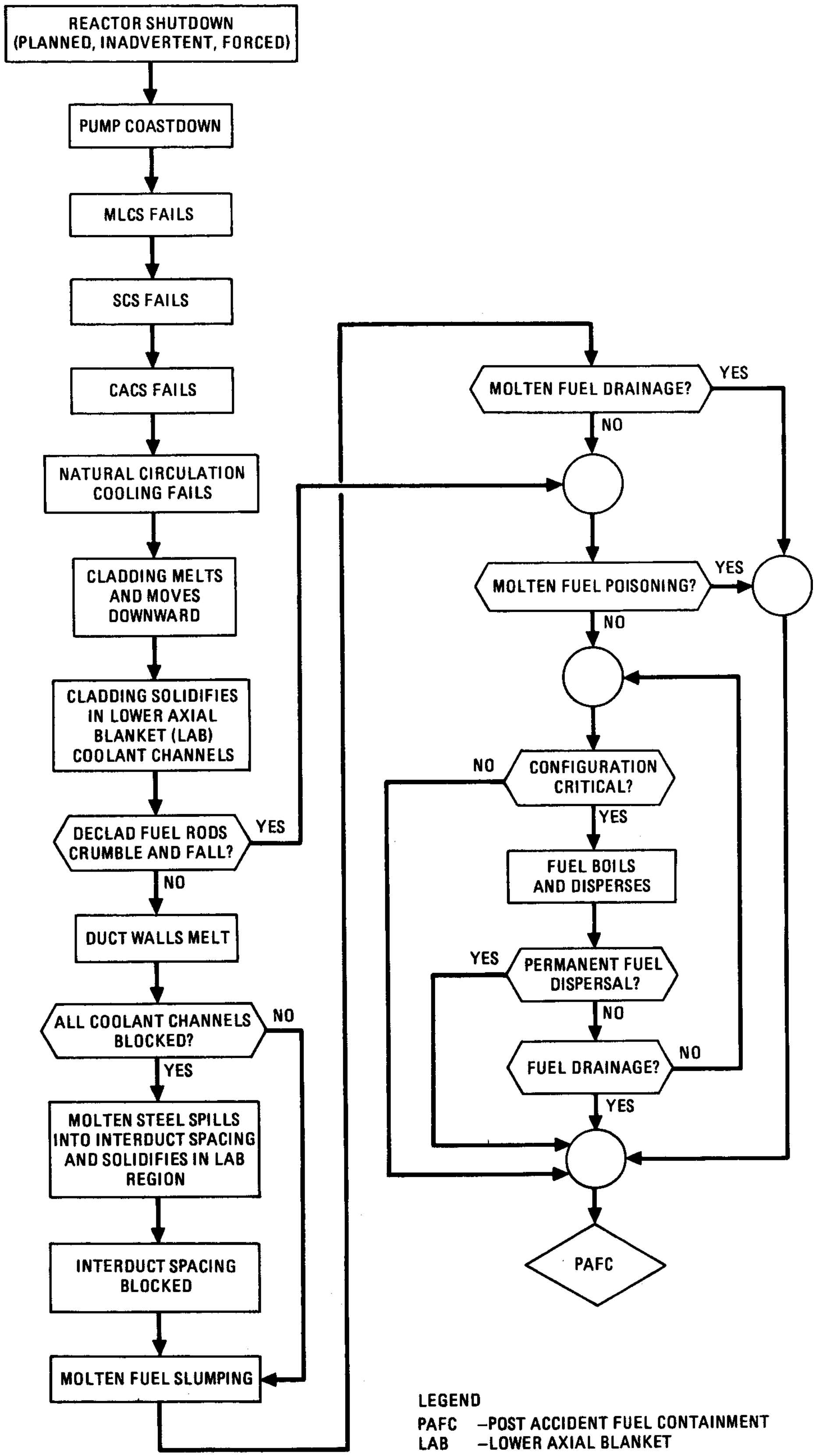

Fig. 3-5. LOSC accident sequence

The following five work packages will meet this objective:

1. LOSC accident sequences. Establish the LOSC accident sequences to be investigated and define the methods development required for the analysis. Group accident sequences into LOSC subcategories which can be analyzed generically.

2. LOSC methods development. Develop and/or adapt analysis methods with the capability identified in item 1. Integrate the methods development for this task with that for task 3.4.4.1.1, item 2 to the extent possible.

3. Analysis. Using the methods developed under item 2, perform analyses of LOSC accident sequences identified in item 1 and define the range of core energy releases, fuel vaporization, fission product releases, and molten debris generated in LOSC accidents. Perform a sufficient range of sensitivity analyses to meet LOP-4 objective 5 for LOSC accidents.

4. Test requirements. Where warranted by uncertainties, define experiment requirements to substantiate the physical models in the analysis methods and data. Define test requirements, if necessary, to verify the integrated analysis prediction. Define experiment/test facility requirements to support the experiment/test needs.

5. Design improvements. Identify improvements in core design or plant operating procedures, where necessary, to meet the task objective.

3.4.4.1.3. Unprotected Loss of Flow (ULOF) Accidents.* The ULOF accident category postulates reduced primary or secondary coolant flow while operating at power in combination with a failure of both the primary and secondary shutdown systems to insert sufficient negative reactivity to bring the reactor to hot standby. Since at most 3 out of 15 control rods or 1 out of 4 backup shutdown rods are required for this purpose, this accident category, in essence, postulates the complete common cause failure of both the primary and the secondary (backup) shutdown systems. While such accident sequences are expected to be extremely unlikely, the current experience base and knowledge of reactor shutdown system reliability does not permit this accident category to be eliminated from LOP-4 consideration.

Primary coolant flow can be reduced by reducing helium mass flow at system pressure or by reducing helium pressure (depressurization) at constant helium volume flow. Since helium mass flow reductions at pressure are by far the more frequent initiator, they will be emphasized. Primary system depressurizations that permit only a short operator action time (to insert control or shutdown rods if the PPS fails) are initiators of sufficiently low probability to be eliminated from LOP-4.

Objective. Investigate ULOF accidents to bound the core consequences of energy release, fuel vaporization, fission product release, and molten debris generation. Mechanistic analyses and sensitivity studies in support of LOP-4 objectives 1 and 5 will bound consequences. Experiments will supplement where necessary.

Work Packages. Two major phenomenological uncertainties in ULOF accident sequences relative to the task objective are (1) the influence of radial fuel homogenization on the reactivity ramp rate when fuel disruption occurs due to loss of neutron streaming and (2) the penetration distance of molten fuel and steel into the lower axial blanket from both high and

---

*In LMFBR terminology, this accident category is frequently referred to as loss of flow (LOF) or transient undercooling (TUC) accidents.

low power assemblies following neutronic subcriticality by initial disruption. Figure 3-6 shows the current understanding of the ULOF accident sequence.

The following five work packages will meet these objectives:

1. ULOF accident sequences. Establish ULOF accident sequences to be investigated and define the methods development required for the analysis. Group accident sequences into ULOF subcategories which can be analyzed generically.

2. Methods development. Develop and/or adapt analysis methods with the capability identified in item 1. Integrate the methods development for this task with that for task 3.4.4.1.4, item 2, to the extent possible.

3. Analysis. Using the methods developed under task 3.4.4.1.2, item 2, analyze ULOF accident sequences identified in task 3.4.4.1.3, item 1, and define the range of core energy releases, fuel vaporization, fission product releases, and molten debris generated. Perform a sufficient range of sensitivity analyses to meet LOP-4 objective 5 for ULOF accidents.

4. Test requirements. Where warranted by uncertainties, define experiment requirements to substantiate the physical models in the analysis methods and data. Define test requirements, if necessary, to verify the integrated analysis prediction. Define experiment/test facility requirements to support the experiment/test needs.

5. Design improvements. Identify improvements in core design or plant operating procedures, where necessary, to meet the task objective.

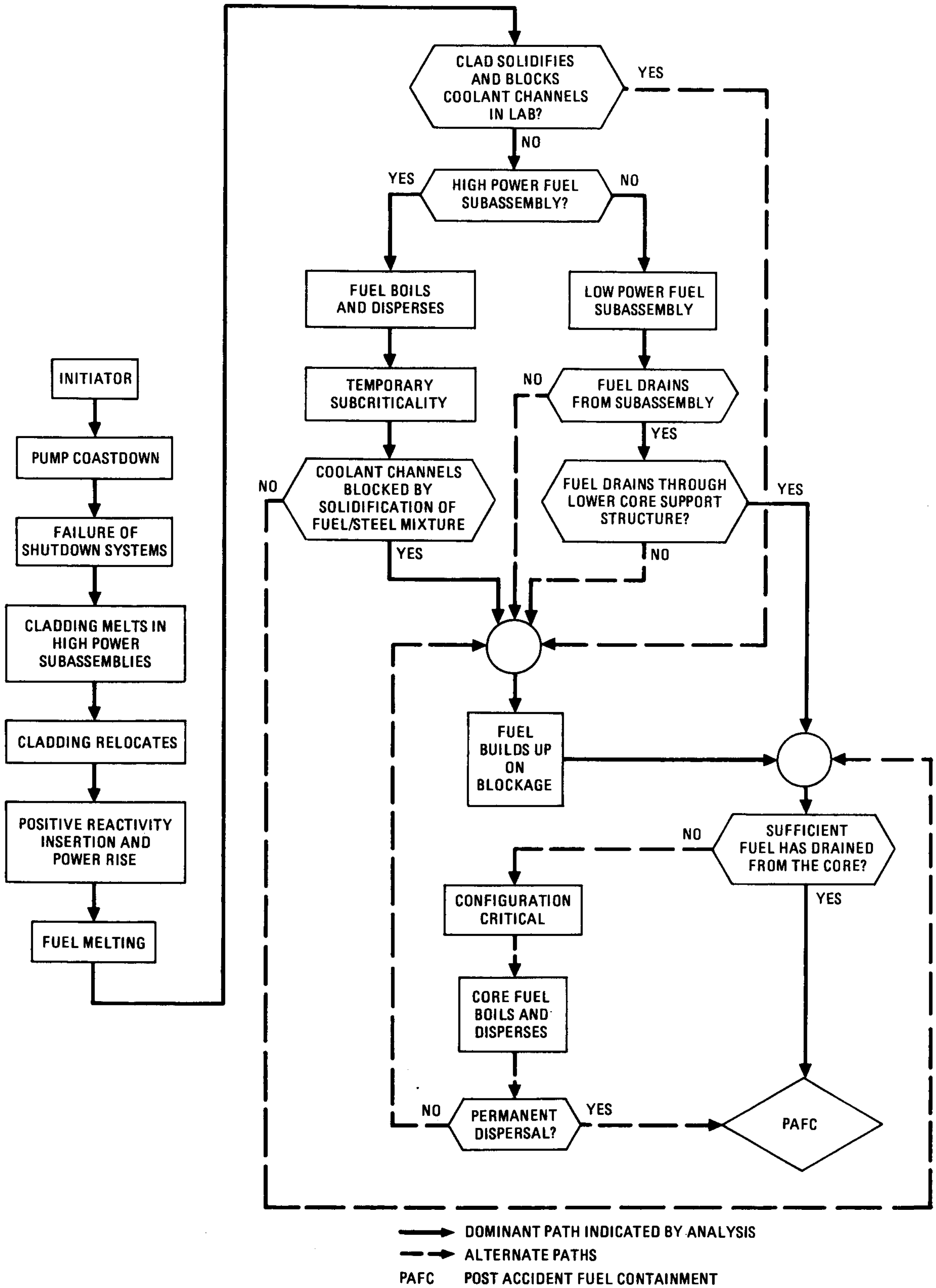

Fig. 3-6. ULOF accident sequence

3.4.4.1.4. Unprotected Reactivity Insertion (URI) Accidents.* The URI accident category postulates an inadvertent and continuous insertion of positive reactivity while operating at power in combination with a failure of both the primary and the secondary shutdown systems to insert sufficient negative reactivity to bring the reactor to hot standby. As in ULOF accidents, the URI accident category, in essence, postulates the complete common cause failure of both the primary and the secondary (backup) shutdown systems. While such accident sequences are expected to be extremely unlikely, the current experience base and knowledge of reactor shutdown system reliability does not permit this accident category to be eliminated from LOP-4. A positive reactivity insertion in the GCFR can occur due to a control rod being inadvertently withdrawn or coolant density decreasing as a result of accidental system depressurization. Since during a depressurization accident the phenomena induced by loss of flow are more controlling for the accident sequence, this accident sequence is treated as a ULOF accident (Section 3.3.4.1.3). Figure 3-7 shows the current understanding of the URI accident sequence.

Objective. Investigate URI accidents to bound the core consequences of energy release, fuel vaporization, fission product release, and molten debris generation. Mechanistic analyses and sensitivity studies in support of LOF-4 objectives 1 and 5 will bound consequences. Experiments will supplement where necessary.

Work Packages. The major phenomenological uncertainty in URI accident sequences relative to the task objective is the fuel fragmentation and sweepout behavior after molten fuel is ejected from the breached cladding. While rapid fragmentation into small particles followed by unimpeded sweepout is expeced on the basis of analysis, this accident characteristic has not been demonstrated.

---

*In LMFBR terminology, this accident category is frequently referred to as transient overpower (TOP) accidents.

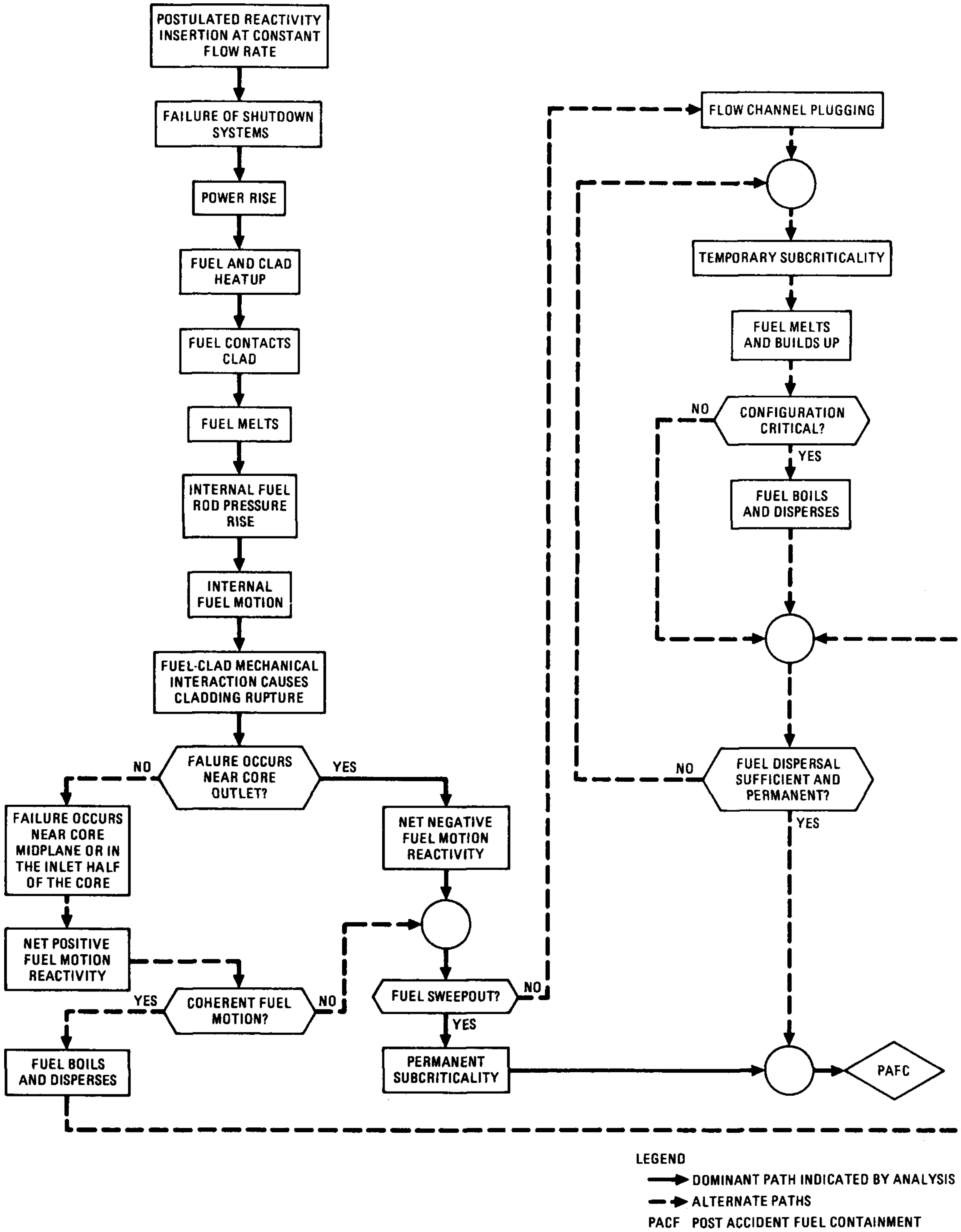

Fig. 3-7. URI accident sequence

The following five work packages will meet this objective:

1. URI accident sequences. Establish the URI accident sequences to be investigated and define the methods development required for the analysis. Group accident sequences into LOSC subcategories which can be analyzed generically.

2. URI methods development. Develop and/or adapt analysis methods with the capability identified in item 1. Integrate the methods development for this task with that for task 3.4.4.1.3, item 2, to the extent possible.

3. Analysis. Using the methods developed under item 2, perform analyses of URI accident sequences identified in item 1 and define the range of core energy releases, fuel vaporization, fission product release, and molten debris generated in URI accidents. Perform a sufficient range of sensitivity analyses to meet LOP-4 objective 5 for URI accidents.

4. Test requirements. Where warranted by uncertainties, define experiment requirements to substantiate the physical models in the analysis methods and data. Define test requirements, if necessary, to verify the integrated analysis prediction. Define experiment/test facility reuqirements to support the experiment/test needs.

5. Design improvements. Identify improvements in core design or plant operating procedures, where necessary, to meet the task objective.

3.4.4.1.5. Reliability Considerations. This task is intended to quantify to the extent practical the probability of accident sequences which are terminated by the successful operation of the LOP-4 barrier.

Objective. Demonstrate that the expected frequency for the combination of all identifiable accident sequences which progress through LOP-4 failure is less than 10^{-7}/reactor yr.

Work Packages. The following three work packages will meet this task objective:

1. Identify LOP-4 accident sequences. On the basis of reliability analyses performed for LOPs -1 through -3, identify those accident sequences which require successful LOP-4 consequence mitigation.

2. LOP-4 failure probability. Quantify to the extent practical the conditional probability of LOP-4 failure for those accident sequences, as identified in item 1, which require LOP-4 mitigation.

3. Cumulative LOP-4 failure probability. On the basis of results from items 1 and 2, quantify the expected frequency for the combination of all identifiable accident sequences which progress through LOP-4 failure. Substitute engineering judgment and justifications where the probability of important physical phenomena cannot be numerically quantified. Demonstrate that the expected failure frequency of LOPs -1 through -4 is less than the LOP-4 probabilistic limit.

3.4.4.2. Core Energy Release Accommodation. This task is intended to demonstrate that objective 2 for LOP-4 is met for the limiting core energy release defined by analysis under task 3.4.4.1. Demonstrating that PCRV boundary integrity is maintained as an LOP-4 barrier serves two important purposes:

1. Activity releases from the PCRV to the containment can be mitigated.

2. No missiles can be generated as a result of PCRV failure which may directly cause a failure or an increased leak rate of the containment building.

While the PCRV structure is generally acknowledged to have an extremely large energy absorption capability, relatively little attention has been focused on the PCRV penetrations and closures and on the possibility that a mechanical energy release in the core may be transmitted directly to a PCRV penetration through structural components such as control rod guide tubes or the instrument tree.

Objective. Demonstrate that objective 2 for LOP-4 is met for maintaining PCRV boundary integrity for the bounding energy release established by analysis under task 3.4.4.1.

The task objective is satisfied if the following criteria are met:

1. All PCRV penetrations and closures remain structurally intact. Small leaks through penetration seals may be considered acceptable.

2. The block valve on the PCRV relief valve remains operable and closing of the block valve will terminate PCRV blowdown into the containment.

3. The PCRV liner remains intact.

4. The PCRV liner cooling tubes remain structurally intact (including the welds to the liner) and operable.

5. The functional requirements for in-vessel molten fuel containment are satisfied.

Work Packages. The following seven work packages will meet this objective:

1. Energy release characterization. Based on the energy release analyses performed under task 3.4.4.1, transform the bounding energy release into the structural loading transient required for analysis in item 3.

2. Analysis definition and methods development. Define the analyses required to satisfy the task objective, accounting for all mechanisms for energy transmission to the PCRV liner, such as direct loading of mechanical structures connected to the PCRV, PCRV internal missiles, and shock wave transmission through the compressible helium. Develop and/or adapt the analysis methods needed for the analyses in item 3.

3. PCRV response analysis. Perform the structural response analyses for the PCRV and its internal structures and quantify the margin to failure of each component whose failure could violate the task objective. Investigate and quantify the response of the PCRV relief valves as a result of the core energy release.

4. Functional requirements. Define any functional requirements to be imposed on the design of the PCRV and its internal structures to assure that the task objective is met.

5. Test requirements. Where warranted by uncertainties and/or small margins, define test requirements to substantiate the physical models in the analysis methods and data. Define test facility requirements to support the test needs.

6. Design evaluation. Evaluate the design of the PCRV and its internal structures to assure that the design meets the LOP-4 related functional requirements defined in item 4.

7. Probabilistic considerations. Quantify to the extent practical the conditional probability that the PCRV will meet the objective for this task given that an LOP-4 limiting core energy release has occurred. Substitute engineering judgment and justifications where important physical phenomena cannot be numerically quantified.

3.4.4.3. In-Vessel Debris Containment. This task is intended to demonstrate that objective 3 for LOP-4 is met for the spectrum of debris conditions determined by analysis under task 3.4.4.1. In-vessel molten fuel containment is necessary to establish the PCRV as a complete barrier for accidents which progress through failures of LOPs -1 through -3. The consequence limits of LOP-4 without in-vessel debris containment may be technically feasible by providing special design fetures which (1) prevent containment overpressurization due to release of CO_2 and hydrogen or hydrogen combustion, (2) provide for debris coolability inside the containment, and (3) prevent the generation of energetic missiles from the effects of molten fuel penetration through the PCRV base mat. The GCFR program has chosen to include in-vessel debris containment as a PCRV design feature for four reasons:

1. In-vessel molten fuel containment establishes the PCRV and the containment as fully separate and independent barriers to accident progression.

2. The PCRV liner cooling system is provided for other reasons and can contain in-vessel molten fuel.

3. If core melt debris is contained inside the PCRV, the PCRV acts as a shielding structure, which makes containment access, in principle, feasible following containment atmosphere cleanup.

Objective. Demonstrate that objective 3 for LOP-4 is met for containing core melt debris within the PCRV for identifiable accident sequences with a probability greater than 10^{-7}/reactor yr. The task objective is satisfied if the following criteria are met for the entire molten fuel containment mission time.

1. The fuel debris configuration is subcritical.

2. The PCRV liner remains structurally intact to the extent required to prevent direct contact of molten fuel with the PCRV concrete.

3. The PCRV liner cooling system remains functional at the level required to maintain the liner temperature below a (to be determined) lining value.

4. Molten fuel debris spillover into the peripheral PCRV cavities (steam generators and CACS) is prevented, or spillover debris is contained in the peripheral cavities.

5. The molten fuel containment mission time is the time following a core melt accident in which the liner cooling system must refreeze the entire debris mass and maintain it in a frozen state.

Work Packages. The following eight work packages will meet this objective:

1. Initial conditions. On the basis of analyses under task 3.4.4.1, define the limiting initial conditions for in-vessel molten fuel containment required for the analyses in items 4 and 5.

2. Functional requirements. Define the functional requirements to design the in-vessel molten fuel containment structures and the removal systems necessary to meet the task objectives. These functional requirements establish the interface between the in-vessel debris containment safety task and the design task for molten fuel containment.

3. Concept/material selection. Define alternative concepts and material compositions for molten fuel containment. Evaluate identified alternatives and select the reference concept and material composition on the basis of (a) ability to meet the functional requirements of item 2, (b) simplicity, (c) material availability and compatibility, and (d) minimum cost.

4. Thermal/structural analysis. Identify the thermal and structural analyses required to demonstrate that the task objective is met. Adapt and/or develop the analysis methods required to perform the thermal and structural analyses. Perform the analyses necessary to support molten fuel containment design development and to demonstrate that the task objectives are met.

5. PCRV internal component response. Analyze the response of the PCRV internal structures during the molten fuel containment mission time, to the extent necessary, to verify that the functional requirements defined under item 2 are met.

6. Experiment and test requirements. Where warranted by large uncertainties and/or small margins, define experiment and test requirements necessary to substantiate the physical models in the analysis methods and data. Define experiment/gest facility requirements necessary to support the test needs.

7. Design evaluation. Evaluate the design of the molten fuel containment structure, the associated heat removal systems, and

any related structures, such as other PCRV internals, to assure that the design meets the LOP-4 related functional requirements defined in item 2.

8. Probabilistic considerations. Quantify to the extent practical the conditional probability that the PCRV will meet the objective for this task given that an LOP-4 limiting core melt sequence has occurred. Particularly emphasize quantifying the success probability for the liner cooling system to remove the decay heat from the liner boundary during the molten fuel containment mission time. Substitute engineering judgment and justification where important physical phenomena cannot be numerically quantified.

3.4.4.4. Activity Release Attentuation. This task is intended to define the release of fuel and fission product activity to the containment for LOP-4 terminated accident sequences.

This task will thus establish the dominant containment activity source term for LOP-4 terminated sequences such that accident dose analyses can be performed under task 3.6.4.3, item 3, to demonstrate that objective 4 for LOP-4 is met. This task will also identify and evaluate design improvements to more effectively mitigate the activity release to the containment, if such improvements are shown to be necessary by the accident dose analyses performed under task 3.6.4.3, item 3. If tasks 3.4.4.2 and 3.4.4.3 are successful, the principal path for releasing core activity to the containment is through the PCRV pressure relief valves. Other potential leak paths exist through closure and penetration seal leakage, helium buffer system and circulator bearing system leakage, failed heat exchanger tubes, instrument lines, and liner leakage.

Objective. Quantify the dominant containment activity source term for LOP-4 terminated accident sequences, accounting for the attenuation mechanisms which can reasonably be expected to mitigate the release of activity from the PCRV to the containment. Identify, evaluate, and recommend design improvements which may be necessary to meet LOP-4 objective 4. This task objective is met when dose analyses for LOP-4 terminated accidents have been completed under task 3.6.4.3, item 3, to demonstrate that LOP-4 objective 4 is met.

Work Packages. The following eight work packages will meet this objective:

1. Initial conditions. On the basis of analyses under tasks 3.4.4.1 and 3.4.4.2, define the limiting initial conditions, including the condition of the relief valve for the release of helium-borne activity from the PCRV to the containment. Limiting initial conditions may have to be defined and releases analyzed separately for accident sequences with the PCRV pressurized and depressurized, respectively.

2. In-Vessel aerosol attenuation. Evaluate the need for and benefit of analyses of in-vessel aerosol attenuation. If necessary, adapt and/or develop analyses methods to quantify the time-dependent reduction of the helium-borne activity source term inside the PCRV. Perform analyses to quantify the time dependency of the limiting helium-borne activity source term inside the PCRV, as required, to support item 3.

3. Activity release to containment. Quantify the release of helium-borne activity from the PCRV to the containment for the limiting initial conditions defined in item 1, accounting for the condition and response of the PCRV relief valve, for other leakage pathways or mechanisms, and for the depletion of the activity source term inside the PCRV.

4. Functional requirements. Define any specific functional requirements necessary to meet the objectives of this task.

5. Design improvements. If the analyses under task 3.6.4.3., item 3, identify a need for additional mitigation of fuel aerosol and fission product activty from the PCRV to the containment, identify, evaluate, and recommend improved release mitigation features. Improved release mitigation features may include the following:

   a. Special procedures for closing all block valves on the PCRV relief valve trains.

   b. Filtration of PCRV relief valve discharge.

   c. Discharge of PCRV relief valves into storage tanks with a secondary relief valve on the storage tank system.

   d. Elimination of PCRV relief valves.

6. Experiment and test requirements. Where warranted by large uncertainties and/or small margins, define experiment and test requirements necessary to substantiate the physical models in the analysis methods and data. Define experiment/test facility requirements necessary to support the test needs.

7. Design evaluation. Evaluate the design features important to mitigate activity releases to the containment to assure that the design meets the specific functional requirements identified in item 4.

8. Probabilistic considerations. Quantify to the extent practical the conditional probability that the activity release mitigating features of the PCRV will meet the objectives for this task given that an LOP-4 limiting energy release has occurred. Particularly emphasize quantifying the success probability that the PCRV relief valve will remain closed, will reclose, or can otherwise be

isolated to prevent a complete PCRV blowdown into the containment and to provide time for the in-vessel aerosol removal mechanisms to substantially reduce the helium-borne activity source. Substitute engineering judgment and justification where important physical phenomena cannot be numerically quantified.

3.5. CONTAINTMENT INTEGRITY

3.5.1. Introduction

LOP-5 evaluates the containment building as a barrier which can successfully delay and control the release of activity to the environment for accident sequences which fail the first four LOPs. Three basic challenges to the integrity of the containment can result from a failure of the LOP-4 barrier.

1. PCRV failure may generate missiles which may impact the containment and cause it to fail as a leak-tight barrier.

2. Core melt penetration into the PCRV base can generate large quantities of CO_2 and hydrogen, although at relatively slow rates. Containment failure can result from overpressurization due to accumulation of noncondensible gases or from the effects of hydrogen combustion.

3. Core melt penetration through the PCRV base slab will release the debris to the containment floor after several days. Continued penetration through the concrete base mat may eventually result in downward containment failure.

Figure 3-8 details these potential containment failure mechanisms. LOP-5 will establish the extent to which the containment can accommodate these effects and satisfy the task objectives.

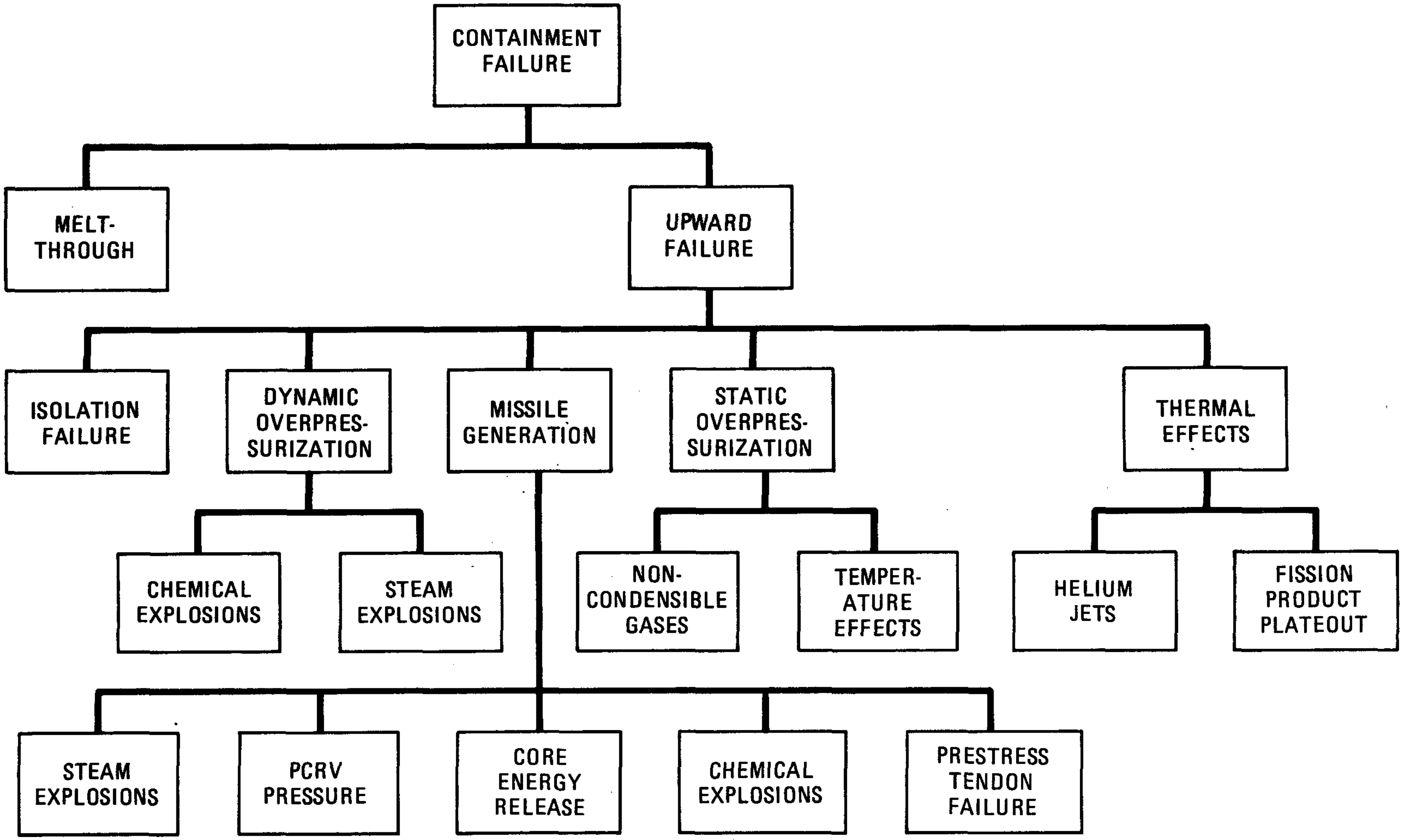

Fig. 3-8. Containment failure mechanisms

3.5.2. Objectives

1. Demonstrate that leak-tight containment integrity is maintained for a minimum period of (to be determined) days for accident sequences progressing through LOP-4 failure with a sufficiently high frequency to require LOP-5 mitigation. Assume that only passive design features are available to delay the containment failure.

2. Demonstrate that, for time periods beyond which passive containment features cannot assure containment integrity, activity release from the containment to the environment can be effectively controlled as necessary to meet the exposure limits of LOP-5.

3.5.3. Scope

This task is intended to quantify the physical phenomena inside the containment building which can be assocaited with the failure of the LOP-4 barrier and which are important to assure that the containment fulfills the objectives for LOP-5. Accident sequences which require LOP-5 mitigation constitute those sequences for which LOP-4 has failed or for which LOP-4 mitigation is not required and which are of relatively higher frequency. Any accident sequences can be exempted from requiring LOP-5 mitigation as long as the combined probability of all exempted accident sequences is less than 10^{-8}/reactor yr.

The containment response will be quantified analytically; however, special purpose experiments will be considered, where necessary, to support the development and/or verification of specific analytical models or to obtain data not otherwise available. Integral experiments to simulate the containment response are not required to meet the objectives of this task. This task will identify containment phenomena to be quantified; develop or adapt the required analysis methods and data and the containment response analysis; define experiment and experiment facility needs; and define LOP-5 specific functional requirements for the containment necessary to meet the LOP-5 objectives.

3.5.4. LOP-5 Task Breakdown

LOP-5 is divided into two tasks. Figure 3-9 shows the task breakdown structure. Task 3.5.4.1 investigates and quantifies the containment response to fuel debris accommodation in the event that in-vessel molten fuel containment is not successful as an LOP-4 barrier. Task 3.5.4.2 investigates and quantifies the mechanisms which could cause upward containment failure. This task derives much of the required input information from task 3.5.4.1, which determines the release rates of flammable and noncondensible gases.

3.5.4.1. Fuel Debris Accommodation. Task 3.4.4.3 considers molten fuel containment inside the PCRV. To meet the LOP-4 objectives, certain functional requirements must be met. Most notably, the PCRV liner cooling system must be restored and/or maintained functional. This task under LOP-5 considers the consequences of a failure to meet the functional requirements and objectives of task 3.4.4.3. Several failure mechanisms for in-vessel molten fuel containment can be identified. All lead to a slow penetration of the molten fuel into the concrete PCRV base with eventual release of a diluted molten fuel-concrete pool onto the containment base mat. In the containment, penetration into and possibly through the containment base mat may occur, which constitutes one containment failure mechanism.

The principal concerns and uncertainties with respect to molten fuel penetration into the PCRV and containment base include the following:

1. The earliest time for PCRV liner failure.

2. The rates of release of steam and noncondensible gases from concrete decomposition.

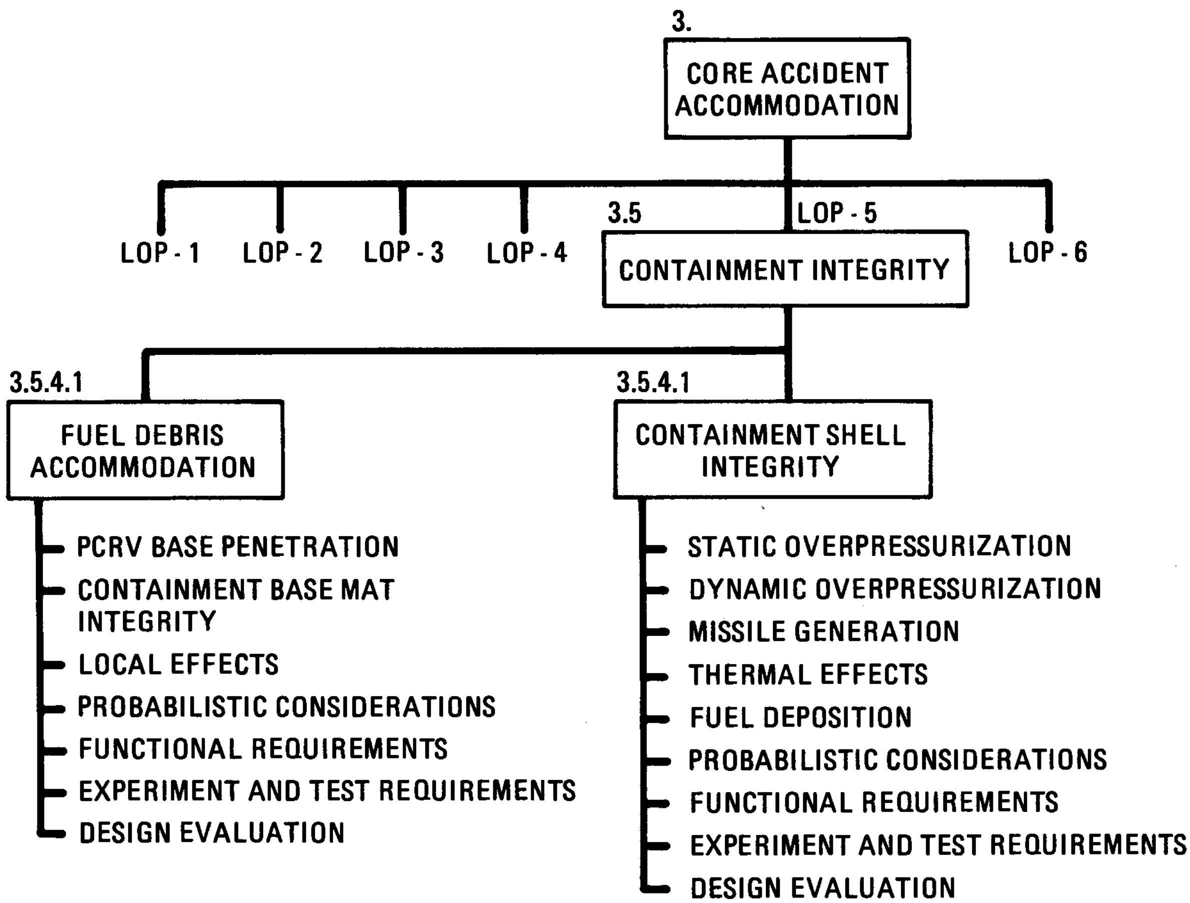

Fig. 3-9. LOP-5 task breakdown

3. The rate of release of hydrogen from steam reacting with molten steel.

4. The rate of penetration downward and sideward into the PCRV base.

5. The time of failure of the first row of axial prestressing tendons as a result of sideward pool growth.

6. The failure mode for the axial prestressing tendons, particularly the potential for missile generation.

7. The response of the central PCRV cavity closure after failure of the axial prestressing tendons if the PCRV is still partly pressurized relative to the containment.

8. The release of additional fission products into the PCRV and containment atmosphere due to effects such as pool sparging.

9. The time and volume of diluted molten fuel/concrete debris pool released onto the containment base mat.

10. The interaction of the released debris with potential water accumulation on the containment floor, the associated steam generation rate, and potential dynamic effects.

11. The uniformity of spreading the debris pool over the containment floor.

12. The penetration rate of the debris pool into the containment base and the potential for permanently refreezing the debris in the containment.

13. The release rate of steam, hydrogen, and CO_2 from debris penetration into the containment base mat.

14. Local effects, such as sump pump wells, which may cause accelerated local penetration of the base mat.

The work packages defined below will investigate and quantify these effects to either provide input data necessary for task 3.5.4.2 or to quantify the time of penetration through the base mat.

Objectives. Objectives 1 and 2 for LOP-5 apply to this task. Passive features already available are believed to be sufficient to meet both objectives. The three specific goals for this task, therefore, are the following:

1. Demonstrate that the time of containment failures due to core debris melting through the containment base mat is longer than the time specified in LOP-5 objective 1.

2. Define the conditions necessary to determine the release of activity and other harmful products from the containment to the environment, if containment base mat melt-through is determined to occur. The environmental consequences (principally population exposures) from this release will be determined in task 3.6.4.3, item 3, to demonstrate that the LOP-5 exposure limits are not exceeded.

3. Provide all the input necessary from this task to quantify the containment shell response in task 3.5.4.2.

Work Packages. The following work packages meet these objectives:

1. PCRV base penetration. Define analyses needed to quantify core melt penetration into PCRV base. Develop and/or adapt the required analysis methods and data. Quantify the core melt penetration into the PCRV base through meltout into the

containment. Quantify all phenomena associated with core melt penetration which are important to meet the LOP-5 objective. In particular, quantify the following:

a. Earliest time of PCRV liner failures.

b. Time-dependent steam/hydrogen release rate.

c. Time-dependent CO_2 release rate.

d. Time of PCRV axial prestress tendon failure.

e. Mode of PCRV axial prestress tendon failure and missile characterization, if applicable.

f. PCRV pressure status at time of tendon failure.

g. Response of central cavity closure plug at time of prestress tendon failure and missile considerations, if applicable.

h. Time of debris pool meltout into containment.

i. Physical condition and quantity of debris drained into containment.

j. Effects associated with upward heat removal during melt penetration.

k. Fission product removal and disposition due to pool sparging and other possible effects.

2. Containment base mat integrity. Define analyses required to quantify containment base mat penetration by diluted fuel/concrete pool. Develop and/or adapt required analysis methods and data. Quantify the debris penetration into the containment base mat. Quantify any effects associated with base mat penetration important to meet the LOP-5 objectives. Particularly quantify the following:

    a. Interaction of draining pool with water on the containment floor.

    b. Melt penetration depth and/or time of base mat melt-through.

    c. Conditions required to permanently refreeze debris pool on containment floor without base mat melt-through.

    d. Time-dependent release of heat, fission products, steam, hydrogen, and CO_2 into containment atmosphere.

    e. Release of fission products and other potentially harmful products from the containment following base mat melt-through, if applicable. These data must be suitable for environmental consequence analysis under task 3.6.4.3, item 3.

3. Local effects. Investigate and quantify the influence of local effects on the analysis in items 1 and 2. Identify ways to prevent such local effects from negating other potentially beneficial design features. Examples of such potential local effects are the following:

    a. Debris drainout along axial prestress tendon channels.
    b. A sump well on the containment floor.

4. Probabilistic considerations. Quantify to the extent practical the conditional probability that LOP-5 objectives are not met due to effects associated with melt penetration, given that an LOP-4 failure of in-vessel molten fuel containment has occurred. Substitute engineering judgment and justification where important physical phenomena cannot be numerically quantified.

5. Functional requirements. Defie any specific functional requirements to be imposed on the design of the PCRV, the containment, or other structures or components necessary to meet the objectives of this task.

6. Experiment and test requirements. Where warranted by large uncertainties and/or small margins, define experiment and test requirements necessary to substantiate the physical models in the analyses methods and data. Define experiment/test facility requirements necessary to support the test needs.

7. Design evaluation. Evaluate the design of the structures and components important to meet the task objective. Verify that the functional requirements specific to this task are met.

3.5.4.2. Containment Shell Integrity. This task investigates the physical response of the containment building to the potential range of conditions which may result from the failure of the LOP-4 barrier (i.e., the PCRV) to terminate an accident sequence. The radiological consequences of LOP-4 and -5 failure will be quantified under task 3.6.4.3, item 3. This task addresses the containment failure mechanisms identified in the right-hand branch of Fig. 3-8, labelled upward failure.

The following principal mechanisms may challenge the containment shell integrity:

1. Dynamic overpressurization (shock wave or impulsive loads) may be associated with (a) the flammable or explosive recombination of hydrogen and oxygen or (b) very rapid steam generation, due molten debris interacting with water on the containment floor.

2. Static overpressurization of the containment may result from (a) the accumulation of noncondensible gases, such as CO_2, or (b) temperature increases associated with hydrogen and oxygen recombining, due to decay heat deposition of air-borne fission products or heat released from the core debris to the containment atmosphere.

3. Containment missiles can potentially result from five sources: (a) core energy release, (b) prestress tendon failure, (c) PCRV pressure, (d) chemical explosions (explosive recombination of hydrogen and oxygen), or (e) steam explosions.

4. Containment integrity may be affected by local effects, mostly thermal, such as helium jet impingement or locally concentrated deposition of fission products. Potential effects associated with the accumulation or concentrated deposition of fuel aerosol particles will also be investigated. Aerosol particles may be discharged into the containment if the PCRV relief valve is failed open.

These effects will be investigated and quantified in the work packages identified below. Passive features alone may not be adequate to meet the objectives of this task. Therefore, this task will identify what optional combination of containment features is adequate to meet the task objectives. Additional features not currently considered may include non-limestone concrete aggregate for the PCRV base and for the containment base mat to minimize the release of noncondensible gases, hydrogen recombiners to prevent the accumulation of a flammable or explosive containment mixture, and/or the filtered venting of the containment.

Objectives. Objectives 1 and 2 for LOP-5 apply to this task. Specifically stated for this task, the objectives are the following:

1. Define the earliest containment failure, resulting from passive mitigation only, for accident sequences requiring LOP-5 mitigation. Demonstrate that this failure time is longer than that specified in LOP-5 objective 1.

2. Evaluate and define the optimum combination of additional containment mitigation features to meet LOP-5 objective 2. Account for the massive concrete confinement building which surrounds the containment with a vented and filtered air space in between.

Work Packages.

1. Static overpressurization. Investigate and quantify the containment atmosphere pressure and temperature response due to all sources resulting from LOP-4 failure, including helium depressurization, release of noncondensible gases, release and/or combustion of flammable gases, and release and condensation of steam and atmosphere heatup due to heat sources and heat sinks in the containment. Demonstrate that, on the basis of passive containment alone, the expected containment failure time due to overpressurization is longer than the time defined in LOP-5 objective 1. Evaluate the need for additional containment features to further mitigate and control the release of air borne activity from the containment for times longer than that provided by passive containment. If such a need is established, evaluate available

alternatives and recommend an optional combination of containment features to meet LOP-5 objective 2.

2. Dynamic overpressurization. Identify all sources which may contribute to a dynamic containment overpressure failure, such as an explosive recombination of hydrogen and oxygen or an energetic steam explosion. Evaluate the need for LOP-5 mitigation against dynamic overpressurization, including available time delays and potential means for preventing such effects or demonstrating that such effects are not sufficiently energetic to cause containment failure. If a need for prevention or mitigation of these effects is established, evalute available alternatives and recommend an optimum combination of containment features to meet the LOP-5 objectives.

3. Missile generation. Investigate the possibility of containment missiles generated from all potential sources associated with failures of the LOP-4 barrier, including core energy release effects, PCRV pressure source effects, prestress tendon failures, explosive recombination of hydrogen and oxygen, and energetic steam explosions. If missile generation is found feasible and if LOP-5 accommodation of such missiles is required because of probabilistic considertions, investigate the potential of such missiles to cause containment failures. If containment failures can occur and if the effects of such a failure need to be mitigated to meet the LOP-5 objectives, evaluate options available for missile effects mitigation and recommend an optional recombination of features to meet the LOP-5 objectives.

4. Thermal effects. Investigate the possibility of thermal effects, such as helium jet impingement or concentrated deposition of decay heat generating products, to cause containment failure. If thermal effects must be mitigated to meet the LOP-5 objective,

evaluate available options and recommend an optimum combination of design features to meet the LOP-5 objectives.

5. Fuel deposition. For LOP-4 failure sequences which result in the release of a significant amount of fuel aerosol to the containment, such as a failed open PCRV relief valve without timely closure of the associated block valve, evaluate potential effects of fuel deposition and accumulation in the containment. Evaluate the need for mitigation of such effects and, if required, evaluate available options and recommend an optimum combination of design features to meet the LOP-5 objectives.

6. Probabilistic considerations. Quantify, to the extent practical, the conditional probability that LOP-5 objectives are not met due to effects which may cause containment shell failure, given that an LOP-4 failure has occurred. Failure of containment isolation should be explicitly included as an LOP-5 failure mode. Support items 1 through 5 with probabilistic considerations, if practical, to determine the need for additional mitigation features. Substitute engineering judgment and justification where important physical phenomena cannot be numerically quantified.

7. Functional requirements. Define any specific functional requirements to be imposed on the design of the containment or on structures and components inside the containment to meet the objectives of this task.

8. Experiment and test requirements. Where warranted by large uncertainties and/or small margins, define experiment and test requirements to substantiate the physical models in the analysis methods and data. Define experiment/test facility requirements to support the test needs.

9. Design evaluation. Evaluate the design of the containment and other structures and components which are important to meet the LOP-5 objectives. Verify that the task-specific functional requirements have been met.

3.6. RADIOLOGICAL ATTENUATION

3.6.1. Introduction

LOP-6 evaluates the radiological attenuation mechanisms both inside and outside the containment which reduce the environmental consequences resulting from a given activity source in the containment. The scope of LOP-6 is somewhat broader than for the other LOPs, because LOP-6 collects in one place all the analyses of radiological consequences required to show that the consequence limits for all LOPs are met, and it includes the radiological consequence analyses required for licensing.

3.6.2. Objectives

The success and completion of the LOP-6 is measured by the following specific objectives:

1. Perform the radiological consequence analyses required for licensing. Demonstrate that the conservative models and assumptions required for licensing analyses meet the applicable dose limits defined in 10CFR (Refs. 3-12, 3-13, 3-14) and in the NRC regulatory guides.

2. Perform the radiological consequence analyses required for the environmental impact report.

3. Perform the radiological consequence analyses to support the objectives of each LOP. Demonstrate that with realistic assumptions and models the public consequence criteria defined in Table 1-3 are met for each LOP.

3.6.3. Scope

LOP-6 is intended to define and complete specific work packages necessary to meet the objectives of this task, including developing specific success criteria, realistic (i.e., best estimate analysis methods, mechanistic consequence analyses, licensing consequence analyses) experiment requirements and plans, functional requirements, and a design review to assure that the design will indeed meet the task objectives.

3.6.4. LOP-6 Task Breakdown

Figure 3-10 shows the LOP-6 task breakdown. Three tasks accomplish the task objectives. Task 3.6.4.1 will analytically quantify the mitigation of activity source terms inside the containment, both for isolated containment conditions (LOP-4 and -5 terminated sequences) and for failed containment conditions if required (LOP-6 terminated conditions). Task 3.6.4.2 will analytically quantify the attenuation of activity releases in the environment. This task will also prepare emergency procedures both onsite and off-site. Task 3.6.4.3 will perform all radiological consequence analyses using the methods developed or adapted under tasks 3.6.4.1 and 3.6.4.2 for all program needs (i.e., operational occurrences, licensing requirements, and severe accidents).

3.6.4.1. Attenuation Inside Containment. This task is intended to establish analysis methods to quantify the attenuation of accident source terms inside the containment and by the confinement building with its filtered interspace discharge. Radiological consequence analyses will be required for routine releases during normal plant operation, for small anticipated accidental releases, and for accident sequences that are terminated by LOPs -4 through -6. This task will develop or adapt realistic analysis methods for all radiological analyses required and the methods required for licensing analyses using the prescribed NRC assumptions and models. These analysis methods will quantify the attenuation of accident source terms inside the containment/confinement for both a normally functioning intact

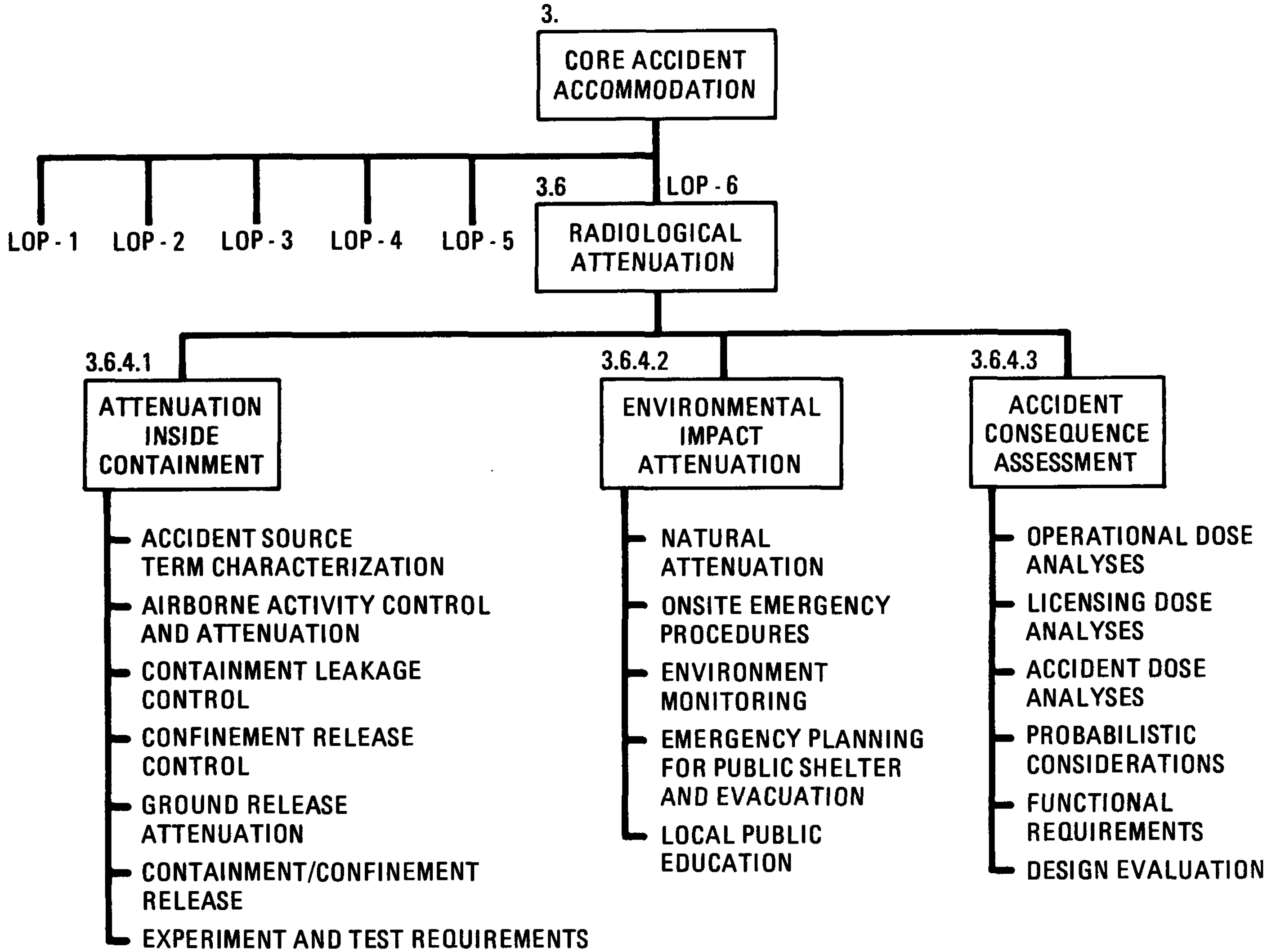

Fig. 3-10. LOP-6 task breakdown

containment and for a degraded containment which may result from LOP-6 failure. The task will consider both engineered attenuation systems: (1) filtered recirculation systems and (2) natural depletion by decay, settling, or plateout.

Objective. This task intended to develop or adapt all analysis methods necessary to quantify the attenuation of accident source terms inside the containment/confinement which are required to meet the objectives of LOP-6.

Work Packages. The following seven work packages will meet this task objective:

1. Accident source term characterization. For each radiological consequence analysis required by this plan, quantify the activity source term in the containment in terms of time of release into the containment, containment condition, quantities of each radionuclide released, and the chemical form of each nuclide released. Base accident source term characterization on the analyses of the specific core response, the PCRV internals and the PCRV under LOP-4, and the specific containment response of the under LOP-5 for each case.

2. Airborne activity control and attenuation. Develop and/or adapt analysis methods to quantify the attenuation of activity source terms defined in item 1 in the containment atmosphere. Consider both active and passive mechanisms for removing radionuclides from the containment atmosphere. Take full advantage of existing analysis methods under the LMFBR, LWR, and HTGR programs.

3. Containment leakage control. Develop and/or adapt analysis methods to realistically quantify the radionuclide leakage rates from the containment. Particularly for activity source terms containing fuel and fission product aerosols, the leakage methods will consider the plugging of small cracks by aerosol deposition.

Utilize methods developed under the LMFBR, LWR, and HTGR programs to the extent practical.

4. Confinement release control. Develop and/or adapt analysis methods to quantify the control or radionuclide releases provided by the confinement building. Consider (a) plateout and settling in the confinement, (b) filtration in the vent exhaust, (c) potential filter bypass releases, and (d) direct leakage through the confinement building structure where the containment/confinement interspace pressure is higher than the atmospheric pressure. Utilize methods developed under the LMFBR, LWR and HTGR programs to the extent possible.

5. Ground release attenuation. Develop and/or adapt analysis methods to realistically quantify the release of radionuclides resulting from a containment base mat melt-through if the analyses under LOP-5, task 3.5.4.1, items 2 through 4, establish containment base mat melt-through as an LOP-5 containment failure mode. Utilize methods developed under the LMFBR, LWR, and HTGR progams to the extent practical.

6. Containment/confinement release. Utilizing the anlaysis developed in items 2 through 5, determine the time-dependent release rate of radionuclides from the confinement boundary to the environment. These release rates will be used in task 3.6.4.3 to assess accident consequence.

7. Experiment and test requirements. Where warranted by large uncertainties and/or small margins, define experiment and test requirements to substantiate the physical models in the analysis methods and data. Define experiment/test facility requirements to support the test needs.

3.6.4.2. Environmental Impact Attenuation. This task is intended to develop or adapt methods and procedures to reduce the environmental and public impact in the event of a significant radioactivity release. Analysis methods will be adapted to quantify radionuclide release impacts on the public and on the environment. This task will also develop requirements for environment monitoring and procedures for the handling of onsite emergencies and emergency plans for public shelter and evaluation. A task will also plan and administer local public education.

Objective. Develop quantitative analysis methods and plans for environmental monitoring, onsite emergency procedures, and emergency procedures for public shelter and evacuation. Develop these emergency plans in sufficient detail for the actual demonstrtion plant site such that the accident consequence assessment can quantify the mitigation of public impact by following these procedures.

Work Packages. The following five work packages meet this objective:

1. Natural attenuation. Develop and/or adapt analytical methods to quantify the natural attenuation of radionuclide releases from the confinement boundary in the environment. Consider attenuation mechanisms for atmospheric and ground releases and model attenuation mechanisms to the extent that they significantly reduce the public and environmental impact from the radionuclide releases defined in task 3.6.4.1, item 6.

2. Onsite emergency procedures. Develop emergency procedures for onsite emergencies from accidents or activity releases not treated in normal plant operating procedures.

3. Environment monitoring. Develop requirements for environmental monitoring to record the exposure levels which may result from accidental releases of activity and to help implement emergency procedures for public shelter and evacuation.

4. Emergency planning for public shelter and evacuation. Develop emergency plans to shelter and evacuate the public in the event of an accidental release of a magnitude which requires such measures to be taken according to Environmental Protection Agency (EPA) guidelines.

5. Local public education. Develop a local public education program in conjunction with local and federal authorities.

3.6.4.3. Accident Consequence Assessment. This task is intended to perform all the radiological consequence analyses required for the GCFR program on the basis of containment/confinement releases defined in task 3.6.4.1, item 6, and to use the provisions for attenuating environmental impact developed in task 3.6.4.2. In this task, the final analysis determines whether the public consequence objectives of each LOP have been met; to this extent, this task supports each of the LOPs.

Objectives. Demonstrate that the public consequences from the release of radionuclides associated with the construction and operation of the GCFR demonstration plant can be expected to be less than the public consequence limits defined for each LOP. Demonstrate that the plant design meets all applicable codes and regulations with respect to radionuclide exposures resulting from normal accidental releases of activity.

This objective is met if the following criteria are satisfied:

1. The plant design meets the intent of all applicable codes and standards for radionuclide exposures with respect to licensing of nuclear power plants. The GCFR Plant Specification for Nuclear

Safety (Ref. 3-2) is the controlling document for meeting this criteria.

2. For all LOP-1 terminated occurrences, the plant design meets the limits of 10CFR50, Appendix I (Ref. 3-13) with realistic assumptions and analysis models.

3. For all LOP-2 terminated sequences, the public exposure is not expected to exceed 50% of the normal annual background exposure.

4. For all LOP-3 terminated sequences, the public exposure is not expected to exceed the annual radiation worker exposure limits defined in 10CFR20 (Ref. 3-14).

5. For all LOP-4 terminated accidents, public exposure is not expected to exceed the dose limits of 10CFR100 (Ref. 3-12), such that neither acute health effects nor significant latent effects are significantly increased.

6. No acute fatalities are expected to result from LOP-5 terminated accidents.

7. The maximum LWR consequences [i.e., the consequences in WASH-1400 (REF. 3-15) at a probability of 10^{-9}/reactor yr] are not exceeded for LOP-6 terminated accident sequences.

Work Packages. The following six work packages will meet the objectives of this task.

1. Operational dose analyses. Perform dose analyses to demonstrate that the GCFR demonstration plant meets all established criteria for expected operational conditions and the dose exposure limits for LOPs -1 and -2. Use realistic analysis models and assumptions for all analyses supporting this task.

2. Licensing dose analyses. Perform dose consequence analyses to demonstrate that the dose consequences defined in the GCFR Plant Specification for Nuclear Safety (Ref. 3-2) are not exceeded. Utilize analysis models and assumptions defined in Ref. 3-2.

3. Accident dose analyses. Perform dose consequence analyses to demonstrate that the exposure limits defined for LOPs -3 through -6 are not exceeded by the containment/confinement releases defined in task 3.6.4.1, item 6. This work package will use realistic assumptions and analysis models.

4. Probabilistic considerations. Quantify, to the extent practical, the expected reliability for each mechanism available to attenuate radiological consequences. Substitute engineering judgment and justification where important physical phenomena cannot be numerically quantified.

5. Functional requirements. Define any specific functional requirements for the containment/confinement and supporting structures, systems, or components to meet the objectives of this task.

6. Design evaluation. Evaluate the design features important to attenuate activity releases to the environment to assure that the design meets the specific functional requirements identified in item 5.

REFERENCES

3-1. "GCFR Plant Specification for Reliability," General Atomic unpublished data, September 9, 1980.

3-2. "GCFR Plant Specification for Nuclear Safety," General Atomic unpublished data, May 30, 1980.

3-3. "GCFR Plant Specification for Availability," General Atomic unpublished data, to be issued.

3-4. "Primary Coolant System Development Plan," General Atomic unpublished data, September 26, 1980.

3-5. "Plant Control System Development Plan," General Atomic unpublished data, to be issued.

3-6. "Reactor Core System Development Plan," General Atomic unpublished data, September 26, 1980.

3-7. "Reactor Vessel System Development Plan," General Atomic unpublished data, August 8, 1980.

3-8. "Reactor Internals System Development Plan," General Atomic unpublished data, July 17, 1980.

3-9. Kelley, A. P., Jr., and T. Taniguchi, "A Reliability Analysis of the Residual Heat Removal Systems for a 300-MW(e) GCFR," General Atomic Report GA-A14653, January 1978.

3-10. "Plant Protection System Development Plan," General Atomic unpublished data, to be issued.

3-11. "Core Auxiliary Cooling System Development Plan," General Atomic unpublished data, September 3, 1980.

3-12. "Reactor Site Criteria," in Code of Federal Regulations, Title 10, Part 100, U.S. Government Printing Office, Washington, D.C., issued April 12, 1962, subsequently amended.

3-13. "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As is Reasonably Achievable for Radioactive Material in Light-Water Cooled Nuclear Power Reactor Effluents," in Code of Federal Regulations, Title 10, Part 50, Appendix I, U.S. Government Printing Office, Washington, D.C., issued May 5, 1975, subsequently amended.

3-14. "Standards for Protection Against Radiation," in Code of Federal Regulations, Title 10, Part 20, U.S. Government Printing Office, Washington, D.C., issued November 17, 1960, subsequently amended.

3-15. "The Reactor Safety Study," U.S. Nuclear Regulatory Commission Report WASH-1400, October 1975 (NUREG75-014).

4. NONCORE ACTIVITY RELEASE ACCOMMODATION

This portion of the GCFR Safety Program Plan is concerned with GCFR ex-core features that could pose hazard to the public and the site personnel. Since the GCFR coolant does not represent a health hazard, principal sources of radioactive materials outside the core include the following:

1. Helium cleanup systems.
    a. Pressure equilization system (PES).
    b. Helium purification system (HPS).
2. Ex-reactor fuel
    a. Fresh fuel.
    b. Spent fuel handling.
    c. Spent fuel storage.
    d. Spent fuel shipping.
3. Radwaste system and facilities.

Except for the radwaste system, these noncore GCFR activity sources represent unique features, particularly since (1) the PES does not exist in other reactor concepts and (2) the vented fuel design is unique to the GCFR.

The hazard from ex-core activity sources is generally accepted not to constitute a dominant risk relative to the core activity. However, these small activity releases might dominate the high frequency risk which is concerned with activity releases that are reasonably certain to occur during the plant lifetime. Therefore, release mechanisms for these ex-core sources of activity should be considered to assure that the risk limit envelope is met over the entire frequency spectrum.

Figure 4-1 shows the top level task breakdown. The detailed objectives and work packages are to be developed at a later date when the design features which will accommodate and contain these activity sources are better defined.

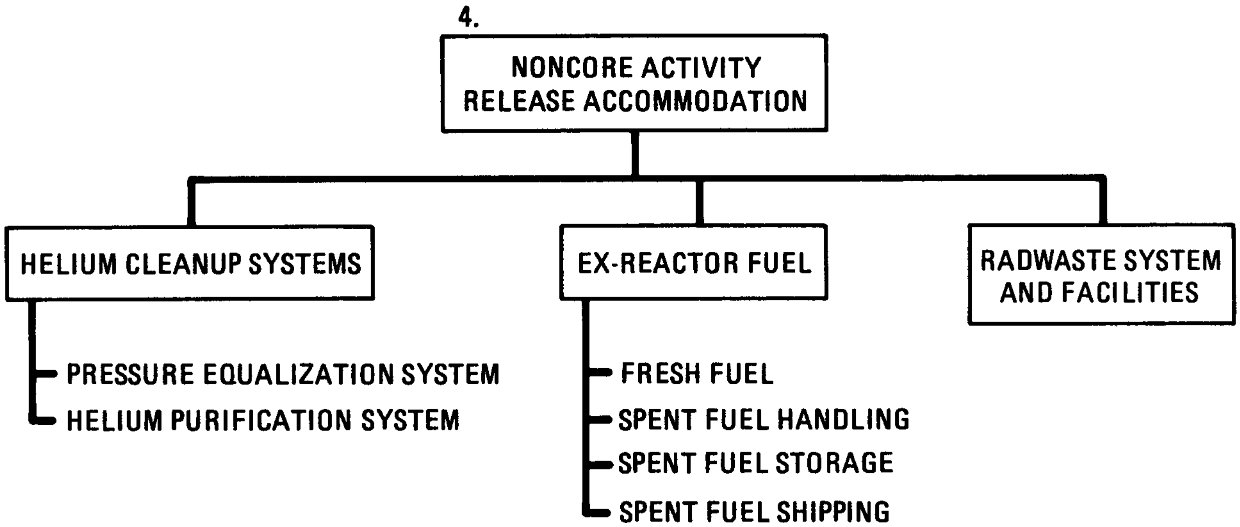

Fig. 4-1. Noncore activity sources task breakdown

5. ACKNOWLEDGMENTS

This Safety Program Plan for the Gas-Cooled Fast Reactor (GCFR) Program was prepared by General Atomic Company and Helium Breeder Associates at the direction and under the support of the U.S. Department of Energy, Contract DE-AT03-76SF71023. In the course of this work, stimulating discussion and valuable direction resulted from meetings with Mr. A. C. Millunzi and Mr. G. Newby of the DOE Nuclear Power Development Division.