ANS Severe Accident Program Overview & Planning Document

by

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ACRONYMS

ABB	Asea-Brown Boveri
ANS	Advanced Neutron Source
ANL	Argonne National Laboratory
ACRS	Advisory Committee on Reactor Safeguards
ALWR	Advanced Light-Water-Reactor
ATWS	Anticipated Transient Without Scram
BNL	Brookhaven National Laboratory
BWR	Boiling-Water-Reactor
BLEVE	Boiling Liquid Explosive Vaporization Event
CPBT	Core-Pressure Boundary Tube
CCFP	Conditional Containment Failure Probability
CE	Combustion Engineering
CET	Containment-Event-Tree
DCH	Direct-Containment-Heating
DSAC	Deterministic Severe Accident Criteria
DOE	Department Of Energy
EPRI	Electric Power Research Institute
EIS	Environmental Impact Statement
FSAR	Final Safety Analysis Report
FCI	Fuel-Coolant-Interaction
HSS	Hydrogen Safety Study
HQ	Head Quarters
JAERI	Japan Atomic Energy Research Institute
LWR	Light-Water-Reactor
LOCA	Loss-Of-Coolant-Accident
MCCI	Molten Core-Concrete-Interaction
NSRR	Nuclear Safety Research Reactor
NRC	Nuclear Regulatory Commission
NPR	New Production Reactor
ORNL	Oak Ridge National Laboratory
PSAR	Preliminary Safety Analysis Report
PDR	Plant Design Requirements
PRA	Probabilistic Risk Assessment
PSER	Preliminary Safety Evaluation Report
RDA	Rapid Depressurization Accident
ROAAM	Risk-Oriented-Accident-Analysis-Methodology
RPI	Rensselaer Polytechnic Institute
SAS	Severe Accident Safety
SAP	Severe Accident Program
SAR	Safety Analysis Report
SP	Safety Panel
SAW	Severe Accident Workshop
SNL	Sandia National Laboratory
UCSB	University of California Santa Barbara

Executive Summary

The Advanced Neutron Source (ANS) severe accident program (SAP) planning document was developed to provide a concise and coherent mechanism for presenting the ANS SAP goals, a strategy satisfying these goals, a succinct summary of the work done to date, and what needs to be done in the future to ensure timely licensability. Guidance was received from various bodies [viz., panel members of the ANS severe accident workshop and safety review committee, Department of Energy (DOE) orders, Nuclear Regulatory Commission (NRC) requirements for ALWRs and advanced reactors, ACRS comments, world-wide trends] were utilized to set up the ANS-relevant SAS goals and strategy. An in-containment worker protection goal was also set up to account for the routine experimenters and other workers within containment. The strategy for achieving the goals is centered upon closing the severe accident issues that have the potential for becoming certification issues when assessed against realistic bounding events. Realistic bounding events are defined as events with an occurrence frequency greater than $10^{-6}/y$. Currently, based upon the level-1 probabilistic risk assessment studies, the realistic bounding events for application for issue closure are flow blockage of fuel element coolant channels, and rapid depressurization-related accidents. Chapter 1 describes ANS SAS goals and implementation strategy.

Chapter 2.1 describes the process for implementing a key element of the ANS SAP strategy. Based upon assessments of eight containment failure modes and mechanisms, it was judged that the areas pertaining to fuel-coolant-interactions and recriticality are the areas with potential for becoming certification issues. Therefore, during the issue closure process, the bulk of the resources will be concentrated upon these two areas. The issue closure process will be dealt with along the lines of the risk-oriented-accident-analysis-methodology (ROAAM) process that has been applied successfully for resolving power reactor severe accident issues.

The balance of areas dealing with potential issues such as containment bypass, static overpressurization, combustion events, direct-containment bypass, molten core-concrete-interactions, and meltthrough of key liners are areas which are not deemed to represent certification issues for ANS. This is especially true for areas concerning direct-containment heating (DCH) and liner meltthrough which are considered inapplicable for ANS conditions, closed, and no further work is planned. Only moderate to little work is envisioned for the (tier-2) areas which, in essence, will be confirmatory in nature. Cold source mechanical failure-related safety concerns are relegated to the future, and will be addressed only if the Level-1 PRA indicates these to be non-negligible severe accident events. Decision points for triggering the conduct of experiments for resolving key uncertainties are presented in conjunction with discussions on treatment of individual containment failure modes and mechanisms.

Modeling, analysis and code module development work done during the issue closure stage will form the backbone for doing the systems analysis work for the preliminary safety analysis report (PSAR) and final safety analysis report (FSAR). A modeling and code development framework centered around existing codes and methodologies has been developed and is presented. Work for supporting Level-2 and -3 probabilistic risk analyses (PRAs) will be based upon sequences identified by the PRA group for development of the containment event tree (CET). Work for PSAR, FSAR, and PRA is described in Chapter 2.2.

The experimental test matrix of relevance to ANS conditions is presented along with a checklist of criteria against which the relevance and need for conduct will be addressed. The experiments primarily investigate separate-effects and strive to resolve key uncertainties related to aspects dealing with closure of issues. Experimental aspects are described in Chapter 3.

Appendix A provides documentation on regulatory positions, ANS, SAW, and SP member comments on a variety of issues identified in Chapter 2.

Appendix B provides review comments received from Prof. Theofanous along with a description of how these have been accommodated.

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1.0 Introduction to ANS Severe Accident Program (SAP), Goals & Strategy

The purpose of this planning document is to provide a coherent and concise mechanism for presenting the ANS Severe Accident Program (SAP) goals, a strategy for satisfying these goals, a succinct summary of the work done to date, and what needs to be done in the future to ensure timely licensability. A document such as this could also be transmitted to invited reviewers for scrutiny before a formal workshop or meeting is held, in order to familiarize participants with work being conducted and planned, and to provide them with sufficient time for considering the potential issues, thereby providing for a more fruitful exchange of ideas and suggestions. It is meant to be as a living document that will be updated periodically as new technical information is derived, certain issues are closed, more work is done on certain others, and keeping pace with the evolving regulatory climate changes.

1.1 Background and Goals of ANS SAP

The Oak Ridge National Laboratory (ORNL) is leading the effort to design the Department of Energy's (DOE's) next generation neutron beam reactor for neutron scattering, materials irradiation and isotope production. The continuing development and improvement of neutron research facilities is essential for the establishment of advanced research and development capabilities in the United States. The Advanced Neutron Source (ANS) is currently planned as a 330 MW research reactor, to be located at the Oak Ridge site, with multipurpose beams and irradiation facilities. The ANS Project is in the advanced conceptual design stage with the reactor expected to become operational in the year 2003.

A defense-in-depth philosophy has been adopted. In response to this commitment, ANS Project management has initiated severe accident analysis and related technology development efforts early-on in the design phase.

The primary objective for the ANS SAP is to provide the means for assessing whether the reactor design meets the ANS Project safety goals (described in Sect. 1.2).

The goals of ANS SAP are as follows _:

- To provide the mechanism and support for designing a robust containment for retention and controlled release of radionuclides in the event of a severe accident.
- To provide the means for ensuring timely licensability via. closure of ANS-relevant severe accident issues, and demonstrating that evolving on- and off-site regulatory requirements have been met by a sufficient safety margin
- To provide best-estimate containment-wide system response and source term analyses for level-2 and -3 Probabilistic Risk Analyses (PRAs) that will be produced.
- To provide the best possible understanding of the ANS under severe accident conditions, and consequently provide insights for the development of strategies and design philosophies or hardware for accident management via training of shift technical advisors (STAs), mitigation, emergency preparedness, and to the extent possible, come up with a design that minimizes the efforts involved with efficient

recovery from severe accidents and consciously-introduced design features that minimize contamination spread to undesirable locations.

It should be noted that DOE Order 5480.23 requires an evaluation of severe accident analysis in the ANS Safety Analysis Report (SAR). The interim guidance accompanying DOE 5480.23 elaborates on the issues to be covered for beyond design basis accidents (BDBAs). This guidance includes evaluation of likelihood, consequences, residual risk and risk reduction against such accidents. The recently completed HFIR SAR includes a treatment of BDBAs in Chapter 15. Furthermore, DOE Order 5480.20 provides guidance that degreed STA training shall include accidents analyzed in the SAR and also the response to and analysis of facility transients and accidents. This forms the regulatory basis for the ANS SAP goal to develop an adequate capability for providing operator guidance for managing and mitigating severe accidents.

The ANS severe accident program incorporates both analysis and supporting experimentation in order to meet the above-mentioned goals. At this stage, the regulatory body that will oversee the ANS has not been determined, despite the fact that, initial discussions with the Nuclear Regulatory Commission (NRC) were held during 11/1993 per directions from DOE. Nevertheless, insights have been gained as to possible requirements from feedback received from two ANS Safety Panel (SP) Meetings, the ANS Environmental Impact Statement (EIS) hearings (1993), and from a recent Severe Accident Workshop (SAW) held during 11/1993. These insights have been used to formulate a succinct set of severe accident safety goals for ANS. The primary focus of the ANS SAP efforts is to meet these goals with margin.

1.2 ANS Severe Accident Safety (SAS) Goals

The overall goal espoused by the ANS Project is that radiological accidents should pose only an insignificant fatality risk to members of the public, and only a small-risk to on-site personnel.

This section will summarize the existing ANS SAS goals and proposed set of goals for the ANS SAP to focus upon. Consistent with current regulatory policy and industry approaches, the goals of this section are directed towards accidents caused by combinations of human and equipment failures, but do not include the possible consequences of beyond-safe-shutdown-earthquake (beyond-SSE) failures. The ANS Project has previously established very stringent seismic standards, leading to the requirement for survival of both reactor core and containment during a SSE of frequency approximately $2 \times 10^{-4}/y$. During the current advanced conceptual design phase and during the planned Title I design phase, an external events PRA will be undertaken and the results will be integrated into the design process with the objective of discovering any significant risk implications of greater-than-SSE earthquakes and evaluating the cost effectiveness of design modifications to enhance the seismic resistance of safety-significant structures, systems, or components. The ANS Project is in the process of finalizing Project level seismic safety goals as specified in the Plant Design Requirements (PDR) document, and the SAP plan will be modified in the future as needed to accommodate those modified goals and plans. Further modification of the SAP will be considered when preliminary Phase I Seismic (External Events) PRA results become available.

The existing set of ANS SAS goals are given in Tables 1.2.1 through 1.2.3, taken from the ANS PDR document¹. These goals have been based upon guidance given by the NRC

Safety Goal Policy Statement² dated 8/21/1986 and a similar draft DOE Nuclear Safety Objectives Statement³ dated 2/1989. Table 1.2.1 lists radiological risk goals for the ANS as a function of distance and type of risk mode (viz., prompt or latent). Table 1.2.2 gives auxiliary risk goals for core-melt ($< 10^{-5}/y$) and for large-release ($< 10^{-6}/y$); these auxiliary goals ensure that the overall goals are met and are essentially geared towards severe accident prevention and avoidance of acute doses from large releases and resulting exposures (viz., > 200 rem) to workers "outside" of containment, respectively. Finally, Table 1.2.3 provides radiological exposure goals as a function of distance and time from the ANS site for the event spectrum category, and have been chosen with conservatism in mind to allow personnel evacuation and dose avoidance on site and to eliminate dependence on evacuation for protecting off-site residents. For this reason, the values given in Table 1.2.3 are much lower than comparable 10 CFR 100 guideline values for site-suitability (viz., 300 rem thyroid dose, and 25 rem whole-body): instead they are determined by (upper or lower, as appropriate) The Environmental Protection Agency (EPA) Protective Action Guide (PAG) values.

The values quoted in Tables 1.2.1-1.2.3 translate into goals for containment integrity along with public and worker protection. For example, the large-release risk goal may be interpreted as being a goal for demonstrating the integrity of the containment and to show that the risk of containment failure is indeed very small. The low risk also translates into the endeavor to avoid prompt fatalities altogether. Table 1.2.2 indicates that there is risk goal for prompt fatalities for on-site workers also, including those who routinely work inside containment. The low exposure goals cited in Table 1.2.3 are an indication that our focus will be to demonstrate enhanced safety by meeting or exceeding standard goals that power reactors abide by for a spectrum of design basis events from normal operation to severe accidents.

At this stage, it is emphasized that the NRC Policy Statement (8/21/1986) clearly stipulates that the statement is about risks and not fatalities. While this tends to skirt the issue by then going on to state what is an acceptable risk of prompt fatalities, it is a clear indication that from the standpoint of perception, prompt fatalities should be avoided to the extent possible, and the related ancillary ANS goal here should be to demonstrate that we meet standard goals with margins appropriate to the phenomenological uncertainties involved.

In Europe, an emerging concept has been the idea of demonstrating deterministically that the containment should withstand challenges⁴. This idea was also promoted by Dr. Alvin Weinberg at the ANS EIS scoping meeting in Oak Ridge, 1993. The promise of demonstrating such a concept sounds impressive, but the concept is being approached with due caution as surprises may occur related to unforeseen loads. The concept of a robust containment is also being promoted by the Advisory Committee on Reactor Safeguards (ACRS), which strives to address uncertainties via conservative design to accommodate the unexpected. German research studies⁵ have focused on this issue of a robust containment for some time now. The recently held ANS SAW panel members also endorse this concept for ANS⁶, in order to accommodate the overall trends being followed for licensing of new nuclear reactor facilities, world-wide. In the recently-released SECY-93-087 document⁷ from the USNRC, we see that the NRC staff are pushing for a deterministic containment integrity goal for power reactors, whereas, the NRC commissioners have yet agreed only to a probabilistic goal, viz., conditional containment failure probability (ccfp) < 0.1 . Nevertheless, for so-called advanced reactors such as CANDU-3 and PRISM [SECY-93-092, and NUREG-13918]), the NRC is requiring^{8,9} a deterministic-type goal that essentially requires maintaining

containment integrity for at least 24 hours following a severe accident, with controlled leakage allowed thereafter. The 24 hour time frame was derived from the best estimate severe accident analyses for power reactor severe accident conditions, and chosen to allow significant aerosol removal from the atmosphere and to allow also for sufficient decay of radionuclides before controlled release to the environment is allowed. We will need to determine a similar estimate for a spectrum of ANS accidents. Fortunately, our past work for gradual pressurization has shown that ANS containment pressure remains below the static design pressure limit of 170 kPa (10 psig). Therefore, barring unforeseen challenges and resolution of such cases (not yet analyzed), it seems very likely that we will be able to demonstrate that the above-mentioned deterministic-type goal can be met.

In SECY-93-087, the NRC staff gives a rationale for accompanying the ccfp goal with a deterministically-based goal. They state (pp. 16-17) that as accident prevention is improved and the core damage frequency (cdf) decreases, ability of containment to withstand events of even lower probability becomes less clear. The ccfp limitations are also evident in uncertainties prevalent in a PRA. Uncertainties in phenomena such as debris coolability and steam explosions have led the staff to conclude that deterministic best-estimate analyses should complement the ccfp containment performance goal. This mindset will be kept in perspective while setting the ANS SAS goals.

In addition to maintaining containment integrity, the ANS needs to consider the significant number of workers and experimenters that will require access to containment, an idea that is foreign to power reactor operation. Several SAW panel members⁶ have indicated that as part of the ANS SAP work, we should develop an acceptable so-called "worker-protection" goal to focus attention on this important issue and demonstrate compliance and closure. This is accommodated via the ANS PDR documents, an on-site worker protection goal (as seen from Table 1.2.2 in conjunction with the large release goal in Table 1.2.1).

The Table 1.2.4 summarizes views related to SAS goals from a wide spectrum of sources. As noted therein, many of the views quoted are directly or indirectly incorporated in the existing ANS SAS goals given in Tables 1.2.1-1.2.3, with the exception of the ccfp containment integrity-related goal, which may be considered a variation of the existing large-release goal. All considered, there is a distinct trend towards focusing on coming up with a goal or goals that have physical attributes and are simple and easy to comprehend. We still have to quantify risk, and a PRA is required and planned for by the ANS Project. Hence, the existing risk goals will be maintained and it will be demonstrated that they are also met by a sufficient margin.

Despite the arguments about coming up with a deterministically-safe containment, it is agreed in the power reactor field that a defensible cutoff frequency has to be employed in conjunction with "visibly" demonstrating to the public that physically-realizable (or imaginable) severe challenges (e.g., missile loads, basemat penetration, bypass) to containment integrity have been accommodated. German safety analysts⁵ (Kreig, et al.), for example, use a cutoff frequency of about 10^{-7} /y, below which they perceive that description by probability numbers is not physically reasonable. A similar cutoff frequency of 10^{-6} /y has been proposed by an ANS SAW panel member (Prof. Theofanous) as given in Table 1.2.4. All else being equal, this proposed cutoff frequency for ANS may be considered analogous to the 10^{-7} /y cutoff frequency for power reactors since the ANS fission product inventory is approximately an order of

magnitude smaller (than for a large power reactor). This is caveated with the recognition that ANS containment will have a large number of experimenters and individuals manning the various user facilities. Event sequence frequencies $> 10^{-6}/y$ are accepted as being within the design basis. The treatment of events below this cutoff frequency should be done to evaluate only the residual risk and to demonstrate that risk (viz., occurrence frequency times consequence) to workers and public are going to be acceptable or bounded by more frequent events.

In addition to the above, new developments¹² (NUREG-1465) in the power reactor field for siting requirements indicate that the best-estimate mechanistically-based source terms should be developed in lieu of the TID-14844 approach¹³. Interestingly, the new values being promoted by USNRC via. work done for NUREG-1150 study¹⁴ and summarized in NUREG-1465, gives the source terms which (for most nuclides) can be significantly greater than those from TID-14844. The prescribed source terms vary significantly between pressurized water reactors (PWRs) and boiling water reactors (BWRs). The aspect dealing with the use of mechanistic approaches is further strengthened from statements in Table 1.2.5 which essentially provides highlights of the NRC staff position on the draft PSER for PRISM. This approach was also highly recommended by the ANS Safety Review Panel (1993).

With the above taken into consideration, the following ANS SAS goals are proposed:

(1) ANS Containment Integrity Goals

1a. Deterministic Goal

The ANS containment shall withstand challenges from all severe accidents of non-negligible likelihood and shall maintain a leak-tight barrier (i.e., leaks below designed leakage rate, and stresses below ASME Service Level C limits) during such events involving gradual pressure buildup. Service Level C limits are applicable only for gradually increasing loads as may result from pool steaming. For dynamic events which generate shock waves and/or missiles the Service Level C limits are not applicable and will be replaced by strain limits (i.e., for surface, membrane and peak strain values) established for steel containment's by the BWR Owner's Group. In this case, severe accidents with non-negligible likelihood are those with a mean occurrence frequency $> 10^{-6}/y$.

1b. Probabilistic Goal

The ANS containment shall satisfy the NRC ALWR requirement for containment integrity, viz., $ccfp < 0.1$. This requirement is conditional upon occurrence of a severe accident. This condition shall be met with frequency weighting for several core damage initiating events which have the capacity to threaten containment integrity, and which have mean occurrence frequencies above $\sim 10^{-7}/y$.

(2) In-Containment Worker Protection Goal

Workers within containment shall not receive exposures greater than 200 rem in the event of any severe accident of non-negligible likelihood.

(3) Ancillary Risk and Exposure Goals

The ANS Level-3 PRA will require risk quantification for a spectrum of severe accidents. Therefore, in addition to the above-mentioned goals, for risk quantification, the ANS Safety Goals given in Tables 1.2.1-1.2.3 shall be met. Any reconsideration's for changing risk goals (e.g., as suggested by NRC in PSER for PRISM to reduce cancer fatality risk to $10^{-7}/y$) will first have to become part of the ANS Project policy and reflected in the ANS PDR. In addition, for site-suitability, exposure goals given in Table 1.2.3 will be met using a combination of mechanistic source terms and conservative meteorology (i.e., Pasquille-F conditions with 1 m/s wind speed for the first 8 h of the transient) at the 95% confidence level.

Besides the above-mentioned SAS goals, we will also meet or exceed ANS-relevant NRC and ACRS prescribed ALWR severe accident requirements for various containment failure modes and mechanisms (e.g., hydrogen concentrations, debris coolability, etc.). These will be addressed individually in Section 2.

1.3 Strategy for Achieving Goals

Having summarized the ANS SAS goals, this section describes the strategy for conducting the overall work for demonstrating we meet or exceed these goals. It should be recognized that meeting these goals will also, in effect, allow us to utilize the derived methodologies for providing input into the areas of accident management, STA training, and for estimation of contamination spread in the event of a severe accident.

As described in the previous section, we have deterministic and probabilistic goals. We have to address deterministic goals by closure of severe accident issues related to containment integrity and in-containment worker protection. In conjunction with this, we need to prove deterministically that containment integrity will be met for a series of containment failure modes and mechanisms where the initiating event is chosen for certain stylized scenarios that meet the cutoff frequency limit of $10^{-6}/y$ (including uncertainties). However, DOE Order 5480.23 also provides guidance that residual risk of the plant should be addressed. In order to accommodate this aspect in the same spirit as we do for realistic bounding scenarios, these lower frequency events will also be evaluated using conservative assumptions to see whether the overall risk would be bounded by the higher frequency events, and to demonstrate that our primary containment integrity goal is met. Details of this process for each containment failure mode and mechanisms are given in section 2.

The addressing of risk goals will require a best-estimate strategy and has been the advice given by SAW panel members, as tabulated in Table 1.2.4. Uncertainties should be addressed via. scaling, experimentation or through expert elicitation process as was done for NUREG-1150. Further details are given in section 2.

The deterministic severe accident criteria (DSAC) approach¹⁵ and the so-called Risk Oriented Accident Analysis Methodology (ROAAM) approach¹⁶ were examined for application to the ANS closure process which has been proposed and used for resolving severe accident issues for power reactors¹⁷. This approach has been successfully used for addressing corium attack on the Mark-I liner, steam explosion-induced containment failure, and DCH issues for the power reactor industry. The work done to close the Mark-I liner attack problem¹⁶ is especially interesting, since it was motivated by the need to resolve controversy from two NRC-sponsored studies which had presented opposite conclusions on failure (and both studies were presented as being conservative estimates).

The overall strategy framework is shown in Fig. 1.3.1. As noted therein, we start by setting up SAS goals and examination of the Level-1 PRA results¹⁷. SAS goals have already been discussed earlier. Table 1.3.2 provides a summary of the results from the Level-1 PRA. Future Level-1 PRA work will include assessment of frequency of cold source mechanical failures to cause core damage. This information will be used when available. Presently, the current Level-1 PRA results were used together with the strategy for selecting initiating events for issue resolution with a cutoff frequency of $> \sim 10^{-6}/y$. Therefore, flow blockage (including manufacturing defects) and rapid depressurization loss-of-coolant accident (LOCA) are the two initiating events to be considered in detail for issue closure purposes. The results of issue closure for individual containment failure modes and mechanisms (e.g., fuel-coolant-interactions, containment bypass, etc.) will necessarily demonstrate meeting of SAS goals for the two initiating events analyzed in a realistic sense (described in section 2). This work is to be done on a separate effects basis and will be closely tied in with methods development, experimentation and front-end thermal-hydraulics.

Work done for issue closure would represent an essential basis for making our safety case to be documented in the preliminary safety analysis report (PSAR) and the final safety analysis report (FSAR). For the PSAR and FSAR, work is planned in which systems analysis will be done wherein ANS system would be analyzed for certain realistic sequences and accidents as shown in Table 2.2.2. Methods development in this area would, in essence, entail integrating several of the models and separate-effects analytical tools developed for the issue closure process into a set of systems-level codes and methods. This framework would complement work done for issue closure and answer the important questions dealing with feedback effects and overall plant response along with core debris / radionuclide distribution. The envisioned modeling framework is described further in section 2. Best-estimate source terms would also be developed for ANS using these tools and utilized for demonstrating site-suitability as is being recommended by the NRC for ALWRs and advanced reactors (e.g., see Table 1.2.5). In addition, additional accidents such as fuel handling accidents with fresh and depleted cores, tritium transport, would also be included if determined to be appropriate. Work for the PSAR and FSAR would form the backbone for making the severe accident safety case for ANS considering non-negligible initiating events (viz., accidents with a frequency $> \sim 10^{-6}/y$) and appropriately chosen plant damage states.

In addition to issue closure and the SAR work which will conclusively demonstrate that we meet of ANS SAS goals for events greater than $\sim 10^{-6}/y$, the best-estimate work for PRA support will have to be done in close cooperation with the PRA group for risk reduction and quantification of lesser frequency events to see whether the overall residual risk has been bounded by the SAR work. That is, one would evaluate lower frequency events to judge whether we have missed a sequence which has a dramatic influence on consequences (viz., cliff-effect) such that the risk from this event is greater than for those considered for the SAR. For PRA work, the strategy calls for use of the basic methods followed for NUREG-1150 where extensive expert elicitation was employed for quantifying uncertainties, coupled with mechanistic severe accident analysis quantification of essential branch points in the containment event tree (CET). Clearly, the iterative approach being followed for issue closure and ANS systems analysis for the FSAR would have already provided the necessary analytical tools for conducting supporting quantification evaluations for the PRA. Therefore, no additional methods or developments are envisioned as being necessary. The PRA group would determine what cutoff frequency will be used for choosing events shown in

Table 1.3.2 with a frequency $< 10^{-6}$ /y. Initial discussions with the PRA group members indicate that we may wish to go no lower than analyzing events with frequencies greater than 10^{-8} /y, and that five or ten sequences within the frequency range from 10^{-6} /y to 10^{-8} /y should be analyzed for overall risk, to demonstrate that the risk has been bounded by events considered in the ANS FSAR.

The overall ANS-system models and best-estimate methodology was developed as a result of satisfying the issue closure, FSAR work and PRA development needs should also provide the necessary tools for addressing severe accident-related accident management, decontamination costing, operator training and emergency planning.

The strategy for closure shall keep abreast of the continual changes to ALWR requirements (SECY-93-087 and, SECY-93-092 for advanced reactors such as PRISM), which, overseas developments and guidance from the ACRS.

Table 1.2.1 Radiological accident risk goals for ANS*

Population	Risk Mode	Comparison basis for goal	Risk to average individual (per year)
Off-site residents within 1.6 km (1 mile) of reservation boundary	Prompt	0.1% of all normal accident risk	4×10^{-7}
Off-site residents within 16 km (10 miles) of the reservation boundary	Latent	0.1% of U.S. average cancer death risk	2×10^{-6}
On-site workers (to be applied to maximally exposed on-site group of workers)	Prompt	10% of average total fatality rate (all causes) of manufacturing and transportation industries in the U.S.	4×10^{-6}
	Latent	10% of latent risk associated with occupational limits for radiation exposure	2.0×10^{-5}

(*) - ANS Plant Design Requirements Document, ORNL/TM-11625.

Table 1.2.2 Auxiliary ANS risk goals*: Core melt and large release

Goal / Category	Statement / Description
Core-Melt Risk Goal	The median probability of severe core damage or meltdown because of internal events shall not exceed $10^{-5}/y$.
Large Release Risk Goal	The median probability of a large release should not exceed $10^{-6}/y$, including both internal and external events. A large release is one that, considering reasonable emergency actions and realistic meteorological conditions, would be capable of causing an off-site exposure in excess of 0.25 Sv (25 rem) at the site boundary and > 2 Sv (200 rem) to on-site workers (for prevention of prompt fatalities).

(*) - ANS Plant Design Requirements Document, ORNL/TM-11625.

Table 1.2.3 Radiation exposure limitations for off-site residents for various accidents*

Event frequency category (per year)	Reservation boundary exposure limit	Basis
Normal down to 0.1	ALARA	Section 6.2.5 of ANS PDR*
Anticipated: 10^{-2} - 10^{-1}	100 mrem	DOE Radiological Control Manual, 10 CFR 20
Unlikely: 10^{-3} - 10^{-2}	500 mrem	10 CFR 20 and DOE "rare year" limit
Unlikely: 10^{-4} - 10^{-3}	1 rem	EPA Lower Protective Action Guideline (PAG)
Extremely Unlikely**: 10^{-5} - 10^{-4}	5 rem	EPA Upper PAG
Extremely Unlikely**: 10^{-6} - 10^{-5}	25 rem	10 CFR 100 Reactor Site Criteria

(*) - ANS Plant Design Requirements Document, ORNL/TM-11625.

(**)- Beyond-design basis for seismically-induced events.

Note: Beyond-design basis events will be evaluated and the total risk associated with them will be assessed against the overall Project safety and risk limitation goals of Table 1.2.1.

Table 1.2.4 Regulatory positions, SAW and SP comments related to ANS SAS Goals

Source	Recommendation / Comment / Suggestion / Requirement
DOE Order 5480.23	Safety analysis report (SAR) should include severe accident considerations
DOE Order 5480.30	Decommissioning and decontamination should be done to minimize radioactive waste and worker exposure
DOE Order 5480.20	Shift technical advisor training shall include accidents considered in the SAR
<u>ANS SAW (11/93) Panel Member Comments</u>	
Frank Chen	Goal of ANS SAP should not be licensability but to come up with design features for containing severe accidents. In comparison to a 747 jetliner, ANS is more like an F-16 jet. The responses of these two will be different to the same external stimuli.
M. Fontana	All previously relevant ALWR severe accident issues should be resolved for ANS or shown why they are irrelevant. ANS severe accident potential should be evaluated without excessive reliance on LWR precedents to assure prevention, mitigation, etc. Most important events for ANS would likely be those leading to early containment failure.
C. Forsberg	Basic licensing approach for ANS should be that containment shall withstand all severe accidents.
T. Fujishiro	Basic licensing approach being followed is appropriate, but ANS should strive to use best-estimate methods coupled with creative application of current regulatory guidance.
R. M. Harrington	Good interaction between thermal-hydraulics and PRA groups should be ensured to assure comprehensive definition of various containment failure states.

S. Hodge	It is important that proposed experimental and analysis plans be conducted, with a focus on dominant threats. Standard research reactor fuels should be considered for ANS.
T. Kress	Set up ancillary goals for ANS (viz., ccfp and dose). For ANS the main issue should be to maintain containment integrity under all severe accident loads. ANS SAP should be used (1) to demonstrate containment robustness, (2) to conduct Level-2,3 PRA best-estimate analyses, and (3) to develop accident management strategies for recovery. ANS SAP has appropriately ruled out DCH issue.
M. Merilo	Licensing approach to satisfy DOE and NRC guidelines is correct. ANS SAP should concentrate on conducting best-estimate work in order to accommodate evolving and changing regulatory climate.
M. Podowski	ANS SAP should develop a clear and consistent set of SAS goals. For best-estimate work uncertainties should be quantified. Some safety issues (such as hydrogen mixing) may be possible to resolve without need for experimentation.
T. Theofanous	Follow the prescribed five step ROAAM approach for formulating the plans of the ANS SAP consisting of: (1) SAS goals, (2) Clear methodology for achieving SAS goals, (3) Technical basis for methodology, (4) Implementation and documentation, and (4) External reviews. Set up a containment integrity goal which assures integrity from realistically-bounding severe accidents, and a unique worker protection goal to protect large number of workers in containment. Use 10 ⁻⁶ /y as a cutoff frequency. Proposed approach is to be used to resolve "issues" and should be a complement to a PRA.
ANS SP member comments	ANS should strive to set up its own SAS goals and not rely too heavily on NRC work for ALWRs. ANS should consider deriving insights from the work conducted for the NPR DSACs. Source terms utilized for ANS should be relevant to ANS and should not be based on TID-14844.

NRC SECY documents	
SECY-93-087	Deterministic Goal: Containment integrity shall be assured for up to 24 hours from severe accident loads, with controlled leakage thereafter. ASME service level C stress limits shall not be exceeded. This goal has not been approved yet by NRC commissioners. Risk Goal: Conditional containment failure probability shall be less than 0.1 under severe accident loads. For advanced reactors, NRC is recommending that the deterministic goal be used in conjunction with ccfp goal. Major shortcomings in the above prescriptions lie in lack of definition of what spectrum of loads and accidents are to be considered. NRC staff state they intend looking at impact of dynamic loads from FCI events, resulting gas generation and combustion events for determining whether ccfp criterion is met.
SECY-91-229	Reactor vendors (ALWRs and new designs) shall assess severe accident design mitigation alternatives (SAMDA) for their designs in addition to safety considerations for severe accidents considered in SECY-93-087. This is required as part of obtaining an operating license.
SECY-93-092	Advanced reactors like CANDU-3 consider design-basis accidents as those with a frequency $> 10^{-7}/y$, and have investigated severe accidents with frequencies as low as $10^{-11}/y$. NRC has questioned validity of such estimates. Staff recognizes that for heavy-water plants, tritium effects will have to be considered. Mechanistic analyses are to be used for deriving information for containment loads and source terms, coupled with sufficient research and development testing data to provide confidence. Events considered for source term evaluations should bound severe accidents and design dependent uncertainties. A set of events shall be chosen deterministically to assess safety margins, to determine and analyze scenarios mechanistically for identifying containment challenges. Overall, NRC staff recommend that new designs will require substantial level of accident mitigation capability for unforeseen events. To account for lack of experience NRC staff strongly recommend using a deterministic containment integrity safety goal rather than a ccfp goal.

Table 1.2.5

HIGHLIGHTS OF THE DRAFT PSER FOR PRISM

Policy Issues

- I. Accident Selection and Evaluation
  - A. General characteristics
    1. Events and sequences selected deterministically, supplemented with PRA insights
    2. Evaluation methodologies and assumptions consistent with LWR practice
    3. External events chosen deterministically, consistent with LWR practice
    4. Extension of the effects of event sequences to more than one module to be determined on a case-by-case basis
  - B. Event Categories
    1. EC-I, Anticipated Operational Occurrences
      - a. Frequency $1.0E-2$ per plant year or greater
      - b. Acceptability based on Appendix I and 40CFR190
    2. EC-II, Design Basis Accidents
      - a. Frequency $1.0E-5$ per plant year or greater for internal events (though DBAs for LWRs typically have frequencies of about $1.0E-4$ per plant year or greater, the staff chose to use $1.0E-5$ per plant year "to increase the confidence that the collective risk of most potential DBAs are (sic) considered in the design and to account for uncertainties, particularly for a preapplication review")
      - b. "Traditional" selection of external events
      - c. Analysis to employ single failure criterion and take no credit for nonsafety equipment
      - d. Conservative analysis methods
    3. EC-III Severe Events.
      - a. Frequency $1.0E-7$ per plant year or greater (based on ensuring that the cumulative risk of several event sequences below $1.0E-6$ per plant year are considered in assessing compliance with the Commission's proposed performance guideline of a frequency of less than $1.0E-6$ per plant year for a large release)
      - b. Rationale for this category
        - (1) Ensure that advanced designs comply with the Commission's Safety Goals and Severe Accident Policies
        - (2) Provide a sufficient test of the capability of the design to allow use of mechanistic source terms for siting determinations and for decisions regarding containment and emergency evacuation plans
        - (3) Ensure that the shift in emphasis concerning defense in depth from accident mitigation to accident prevention, as compared to LWRs, does in fact provide a design with safety at least equivalent to that of current-generation LWRs
      - c. Best-estimate analysis methods are to be used with conservative meteorology
      - d. Includes also "bounding events"
        - (1) Chosen using engineering judgement

- (2) Provides "a sufficient test of conceptual design so that accurate knowledge of the failure modes would not be critical to assessing or understanding its safety"
- (3) Not rigorously quantified in terms of probability - believed to be near $1.0E-7$ per plant year.

II. Source Term

- A. Mechanistic source term allowed
- B. Must ensure that sufficient data exist (through an R&D program or prototype testing) on reactor and fuel performance under EC-II and EC-III conditions to produce adequate confidence in the mechanistic analysis methods used
- C. Must ensure that uncertainty or slight change in assumptions would not cause large change in source term
- D. Dose criteria using mechanistic source term
  - 1. EC-II - 10% of 10CFR100
  - 2. EC-III - 100% of 10CFR100

III. Containment

- A. Design must be adequate such that dose criteria are met in all event categories
- B. Must be capable of limiting leakage for the first 24 hours after onset of core damage to value used in evaluation of event categories; must "control" leakage thereafter.
- C. Must be protected from sabotage and external events at least as well as current generation LWRs.

IV. Offsite Emergency Planning

- A. NRC believes serious fuel damage accidents can take place despite design features and other precautions; therefore emergency planning is necessary
- B. Relaxations in emergency planning will be considered after accident evaluation information is received from applicants; relaxations may be considered in the following areas:
  - 1. Size of the EPZ
  - 2. Frequency of exercises
  - 3. Notification requirements
  - 4. Other areas as appropriate

NRC Conclusions and Comments on PRISM PRA

I. Conclusions

- A. All risk results are much lower than safety goals
- B. Seismic risk is the largest component of total risk

II. Comments

- A. Data lacking to support designer's "optimistic" estimates of system reliability
  - 1. Vessel failure $1.0E-13$ per year (LWRs are $1.0E-7$ per year)
  - 2. Reactor protection system failure probabilities as low as $4.0E-10$ per demand (LWRs are above $1.0E-5$ per demand)
  - 3. Station blackout frequency of $3.0E-5$ per year, considering that the design does not include safety-related diesels
  - 4. Shutdown heat removal failure probability of $3.0E-16$ per demand
- B. External events other than seismic not quantified

- C. Systems interactions not assessed
- D. Best-estimate values used instead of mean values as specified in safety goal policy statement
- E. Role of the operator not explicit-credit taken for operator action without specifying actions or evaluating their feasibility
- F. Greater effort will be needed to achieve adequate completeness at the lower end of the event frequency spectrum in order to substantiate low risk estimates; reliance on existing studies that truncate at relatively high frequencies is insufficient

Containment Challenge Event

- I. Postulated core disruption accident leading to a sodium fire in the containment - no probability of this event computed.
- II. Demonstrated that containment dome was not threatened (BNL performed an analysis for the NRC to confirm GE's conclusion)
- III. Staff said calculations showing acceptable dose consequences rely on incomplete release rate data
  - A. Release rates into containment dome area based on oxide fuel data--extrapolation to metal fuel and sodium pools is uncertain
  - B. Staff said applicants' assumptions regarding release rates appear reasonable

Figure 1.2.1 Framework & strategy for ANS severe accident safety work related to issue closure, PSAR, FSAR and PRA

Notes:

- 1) Cutoff frequencies quoted for flow blockage and depressurization events are median (not mean) values as quoted in ANS Phase I Level-1 PRA and given in Table 1.3.2.
- 2) Methods development and experimentation activities will be driven by needs identified during issue closure and PSAR/FSAR work areas. Information derived will feed back into issue closure and SAR work.

Table 1.2.6 ANS Severe Accident Safety (SAS) Goals & Criteria

<u>Goal / Category</u>	<u>Statement / Description</u>
Deterministic containment integrity goal	The ANS containment shall withstand challenges from all realistically-bounding severe accidents and shall maintain a leak-tight barrier (i.e., leaks below designed leakage rate, and stresses below ASME Service Level C limits for nonexplosive situations) during such events. Realistic severe accidents are defined as those with an occurrence frequency $> 10^{-6}/y$.
Probabilistic containment integrity goal	The ANS containment shall satisfy the NRC ALWR requirement for containment integrity, viz., $ccfp < 0.1$.
In-Containment worker protection goal	Workers within containment shall not receive exposures greater than 200 rem in the event of any realistically-bounding severe accident. Realistic severe accidents are defined as those with an occurrence frequency $> 10^{-6}/y$.
Ancillary Risk & Exposure Goals	The ANS Safety Goals given in Tables 1.2.1-3 will be met and severe accident analysis will attempt to demonstrate adequate margin (to these goals). Any changes to stated goals (such as has been suggested by NRC for PRISM) will be first introduced in the ANS PDR. In addition, for site-suitability, exposure goals given in Table 1.2.3 will be met using a combination of mechanistic source terms and conservative meteorology (i.e., Pasquille F conditions with 1 m/s wind speed for the first 8 h of the transient) at the 95% confidence level.
ACRS Recommendations & ANS-specific criteria for various containment failure modes and mechanisms.	Besides the above-mentioned SAS goals, we will also meet or exceed <u>ANS-relevant</u> NRC, ACRS prescribed ALWR severe accident requirements, and ANS-specific criteria for various containment failure modes and mechanisms (e.g., hydrogen concentrations, debris coolability, etc.)

Table 1.3.1 Key Features of Risk-Oriented-Accident-Analysis-Methodology (ROAAM)

Substantive components of ROAAM	Minimum standards of acceptable quantification are set up. Models must be explicit, physically based and testable. Any experiment proposed must be clearly applicable
Formal aspects of the ROAAM approach	The closure process is accomplished via 5-step approach: 1) Complete documentation or quantification by group doing analytical work to characterize the various issue-related quantification is done for external review. 2) The group makes a presentation to the whole panel after expert comments are invited given "only" for clarification of written material. 3) Expert panel comments are received and responses from authors are prepared. 4) The process continues with a general Workshop where comments and responses are openly discussed, special working group meetings are set up for in-depth discussions, and additional research (viz., improved analysis or experiments) is proposed for issue resolution. 5) Issue closure documentation is prepared (if issue closure is agreed upon based upon sufficient reduction of intangible uncertainty and meeting of safety goals) and sent to expert panel for review and agreement. If not, step 1 through 5 are repeated.

Example definition of probability levels for use in ROAAM

Process likelihood	Process characteristics
1/10	Behavior is within known trends but obtainable only at the edge-of-spectrum parameters
1/100	Behavior cannot be positively excluded, but it is outside the spectrum of reason
1/1000	Behavior is physically unreasonable and violates well-known reality. Its occurrence can be argued positively.

Table 1.3.2 Rank-Ordered List of Leading Core Damage Sequences

Rank	Frequency (per year)*	Description
1	5.00-5	Flow blockage due to material originating downstream of the primary system strainer
2	2.32-5	Fuel damage due to undetected fuel manufacturing defects.
3	1.50-5	Flow blockage due to material originating upstream of the primary system strainer.
4	4.18-7	Rapid Depressurization LOCA near the core and in excess of the mitigation capability.
5	2.00-7	Flow blockage due to material originating outside of the primary system.
6	7.44-8	Degraded secondary cooling followed by scram failure. No flux/flow or high temperature signals; the operator fails to manually scram the reactor.
7	4.42-9	Single rod insertion followed by runback failure; failure to control reactivity.
8	2.40-9	Rapid depressurization LOCA similar to sequence 4. Offsite power supply from 161 kV grid is lost.
9	8.58-10	Beam tube failure followed by failure of the inner control rods to insert. Reflector vessel cooling is assumed to be lost.
10	8.51-10	Slow depressurization LOCA followed by failure to control reactivity. No scram signal is generated and the operator fails to manually scram the reactor.
11	4.26-10	This sequence is similar to sequence 6, except offsite power supply from 161 kV grid is lost.
12	3.69-10	Degraded reflector vessel cooling followed by runback failure; failure to control reactivity. No scram signal is generated and operator fails to manually scram the reactor.
13	3.69-10	Degraded primary flow followed by runback failure; failure to control reactivity.
Total	8.89-5	
* Note: Exponential notation is indicated in abbreviated form; e.g., 5.00-5 = 5.00×10^{-5}		

Fig. 2.2.1 Severe Accident Modeling Framework for Systems Analysis and PRA Work

2.0 ANS SAP Workscope and Plans

Section 2 addresses in greater detail how individual containment failure modes and mechanisms will be addressed following the general ideas from ROAAM for closure of individual issues (if found) to occur and the systems-analysis work for the PSAR and FSAR. Further work to satisfy the best-estimate analysis requirements for the PRA will also be addressed. In addition, for each containment failure mode and mechanism, a summary of work that has already been done for ANS and what the relevant findings were related to uncertainties, and work yet to be done. Simultaneously, a list is also provided of relevant comments and suggestions made by various review bodies.

2.1 Containment failure modes and mechanisms - status of past work and work for closure

Table 2.1.1 categorizes the various potential containment failure modes and mechanisms which will form the basis for the subsections in this chapter. Based on experience from LWRs, this list was chosen from a phenomenological perspective, in order to note whether such an area would lead to "issues" requiring further focus with the ROAAM-type approach. It does not necessarily represent work related to systems analysis, but rather, an effort to screen for potentially significant problems that may require focused attention.

Based upon preliminary assessments, it appears that only two of the afore-mentioned eight containment failure modes and mechanism concern areas constitute potential certification areas (viz., explosive FCIs and recriticality threats) will require in-depth work for closure. The other seven concern areas are not deemed to have the potential to become certification issues because the ANS geometry, physics, decay heat are so different from those of power reactors and the associated potential for violating ANS SAS goals is considerably less. That is not to say that the phenomena behind these "concerns" will be ignored. Confirmatory work to clarify uncertainties and resolution of review comments / concerns will be done. Typically, a modeling approach will need to be developed to estimate the effect of each phenomenon on the transport of severe accident debris; or for others, ANS SAP will provide analytical support to and give guidance-cum-advice to the engineering systems designers. This is done so as to ensure that the mitigation systems act in the manner asserted in order to prevent the phenomena from becoming certification issues.

For many of the issues in this section, we need first to investigate a series of enveloping scenarios based upon the cutoff frequency of $10^{-6}/y$. Results of the Phase I Level-1 PRA (tabulated in Table 1.3.2) indicate that only flow blockages (resulting from debris in the primary coolant or from fuel failure due to manufacturing defects) and rapid depressurization loss-of-coolant-accident (LOCA) initiated accidents fall in this category. The characteristics of flow blockages and manufacturing defect-related damage events are similar. Both could cause channel dryout and rapid melting. For ANS, the term LOCA takes on a different meaning. The entire core and major portions of the RCS are submerged in large volumes of water. Therefore, it is not credible to have situations wherein the RCS is voided. Rather, by rapid depressurization accident (RDA) is meant a break at or near core entrance such that water exiting will lead to a loss of pressure that leads to onset of core damage, calculated to happen at the hot spot. Core behavior during a RDA event is less clear. For example, even though such a RDA

event may cause core damage, it is not certain whether this damage or melting is confined to the hot spot or whether the depressurization event will result in simultaneous damage to significant portions of the core. The type of analysis post-damage is expected to be different in magnitude and scope for the flow blockage-type initiating event compared to that for the RDA event. Therefore, unless otherwise stated, for the issue closure process we will use a localized flow blockage and the rapid depressurization LOCA to represent the non-negligible severe accident events to evaluate consequences as they relate to the ANS SAS goals.

Other initiating events falling in the non-negligible category will be included only if the Level-1 PRA provides the necessary basis for doing so. For example, occurrence frequencies for cold source mechanical failures with the potential for causing core damage have yet to be assessed. If such failures prove to be non-negligible initiating events for core damage, the issue closure process will be upgraded to reflect such developments.

It should be realized that the 0.4×10^{-6} mean occurrence frequency for rapid depressurization LOCA accident (as seen from Table 1.3.2) was obtained using a simple analysis method for piping flaw growth. Fracture mechanics-related work has yet not been done (which may show a considerable reduction in the occurrence frequency). If this happens, the rapid depressurization LOCA may be removed from the non-negligible bounding event category. For the time being, as mentioned above this event has not been analyzed beyond onset of core damage to evaluate extent of core melt progression. The consequences from such an occurrence may be more or less severe compared to core flow blockage events which have an occurrence frequency in the 10^{-5} range.

Other, lower frequency events (with much lower frequency of 10^{-7} /y and less) will be examined only to evaluate effects of uncertainties with regard to risk reduction for BDBAs.

2.1.1 Direct bypass and failure to isolate

Direct bypass and failure to isolate concerns arise from situations wherein fission products from failed fuel (e.g., during severe accidents) find a pathway to the environment and people without filtration through filter banks. Overall, it is judged that this is an open area which requires some work but does not represent a certification issue.

Details regarding past work, regulatory positions, ANS SAW and SP member comments are given in Appendix A.

2.1.1.1 ANS acceptance criterion - Containment bypass and failure to isolate

The proposed ANS acceptance criterion for this issue is given below:

"Direct bypass in conjunction with severe accidents must have a total conditional probability (ccfp) less than 0.1, such that ANS SAS deterministic containment integrity, risk and dose limit goals are met for non-negligible severe accident events."

2.1.1.2 ANS SAP workscope - Containment bypass and failure to isolate

ANS SAP strategy for analysis is shown in Fig. 2.1.1.2. As mentioned in the description of past work (see Appendix A), CSAR work included bypass to the environment directly and did not consider radionuclide transport to volumes occupied by experimenters and users of various facilities. Explosive FCI events which may open up bypass flow paths to the experimenters on the first and second floors will be treated separately in Section 2.1.2. For nonexplosive events, ANS SAP work in this area for closure starts with performing core debris melting, dispersion, transport, and consequence analysis. Various pathways for bypass to experimenters and users within containment and the environment will be evaluated.

Work in this area would provide the necessary phenomenological data needed for completing the containment reliability study, as well as for providing the necessary information (viz., timing and magnitude of source term) to Project designers and engineers concerned with safe evacuation of experimenters and users on various floors. The information would also be used by designers for the determination of actuating signals for valves on lines leading from heat exchangers to the areas outside of containment. This would include examination of users and operations on the operating floor connected with NAAF-1, hot cells, and gamma radiation facility, etc. If this work, in conjunction with the reliability study and the Title 1 evaluation of design alternatives indicates that ANS SAS goals are met, issue closure would result. If significant bypass potential is demonstrated and first cut analysis indicates that containment performance goals have been exceeded, this area shall be focused upon via ROAAM approach to do what is necessary (e.g., design modification, improved analysis, experimentation) to bring it to closure. Work done here would act as a possible trigger point for experimentally determining basic information to quantify possibility and extent of damage propagation and dispersion, fission product release and chemistry if conservative approaches show possibilities of containment bypass.

ANS SAP will work with containment designers and PRA engineers to obtain a very high degree of reliability to ensure, at the very least, that containment isolation is

available on-demand with high-enough reliability ($ccfp < 0.1$), and that experimenters and users are evacuated safely so that this issue does not dominate risk. For experimenters and users reliable separation (from process areas where feasible) and rapid-enough evacuation will be demonstrated such that these individuals might have well been outside of containment.

Overall, this area of concern is not deemed to be a certification issue. ANS SAP support will be provided to guide the Title I evaluation of design alternatives related to containment isolation reliability, rapid evacuation and containment ventilation alternatives. Direct bypass flow paths becoming available during explosive events will be addressed via closure of FCI and detonation issues. Work done for closing this topical area will be included in the PSAR and FSAR.

2.1.2 Energetic Fuel-Coolant Interactions (FCIs) from blast loads, and missiles

Challenges to containment from FCI events present a classical threat which has been analyzed extensively for power reactors as well as for DOE's NPR. Briefly, challenges from FCIs arise from their potential to cause damage from blast loads and missiles. For the ANS, FCIs are roughly characterized as occurring between core debris interacting with water, and between cryogenic deuterium (D₂) interacting with D₂O in the reflector tank. For D₂-D₂O interactions, the relatively hot D₂O can be thought of as being the fuel transferring its energy to cause vaporization of the cryogenic D₂. Note that, the situation of D₂-D₂O interaction dynamics in ANS has no parallel with conventional power or production reactor scenarios. This aspect is covered separately in Section 2.1.9.

This section briefly describes what is felt needs to be done for addressing this important topic. Description of past work, regulatory positions, ANS SAW and SP member comments are described in Appendix A.

2.1.2.3 ANS acceptance criterion - FCI related concerns

Considering the above, the proposed ANS acceptance criterion is as follows:

"For non-negligible severe accident events (viz., core flow blockage and rapid depressurization LOCAs) postulated to result in explosive FCI, resulting loads and consequences shall be demonstrated to meet ANS SAS deterministic containment and ccfp goals."

2.1.2.4 ANS SAP workscope in this area - FCI related concerns

Approaches need to be developed for addressing FCI events involving core debris-water interactions. The approach for attacking this issue for ANS is displayed in Fig. 2.1.2 for the situation where a steam explosion caused by a flow-blockage event under full power conditions will be analyzed. As noted therein, it first entails making mechanistic estimates (which, at this stage of development, are fraught with uncertainties, and are bounded by making suitable assumptions) for core melt propagation, core mass melting and material superheating prior to a steam explosion. The resulting values of core melt mass and temperatures are used to provide an estimate of energetics and missile generation. Energetics calculations are done via thermodynamic estimation of maximum bounding values and also via use of two-dimensional simulation of shock wave generation and propagation utilizing the shock-physics code, CTH, which has been successfully used in the past for analyzing a similar event in HFIR. Thermodynamic estimates for energetics also provide a useful check for CTH calculational results, which can, at times be difficult to analyze. These energetics calculations would be done in a step-wise fashion to isolate the relative contributions from effects of chemical interactions, and also the additional effect (i.e., loadings) from cryogenic-water interactions. The pressure loads would be used to evaluate whether energetic missiles may form, or if key structural boundaries are compromised. Key structural boundaries would include beam tubes or structures which, provide the barrier between the core and experimenters. The breaching of these barriers could cause containment bypass to occur resulting in potential violation of ANS SAS goals.

In any case, if thermodynamic results in conjunction with simple dissipation calculations give rise to a range of loads (viz., missiles, rupture of key structural boundaries) which do not threaten containment or ANS SAS goals, an issue may not exist. CTH calculations would, in essence, be confirming this finding with a mechanistic methodology, to account for multidimensional effects. The findings would be documented, and presented to an expert panel for scrutiny to make sure something important is not missed.

If, on the other hand, bounding evaluations (and CTH results) are found to give rise to the possibility of containment failure or exceedance of the ANS SAS goals, this problem would require formulation in the ROAAM framework for resolution. Resolution may require insightful experiments to look for bounds on propagation, bounds on energetics (including Al-H₂O chemistry), likelihood of triggering of ANS fuel, improved modeling methodology, and making probabilistic estimates to evaluate whether we have been too conservative. Probabilistic estimates would also help in showing, at the very least whether we have met the ccfp-related ANS SAS containment performance goal. Work done at this juncture would act as a possible trigger point for experimentally determining basic information to quantify possibility and extent of aluminum ignition, FCI energetics, and damage propagation, if conservative approaches for melting and transport give results indicating that ANS SAS goals are not met.

2.1.3 Containment overpressurization loads (quasi-static pressure rise)

The challenge in this section concerns potential issues dealing with the containment overpressure beyond a certain limit from quasi-static pressurization such as steaming loads. Pressure rise in this case would occur over several minutes or hours as may be possible if all pool cooling were lost and core debris were dispersed into the reactor pool. This concern area does not deal with dynamic pressure loads such as occurring from blast waves from FCIs or hydrogen detonation events.

Overall, this is not deemed to be a certification issue for ANS based on work done to date, coupled with resolution of associated issues elsewhere. Additional work in this area will deal primarily with confirming the adequacy of MELCOR calculations via comparisons with codes such as CONTAIN and COMMIX.

A summary of past work, regulatory positions, ANS SAW and SP member comments are given in Appendix A.

2.1.3.1 ANS acceptance criterion - Static overpressurization challenges

The ANS acceptance criterion is given below:

"For non-negligible severe accident events (viz., core flow blockage and / or fuel defects) derived from Level-1 PRA insights with frequency $> 10^{-6}/y$, containment overpressurization will be shown to meet ANS SAS deterministic containment integrity and ccfp goals."

2.1.3.2 ANS SAP workscope in this area - Static overpressurization challenges

Based upon work done to date (and summarized in Appendix A) it appears that future work in this area would mainly be towards quantification of some heat transfer-related uncertainties. Thereafter, it must be demonstrated that the ANS containment encompasses sufficient margin to comfortably meet ANS SAS goals. The nature of work for closure is to be confirmatory in nature (viz., to settle uncertainties and questions brought up by various reviewers). This failure mode should not be a certification issue. With this in mind, ANS SAP workscope in this area would consist of specific tasks that include the following:

- Perform additional MELCOR calculations for containment pressurization for cases where additional energy may be generated from a pulsating recriticality event. This evaluation, nevertheless, should be done only if long-term pulsating recriticality potential is found for particular ANS containment locations. The only credible locations are considered to be the ANS RCS piping and heat exchangers, and the subpile room.

- If directed by PRA considerations, additional MELCOR calculations will be performed for the case where the secondary annulus exhaust fans are inoperable to evaluate source terms, with and without primary containment failure. These evaluations should be done in as realistic a manner as possible with specific modeling of failure at penetration, and leakage from potential cracks.

- Conduct a separate MELCOR calculation to note the effect of containment venting, if the need arises for providing time relief for workers to exit from the containment.

- In conjunction with the above tasks, perform a multidimensional modeling of the high-bay volume and secondary annulus volume thermal-hydraulics to confirm or improve the bases for MELCOR predictions of heat transfer and containment pressurization. Effects of mixing of gases should also be considered. For this to happen, an industry code such as COMMIX or GOTHIC must be acquired, made operational and employed with sufficient expertise for making intelligent use of the results. Alternately, a well-known code such as PHOENICS may be used.

If MELCOR calculation results are found to be bounding, the results should be documented, critiqued, and the issue closed. If the results deviate significantly in the wrong direction, and significant pressure buildup close to 14 psig is noted, then, this could act as a potential trigger point for qualification via scaled experimentation. The issue should be then resolved in a focused manner using the ROAAM approach wherein front-end scenarios would also be analyzed further, and a more best-estimate approach would be taken successively, as mentioned in section 1. During the ROAAM exercise, the question on containment robustness brought up by the SAW panel member T. Kress should be addressed also, as a way to potentially increase the ANS containment robustness significantly, at a modest cost increase, if any.

The closure approach is shown in Fig. 2.1.3.1. As seen from Figure 2.1.3.1, additional MELCOR-type evaluations for assessing pressurization from recriticality and the impact of secondary containment failure would be undertaken only if the need arises. Most of the work would be focused on validation of MELCOR predictions, via. comparisons with CONTAIN and multidimensional codes such as COMMIX, GOTHIC or PHOENICS.

Based upon work done so far with MELCOR, and the fact concerning containment bypass and recriticality-induced loads will be resolved separately it is felt that this area does not represent a certification issue for ANS. Only confirmatory work is needed for closure.

2.1.4 Containment loads from combustion processes

This area deals with containment loads derived from combustion events involving combustion of gases such as hydrogen, deuterium, and carbon monoxide. Hydrogen or deuterium sources would be Al-H₂O or Al-D₂O reactions, as well as molten core-concrete-interactions (MCCIs). Carbon monoxide sources would be primarily MCCI events. For ANS, the conscious choice for use of alumina cement concrete eliminates CO generation potential. Hydrogen generation potential is reduced (due to lack of absence of rebar oxidation) but not eliminated during MCCI with alumina concrete since unoxidized aluminum in the core debris can react with overlying water during the MCCI event to produce hydrogen.

Overall, due to the large size of the ANS containment; even if all of the hydrogen that can be generated from MCCI, cold source rupture, or core aluminum oxidation) were to be released and uniformly mixed, the volume concentration would be less than ~ 1 vol/o. The only situations that need to be looked at concerning conditions whereby localized concentrations exceed the detonability limit. Due to this feature, this failure mode is not considered to be one that leads to certification problems.

2.1.4.1 ANS acceptance criterion - Combustion related concerns

The NRC does not currently have a requirement for looking at stratified configurations, but indirectly enforces this via. requiring containment-wide hydrogen control (as mentioned above for ABB's System 80+ design). The ACRS advocates investigation of stratification (or pocketing). World-wide trends are also in the direction professed by the ACRS (viz., to address stratification potential and accommodation via. igniters or recombiners). Therefore, it is recommended that the ANS position should be to follow the ACRS guidelines resulting in the following acceptance criterion:

"ANS containment-wide hydrogen concentrations will be kept below the appropriate detonability limit (tentatively set at the USNRC-specified 10 v/o at room temperature conditions). Detonation potential from stratified configurations will be evaluated with best-estimate methods and accommodated as needed based on risk. Containment integrity will be maintained from combustion-related loads (generated during realistic scenarios leading to missiles or blast waves)."

2.1.4.2 ANS SAP workscope in this area - Combustion related concerns

The ANS SAP work plan in this area is shown in Fig. 2.1.4.1. ANS SAP work will investigate the front-end of the scenarios to evaluate hydrogen release magnitude and timing from the source locations to different containment locations. Scenarios to be evaluated would entail looking at locations such as the subpile room, the region above the reactor pool, the cold source safe room, the letdown tank regions and regions in the high bay where hydrogen is to be released for dilution, viz., design features for adequate high point vents to prevent pocketing as required by 10 CFR 50.44. Examples of high bay regions would be where the line from subpile room vents to the high bay and region in the high bay to which the letdown tank overpressure relief line is vented. This work would be in direct response to needs identified in the ANS CSAR. Release amounts and timing would be evaluated for mixing behavior, if need be using

established codes such as GOTHIC, COMMIX or COBRA-NC. This quantitative information on timing, magnitude, and mixing would be provided to system designers with suggestions for incorporation of engineered prevention and/or mitigative design features. This approach will be employed during Title 1 design to prevent uncontrolled combustion in key locations such as the letdown tank room and two relatively small rooms associated with the cold source cryogenic system. The ANS CSAR also indicates a need for evaluation of the post burn (deflagration) atmosphere to see if a fire sprinkler system is required for those regions that may experience a burn.

If adequate assurance is reached for prevention and mitigation of combustion, closure would result. If not, the impact of pocketing would need to be gauged for missile generation and blast loads. Work in this area would include assessment of the combustion process and structural response in the subpile room, to see whether any bypass pathway may result. If ANS SAS goals for containment integrity are met, closure would result. If not, closure would be achieved via employment of ROAAM.

A potential trigger point for experimentation would occur if we need to evaluate the cold source rupture characteristics combined with cryogenic deuterium-water interactions. Limited, highly focused scaled experiment for mixing behavior of hydrogen released from the reactor pool or via other flow paths in the large ANS containment may be necessary if investigations of past work done for power reactors does not provide such information.

It is our impression that due to the combination of large containment and low deuterium gas production potential this should not represent a certification issue. Analysis work in this area would be focused towards the supporting and qualifying incorporation of any ESFs needed to prevent hydrogen detonation conditions, a concept which the NRC has followed for System 80+. Overall work in this area will be done in conjunction with well-defined conditions for the release of D₂ at various locations, for selected sequences.

2.1.5 Molten Core-Concrete-Interactions (MCCIs) - basemat penetration, structural failure

This area revolves around the concept of core meltdown and relocation to the subpile room floor, where it interacts with the concrete floor. The MCCI interaction can be postulated to occur with or without flooding via. overlying pool of water. The case of MCCI with flooding may occur if flooding is intentionally engineered into the ANS containment system or if light water from the ANS reactor pool or heavy water from the RCS accompanies the relocating core debris. The ANS Project has presently decided to utilize alumina cement as the material of the subpile room basemat, along with strategic flooding for the subpile room in the event of a severe accident. Therefore, MCCI cases without flooding need not be considered any further. The case of MCCI with conventional concrete was considered with and without flooding for the CSAR work. This was done before the Project made a conscious decision to employ alumina cement with flooding capability. For the CSAR work, the case of MCCI without flooding was postulated to occur if intentional flooding is not engineered, or if the breach through which core debris relocates to the subpile room is closed due to metal freezing (an unverified assumption).

For this area, our past work and experience with reviewers suggest that while some uncertainties need clarification and reviewer suggestions need to be followed through, the bulk of future work in this area would be largely confirmatory in nature. Sufficient evidence exists to indicate that the MCCI concern for ANS will be dealt with successfully so as to not become a certification issue.

2.1.5.2 ANS acceptance criterion - MCCI related concerns

Based upon the above, the proposed MCCI issue resolution criterion for ANS is:

"Sufficient cavity floor space will be provided in conjunction with an appropriate basemat floor material and flooding capability to prevent significant ablation and combustible gas generation from non-negligible severe accident events. A long-term safe, stable state should be achieved."

Meeting the above criterion would meet the intent of current requirements for ALWRs in that we would be concentrating not on the secondary related issue of coolability but on the primary issue of MCCI with a focus on maintaining containment integrity.

2.1.5.3 ANS SAP workscope - MCCI related concerns

As mentioned previously, work done to date and responses received so far indicate that the MCCI concern for ANS should be possible to address as part of a confirmatory process. As such, this concern area should not constitute a certification issue. The proposed MCCI concern resolution criterion for ANS should be possible to meet without significant expenditure of resources.

ANS SAP workscope towards demonstrating that the proposed MCCI criterion is met is shown in Fig. 2.1.6. As noted therein, the tasks consist of utilizing the modeling

approach suggested by SAW panel member T. Kress to evaluate whether debris would overheat to ablation temperature of concrete. Modeling evaluations would be done for dispersal from degassing loads (essentially from steam unless debris temperature exceeds the concrete ablation temperature at the interface). Different depths of stratified core debris will be considered with limestone-common sand and alumina cement. Only flooding cases will be considered since the ANS Project has made a conscious decision to provide flooding capability, either via use of techniques used for the ABWR or from a separate water source as done for ABB CE System 80+, which uses the Refueling Water Storage Tank (RWST). The precise mechanism is yet to be worked out by ANS designers. This should help in evaluating whether a cliff-effect exists regarding depth of debris. If it is found that considering whole core debris onto the subpile room during MCCI results in a concrete ablation onset, improved evaluations for obtaining realistic core debris on the subpile room floor would be performed. This should result in reduced core masses being relocated on to the subpile room floor over a long-enough period of time, and may also be accompanied with structural materials which would go towards reducing the debris decay power density. Renewed evaluations would be done at this stage. If debris coolability is demonstrated, such that concrete ablation is prevented with considerable margin, the work would be documented and critiqued for closure. If significant margin to ablation is not possible to show, scaled experiments to provide confirmation and information should be conducted. A trigger point for experimentation would exist here, focused towards evaluating heat transfer characteristics of water-covered stratified core debris over alumina cement (or conventional concrete) to back up heat transfer assumptions and to provide demonstrative proof of having enough justification for closure. It is expected that ROAAM approach if utilized for this closure, would not involve multiple loops. However, at this juncture, if for some reason we do not achieve closure the use of other severe accident mitigation devices (e.g., COMSORS) should be investigated as part of the overall ROAAM approach. Adopting other mitigation designs, such as COMSORS, would, very likely require experimental confirmation.

Work in this area would also involve an investigation of alternate means by which core debris may compromise containment barriers, either by melting some critical liner, or exiting through drains (if they exist in the final design). Work in this area would also have to consider the possibility of a gaseous detonation event in the subpile room to evaluate whether damage can occur to the containment integrity. At this stage of development, a detonation in the subpile room may cause damage to ceilings and walls. It is not clear whether containment bypass will result, but this is judged unlikely. However, ANS Project has taken a proactive stand on these subjects by doing what is necessary to prevent and/or mitigate detonations. The topic of containment bypass is to be effectively addressed via closure in this area, as discussed earlier in Sect. 2.1.1.

2.1.6 Recriticality

This area deals with accidental recriticality of the ANS core debris material during severe accident conditions. The two and three-element baseline ANS core fuel designs utilize about 21 kg of 93% enriched (U-235) uranium, and about 19 kg of 50% enriched uranium which may cause recriticality to occur under accident conditions if reconfigured appropriately. Accidental recriticality may occur under dispersed or possibly, with less likelihood under lumped configurations. Recriticality potential under core melt conditions that cause portions of the core material to relocate also may become important. It is expected that the closing of this will represent another certification-related topic for ANS.

A description of past work, regulatory positions, and reviewer comments are given in Appendix A.

2.1.6.1 ANS acceptance criterion - Recriticality concerns

The proposed ANS SAS criterion associated with recriticality events is given below as follows:

"Recriticality event for the ANS under severe accident conditions will be prevented by design to be well beyond design basis (viz., $< 10^{-7}/y$). Inadvertent recriticality, in any event will be shown to not threaten containment integrity such that ANS SAS goals are met."

The above criterion should allow us to develop the necessary framework for focused efforts.

2.1.6.2 ANS SAP workscope - Recriticality concerns

ANS SAP workscope in this area is depicted in Fig. 2.1.6. The essential task components have been derived based upon reviewer comments and suggestions. As seen in Fig. 2.1.6 the process is initiated by evaluating dispersion potential via realistic bounding calculations for a flow blockage-type event. Evaluations would include dispersion potential from entrainment by high-velocity steam-water flows and explosions, as well as turbulent mixing and transport. For the test matrix of debris configurations in the RCS (where only dispersed situations are to be analyzed), k_{eff} values shall be evaluated using the Monte Carlo methods outlined in ORNL/TM-12302. For affected regions, if subcriticality is assured (i.e., if k_{eff} is below 0.95), results would be documented and the resulting analyses would be put up for review and closure. If k_{eff} is above 0.95, discussions shall be held with containment designers to evaluate whether borated sleeves will be possible to engineer within the constraints of the overall system design. If this is accepted, it shall be demonstrated that k_{eff} is significantly below 0.95, the analysis shall be documented and reviewed prior to closure. If a design fix is not feasible, transient aspects would be evaluated to gage the potential for generation of threats to containment integrity. If containment integrity is not threatened and the ANS SAS goals are met, results shall be documented and reviewed prior to closure. If containment integrity goals are not shown to be met, or if design fixes are not feasible, or if uncertainties in computed debris states are too large, decision making about closure of this issue would be initiated via ROAAM-type approach which could involve improved modeling of the core disruption mechanism supported with carefully

engineered and scaled experimentation. A trigger point would exist here for experimentation to evaluate the dispersion potential from melting ANS fuel plates.

A process similar to that shown above would be employed in conjunction with non-negligible severe accident events leading to core disruption where the potential for debris recriticality shall be evaluated for regions away from the RCS, principally in locations such as the subpile room. An interesting point regarding recriticality in the subpile room would involve addressing the question of what would happen if levitation of the core debris mixture occurs in the form of small particles above the basemat in a water medium. The levitation-related postulate was made by several ANS SAW panel members, and will therefore, be evaluated for its potential physical relevance on recriticality in light and heavy water fluid mixtures over the alumina concrete basemat. If it can be shown that the relatively heavy UAlx particles are not possible to levitate either from degassing or steaming loads, the possibility of recriticality in the subpile room could then be ruled out. In any case, even if a potential for recriticality were determined to be possible, this could be addressed satisfactorily by a number of designed features (e.g., adding neutron poison to basemat material or to the water used for flooding).

2.1.7 Meltthrough of containment barriers (e.g., liner meltthrough)

This area deals with the concerns associated with core melt progression whereby core debris may melt critical structures to cause escape of the fission products out of containment. For ANS, this topical area of relevance to some power reactors is not considered a certification issue.

Overall, with the current knowledge and state of the ANS containment design it is not considered realistic to even postulate core debris contacting the primary containment steel shell. The melt through of the ANS basemat is considered an MCCI-related issue and is closed separately. Therefore, the Mark-I liner-type issue can not exist for ANS. Further ANS designers have consciously provided redundant automatically-actuated isolation valves on the heat exchanger secondary coolant supply and return lines (as detailed in Sect. 2.1.1 on bypass issues). Therefore, the liner meltthrough issue as it applies to some power reactors such as the BWR Mark-I containment is judged to be inapplicable for ANS conditions.

2.1.7.2 ANS SAP workscope - Barrier meltthrough

Based upon the above-mentioned discussion, liner meltthrough issues as they apply for power reactors are not considered relevant for ANS.

Therefore, the liner meltthrough issue for ANS is considered closed, and no further work is planned.

2.1.8 Direct-containment heating (DCH)

Direct containment heating (DCH) is the hypothesis that upon core melting in a PWR the primary system pressure remains at a high value around 17.2 MPa (~ 2500 psi) which is just below the ASME code safety valve set point. Thereafter, overheating with subsequent melting of the reactor vessel lower head causes ejection of molten corium into the reactor cavity. This high pressure ejection presumably causes fine particulate formation clouds which may disperse, rapidly overheat, and overpressurize the containment atmosphere, leading perhaps, to containment failure.

The ANS primary system, on the other hand, has an operating pressure limit of 4 MPa which clearly indicates that the ill-effects of forced ejection (if at all), should be very much lower than that of a PWR. The reactor is also located under water in a deep pool. Based on these aspects, and using engineering judgment DCH concerns as they apply to power reactors do not constitute a certification issue for the ANS.

2.1.8.2 ANS SAP workscope - DCH

As mentioned previously, DCH as thought of as a concern for power reactors does not apply for ANS. The effects of forced ejection into the subpile room may cause rapid transport of radionuclides to other parts of the containment, thereby affecting timing aspects dealing with evacuation of personnel. However, this aspect is considered a containment bypass issue (which is addressed separately via. closure of containment bypass concerns in Sect. 2.1.1), and is not a DCH-related issue as commonly thought of for power reactors.

Therefore, no work is planned for ANS in this area. The issue is considered inapplicable and closed.

2.1.9 Concluding remarks on concern / issue closure

To conclude, it appears that only two of the afore-mentioned nine containment failure modes and mechanism concern areas constitute potential certification areas (viz., explosive FCIs and recriticality threats) will require in-depth work for closure. Based on work done so far coupled with engineering judgment, the other six concern areas are not deemed to have the potential to become certification issues because the ANS geometry, physics, and decay heat are so different from those of power reactors and the associated potential for violating ANS SAS goals is considerably less. This is particularly true for the liner meltdown and DCH areas, which are inapplicable for ANS and are considered closed. That is not to say that the phenomena behind the other four non-certification related "concerns" will be ignored. Confirmatory work to clarify uncertainties and resolution of review comments / concerns will be done. Typically, a modeling approach will need to be developed to estimate the effect of each phenomenon on the transport of severe accident debris; or for others, ANS SAP will provide analytical support to and give guidance-cum-advice to the engineering systems designers. This is done so as to ensure that the mitigation systems act in the manner asserted in order to prevent the phenomena from becoming a certification issue.

2.2 Work for PSAR, FSAR, PRA, accident management, operator training / actions, damage estimation for use in impact assessments

The previous section focused primarily on the specific severe accident issues related to containment failure modes and mechanisms. This section focuses on the overall modeling and analysis workscope of the ANS SAP in terms of providing best-estimate analyses for the PSAR, FSAR, Level-2 & 3 PRA, developing best-estimates of source terms for site-suitability, developing emergency preparedness and accident management / recovery strategies, and in general, for use of the tools to provide DOE with estimates of overall impacts related to severe accidents.

In order to meet the needs of the best-estimate work needed to quantify the Level-2&3 PRA, etc. as mentioned above, an overall system model of the ANS containment and RCS needs to be developed as also suggested by several ANS SAW panel members (viz., T. Kress, S. Hodge, M. Merilo, F. Chen and M. Z. Podowski). Key features of such suggestions are tabulated in Table 2.2.1.

The description in this material will be focused around the overall ANS SAP framework for severe accident safety work highlighted in Figure 1.3.1.

2.2.1 ANS Severe Accident Safety (SAS) methods development, validation and assessment

2.2.1.1 ANS System Model (for nonexplosive situations)

Principal components of the system model for non-explosive events would involve developing a core melt progression module for the ANS RCS and coupled to a system model for the ANS containment. At this stage, two options can be considered. The first option would be to develop along modular lines, with a separate module each for capturing physics of core melt progression and transport through the ANS RCS, and for evaluating fission product chemistry effects. Results from these separate code modules would be linked to an overall containment model for ANS similar to that has been developed with MELCOR. The second option is to consider development of an integral capability similar to that developed for the Savannah River Production (SRP) reactors in which the core melt progression module of MELCOR (developed for power reactors) was replaced with an equivalent module developed for SRP reactors. The resulting code was called MELCOR/SR. Other options involve modification of the well-known SCDAP/ RELAP5⁵¹, SAS-HWR⁵², or TRAC⁵³ (which had also started to be modified to include melt progression capability for the NPR, but has been canceled). As seen in Table 2.2.1, ANS SAW panel member S. Hodge recommends developing a separate core melt progression module linked to the CONTAIN code. It is not clear whether CONTAIN is the appropriate choice for ANS applications over MELCOR. Other ANS SAW panel members M. Merilo as well as T. Kress suggest developing an appropriate best-estimate analysis capability but they do not make strong recommendations, one way or the other (see Appendix A).

2.2.1.2 ANS System Model (for containment transport analyses)

Since we already have tools developed for power reactor containment transport analyses, no significant efforts are considered essential for ANS applications with the exception of accommodating salient aspects of fission product chemistry, tritium generation and transport, and if necessary, core-concrete-interactions. Further, significant pieces of information would also have been developed from focused studies on resolution of various issues related to containment failure modes and mechanisms described in section 2.1. Several of these pieces of information, separate-effects models, etc. would be the building blocks for best-estimate representation of physics of core-melting and relocation within the RCS. Such an integral capability would be in a continuous stage of development as the individual separate-effect models (at various levels of sophistication) are developed to address issues. The module for core-melt progression would take these models, improve upon the physics as necessary based upon time limitations, and integrate them for permitting systems analyses.

2.2.1.3 ANS System Model (for explosive situations)

For explosive severe accident situations, the core-melt progression module would be utilized to provide best-estimates of melting and possible superheating if energetic reactions are shown to be possible in non-negligible severe accident events. The results would be utilized in two ways.

The first step would (accounting for various plant damage states, and for system-wide behavior) compare the resulting estimates of heatup and melting to confirm that the work done for FCI, bypass and recriticality issue closure is indeed bounding.

The second step would involve performing best-estimate evaluations of energetics using tools developed for characterizing mixing, propagation, and fluid-structure interactions. These evaluations of fluid-structure interactions will provide information related to failure of the key components which may change the plant damage state (viz., impact radionuclide pathway definition). Obviously, portions of evaluations that have already been conducted for issue closure (i.e., work done in narrowly focused areas for a given plant damage state) will not be revisited. These energetics calculations would be coupled interactively with the ANS system model (developed with codes such as MELCOR or CONTAIN) for characterizing radionuclide transport for containment response evaluations. Comprehensive feedback phenomena related to nuances associated with various plant damage states will be possible to take into account with this approach. Best-estimate energetics would also provide information on whether cold sources will be damaged. Cold source mechanical rupture and related transient cryogenic D₂-water interaction loads would be evaluated. This analysis would then be coupled with state of the RCS and bypass flow paths to evaluate on a systems basis best-estimate mixing and transport analysis for ascertaining hydrogen stratification potential in key locations of the ANS containment (e.g., letdown tank).

For situations involving (detonable) combustible gases the mixing evaluation tools utilized previously (e.g., GOTHIC or COBRA/NC) for resolving combustion issues would be used here also. If igniters or recombiners have been introduced in ANS containment their effects would be possible to simulate in an integral fashion via. containment transport calculations using MELCOR or CONTAIN. This information, coupled with ESFs will be utilized for characterizing combustible gas explosions / detonations and the associated system behavior. The actual conduct of detonation

calculations (if found necessary for PRA or FSAR work to characterize system-wide behavior and / or various plant damage states not considered during issue closure) would be done with CTH and CET89, the same tools used for the ANS HSS only after characterizing potential for deflagration-to-detonation transition (DDT). This is proposed since the goal here is to first, if possible, prevent and mitigate hydrogen detonations. Therefore, it is considered more useful to mechanistically evaluate the possibility of DDT (instead of merely relying on a given fixed hydrogen concentration limit). It is not definite that a detonation shall result merely because the concentration limit has been exceeded.

2.2.2 Work for PSAR & FSAR

Work done for issue closure would represent an essential basis for making our safety case which is documented in the preliminary safety analysis report (PSAR) and the final safety analysis report (FSAR). For the PSAR and FSAR, work is planned in which systems analysis will be done wherein ANS system would be analyzed for certain non-negligible severe accident accidents as shown in Table 2.2.2. As seen from Table 2.2.2 FSAR work would include investigation of flow blockage and RDA events. Events involving fresh and spent fuel handling will be analyzed for the PSAR and FSAR if they are determined to result in severe fuel damage at a frequency greater than $10^{-6}/y$. The aspect of tritium leakage and transport will be investigated if called upon by system engineers for the ANS containment or the detritiation facility since the requisite tools and analytical expertise would already have been developed by ANS SAP. Localized fuel melting (which does not propagate) would also be included for completeness.

Methods development in this area would, in essence, entail integrating and conducting minor improvements to several of the models and separate-effects analytical tools developed for the issue closure process into a set of systems-level codes and methods described earlier in this section. This framework would complement work done for issue closure and answer the important questions dealing with feedback effects and overall plant response along with core debris / radionuclide distribution. Such integral evaluations are necessary to make sure that feedback effects do not cause results which have not been bounded during the issue-closure process. This capability is necessary to understand how the plant would behave during severe accident conditions. Areas to be evaluated and affected during such sequence calculations include: containment bypass, melthrough of containment liners, MCCI, recriticality, debris coolability in various sections of the RCS and containment, steam explosion potential and range of containment threatening missiles, containment pressurization, degree of system-wide dispersal, design and conduct of mitigative features, accident management strategy development, and operator training. Recriticality-related energy loads would need to be set as user input, stylized control or forcing functions for containment loads evaluation.

Best-estimate source terms would also be developed for ANS using these tools and used for demonstrating site-suitability as is being recommended by the NRC for ALWRs and advanced reactors (e.g., see Table 1.2.5). As mentioned previously, best-estimate source terms developed by the NRC (in NUREG-1465) from NUREG-1150 studies have indicated that source terms can be significantly different for PWRs and BWRs, which also vary with timing of the containment failure. These source terms were found to be quite different and for certain nuclides to be significantly larger than the classical source terms given in TID-14844. Mechanistically-derived source terms derived for ANS

severe accidents which, shall be utilized with conservative site meteorology (as indicated in Table 1.2.5 for the PRISM PSER) to demonstrate site-suitability for ANS.

Overall, the work for the PSAR and FSAR would form the backbone for making the severe accident safety case for ANS, considering credible initiating events (viz., accidents with a frequency $> \sim 10^{-6}/y$) and appropriately chosen plant damage states.

2.2.3 Level-2 PRA best-estimate calculations

As shown in Fig. 1.3.1, work done for issue closure and the FSAR would provide the necessary bases for making a safety case for the initiating events falling in the $> \sim 10^{-6}/y$ category for a given set of plant damage states specified by the ANS Level-1 PRA. For the Level-2 PRA, work is focused to determine that overall risk goals have been met (or enveloped) and to see whether risk-cliffs exist wherein lower frequency events have disproportionately higher consequences. At this stage, it is not clear what the lower cutoff frequency should be, although a tentative value of $\sim 10^{-7}/y$ is being debated for appropriateness. Therefore, it is not certain what precise sequences and plant damage states would need to be analyzed further. Table 2.2.2 lists a few low frequency events which may lead to higher consequences (e.g., due to larger fractions of core melting, increased energetics, availability of containment bypass paths, etc.).

A Level-2 PRA containment event tree (CET) can give rise to several thousand end states. Again, unlike a Level-1 PRA, Level-2 PRA work is largely ($> 90\%$) driven by the need for phenomenological guidance and clarifications for appropriate quantification. Due to the significant resources to perform meaningful mechanistic evaluations for severe accident-related phenomenology, it is also clearly impractical to require mechanistic modeling for every branch point. Significant binning of sequences and paths needs to be performed to make the problem more manageable. Mechanistic modeling and analyses are done only for selected critical and appropriately grouped sequences and branches to provide phenomenological guidance and meaningful split-fractions leading to the appropriate quantification of the CET. Since such a CET has not yet been developed, not much can be done in terms of workscope planning.

In any case, it is anticipated that further methods and model development will be unnecessary to satisfy the Level-2 PRA needs. Methods and models developed for the FSAR would suffice. Detailed workscope definition in this area (including impact of external events) will have to await further developments in the Level-2 PRA.

2.2.4 Dropped fresh and spent fuel accident analysis

It was mentioned earlier that the dropped fuel assembly work would be done as part of the ANS FSAR documentation. Due to its potential importance we provide some additional clarifications which are given in this section. ANS SAW panel member S. Hodge recommended that the dropped fuel accident (in the transfer canal) be analyzed for judging its potential for threatening containment integrity. Fuel handling accidents are of high interest to the ANS Project if for no other reason than the reactor is refueled about 13 times a year. Since the source terms would be relatively low, the main hazard of potential interest is the possibility of in-containment release or (in the event of a prompt criticality) explosion. There is the possibility of having to analyze the dropout of a fresh fuel assembly in the refueling stack which undergoes

prompt criticality [if it falls within the design bases (viz., $> 10^{-6}/y$)]. Both of these tasks would involve conducting a heat transfer evaluation for best-estimates of the fuel mass melt and temperature for fission product releases, and energetics of an FCI. The FCI energetics would be conducted to evaluate (with and without aluminum-water chemical interactions) whether ANS SAS goals for containment performance are met. An integral calculation would need to be done for evaluating source terms and off-site consequences, coupled with an analysis of potential missiles. If this evaluation indicates a potential problem area, it would be cast as an issue as done for the other potential mechanisms for containment failure described in section 2.1, and then do the necessary tasks to meet ANS SAS goals.

2.2.5 Radiation doses to in-containment workers during severe accident conditions

A final area dealing with the fact that workers within containment would involve investigating radiation doses to experimenters within containment say from the high bay region radionuclides. Such work will necessarily be integrated with the overall aspects and practicalities of evacuation schemes. If sufficiently fast evacuation is guaranteed, shielding calculations would not be necessary. If performed, this would involve shielding calculations using either simple techniques (for scoping studies), or with the well-known SCALE code system in conjunction with integral containment response calculations to make sure the ANS SAS goals regarding worker protection are met. SCALE is already operational on ANS SAP workstations. Therefore, no additional code development is envisaged for this purpose. Work in this area would involve developing input models for the ANS containment regions with appropriate codes in SCALE, and exercising them with input functions developed from containment radionuclide transport calculations. ANS SAP would interface and work closely (to make best use of existing capabilities) with other groups in the ANS Project (including the chosen reactor vendor and architect-engineering firm). For situations requiring sophisticated best-estimate modeling for shielding calculations, ANS SAP could provide source terms to relevant shielding experts.

2.2.6 Need for upgrade of MACCS capabilities

One other area specified by the ANS SAW panel member T. Kress involved the aspect of addressing on-site dose evaluation utilizing the well-known MELCOR Accident Consequence Code System (MACCS). In order to address this issue for ANS, the modeling of on-site effects from radionuclide releases will need to be looked into. This task has already begun via communicating with relevant researchers in the field and collecting information on how best to model on-site effects. MACCS modeling for ANS has not included terrain effects as well the effects of transient populations. This is because MACCS does not incorporate such a capability, but can (as has been done for some power reactor analyses) be utilized with care to simulate the essential features of terrain effects. An alternate to MACCS, which also builds upon the classical CRAC-series codes is the CRAC-EZ code developed by PLG Co. CRAC-EZ has models to account for terrain effects. This model, however, also requires data from multiple towers around the site to map out the flow field surrounding the site in question. Alternately, experiments would be required for obtaining such information using tracers. Upon discussions with experts in the field (such as D. Chanin and V. Mubai) without such mapping or information of the flow field, it is doubtful whether CRAC-EZ provides meaningful and defensible information. Investigations will be done to judge the relevance of using CRAC-EZ further to see if it will be necessary for ANS applications to switch from using MACCS. The ANS SAP plan includes an evaluation phase in which the literature is surveyed and experts are consulted (and not just the experts that

developed MACCS or CRAC-EZ). This evaluation and literature search phase will have the objective of finding the experiments and on-site radiation exposure calculations conducted in the past for several of those other nuclear facilities that utilize buildings and are interested in calculating the on-site exposure to people.

Two other elements shall be addressed in this area. The first is the systems-engineering aspect. Every effort shall be made to try and ensure an elevated release, since this has the greatest chance of helping on-site workers. Highest priority in this area shall be devoted to predict and model the radionuclide transport within the primary containment, secondary containment, or support building and subsequent transport to whatever stack or opening might be involved in the release. It is expected that, most of the credible primary and secondary containment failure release points will be into the reactor support building. Again, nuclide transport between secondary containment and the release stack needs attention because of the ~0.33 km distance between the containment and the "remote release" stack. SAP shall provide for the development of radionuclide transport in these areas that lie between the primary containment and the environment.

The desired code-mix for best-estimate integral evaluations is shown in Fig. 2.2.1.

Table 2.1| List of possible severe accident-related containment failure modes and mechanisms for consideration

Group-I Potential certification issues

- Explosive fuel-coolant-interactions (FCIs) that cause blast loads and missiles
- Recriticality-induced loads (affecting FCI and overpressurization areas)

Group-II ALWR Certification Issues Being Addressed for Applicability for ANS

- Direct bypass and failure to isolate
- Containment overpressurization loads (quasi-static pressure rise)
- Containment loads from combustion processes (deflagration and detonation)
- Molten core-concrete-interactions (MCCIs) - basemat penetration and structural failure
- Direct meltthrough of containment barriers (e.g., liner melt-through)
- Direct containment heating (DCH)

- Table 2.2.2 Sequences and accidents analyzed for PSAR & FSAR

<u>Number</u>	<u>Sequence / Event</u>
1	Core flow blockages (large and small) induced core damage
2	Fuel manufacturing defects induced fuel failure and damage
3	Rapid depressurization LOCA or large break LOCA
4	Localized fuel melting*
5	Catastrophic cold source failure*
6	Fuel handling accident with fresh core*
7	Fuel handling accident with spent core*
8	Tritium release and transport
9	Major structural failure of experiments causing fuel failure*

(*) - If Level-1 PRA indicates occurrence frequency is $> 10^{-6}$ /y. If not, these will be addressed as part of Level-2 PRA and summarized in the PSAR & FSAR with regard to risk reduction against BDBAs.

Fig. 2.1.1.2 ANS SAP strategy for closure of containment bypass issue

Fig. 2.1.2.1 ANS SAP strategy for closure of fuel-coolant-interaction issue

Fig. 2.1.3.1 ANS SAP strategy for closure of containment overpressurization issue

Fig. 2.1.4.1 ANS SAP strategy for closure of combustible gas-related issues

Fig. 2.1.5.1 ANS SAP strategy for closure of core-concrete-interaction issues

Fig. 2.1.6.1 ANS SAP strategy for closure of recriticality-related issues

3.0 Supporting Experimentation

This is an area wherein not much planning has been possible due to the need for focusing limited manpower in various analysis areas. Nevertheless, some work has been done in trying to scope out what might be needed to support model development and to shed light into the many phenomenological uncertainties involved for ANS-type conditions. It should be recognized that significant experimentation was planned for the NPR-HWR, and much severe accident experimentation was done for the Savannah River production reactors in the past. However, most of these experiments were conducted with U-Al alloy fuel materials. Therefore, even though insights may be possible to draw from careful examination of past results of experiments for other reactor situations, there is an almost total dearth of experimental data for behavior of silicide fueled reactor materials under severe accident conditions. This lack is tempered with the fact that, ANS reactor has an order of magnitude lower power level, aluminum mass and uranium inventory. Therefore, the same degree of depth in data necessity should not be necessary for ANS compared to the NPR or other power reactors.

This section describes what work has and is being done related to severe accident experimentation, as well as comments and suggestions made by various reviewers.

3.1 Proposed experiment test matrix and selection criteria

A series of separate effects-type experiments were proposed as a first-cut attempt at determining what would be needed in terms of future resources. Due to limited resources, all experiments were chosen to represent separate-effects which endeavor to answer specific questions dealing with severe accident phenomenology. The need for these experiments will have to be tied in with the issue closure process outlined in Section 2 as a means for doing the necessary research and development for satisfying ANS SAS goals. As mentioned previously, the proposed experiments (or experiments close to these) will be triggered based on the outcome of various stages of the issue closure process, thereby forming a robust basis for conducting relevant experiments rather than for improving the state of our knowledge alone.

The experiment test matrix is shown in Table 3.1.1. As noted therein, the experiments range over the phenomenological areas outlined in Section 2 dealing with core response during severe accident conditions. Table 3.1.2 presents the experiment-selection framework and checklist which asks the tough questions related to selection criteria. This framework is to be used in conjunction with issue closure and modeling needs for various experiment groups given below. It should be recognized that the proposed experiments (as of 11/93) represented very limited planning work for various categories and are to be used only as a guide for developing more detailed design and conduct plans. The initial planning work for selected experiments included determination of why and where the proposed experiments were needed and also what form and geometry the experimental rig would assume. Salient information is provided in conjunction with presentations (describing most of the details) made at the past ANS SAW (11/93). Pending further developments, these presentations are included as part of the planning document in tabular format.

3.1.1 Physical Properties Evaluation Experiments

Knowing the physical properties of materials being modeled for analyses presents one of the most fundamental aspects of realistic modeling. For example, steam tables are a must for any thermal-hydraulic evaluation. Similarly, materials need to be characterized for severe accident conditions. For power reactor conditions, MATPRO was developed. However, for U-Al fueled reactors, very little information exists for material properties under severe accident conditions. A study along these lines was conducted for the NPR-HWR effort at Argonne National Laboratory (ANL), a summary of which is given in Table 3.1.1.1. As noted therein, very little is known about the physical properties even for aluminum alloys at temperatures beyond melting conditions, and almost no information is available for properties of U₃Si₂-Al mixtures under severe accident conditions.

A presentation was made to the ANS SAW⁶ (11/93) on this subject which provides preliminary assessment of the needs and approach for tackling this area. It is expected that, this would be one of the first areas for experimental investigation during FY95 and onwards. Further details can be found in Ref. 6.

3.1.2 NSRR Experiments

The Nuclear Safety Research Reactor (NSRR) at JAERI is a research reactor which operates in a pulse-type mode. A collaborative arrangement has been set up between ORNL and JAERI whereby ANS miniplates are subjected to bursts of nuclear energy to evaluate plate response characteristics ranging from cracking and bowing to dispersion and thermal-to-mechanical energy conversion. This effort is supplemented with an ongoing analytical effort to evaluate thermal-hydraulic-cum-structural response. Presentations were made to the ANS SAW⁶ (11/93) on this subject. It is hoped that in the future, similar tests will be done with pre-irradiated miniplate specimens.

3.1.3 FCI Experiments

Two experiments are planned in this area. The first experiment evaluates the propensity and extent of ANS fuel mixtures to undergo ignition using very small and thin coupon samples with masses in the gram range. Coupons are heated via capacitor discharge to cause melting and high temperatures. These are extremely well-characterized experiments wherein boundary conditions, etc. are carefully monitored to provide an in-depth understanding of ignition under explosive FCI conditions, for ANS fuel. A presentation was made on this experiment to the ANS SAW⁶.

The second proposed experiment utilizes a shock-tube geometry wherein a water column impacts a pool of U₃Si₂-Al mixture, mixes with it and leads to energetic explosions. Such a geometry has been used in the past by R. Wright⁵¹ and a variation of this is being used in Europe and Japan also for power reactor FCI investigations⁶⁰. The proposed geometry for ANS endeavors to capture in-core conditions to the extent possible, derive information on triggerability and energetics, and use data in a somewhat empirical fashion for conducting safety studies. The planar front at impact between fuel and water closely resemble conditions in the ANS core which can be considered well-mixed in which fuel plates contact water in a planar fashion. It is also expected that this geometry would provide an upper bound for thermal-to-mechanical energy conversion. This is based upon the availability of only one-degree of freedom for the high-pressure vapor to expand in the experiment (viz., upwards) compared to several directions in the actual core situation. It is recognized that this geometry, nevertheless does not answer the question dealing with fuel injection into water or the other way around. Therefore, this experiment setup may need to be revisited for appropriateness.

Based upon further guidance provided by T. Kress (see Appendix A), the setup of Fig. 3.2.1 was proposed. This setup provides the means for injecting melt-fuel mixtures at various temperatures and back pressures into a pool of liquid. Upon scrutiny, this appears to be the most desirable experiment to investigate triggerability as well as energetics in the ANS core region. This is based on the expectation that core melting would take place under full-power and flow conditions whereupon limited portions of the core might melt and be ejected into the downstream coolant pool. The masses involved in this scenario would be relatively small (i.e., a given number of fuel plates around flow blockages). In any case, a parametric study could be conducted with different melt masses being ejected through slots at given temperatures and a set back pressure (evaluated a priori based on the core melt progression situation being analyzed).

A presentation on the proposed FCI shock tube experiment based upon limited planning was made to the ANS SAW (11/93)⁶.

3.1.4 Melt Dispersion Experiment

Fuel plate melting and dispersion affects recriticality potential and coolability. In the past, experiments to determine the dispersion behavior of uranium-aluminum alloy tubes (for production reactors) have been undertaken to determine this important feature. Results have indicated significant dispersion potential. However, this may be misleading for ANS conditions wherein the fuel, upon melting forms a viscous mass and may tend to display significant resistance to dispersion in a flowing medium.

Key ideas on a proposed experimentation effort along with past work results were presented to the ANS SAW (11/93)⁶.

3.1.5 Debris Coolability Experiments

Debris coolability aspects are central to knowing how core debris relocates and gives off fission products. Thermal-hydraulic behavior of core debris is the central aspect under investigation which further, goes towards providing the bases for realistic modeling and analyses. These experiments are focused towards resolving heat transfer-related uncertainties, esp. at the surface between a stratified molten debris and water. A presentation on this subject including the proposed separate effects experiments was made to the ANS SAW (11/93)⁶.

3.1.6 Debris-Structure Interaction Experiments

An important feature of core-melt progression relates to how core debris interacts with various structures in its path to cause ablation, melting or chemical attack. It is the purpose of proposed selected experiments to evaluate this potential and allow more realistic modeling of core-melt progression and answer related questions of structural failure and relocation to the subpile room. A presentation on this subject including the proposed separate effects experiments was made to the ANS SAW (11/93)⁶.

3.1.7 Fission Product Release & Chemistry Experiments

The key aspect of fission product release from heated ANS fuels under severe accident conditions plays an important role in a variety of circumstances. ORIGEN code evaluations for energy distribution between the various fission products have revealed that about 50% of the decay energy from fission products is associated with the volatile groups. Therefore, it is important to know how these volatile fission products get released as a function of temperature, which further determines whether debris stays cool or it relocates and what the potential for ablation of various

containment boundaries is. A related aspect deals with fission product chemistry which also can play a major role in the release and volatility characteristics of different forms of iodine, which is strongly affected by the forms of compounds formed during melting, pH level of pools, temperature, radiation levels, etc.

Fission product release data from silicide fuel plates with burnups in the 24 % and 66% have been taken at JAERI, albeit in air. At temperatures above aluminum melting, the interface between core debris and water pool would necessarily have to be steam-blanketed. However, the important effect of ambient steam was not studied by JAERI experimenters. Past data taken with U_3O_8 -Al plate samples have demonstrated that the release of the important volatile fission products at temperatures above 660 C can be greater by more than a factor of two in steam versus air. Therefore, JAERI data may be providing values at the lower bound. This shortcoming needs to be clarified. These aspects were discussed and presented at the ANS SAW (11/93)⁶.

3.1.8 Damage Propagation Experiment

An experiment to study damage propagation was proposed as it can significantly affect the degree of core-melting and superheat (and therefore, steam explosion energetics) specifically during core flow-blockage initiated events. However, due to funding constraints no significant work to date has been possible in this relatively important area. Wherever possible, it is planned that existing apparatus being utilized by other ANS Research & Development (R&D) functions would be studied for applicability with some modifications. For example, it has been speculated that the fuel plate stability testing apparatus may be possible to use with some modifications (such as after connection to pressurized accumulators which discharge pressure pulses with the actuation of a fast-opening valve) to simulate possible damage propagation via dynamic loads.

For this area, it is felt that meaningful experiments would have to be driven by need to isolate separate effects, the postulates and analysis of which have not yet been done. Various failure modes and mechanisms will need to be postulated, studied for relevance and the need for supporting experimentation which reduce uncertainties in phenomenological areas.

3.1.9 MCCI Confirmatory Experiment

An experiment to conduct confirmatory work which demonstrates that MCCI can be prevented (viz., elimination of runaway ablation and significant combustible gas generation) was proposed to show the appropriateness of using alumina cement with strategic flooding. If indicated by the planned additional calculations or subsequent reviews, an experiment along these lines would be done for demonstrating the appropriateness of alumina concrete. This experiment would build upon work done for debris-structure interaction tests (which would have been conducted without water). Further details investigating how exactly this proposed experiment (if conducted) would be set up and executed have not been addressed due to funding considerations.

The actual conduct of this experiment would be driven as part of the closure process for MCCI as discussed in Section 2. An experiment along these lines would be done for demonstrating alumina cement as the appropriate choice for ANS. The MCCI confirmatory experiment may also involve investigating the potential for using COMSORS as the appropriate basemat material. Other mitigating devices (floor liners or bed of alumina marbles, etc.) may need some experimental testing if they are adopted.

3.1.10 Containment Thermal-Hydraulics Experiments

This experiment category was proposed as part of the potential need for evaluating the general area of containment thermal hydraulics related to condensation-cum-convection heat transfer in large volumes, stratification of gases and radionuclides, evaluating mixing patterns, etc.

Current best-estimate MELCOR calculations have indicated that ANS containment should have significant margin due to overpressurization failure from gradually-rising pressures. Therefore, containment thermal-hydraulics experiments will be conducted only if the complementary multi-dimensional CFD calculations (to assess MELCOR evaluations) indicate problems with the relatively simplistic MELCOR calculations and identify the need to clarify some important, yet unidentified phenomena.

It is anticipated that experiments in this category, if done, would be conducted using carefully designed bench-scale models of the dome and annulus regions such that visualization of thermal-hydraulic behavior would be achievable to the practical extent. The main drivers for these scaled experiments would be the need for benchmarking against multi-dimensional code calculations, MELCOR or CONTAIN predictions for global behavior, extent of need for backing up the mixing and holdup fractions in the dome and annulus regions, and also the containment-related issue closure process. However, due to constraints and other considerations, it has not been possible to conduct even preliminary planning for these experiments.

Table 3.1.1 Proposed ANS Experiment Test Matrix

Number	Test	Need Addressed / Info. Desired
1	FCI shock Tube test	Basic data on triggerability, thermal-to-mech. conversion (with and without ignition)
2	FCI -Small scale testing	Look at onset of ignition, dispersion
3	Physical Properties	Additional physical properties for T/H severe accident calcs.
4	Transient Fuel Perf.	Design basis, and thresholds for disruption, energetics
5	Debris-Structure Interac.	Characterize fundamentals of interactions between debris and structures
6	Flow Blockage / Damage Prop.	Evaluate damage initiation and propagation
7	FP Release / Chemistry	Evaluate need and information related to FP release and chem. of silicide fuels
8	Debris coolability	Evaluate what constitutes coolability (viz., T/H of cooling)
9	Fuel Melting/Dispersion	Determine how the fuel comes apart and disperses in flowing medium essentially in intact mode
10	MCCI Mitigation Test	Similar to debris-structure interaction test, but with a larger scale apparatus and including water
11	Containment T/H	Evaluate natural conv. heat transfer, transport of FPs, flow patterns, holdup

Table 3.1.2 ANS Severe Accident Experiment Planning Guideline Question Checklist

The following guidelines checklist is to be used for conducting the detailed planning for each experiment. The list of considerations is by no means complete. Therefore, more may be added at a later stage. We will be doing so on an ongoing basis. For the purposes of this planning work, each experiment will be kept separate even though in reality, many of the questions, if answered once may not be necessary for address again.

1. In general, why is this experiment needed? What impact would it have on the overall and specific safety aspects (design, and risk) of the plant?
2. What would happen if the data from this experiment were not available?
3. Are we aware of other places where similar experiments have been conducted, even if the materials were different? What was the motivation for the others to conduct this experiment? How much could we learn from the stuff already conducted? Can we analyze this data and scale to our needs with reasonable assurance?
4. What would be the best place in the country and out of it for conducting these tests from the standpoints of:
  - Quality of personnell
  - Past experience
  - Available test rigs which may be used with little or no modification
5. Could this experiment be done at UT (i.e., off-site) in a high-quality fashion? If so, who would it be with?
6. If the test would need to be done at ORNL in general, would it be feasible to conduct the same test in ETD buildings at Y-12? If not, what is the best place at ORNL?
7. Does the experiment involve working with radioactive materials?
8. Personnell needed for conducting the experiment.
  - Scientific staff members (planning, design, qualification)
  - Technicians (for putting the apparatus together, and maybe, operating apparatus)
  - Arts and Craft (for manufacturing)
  - Rad worker training necessary?
9. Experiment work area
  - Does it already have radiation work permit?
  - What special OSHA/NEPA requirements will be needed for safety documentation?
  - Is the space available adequate?
10. Experimentation Aspects

A) Hardware

- Develop a conceptual picture of what the experiment setup would look like
- Evaluate the maximum capacity of materials (to be tested for) to be handled
- Look at long range (would the test rig be useful for spinoffs)
- Come up with a detailed set of necessary instrumentation
- Come up with a detailed set of data acquisition systems
- Identify those hardware items which may be available for tests already being conducted for ANS
- Identify manufacturers from whom the slated hardware will be ordered
- Evaluate the time required for ordering and actually receiving the components
- Evaluate what mode will be used for ordering (petty cash, etc.)

B) Shakedown

- Evaluate what needs to be done to conduct a thorough shakedown to assure test rig will provide needed data

C) Data Acquisition

- Evaluate and document exactly what is to be measured or acquired
- Document the exact procedure by which each data point is to be acquired
- Document what if any analysis shall be done to ensure that the next data point will be taken (i.e., from previous data already acquired)
- In what medium will the data be stored and analyzed

D) Time & Cost

- Estimate the time required (manhours, manweeks, mmonths) per data point for:

planning,
safety documentation preparation,
obtaining necessary permits,
design,
ordering and acquiring necessary hardware and software
manufacturing components,
test rig assembly,
test rig shakedown
data acquisition
ongoing data analysis during testing
data analysis after testing
documentation (archival aspects of data acquisition, & report preparation)
disposal (waste, decontamination if necessary)

- Would we be using consultants ? If so, who, for how long, and daily rates ?
- Develop a bar chart showing when specific features of the experiment will be completed
- Estimate the dollar costs for each of the following in conjunction with time aspects:
  - manpower (from different divisions, and levels such as technicians, engr, etc)
  - estimate costs associated with getting approvals
  - hardware costs

assembly costs
shakedown costs
data acquisition costs
data analysis costs
documentation costs
disposal costs

Table 3.1.1.1a Available Property Data for Al, U, and Al-U Species

Property	Al	Fuel U-Al Alloy	UAl ₄	UAl ₃	UAl ₂	U
Solid						
h_f	y	d	---	---	---	y
T_s	y	y	UAl ₄ → UAl ₃ + Al (1005 K)	UAl ₃ → UAl ₂ + Al (1632 K)	---	y
T_L	y	y			---	---
C_p	y	---	y	y	y	y
ρ	y	e	298 K	298 K	298 K	y
k	y	f	---	---	---	y
Liquid						
T_B	y	---	---	---	---	y
T_C	a	---	---	---	---	---
C_p	y	---	---	---	---	1408 K ≤ T ≤ 1579 K
P	b	---	---	---	---	---
ρ	y	---	---	---	---	1408 K ≤ T ≤ 1600 K
k	y	---	---	---	---	---
μ	933 K ≤ T ≤ 1500 K	---	---	---	---	---
σ	933 ≤ T ≤ 1500 K ^c	---	---	---	---	---
ϵ	y	---	---	---	---	---

a Estimated

b Values calculated from thermodynamic relations

c Uncertainty in temperature coefficient is 50%

d Estimated from phase diagram and limited data available for UAl₃ and UAl₄

e Data available for 30.5 wt% U but not 35 wt% U

f Data available for 30.5 wt% U at 425 K ≤ T ≤ 625 K; for 32.9 and 42.5 wt% U at 338.2 K

Table 3.1.1.1b Available Property Data for Structural Material

Property	Cladding		Stainless Steel
	8001 Al	1100 Al	304L
Solid			
h_f	---	---	y
T_s	---	(Melting	a
T_L	---	Range)	a
C_p	---	373 K	y
ρ	---	298 K $\leq$ T $\leq$ 573 K	y
Liquid			
T_B	---	---	y ^b
T_C	---	---	---
C_p	---	---	c
P	---	---	y
ρ	---	---	y
k	---	---	y
μ	---	---	y
σ	---	---	---
ϵ	---	---	---

a Estimated from phase diagram.

b Calculated from vapor pressure.

c Estimated

Fig. 3.2.1 Schematic representation of FCI experimental rig (proposed by T. Kress, 11/93)

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Appendix A

Summary of past work, regulatory positions, SAW and SP member comments

The purpose of this appendix is to provide a summary of past work done in various areas concerning experimentation and issue closure, as well as comments on regulatory positions relating to various concern areas. This is supplemented with discussion of various review comments received from the ANS SAW and SP members. Material in this appendix is divided into two parts. Part A.1 is devoted to discussions concerning various containment failure modes and mechanisms cited in Section 2.1 of this report. Part A.2 contains discussions related to experimentation.

A.1.1 Past work, regulatory positions, SAW and SP comments - Containment bypass and failure to isolate

This has been the area very strongly emphasized at the ANS SAW by panel members⁶, as well as by the ANS Safety Review Panel¹⁸. This is clearly a potentially crucial area, and will require close examination of various possible system flow paths which could provide a direct flow path for radionuclides to the environment. Possible areas to be investigated include ventilation ducts, lines penetrating containment, and seals, etc.

A series of calculations were performed during pre-conceptual design to define the overall functional requirements for containment leak rate and filtration efficiency (D. L. Selby et al, "ORNL Contributions to the ANS PProject for October 1986 - March 1987," ORNL/TM-10579, 4/1/87). The requirements adopted-- maximum containment leak rate of 0.5% of containment free volume per day with a filtration efficiency of at least 99% for secondary containment air treatment-- go beyond the minimal requirements identified in the study. The relevance of the study in the present discussion is to define in a qualitative sense how much containment bypass or degradation could be tolerated within accepted guidelines. The calculations assumed the occurrence of a whole core melting accident with the TID-14844 source term release into the atmosphere of primary containment, subsequent leakage from the primary containment at a constant leak rate into the secondary containment, and thence, without delay, to the environment through a filtration stage.

Conservative calculations (>95% confidence level) of atmospheric dispersion allowed determination of the locus of limiting combinations of leak rate and filtration efficiency that meet the 10 CFR 100 radiation exposure guidelines at the site boundary. For 97% or higher filtration efficiency, it was found that significant primary containment leakage could be tolerated: up to 20% per day. Dropping filtration efficiency to 90% reduced the allowable containment leak rate to about 5% per day, whereas a filtration efficiency of only 50% further reduced the tolerable containment leak rate to about 1% per day. These results show that the evaluation of containment failure modes, especially partial failures or degradation, must be considered whether the failure is to primary containment only, secondary containment only, or to both primary and secondary such that leakage goes directly to the environment. Clearly, the acceptability of a given containment failure or degradation event will also be heavily dependent upon specific sequence and source term characteristics.

Table A.1.1 summarizes regulatory positions, along with SP and SAW member comments on this subject.

In SECY-93-087, the USNRC staff recognize that leakage paths may exist, e.g., vacuum relief valve leakage, cracks in structures and penetrations. One point made by the NRC staff which may have bearing on ANS deals with heat exchangers. If heat exchangers are external to the containment, core debris from the reactor vessel may threaten integrity of tubes and therefore, provide a pathway for release of fission products. For ANS, the heat exchangers are within containment. Therefore, this

concern does not apply directly. We only need to consider lines connected from the heat exchangers to the secondary side heat exchangers / coolant basins which are out of the containment. This aspect was discussed with ANS Project engineer B. Maxon who indicated that support would be necessary to evaluate design parameters and limits for actuating signals, etc.

SAW panel member T. G. Theofanous recommends demonstration of a high-enough reliability such that frequency of bypass may be considered negligible (which may require on-line monitoring and special procedures). Another SAW panel member C. Forsberg makes the claim that since we have a large number of workers within containment, isolation may be far more difficult to ensure. He goes on to suggest that if we provide a dedicated vent line in the design to relieve pressure buildup and thus the driving potential for radionuclides to areas where experimenters may be working. In SECY-93-087, the USNRC staff have approved a dedicated vent for ABWRs, and the staff recommends that the need for a containment vent for passive designs should be evaluated on a design-specific basis. The SP also recommends looking at this important issue to see whether flow paths may be available for radionuclides to bypass the reactor pool. All these suggestions are considered valid and should be evaluated for incorporation, except possibly for the dedicated vent line which will not be necessary if rapid-enough evacuation of personnel can be assured from containment. Past scoping evaluations have shown that for significant steaming events, the ANS containment pressurization is far from the design pressure and hence, ANS may not need a separate vent line to relieve pressure buildup but may need it to drive fission products away from workers to allow sufficient time for evacuation.

Past work in this area concerned evaluations by for the ANS Conceptual Safety Analysis Report¹⁹ (CSAR) and documented in ORNL/TM-12333¹⁰ included in which were cases wherein a 0.5 m diameter hole was postulated to be present for direct transfer of fission products to the environment (possibly from failure to isolate). As expected, due to the rapidity of events and due to the very improbable set of assumptions made for one of the sequences, early fission product release to the environment outside of containment with bypass of filter banks led to prompt (8 on-site) fatalities. Individuals within containment were assumed to safely evacuate such that no user or experimenter would be subjected to radionuclides during their presence within the containment. The estimated frequency of the event (viz., Core-concrete-interactions with containment bypass) was in the $10^{-7}/y$ range. MELCOR code²⁰ modeling was nonmechanistic for core melt progression and did not include modeling of fission product chemistry which may play a major role in attenuating release of fission products to the high bay region.

The ANS Project has conducted a containment reliability study (ref:ORNL/ANS/INT-30) to assess the conditional containment failure probability (ccfp). This study looked at containment failure due to equipment (i.e., air-locks, penetrations, ventilation ducts) failures only and did not include phenomenological effects of severe accidents. The study concluded that the ccfp would be about 0.056 (excluding phenomenological effects) which does meet the overall ccfp goal of being < 0.1 , but falls short of the internal Project goal of < 0.01 (for equipment reliability).

Past work also included a study involving a systems-engineering approach to identify a list of potential leakage paths within containment to experimental locations on the first and second floors, as well as from the containment to the environment. Various potential radiation release pathways are shown in Figure A.1.1. Reactor containment boundary external leak paths are tabulated in Tables A.1.2 and A.1.3 respectively (ref. ISDD-21). ANS Project designers have made considerable efforts to ensure that experimenters on the first and second floors are physically separated from the process areas. The first and second floors have their own personnel air locks and ventilation systems, with two stairwells between these floors located at opposite ends of the containment (for redundancy). One direct pathway for radionuclides to the first and second floors is via. rupture disk between these floors and the main operating floor in the high bay, which opens up under a 14 kPa (2 psid) pressure difference. Another potential pathway for radionuclides to directly bypass filters and

subject experimenters on the first and second floors to radionuclides is via. rupture of cold sources and /or beam guides. Such a rupture may only occur from explosive events in the core region.

The operating area consists of various user facilities, viz., the hot cells, the National Activation Analysis Facility (NAAF-1), gamma radiation facility and isotope handling facility (ref. ISDD-47). The precise number of personnel associated with these user facilities may range from 5 to 40 on a continuous basis (per ANS Project staff engineer T. McManamy). NAAF-1 would include rabbit tubes and capsule transfer paths to and from the reactor core region. According to McManamy, some of these tubes may lead outside of containment. Design details are not yet available. Work has not been completed to evaluate the risk to users (from radionuclides during severe accidents) on the operating floor.

A.1.2.1 Summary of past work and present status - FCI related concerns

A scoping study was conducted²¹ (documented in ORNL/TM-11324) to evaluate the potential for steam explosion events in the ANS core region to fail the core-pressure-boundary tube (CPBT) and the reflector tank. Three different non-mechanistic initiating scenarios were considered. The first two were large reactivity insertion and a loss of primary coolant pumps without control rod insertion by either of the two redundant, independent scram systems. Core melting and superheating were considered for these two events, for which we now know the occurrence frequency to be less than 10^{-7} /y range (from the Phase I Level-1 PRA results tabulated in Table 1.3.2). The third scenario was postulated to occur due to the core having become molten without superheat. A model was developed to predict pressure rise which was then coupled with finite-element ADINA code²² calculations. ADINA calculations resulted in failure envelopes for the CPBT and reflector tank, respectively. It was judged that, the CPBT would fail under loads from all three scenarios if a whole-core steam explosion takes place. The reflector tank would also fail from loads generated from the first two scenarios, but not from the third one. These evaluations were conducted without contribution from aluminum-water chemical reactions.

Modeling and analyses capabilities have since improved considerably due to work done for ORNL's HFIR and documented²³ (in ORNL/TM-12220). Past work done for ANS indicate that bounding or highly conservative analyses may not be the way to proceed in the future. However, we also cannot make the blanket assumption about absence of chemical energy on an explosive time scale. For future analyses, accounting has to be done for chemical energy from Al-H₂O reactions which can add about 20 times more energy (viz., 17 MJ/kg) compared to the thermal energy content of the melt itself (viz., 1 MJ/kg).

In order to evaluate the steam explosion triggering potential for mixtures of silicide and oxidic fuels mixed with aluminum, steam explosion experiments were also sponsored and conducted (Ref: Y. Jin and M. L. Corradini, "Single Droplet Vapro Explosion Experiments Effect of Fuel Composition (Aluminum, Cermet and Silicide Fuels), ARS'94, April 17-21, 1994, Pittsburgh, pp. 133-141) at University of Wisconsin at Madison (UWM). These were simple experiments involving ~3-6 g droplets of aluminum or mixtures of aluminum with U₃Si₂ or U₃O₈ with fuel fractions ranging from 5 to 10 w/o. The droplets were melted in a crucible and dropped into a pool of subcooled water under atmospheric pressure conditions. An impact hammer-type triggering device was utilized to trigger an explosion. Aluminum was used as a baseline fuel composition for the experiment. The molten droplet temperature range was between 700 C to 1220 C. It was found that an external trigger causing a shock pressure of > 0.8 MPa was necessary to initiate explosions of droplets. For the experiments involving aluminum alone, almost all of the melted aluminum was possible to drop into the water pool. However, when the fuel-aluminum mixtures were utilized a highly viscous solid-liquid slurry composition is formed principally due to the exothermic chemical interaction between the fuel and aluminum. Thereafter, the highly viscous part still remained within the crucible

and the metallic portion drained down to the water pool (which was possible to trigger as easily as aluminum alone). For ANS conditions involving steam explosions during flow blockage events, such a contact mode is thus not relevant since much of the viscous material would also be involved in the explosion process (which might be far more difficult to trigger, if at all). However, the tests did provide some important information related to behavior of core debris relocation characteristics. A distinct potential for separation was observed, which might become important to keep in mind for certain melt relocation scenarios.

A.1.2.2 Summary of regulatory positions on issue, SAW & SP Comments - FCI related concerns

Table A.1.2.1 summarizes regulatory positions for power reactors on the vapor explosion issue, as well as many comments received from ANS SAW and SP meetings. As noted therein, for power reactors the NRC provides guidance in SECY-90-016²⁴ that staff will evaluate FCI dynamic loads due to ex-vessel FCIs on containment in conjunction with the ccfp goal. In the past, the so-called alpha-mode containment failure issue (viz., steam explosion leading to an energetic missile causing containment failure) has been analyzed quite extensively for power reactors, and has been found to be a low probability event. For the alpha mode failure, an LWR vessel is largely voided through which a water slug accelerates prior to impact against the reactor vessel top head (which then breaks off and forms a potentially containment-threatening missile). For credible steam explosion events in ANS, a situation such as that for LWRs (viz., voided pressure vessel) is not possible due to the large amounts of water surrounding the entire RCS. Some of the uncertainties dealing with steam explosion phenomena concerning mixing, triggering, chemical energy contributions from metal-water reactions and propagation are still under world-wide research. Some of these research areas are generic in nature and also apply to ANS. For ex-vessel steam explosions, new power reactor applicants strive to prevent steam explosions by ensuring vessel coolability (viz., accident management). Ex-vessel steam explosions for ANS would need consideration in the subpile room if core melt relocation to the subpile room and interaction thereafter with water, lead to appropriate conditions for onset of steam explosions.

For the DOE's New Production Reactor (NPR), aluminum ignition was a key issue (for the heavy water concept) and has been investigated as part of the DSAC approach, where a major question wrestled with involved the extent of chemical reaction between Al and H₂O during an explosive FCI. To shed some light, three series of experiments were conducted as part of this effort. One experiment²⁵, conducted at Sandia National Laboratories (SNL) with small droplets of molten aluminum dropped into a pool of water resulted in onset of ignition at 1700 K, with the extent of chemical interaction approximated at about 4% (of mass). These were triggered experiments. A second series of experiments²⁶ conducted at SNL consisted of about 5 kg melt masses dropped into a tank of water. Both, triggered and untriggered experiments were conducted which indicated onset of large-scale chemical interactions (i.e., more than 30-40% aluminum ignition) at temperatures as low as 1300 K. A third set of experiments²⁷ were conducted at Argonne National Laboratory (ANL) in which the radial contact mode was forced to simulate interactions between molten tubes of aluminum and water in a production reactor. The melt mass used was about 1 kg at 1100 K. Only one triggered experiment was possible to conduct wherein the hot cell was damaged in the process, and the experiment series was stopped. Indications are inconclusive whether Al-H₂O chemistry played a role in the ANL experiments. Unfortunately, none of these experiments were conducted with fuel materials. Hence, the effect of mixtures on occurrence and energetics is largely unknown.

At this stage, it should be mentioned that past experiences with steam explosions in research reactors such as SL-1, and SPERT have indicated that only a small amount of the molten aluminum would undergo ignition (viz., Al-H₂O chemical reaction on an explosive time-scale). The amount of melted (with superheat) aluminum which ignited is difficult to quantify with good accuracy since much of

the debris was widely dispersed. Estimates are in the $< 5\%$ range for portions of the core which melted and superheated. These estimates are considerably smaller than the Sandia large-scale tests. These were reactivity excursion-driven events wherein fuel plate temperatures were rapidly taken up to aluminum vaporization in the hot spot region of the core. Reactivity excursion-type events are not considered realistically bounding events for ANS. However, due to its very high power density, adiabatic-type heatup (following steam blanketing) under full power conditions could be very rapid in the ANS (i.e., 3000-7000 K/s). As such, only limited parallels can be drawn. The actual amount of ignition for ANS conditions will vary with the thermal-hydraulic conditions, material considerations and contact mode prevalent during the explosion event.

From Table 2.1.3 we see that, unlike power reactors, as mentioned by SAW panel member T. Kress, for ANS we cannot avoid the FCI issue, a priori, on grounds of low probability of occurrence. The SAW panel members as well as SP members have made it clear that a system-wide scoping analysis of this area for ANS should be done as rapidly as possible, preferably, in a manner similar to that done for HFIR to judge whether this is a relevant issue for ANS. If it is found to be an issue, it can then be addressed via. focused approach (e.g., ROAAM) coupled with selected experimentation.

A.1.3.1 Summary of past work and present status - Static overpressurization challenges

Work done in this area constituted the bulk of severe accident modeling and analysis for the ANS Conceptual Safety Analysis Report (CSAR)^{10,19}. Briefly, the MELCOR code was used to model the ANS containment, but not the RCS. All major compartments, and heat structures along with control logic were modeled. Two different types of scenarios were modeled in conjunction with conservative assumptions of whole-core melt, and fission product inventories at end-of-cycle (EOC). The first scenario looked at steaming events, wherein, it was postulated that core dispersal would occur leading to debris being sourced into the reactor pool, which had drained down to the beam tube level. Pool cooling equipment was considered to be inoperative. Noble gases were assumed to escape to the high bay volume, whereas other volatile fission product groups (iodine, cesium, tellurium) were assumed dissolved into the pool water. Upon pool heatup, and subsequent increase in vapor pressure the volatile fission products get released into the high bay volume. Steaming loads were evaluated, which indicated that the containment-wide pressure would rise to about 122 kPa (3 psig). The second scenario investigated containment pressurization loads from an MCCI event in the subpile room, with and without water. It was found that, containment-wide pressurization levels were bounded by those from scenario 1. Details of these investigations have been documented¹⁹ (in several technical papers and in ORNL/TM-12333).

A variation of scenario-1 based upon 10 CFR 100 guidelines for splitting fission products, was also conducted with an extensive parametric study to investigate heat transfer effects, and other uncertainties, for which results are summarized in Table A.1.3.1. Details have been documented in a letter report and in a technical paper¹¹. In essence, it was found that, containment pressure is a strong function of heat transfer characteristics between steam-air mixtures and the containment steel shell. The present MELCOR model utilizes one single control volume for the very large high-bay volume, and assumes perfect mixing and contact with steel primary shell. Convective-cum-condensation heat transfer modeling uncertainties become crucial, especially since ANS design pressure is only 170 kPa (10 psig). For power reactor containment's where significant credit is taken for such passive heat transfer capability, the USNRC has required extensive experimentation, as is being done for the AP-600 design. Overall, the MELCOR evaluations indicate for the most part that, margin to failure exists for steaming loads based upon conservative-modeled scenarios following the 10 CFR 100 guidelines. What remains to be confirmed is the validity of MELCOR evaluations.

A.1.3.2 Regulatory positions, SAW and SP comments - Static overpressurization

challenges

The regulatory aspects, along with SAW and SP member comments for this area are given in Table A.1.3.2. As noted previously also, SECY-93-087 specifies that the containment pressurization should be low enough during severe accidents so as to make sure that stresses in the shell do not exceed service level C limits. As mentioned previously, the ANS design pressure is stated as being at a level of 170 kPa (10 psig). This is not the pressure which will cause service level C stresses to form in the steel shell. Discussions with E. Stone of MMES engineering were held on this topic. E. Stone, upon conducting some evaluations came up with a containment pressure limit of about 197 kPa (14 psig), at which the service level C stress limits would be reached. This value is 40% higher than the design pressure, and gives an additional margin to safety, which may become useful to have considering uncertainties related to heat transfer and steaming potential. Perfect mixing models such as the ones used with MELCOR need to be validated against multi-dimensional evaluations. For example, the NRC has recently indicated (letter²⁸ from B. Sheron to C. Pugh, 2/1994) that it intends using the three-dimensional capability of COMMIX²⁹ to benchmark and validate results predicted by CONTAIN³⁰, especially for the AP-600 design which claims considerable benefit from passive heat removal by the steel shell primary containment. ANS Project has also committed (in response to SP recommendations) to perform such calculations to validate MELCOR predictions.

SAW panel member C. Forsberg postulated that a pulsating recriticality event could in effect, provide a significant additional source of energy for steam formation, which may overshadow the energy level from decay heating alone. The same reviewer also recommends maintaining the option of having a dedicated vent to provide pressure relief, so as to divert radionuclides away from populated areas and allow meeting of ANS SAS worker protection goal. At this stage, evidence from the ANS CSAR work for reactor pool steaming events suggests that there will be enough time for evacuation of experimenters in the first and second floors. On the other hand, MCCI events as modeled will not provide such a time span. Overall, both the review groups demonstrated skepticism regarding core melt scenarios involving a dry MCCI event in the subpile room. For the future, such an event will be investigated only if found to be mechanistically feasible. Sensitivity study results from Table 2.1.4 suggest the possibility of higher pressures for which a vent pipe may prove useful. This recommendation, therefore, has merit and we should reserve the possibility for incorporation of this vent pipe. Overall, the issues related to safely evacuating workers will be dealt with separately in Section 2.1.1 dealing with containment bypass. Pulsating recriticality issue will be dealt with separately in Section 2.1.6 via. closure of recriticality-related topics. Again, MCCI-related threats are also to be closed separately (Section 2.1.5). Once accomplished, concerns and comments made by Forsberg will be adequately resolved..

SAW panel member S. A. Hodge recommended the use of CONTAIN for modeling ANS containment similar to that done with MELCOR. This recommendation has merit from the standpoint of CONTAIN code's pedigree and longer history of validation and acceptance. From a technical perspective, a scrutiny of CONTAIN code manual and recent (3/1994) discussions with K. Washington (CONTAIN code development manager at SNL) reveal that the CONTAIN code has incorporated special-purpose models for modeling specific features of PWR and BWR systems in addition to also modeling heavy water, deuterium and tritium gases. Compared with MELCOR, CONTAIN also has superior radionuclide transport and modeling capabilities, the principal difference being attributed to the allowance of fission product transmutations. It is not clear, at this stage whether the unique capabilities of CONTAIN are significantly different from those of MELCOR. K. Washington also stated that a knowledgeable user of MELCOR would most likely not see much difference in predicted results, if MELCOR is used carefully. Another development along these lines involves ongoing debates within the NRC on whether CONTAIN and MELCOR are both to be supported financially, with the NRC's preferred choice being MELCOR for further development, and CONTAIN to be archived. For ANS, it should be recognized that building a new input model with CONTAIN similar to that done for MELCOR, is not expected to be a major

expenditure of funds and time, considering the many similarities of the two codes. Presently, the only advantage we see deals with CONTAIN code's enhanced credibility and the fact that it represents heavy water, tritium and deuterium transport. Tritium gas decay is also modeled. Additional discussions with S. Hodge also reveal that experience shows that there are a lot less problems encountered with using CONTAIN vs MELCOR. This option will be kept open.

S. A. Hodge also recommends investigating what would happen if the fans or blowers for the secondary containment are inoperable, such as during a station blackout. The fans causing suction on the secondary containment are considered to be safety grade equipment, are always on, and are backed up with battery-banks, and can be operated off the standby diesel generators. Therefore, the suction would be available during loss of off-site power (which has a frequency of about 0.1 / yr). The loss of safety-grade diesel generators will further lower the frequency of occurrence to about 1 in a 100 therefore, bringing the probability of a large release below the ANS SAS goal of $10^{-7}/y$. From a consequence standpoint, the loss of secondary fan exhaust system could cause pressure buildup and significant leakage out of the secondary containment via cracks and penetrations. Once negative pressure is lost, discussions with ANS designer G. McNutt reveal that consequence evaluations would likely be forced to be done without decontamination from filter banks. A design option may be pursued in which we line the inside of the concrete shell with a steel liner, and make the penetrations safety grade also. However, it may not be cost-effective. ANS SAP will tackle this subject only if required by PRA studies. Currently, since the estimated frequency of occurrence is in the $10^{-8}/y$ range, it is not considered worthy of inclusion for issue closure purposes.

SAW panel member T. Kress made a remark that the geometry and sizing of the ANS primary containment has not been done in a comprehensive manner to evaluate gains or losses on containment robustness from small parameter (e.g., thickness) changes. This is a valid question. Containment shell thickness was based on fabricability concerns having to undergo time-consuming in-place heat treatment and also from early severe accident calculations with CONTAIN and MELCOR which had predicted pressure buildup significantly below 170 kPa (10 psig). The ANS CSAR evaluations using MELCOR also demonstrated peak pressures well below the design pressure. Therefore, this issue has not been addressed further, but will be looked into if detailed containment thermal-hydraulic evaluations give rise to pressure buildup concerns. This recommendation will be addressed in conjunction with the suggestion for incorporation of a vent pipe.

Finally, SAW panel member M. Merilo displayed concern regarding overtemperature failure of penetrations. For the MCCI and steaming scenarios analyzed so far, we find that the primary steel shell temperature does not rise above 400 K, a value far below any known temperature at which penetrations may fail (viz., ~600 K). However, this concern may become important for the situation involving combustion of gases, where very high temperatures may be evaluated in detonation regions. This aspect is addressed further in section 2.1.4. The likelihood of such situations involving detonations has not been evaluated, yet. The issue of high temperature-induced penetration failures will be closed separately via. closure of topics related to combustible gas loads. Therefore, this subject will not affect containment overpressurization-induced failure.

The first round of SP comments indicated that the SP was concerned about the lack of containment sprays, and also the lack of an active containment heat removal system. The SP was particularly interested in bypass pathways around reactor pools (which has been addressed separately in Sect. 2.1.1). The second round of SP comments indicate that not having active sprays and containment heat removal systems should not be of much concern for ANS due to the large containment volume coupled with a relatively small decay heat source. However, the lack of need for active systems of heat removal and steam/aerosol/volatile fission product condensation is predicated upon the accuracy of MELCOR predictions. The SP has stated " It is important that an accurate thermal-hydraulic analysis of containment response be performed. In particular, detailed 3-D computational fluid

dynamic (CFD) analysis should be performed to assess the accuracy of the somewhat simplistic MELCOR calculations done to date." We agree with this conclusion and advice, which was positively commented upon by the SP during the second round of comments.

A.1.4 Past work, regulatory positions, SAW and SP comments - Combustion related concerns

Past work done in this area has been conducted in conjunction with the ANS CSAR and the ANS hydrogen safety study (HSS)³¹. To summarize up front, work in both areas indicated that potential for very high temperatures and loads would exist only for a few selected cases (wherein highly conservative assumptions are made). For the majority of cases, past analyses indicate that the combination of a large containment and small quantities of hydrogen will likely not make this a certification issue. Only confirmatory work will be needed dealing with incorporation of design fixes to prevent detonations in areas likely to create hydrogen pocketing.

For the CSAR, combustion events were modeled¹⁹ as deflagrations due to hydrogen and carbon monoxide combustion in the subpile room during MCCI events. It was found that an MCCI event without flooding would lead to high temperatures (of about 4000 K) in the subpile room, a point of concern brought up by a DOE reviewer of the ANS CSAR³². For an MCCI with flooding, the resulting temperatures are much lower (about 1000 K). ANS Project shall incorporate alumina concrete as the basemat material coupled with strategic flooding in the subpile room. Therefore, CO and H₂/D₂ generation as a result of concrete decomposition will not be of concern any more. Detonation of gases was not possible to model with MELCOR (which does not incorporate shock-physics capability). In any case, ANS Project shall incorporate igniters and/or other engineered safety features (ESFs) in the subpile room to prevent detonations.

As part of the HSS, it was conservatively postulated that deuterium from the cold sources and aluminum-water reactions would bubble up to the surface of the reactor pool and undergo a detonation event there. Various bubble sizes corresponding to the upper detonability limit, stoichiometric limit and lower detonability limit were analyzed for detonations right above the pool surface. Loads at various containment locations in the high bay were derived using a 2-D system model developed with CTH³³ (coupled with CET89³⁴ which provided the necessary shock parameter inputs for CTH). These values were used to drive a finite-element model of the primary steel shell using the ANSYS code³⁵ a task done by Gilbert-Commonwealth (GC). GC personnel also analyzed response of the floor. Overall, it was found that sufficient margins to failure existed. A separate evaluation was also conducted to look at a stratified detonation event (at the lower detonation limit) right below the top dome surface of the primary steel shell. Results combined with ANSYS model indicated that failure from pressure-induced deformations would not occur, albeit the margin to failure was much smaller. However, very high peak transient temperatures were evaluated (~4500 K) in the detonation mixture. The effect of such high temperatures on penetrations, etc. were not characterized. An examination of conceptual design drawings for the dome region under investigation for the HSS, revealed that the ANS containment does not have any penetrations near the top of the high bay region of the containment. Therefore, failure from high temperatures in the dome region is not a concern for ANS. Separate loading-response calculations were also conducted for the experiment room to simulate a deuterium spill on the floor. It was found that the floor and ceiling boundaries would not be able to withstand derived loads, and that stirrups would be needed to accommodate an explosion there. Results of the above-mentioned calculations have been documented.

Upon completion of this analysis for the experiment room, a later review at the cold source workshop (June, 1993) revealed that the analysis for the experiment room would have to be revisited since the study did not account for air treatment systems in the cryogenic enclosure room. The air treatment systems will be designed to reduce the available amounts of deuterium gases mixing with

air. Therefore, a scoping study shall be needed to provide designers appropriate conditions for air flow such that concentrations are well below detonation limits, and therefore, detonations in the cryogenic enclosure room will not represent a credible challenge.

The ANS HSS did not include assessment of potential for containment-threatening missile generation. This should not represent a hazard for ANS since focused efforts will be made to prevent and mitigate detonations in the first place.

Regulatory positions, coupled with SP and SAW panel comments are given in Table A.1.4.1.

The analyses for the ANS HSS go beyond current NRC requirements, since all cases evaluated so far involve stratified configurations. Fortunately, all cases evaluated so far, with the methods utilized indicate that containment failure from blast loads is not likely. However, SAW panel member S. Chen brought up the issue of considering hydrogen detonation event at the side walls of the primary containment, claiming that the structural load bearing capacity of the ANS containment from such an asymmetric loading could be much lower than from events presented from the ANS HSS. While this conclusion is debatable in the absence of a structural analysis, it does bring up an important point dealing with potential pitfalls associated with making postulates without having a physical basis for the same. This point was made more strongly by SAW panel members M. Merilo and T. Theofanous who recommend that a good front-end analysis be conducted which looks at how hydrogen gets released and evolves into detonable configurations in the ANS containment. This would give a physical basis for conducting any further detonation calculations, if locations are found where the possibility of stratification exists. These evaluations should also help to locate igniters or recombiners. M. Merilo also recommends using GOTHIC³⁷ or COBRA-NC³⁸ codes to investigate the mixing and stratification potential.

At this stage, the NRC regulatory criterion for hydrogen concentrations should be noted. The NRC position as stated in SECY-93-087 clearly states that containment-wide hydrogen control should be maintained to assure concentrations of uniformly distributed hydrogen are less than 10 v/o, assuming 100 % fuel rod zirconium-water reactions. However, selected regions may exceed the detonability limit. If so, ANS containment may need to install igniters (as ABB has done³⁶ in the CE-System 80+ containment). As mentioned previously, ANS subpile room is already slated to incorporate hydrogen control ESFs in the subpile room. Other regions such as at lines from the letdown tank will be investigated as necessary. On a containment-wide basis, for ANS, if all the hydrogen from aluminum-water reactions and from cold sources were mixed evenly in the ANS containment, the NRC's 10 v/o limit is met by a wide margin (viz., for ANS the concentration would be less than 1v/o).

A.1.5 Summary of past work, regulatory positions, SP and SAW comments - MCCI concerns

Past work in this area consisted of a scoping study⁴⁰ (documented in ORNL/TM-11761). Principal focus of this work consisted of considering MCCI situations for various concrete types (viz., limestone-common sand, basaltic, and alumina cement). Debris spreading was evaluated based upon experimental data taken with molten aluminum. For all configurations, we assumed that the ANS core debris would spread to form a pancake geometry over a given concrete type. The best-estimate spreading radius was about 1.1-m, whereas, the subpile room radius is about 3-m. CORCON.MOD2 code⁴¹ was used for MCCI simulations. Aluminum properties and eutectic information were added into the code. Silicide fuel was treated as being uranium-dioxide since silicide fuel properties and chemistry effects have not been modeled in the code. Cases were studied with and without water availability. The cases without water availability involve assuming that the molten debris relocates downward to the subpile room, but the primary coolant is blocked (due

possibly to metallic components freezing). Overall results obtained indicated significant ablation and combustible gas production potential with and without water if conventional concrete types were to be used. Significant ablation potential was also observed if alumina cement was used without an overlying water pool. These cases with conventional concrete will not be a concern in the future since ANS Project has committed to use of alumina cement concrete coupled with ensuring strategic subpile room flooding. However, if a water pool were present, alumina cement showed considerable promise. Little or no ablation was evaluated. Combustible hydrogen gas generation was limited to that produced from aluminum-water chemical reactions only, which is an order of magnitude lower than a comparable amount from conventional concrete cases. ORNL/TM-11761 includes a list of potential shortcomings and needs for further investigations.

This topic is an important regulatory issue for large LWRs. The regulatory position is given in Table A.1.5.1 along with recommendations from SAW and SP individuals. The NRC and industry position on this issue is closely tied to debris coolability such that MCCI is prevented. Prevention of MCCI results in significantly lower threats to containment failure from gas generation and runaway ablation. The debris coolability criterion is that sufficient cavity area should be provided in the design to allow long term debris coolability, specified by satisfying the $0.02 \text{ m}^2/\text{MW(t)}$ criterion and means for assuring quenching via cavity flooding. The quantitative criterion of $0.02 \text{ m}^2/\text{MW(t)}$ was derived from the Lipinski model for debris bed quenching. The debris bed is assumed to be composed of heat-producing solid spherical particles packed in a bed with an overlying water pool. Cooling occurs via water ingress into the debris bed and limited by counter-current-flooding-limitation (CCFL). The $0.02 \text{ m}^2/\text{MW(t)}$ criterion is correlated with the CCFL limit for a packed debris bed. It is important to understand the implications of this model, which requires that the packed bed have significant porosity, so as to allow sufficient water ingress. A case with little or no porosity for the heat producing fuel-aluminum mixture would closely approximate the ANS situation which is dominated by a metallic core debris with a power density which is more than 100 times that for power reactor debris. The metallic core debris would be composed of heat producing metallic compounds of UAl_x that are formed when ~ 100 micron size U_3Si_2 particles exothermically interact with aluminum upon onset of aluminum melting. The resulting mixture would represent a form of slurry-flow wherein the heat producing particles are separated from one another by an aluminum matrix. This is different from a porous bed of heat producing UO_2 pellets. Therefore, the power reactor criterion is physically inapplicable for ANS applications and would require far more spreading than is physically possible. For example, a straightforward application of the power reactor debris coolability criterion would require about 6 m^2 of spreading surface area. This is about twice the maximum spreading area we obtain using actual experimental data. Further, the surface heat flux for a decay power level of 9 MW would still be about three times the pool boiling critical heat flux (which is 0.5 MW/m^2). Since the Lipinski model for a porous bed has no physical application to ANS in the first place we need another alternate criterion or position for ANS applications which achieves the same purpose of preventing the occurrence and consequences of an MCCI (viz., runaway ablation and large-scale combustible gas generation).

Table A.1.5.1 also lists the range of comments received from ANS SAW panel members. As may be seen, this issue, more than any other gave rise to the most number of comments and criticism. Most of the criticism centered around the inapplicability of CORCON code for capturing the physics of the situation, which in their minds was modeled in an unrealistic manner. Panel members criticized the concept of the existence of a 8-mm thick layer of core debris eating away at concrete in the presence of degassing loads. This reaction was probably motivated from power reactor-related experiences where one is used to seeing situations where core debris depths are more than ten times the depth for ANS for power reactor situations. Several panel members thought that the relatively thin core debris layer would be blown out and dispersed by the degassing loads, thereby, attaining a large-enough surface area to ensure coolability. Therefore, the concept of a thin molten core layer over concrete undergoing MCCI without water was considered unrealistic by the reviewers. This

physical process, if it were to occur, clearly has not been modeled in CORCON, which was originally developed for power reactor situations. Upon further questioning about what minimum depth should be required after which CORCON validity would not be of much concern, an engineering estimate of about 32-40 mm was quoted by Prof. T. G. Theofanous. Separate inquiries to G. Greene (of Brookhaven National Laboratory who has obtained much of the aluminum spreading data) gave information stating that these experiments were also conducted with degassing and no such dispersion was observed over the range of experiments conducted. Documentation of this information has been requested from G. Greene, who said he would send it to ORNL. In his aluminum debris spreading experiments G. Greene also injected heated gases through molten pools of various depths, which he said caused bubbling to occur. The bubbling effect and heat transfer therefrom is already modeled with CORCON, but may not be done accurately for very thin layers. Considering this information, on an overall basis it does not seem like the evaluations with CORCON were much out of line. They represented the best effort at the time.

At this stage, it seems logical to assume that CORCON does not have all the correct physio-chemical models for analysis of ANS MCCI conditions. Nevertheless, it is not clear that the modeling and analyses done so far are significantly nonconservative, with the exception of dispersion. The final outcome of the scoping study was to demonstrate conservatively that use of alumina cement along with flooding would provide a good mechanism for meeting the ANS MCCI criterion as stated above. For the case with alumina cement coupled with flooding where little ablation was predicted, CORCON was effectively used as a heat transfer code only. For this condition where alumina cement does not give rise to degassing loads, the dispersion issue becomes irrelevant. A safe, stable state may be possible to reach wherein "runaway" ablation has been prevented, at least based on code calculations. The only difficulty concerns alumina cement structural integrity at very high temperatures. Past work⁴² done for the NPR indicates that beyond 2000 K alumina cement integrity gets degraded such that the debris temperature should be made to remain below this threshold value, even though the ablation temperature is a few hundred degrees Kelvin higher. In ORNL computations, the interface temperature was above the alumina cement ablation temperature during the initial stages due to the combination of high decay power level and aluminum combustion with water. When these computations were conducted it was not known what quantity of core aluminum mass would be unoxidized aluminum. Therefore, it was conservatively assumed that aluminum at start of MCCI would be unoxidized. Future work dealing with core melt progression should provide appropriate estimates of degree of aluminum oxidation prior to relocation on to the subpile room floor. Although the amount of ablation CORCON predicted was small, the high temperatures reached (viz., > 2000 K) could lead to questions dealing with structural integrity. Therefore, this point would need to be addressed to see if it constitutes a difficulty in using alumina cement as a choice for lining the subpile room floor. It should be recognized that the use of alumina cement is an established practice in the foundry industry where furnaces are lined with this material. Alumina cement is largely aluminum-oxide, is relatively cheap and should not raise the cost significantly above that which would be used to pour regular concrete in the subpile room.

SAW panel member T. Kress has suggested that instead of using CORCON, we utilize a well-established heat transfer code HEATING⁴³ to simulate the essential characteristics of the MCCI process. We feel like this suggestion has merit and should act as a useful cross-check for earlier CORCON calculations. SAW panel member M. Podowski suggested that we also look at the possibility of debris agglomeration which may cause uneven concrete ablation, and a possible pathway for containment bypass. This is a useful comment, but may be questionable considering the very high power density of ANS core debris which would cause melting and rolloff. However, for example, if the subpile room floor is uneven, or has locations where the debris pool can accumulate, this may become important to address. SAW panel member F. Chen questioned the proposition of core debris on the floor in the first place, without having conducted melt progression calculations. While this is a valid question, our initial scoping calculations for heat transfer and structural ablation have indicated a strong possibility of relocation. These 1-D scoping ablation evaluations were made

for core debris configurations over steel plates at various levels of decay power. Due to the confined geometry at the bottom of the CPBT and high decay power density, rapid ablation of steel was predicted. The precise extent and timing of such an event will have to await more detailed core debris melt progression modeling and analyses which account for the present ANS RCS geometry and core design.

S. Hodge recommended using the latest version of CORCON.MOD3 claiming that the NRC would most likely require this exercise anyway. Our investigations on the status and capabilities of CORCON.MOD3 reveal that this newer version does have improved heat transfer modeling capabilities, but does not incorporate capabilities for modeling aluminum-based fuels. Further modeling changes would be necessary, since aluminum-water reactions can generate significant amounts of chemical energy and may also affect the chemistry processes during ablation. This suggestion will be kept in mind. SAW panel member M. Merilo provided caution that core debris in the subpile room should be analyzed to see whether a possibility of failing some important containment barrier exists (e.g., liner-melting as in a BWR Mark-I containment). The ANS containment design has not yet progressed to this stage yet where such details can be addressed. Based on current drawings, we do not see any possibility of breaching a containment barrier except the concrete basemat, which may have drains, etc. which will have to be looked at carefully. Finally, SAW panel member C. Forsberg makes a suggestion that we consider the use of the core-melt source reduction system (COMSORS) as a backup method for addressing the MCCI issue for ANS. COMSORS is already being considered as a backup for ALWR MCCI concerns, and is under gradual development. This approach may become very attractive if it is found that debris accumulation leads to not being able to defend the use of alumina cement. COMSORS depends on a number of mechanisms for it to work as intended, such as dissolution rates, convection currents, dependence on lead as a final barrier to further transport. These are currently being investigated for $\text{UO}_2\text{-Zr}$ combinations at ORNL by C. Forsberg. Compared to use of alumina cement, COMSORS seems far more complex an approach both from a physics standpoint and also from an analysis standpoint. However, it also holds significant promise, and should be held as a strong backup with some support provided by ANS SAP towards obtaining key data for $\text{U}_3\text{Si}_2\text{-Al}$ mixtures, in conjunction with current tests.

A.1.6.1 Past work, regulatory positions, SAW and SP comments -Recriticality concerns

Past work in this area has involved benchmarking against data, establishment of a threshold (value for determining onset of criticality, viz., $k_{\text{eff}} = 0.95$) based on these comparisons and scoping calculations for recriticality potential of ANS core debris in lumped and dispersed configurations. Calculations were limited to recriticality occurrences within the ANS RCS alone. The geometry studied was one wherein the core debris was located within a steel pipe filled with D_2O , and surrounded with H_2O on the outside of the pipe. Overall, for the situations studied, it was found that lumped core debris configurations even in the limiting sense (i.e., including entire core mass) would not lead to recriticality. Dispersed configurations of various lengths, and conditions of void fraction, mixture temperature, light water contamination to various extents and boron content were also studied. It was shown that dispersed configurations have "potential" for recriticality. Light water contamination could significantly increase the predicted k_{eff} value. Temperature dependence was seen to be weak. The effect of boron and void fraction were quite large, as is to be expected for a light-water reflected system. Significant decrease of k_{eff} with increases in void fraction indicate that the event would tend to shut itself down, and not escalate. However, this also gives rise to the possibility of having a pulsating energy source with a significant enough potential for producing steaming loads in the containment. The effect of boron was indicative of the possibility of introducing a design-fix to prevent recriticality from occurring in the first place. The scoping study included a case in which small amounts of boron were added to the steel piping to reduce the k_{eff}

levels considerably below 1.0. Results of the scoping study have been documented in technical papers and in a report, ORNL/TM-12302. One significant development since the publication of ORNL/TM-12302 is that the ANS fuel loading for the two-element baseline core was changed to increase the amount of U-235 from 15.5 kg to 23.7 kg, an almost 50% increase. The present three-element baseline core employs about 19 kg of U-235 with a 50% enrichment. These changes indicate that the previous estimates of recriticality potential may have to be revisited.

Table A.1.6.1 lists current regulatory positions on recriticality, along with SAW and SP comments. The issue of recriticality for ANS has no parallels in the power reactor world from a regulatory viewpoint. There have been situations which have been analyzed for recriticality during severe accidents (e.g., for BWRs) but nothing could be found in NRC documents regarding regulatory positions on this issue. For example, SECY-93-087 and SECY-93-092 make no mention of recriticality as being an issue for power reactors. For ANS, the SP has recommended that recriticality during severe accidents would not be acceptable from a "traditional licensing" perspective, and that we should do whatever is necessary to design this problem away. This was a major work area for the NPR. The ANS SAW panel members make no such statements on licensing-related sensitivity, arguing that the recriticality problem, if shown not to endanger the containment performance goals, should be of no major concern. Clearly, where possible, problems should be designed out of the system. However, this can not be done arbitrarily and can become expensive. Therefore, a goal or criterion needs to be developed which will allow the necessary framework for focused efforts. The relevant ANS SAS criterion in this area is provided in the next subsection.

SAW panel member T. Theofanous points out that the cases for dispersed configurations need to be defensible on grounds of being physically-based. The concern here is that we have yet not shown whether the configurations considered in the ANS scoping study can be physically possible to attain in a realistic sense, if one considers actual physics of the core disruption process and transport thereafter. This is a valid point and should be addressed by ANS SAP. Theofanous also suggested the use of borated sleeves over RCS piping in vulnerable locations. This too, is a valuable comment and will be kept in mind as a possible preventive design feature. As has been mentioned in earlier sections, SAW panel member C. Forsberg claims that recriticality events, even if they are shown not to be explosive, could form a pulsating energy source. It is clear that the transient aspects of debris recriticality in a water-filled RCS of the ANS will need to be characterized to see if the thermal-hydraulics of the event causes significant void formation so as to cause early termination of a power excursion. Forsberg also recommended looking at alternate locations for recriticality potential, especially in the subpile room. SAW panel member S. Hodge questioned the feasibility of having a dispersion potential for core flow blockage events. This point makes intuitive sense, but is not correct for ANS since flow blockage events are not necessarily ones wherein a flow channel is completely blocked. Partial flow blockages large enough to cause post-CHF fuel melting in the presence of flowing medium (albeit at a lower mass flow) are clearly a plausible mechanism for gradual core dispersal (in the absence of a steam explosion wherein core dispersal would occur rapidly). What remains to be characterized are the dynamics of a realistic scenario whilst accounting for many uncertainties associated with damage propagation, ANS fuel behavior upon melting, etc. SAW panel member M. Merilo suggested that recriticality issues may be possible to analyze away. In light of what has been mentioned above, this may be possible but not guaranteed. As per suggestion made by Theofanous, we need to investigate front-end of accident scenarios for evaluating physically-based dispersed configurations. Finally, SAW panel member M. Podowski makes the suggestion that evaluation of the front-end realistically will need to be accompanied with carefully scaled experimentation since there is no information available on dispersion potential for molten silicide fuel plates. This point may have more merit than the statement suggests, as any evaluation for a situation as novel as this for ANS will require making a good number of unverified assumptions. In any case, these types of evaluations are necessary and if done with good engineering judgment may provide some useful and definitive information, and will provide the necessary impetus or trigger for deriving insightful experimental data.

A.1.7 Past work, regulatory positions, SAW and SP comments - Barrier meltthrough

ANS SAP has not done any work in this area. However, there appears to be (based on the current state of ANS containment design) no issue related to liner melt-through¹⁷ (such as that for BWR Mark-I containment's). The issue of meltthrough of basemat barrier via MCCI has been addressed separately, and hence, is not included here. Again, the issue of failing containment barriers via meltthrough of heat exchanger tubes and escape before closure of isolation valves is a containment bypass issue, and hence, is also not included here.

Regulatory positions along with reviewer comments on this are given in Table A.1.7.1. In SECY-93-087 (related to containment bypass) the NRC staff give guidance related to the possibility of meltthrough of heat exchanger tubes, thereby leading to containment bypass. SP and SAW panel members have provided due caution to address these seemingly innocuous areas which may have the potential to become issues. ANS SAW panel member M. Merilo was particularly concerned about investigation of potential for meltthrough of containment barriers from the high power density ANS fuel debris.

A.1.8.1 Past work, regulatory positions, SAW and SP comments - DCH

No significant work has been done in the DCH area for ANS except for a MELCOR calculation to evaluate what the maximum pressure rise in the containment would be if all of the core decay heat were sourced into the containment atmosphere. These calculations were conducted for an earlier design, and hence are not considered valid. Nevertheless, DCH-type concerns of overheating and rupture of the containment for ANS does not apply due to the large containment size and much smaller core mass and fission product inventory.

Table A.1.8.1 provides current regulatory positions on the DCH issue, along with ANS SAW and SP member comments. Regulatory position on this issue as given in SECY-93-087 gives guidance that reactor cavities should be shaped such that a direct flight path to containment atmosphere is avoided, coupled with RCS depressurization.

The general guidance received from the ANS SAW panel members is to address scenarios that are realistic (which we accept). Rather than postulate a DCH scenario for ANS, it is concluded that this condition is highly unrealistic for ANS, considering the primary fact that ANS (unlike a commercial PWR) is a low pressure system located under water. One may postulate ANS core debris being ejected into the subpile room. However, being a solid water-filled system the ANS RCS would very likely depressurize rapidly if core debris were to melt the seals separating the RCS from the reactor pool, prior to being ejected into the subpile room. Initial scrutiny of the system parameters by the ANS SAW members has resulted in comments to the effect that DCH issue for the ANS is to be considered a very improbable event.

A.2 Severe Accident Workshop (SAW) and Safety Review Panel (SP) Comments & Recommendations on Experimentation Aspects

The test matrix, selection criteria and associated elaboration's for many of the experiments cited previously in Table 3.1.1 were presented to the ANS SAW panel members, and to the ANS SP.

Comments and suggestions received from individual members are tabulated in Table A.2.1. SP members state "Vapor explosion experimental work is important. Be sure to include in these investigations substantial superheat (~500 C) and the mechanisms of melt dispersal." We agree with this advice.

SAW panel member F. Chen suggests "We should not blindly rush into doing experiments based upon artificial scheduling or resources related requirements. If we do not spend a little bit of effort up front to define what assumptions or parameters are most important for ANS we may end up wasting a lot of time spending a lot of resources studying issues that benefit the LWR program and of little or no value to ANS." We agree with this statement, which in effect, corroborates our position that experimentation will be tied in with the need for satisfying ANS SAS goals during the issue closure process.

T. Fujishiro suggests that the NSRR experiments can provide prototypic data only to a certain extent and should not be expected to represent ANS in-core phenomena exactly (a relatively impossible task). This is also a very valid comment, but should be read in the correct context. NSRR experiments with prototypic fuel plates and actual nuclear heating, the rates of heatup which come very close to representing ANS conditions, quite possibly are the only ANS experiments in the test matrix which come closest to shedding prototypic light in crucially-important phenomenological areas. These areas are essentially the ones dealing with the front-end core melting, superheating and propagation. Therefore, every effort will be made to obtain fuel performance data under geometry conditions which are close-to actual ANS geometry's.

S. Hodge strongly recommends that it is very important that all proposed small-scale experiments be conducted to the extent that budgets will permit. Hodge argues that the evidence of data to back up claims made for safety is almost always asked for by regulatory review bodies. In the absence of data, how can one possibly answer questions dealing with ANS fuel behavior in relation with the many phenomenological uncertainties during severe accidents. We agree with this philosophical argument. Carefully conducted experiments can reduce uncertainties considerably, leading to improved modeling and understanding of system behavior, to making an adequate safety case, and to answer questions dealing with ancillary issues concerning degree of contamination, need and validation of designed features. Hodge suggests that the need for conducting MCCI experiments would be questionable if we utilize alumina cement with flooding for the subpile room floor. This suggestion is questionable since work for the NPR with alumina cement has suggested that the interface temperature should not go above 2000 K. For the ANS calculations, interface temperatures were several hundred temperatures higher, albeit not beyond the first half hour. Whether alumina cement structural integrity is compromised for temperatures above 2000 K even for a short time will be investigated and clarified via literature survey. Again, our evaluations were done for one fixed spreading radius whereas, the elimination of the MCCI threat would require also investigating effects of accumulation and clumping. These aspects should first be investigated analytically using best-estimate techniques and then decided whether the confirmatory MCCI experiment needs to be done for demonstration of the soundness of assumptions and modeling aspects. The need for experiments for clumping situations may be negated if an effective mitigative fix were employed, such as the use of alumina marbles or some other severe accident mitigation design feature.

T. Kress suggests that ANS severe accident experimentation should be tied to proving robustness of containment features and for setting up the necessary database for conducting a Level-2 PRA which demands a best-estimate approach. For FCI-related topics, Kress specifically states "I believe this issue presents you with your biggest SA headache. You cannot rule it out on low probability grounds (as was done for LWRs) and, I suspect the bounding energetics will be too much for the containment (although this should be the first thing to check). Kress separates FCIs into Type-1 FCIs (viz., power excursion driven fragmentation and dispersal into the surrounding D₂O), and Type-2 FCIs (viz., molten fuel and cladding contacting D₂O and/or vice versa). Kress suggests using the existing NSRR experiments to the fullest extent for resolving Type-1 FCI energetics, making sure that we pay particular attention to similarity in geometry, rates of heatup, etc. This is a very valuable suggestion, and is being pursued to the maximum possible extent. For Type-2 FCIs, Kress suggests that the proposed shock-tube geometry is not representative enough for the various

contact modes possible during core melting and heatup. That is, the fuel material is too constrained and does not allow for evaluating behavior in which fuel is injected into water. Kress goes on to provide a suggested approach based on a fundamental assumption that small-scale explosions are more energetic than large-scale ones. The suggestions for Type-2 FCIs have merit and should be pursued further, with the exception that the assumption of small-scale explosions being more energetic than large ones is not supportable by evidence to date, which is quite the opposite. Larger masses provide the physical possibility of a propagating explosion. However, thresholds may exist beyond which scale effects become negligible, and again, it may be feasible to consider large-enough masses for representing energetics of the relatively small ANS core. Therefore, the essential ideas provided by Kress may still be possible to utilize. At this stage, it should be mentioned that our feeling is that the shock tube geometry will provide for conservative values for energetics. This is due, primarily, to the fact that there is only one degree of freedom available for motion of expanding steam compared to two or more in the ANS core geometry. Therefore, values for energetics obtained in the shock tube may be applied with the label of conservatism for thermal-to-mechanical energy conversion. While these values may be higher than actual values for ANS conditions, it is expected that they will be a factor of ten lower than those predicted using bounding thermodynamic approaches and possible to use comfortably in making an adequate safety case. This perspective was not thought of at the time presentations were made to the ANS SAW panel, but will be discussed at the next opportunity. Discussions with several panel members (M. Fontana, T. Kress, M. Podowski, T. G. Theofanous) after the workshop have resulted in general agreement on this point. Despite the appropriateness of this argument, the shock tube geometry may not answer the issue of energetics of fuel being injected into water. The test apparatus suggested by Kress and reproduced in Fig. 3.2.1 may prove useful and will be assessed.

M. Merilo suggests that the experiment selection criteria developed are excellent, and that following the guidelines would provide a sound basis for conducting necessary experiments.

M. Z. Podowski suggests that inevitably, we would need to perform damage propagation and melt dispersion experiments to shed light on the basic question of whether this is even possible, and to what extent. Otherwise, modeling aspects would always be challenged for credibility. This point is valid, and will be accounted for during the issue closure process where the need for such experiments will be looked into. Podowski also suggests that elimination of MCCI threats with alumina or other cement concretes should consider the possible effects of clumping or accumulation of debris. This is a valid point and has been discussed earlier in relation the comments made by S. Hodge.

T. G. Theofanous makes the point that experimental program is not put in proper focus, arguing that the conduct of experiments should be intimately tied into needs identified with closure of issues. This is generally valid in the sense experiments should be conducted only if needed. The important issue deals with answering the need question with perfect certainty. The aspect of needs is being addressed via. answering basic questions in Table 3.1.2 and in conjunction with the issue closure process wherein experiments will be triggered only to answer specific questions dealing with phenomena under specific situations where uncertainties may be very large, and cannot be put to rest via bounding them or via analysis alone. The lack of strong positions on several proposed experiments is currently tied into the lack of a focused effort investigating the details of various scenarios connected with containment failure modes and mechanisms. The disciplined and focused issue closure approach documented in Section 2 will address this need whereby the need and type / nature of individually proposed experiments will become self-evident. At this stage, we hasten to add that the currently envisioned experiments are not considered by the ANS SAP to be the exact experiments that will be conducted, rather, they are a best-estimate representation of current thinking which will be polished as events progress and more information becomes available.

Table A.1.1.1 Containment bypass issue - regulatory positions,
SAW and SRP member comments

Source / Comment or Guideline	Plans for Accommodation
<u>NRC SECY-93-087</u> Containment bypass / leakage paths may exist (e.g., cracks in structures and around penetrations). Dedicated vent lines (e.g., as approved for ABWRs) will be evaluated on a design-specific basis. Vendor or licensee should consider areas such as heat exchangers which provide a secondary path out of primary containment	<u>Proposed ANS acceptance criterion</u> ANS SAP will, in conjunction with containment designers will address this issue so as to make it a very unlikely event, so as to meet intent of ANS SAS deterministic containment integrity, risk and dose limit goals. Direct bypass flow paths resulting from accident loads (e.g., FCIs) will also be evaluated.
<u>ANS SAW Panel member comments / suggestions</u> <u>C. Forsberg</u> Include piping to maintain future option of installing containment filter venting system. This will also help if containment isolation becomes an issue. Assuring containment isolation for ANS will be more difficult than for power reactors <u>T. Theofanous</u> Address containment bypass up front, such that frequency of such events is negligible. This may require on-line monitoring	This suggestion will be discussed with containment design task team and given due consideration for inclusion. This will be addressed on an ongoing basis as the containment design matures.

ANS SP member comments / suggestions

Lack of containment spray system may be a certification issue. Vapor/gas bypass pathways may exist around water pools during severe accidents. Reliance on secondary containment and secondary filter system is probably acceptable but will require defense.

Lack of containment heat removal system may also be a certification issue. As noted above, bypass pathways may exist around water pools in severe accidents, and detailed analysis will likely be required to determine the containment pressurization transient. A SBWR-type passive heat removal system should be considered to mitigate pool bypass issues.

The "super-containment" concept has pitfalls. The informed public is justifiably skeptical of absolutes. The DSAC approach taken for the NPR provides a more practical approach to assure functional reliability of containment during severe accidents. The Panel recommends a focus on bypass and penetration failure modes as well as on pressure and missile loads.

Due to large containment size, it is not felt like containment sprays will be required to assure low-enough pressure buildup.

The general issue of addressing bypass flow pathways is being taken very seriously by ANS Project, both for locations internal (viz., affecting experiment staff) and external (viz., affecting the public) to containment.

Our present evaluations for containment pressure buildup has indicated margins. These calculations done with MELCOR shall be complemented / checked via more detailed multidimensional calculations. If sufficient margins are found, no consideration of active ESFs (as recommended) is needed. If adequate margins do not exist, recommendations will be followed upon.

The super-containment-related pitfalls have been accepted and taken into account while deriving ANS SAS goals in Section 1. As mentioned above, bypass issues are being taken very seriously, and also in conjunction with FCI-type loads (which is addressed separately in Section 2.1.2)

Table A.1. 1.2 Reactor Containment Boundary External Leak Paths

-
1. Reactor fuel clad to
    - 1.1 primary coolant
    - 1.2 reflector vessel coolant
  2. Primary coolant to
    - 2.1 primary piping/equipment cells
    - 2.2 reflector vessel
    - 2.3 reactor pool
    - 2.4 subpile room
  3. Reflector vessel coolant to
    - 3.1 beam tube to first experimental floor
    - 3.2 reactor rabbit tube to second experimental floor hot cell
    - 3.3 cold source
  4. Cold source beam guide to second experimental floor
  5. Reactor pool light water to
    - 5.1 first floor
    - 5.2 second floor
  6. Fuel pool light water to
    - 6.1 first floor
    - 6.2 second floor
  7. Primary cell light water to
    - 7.1 first floor
    - 7.2 second floor
  8. Secondary light water to
    - 8.1 first floor
    - 8.2 second floor
  9. Experimental zone to operating zone
[Only when differential pressure exceeds 14 kPa (2 psid)]
  10. Operating zone to experimental zone
[Only when differential pressure exceeds 14 kPa (2 psid)]
  11. Cold source deuterium to
    - 11.1 operating atmosphere
    - 11.2 experimental atmosphere
  12. Core water or concrete reaction products to
    - 12.1 operating atmosphere
    - 12.2 experimental atmosphere

Table A 1.2.1 Fuel-Coolant-Interaction (FCI) Issue - regulatory positions, SAW and SRP member comments

Source / Comment or Guideline	Plans for Accommodation
<u>NRC SECY-90-016</u> NRC staff will evaluate consideration of containment loads from FCI dynamic loads with respect to ccfp goal	<u>Proposed ANS acceptance criterion</u> ANS SAP will, in conjunction with containment designers, address this issue using a realistic bounding initiating event (viz., core flow blockage and/or fuel defects) so as to meet intent of ANS SAS deterministic containment integrity and ccfp goals
<u>ANS SAW Panel member comments / suggestions</u> M. Fontana Critical issue deals with fuel-coolant-interaction (FCI) coupled with ignition. Bounding calculations are not recommended for this issue. T. Fujishiro The FCI issue with aluminum ignition should not follow bounding methodologies, but should investigate related issues dealing with incipience. R. M. Harrington Set up realistic bounding scenarios at front-end to evaluate explosion loads on containment. S. Hodge Address a dropped fuel element accident during refuelling. T. S. Kress FCI issues will predominate risk for ANS and may pose the greatest challenge for effective closure. Relegate power excursion driven (Type 1) FCIs to a low-enough probability and use existing collaboration with Japan Atomic Energy Research Institute (JAERI) to the fullest, and strive for prototypic testing conditions in order to use the data to directly justify any limitations placed on energetics. For conventional (Type 2) FCIs involving molten fuel/cladding (including liquid D₂) contacting water, resort to a completely empirical approach using geometries close to the suggested shock tube. For Type 2 experiments evaluate several contact modes statistically.	Agree. Bounding analyses, where done, are being conducted with an eye towards realism and for evaluating if resulting loads are going to be tolerable for ANS situations. We are taking a graded approach for this issue. If our first-cut analyses with realistic bounding loads indicates containment failure, more refined modeling coupled with experiments will be done to reduce conservatism. This is being done currently for flow blockage initiated core melt events. Flow blockage is assumed to occur in one core element only rather than assuming a whole-core melt event apriori. Agree. This feature is the subject of continuous scrutiny. Such an accident and resulting recriticality potential will be designed against. This recommendation is being followed for Type 1 FCIs in the Nuclear Safety Research Reactor (NSRR) experiments. Tests are being conducted to simulate temperature excursions during flow blockage events in confined geometries to strive for prototypic conditions to the extent feasible. Experiments to address Type 2 FCIs will plan to accommodate advice given to assure statistical fidelity. This will be decided based on results of scoping study looking at FCI loads from steam explosion events driven by flow blockages.

Table 1.1.3 Reactor Containment Boundary External Leak Paths

-
1. Reactor primary containment—steel
    - Source SSST in air
    - Leakage—welds/penetrations into annulus
  2. Reactor secondary containment—concrete
    - Source leakage from primary containment
    - Leakage—from primary containment and outside environment into annulus vented to exhaust system
  3. Air lock doors
    - Source SSST in air
    - Leakage—elastomer seals
  4. RCB HVAC isolation valves
    - Source SSST in air
    - Leakage—valve seats into primary-secondary annulus or HVAC exhaust system
  5. Reactor cooling secondary isolation valves
    - Source SSST in water
    - Leakage—valve seats into cooling tower and outside environment
  6. Guide tube isolation valves
    - Source SSST in water or air
    - Leakage—valve seats into guide hall equipment
  7. Electrical penetrations
    - Source SSST in air
    - Leakage—electrical penetration into annulus from primary and reactor support building
  8. RCB basemat
    - Source SSST in air/water or core debris above basemat
    - Leakage into ground
  9. Other penetrations
    - Closed inside primary containment
    - Closed outside primary containment
    - Closed both inside and outside primary containment
-

The design and performance requirements of the reactor containment penetrations, including air locks for all systems that penetrate the containment, are defined and controlled by SDD 52, Reactor Building. The isolation valves, which are provided by systems that penetrate the containment, are controlled by SDD 69 where remote operation is required.

M. Merilo

It is not clear if proposed FCI experiments will meet the needs of closure. Analysis similar to that done for HFIR should be conducted.

M. Podowski

FCI experiments (including ignition phenomena) should be carefully formulated based upon separate effects evaluations, coupled with in-depth scaling analysis. The important aspect of melt-water interaction physics prevalent in ANS core should be bounded.

T. G. Theofanous

Mechanistic bounding analysis similar to that done for HFIR should be done to scope out the issue. Chemistry effects would most likely drive end results.

Analysis similar to that done for HFIR is being done currently. Enhanced planning and articulation of experimentation needs will follow.

Agree. The important aspect of clearly emphasizing the applicability of suggested test methods to simulate physics of the FCI process relevant to ANS conditions needs to be better clarified and advocated. This will be done using additional reviewer comments received to date.

Agree. This task is currently under way. Chemistry effects are most likely to drive end results.

ANS SP comments & suggestions

ANS Project should address steam explosion events and their impact on containment integrity, in conjunction with a well-planned experiment program

ANS project is addressing FCI events for realistically bounding initiating events. Scaled experiments will be conducted where applicable

Table A 1.3.2 Containment overpressurization from quasi-static pressure buildup - regulatory positions, SAW and SRP member comments

Source / Comment or Guideline	Plans for Accommodation
<u>NRC SECY-93-087, SECY-90-016, SECY-93-092</u> The containment shall maintain its role as a reliable, leak-tight barrier by ensuring that containment stresses do not exceed ASME service level C limits for a maximum period of 24 h following the onset of core damage, and following this 24 h period shall continue to provide a barrier against the uncontrolled release of fission products In view of low probability of severe accidents that would challenge integrity of containment, the probability of failure of the mitigation systems (those systems which can reduce the consequences of a core damage accident), from the onset of core damage to loss of containment integrity resulting in an uncontrollable leakage substantially greater than the design basis leakage, should not exceed 0.1. The staff, however, intends to ensure that the containment can deal with all credible challenges and does not intend to apply this ccfp guideline to detract from overall safety For advanced reactors, the staff proposes to utilize a standard such that containment designs must be adequate to meet the onsite and offsite radionuclide release limits for various event categories. Events and sequences will be selected deterministically supplemented with insights from PRAs. For new reactor designs with limited operational experience will require a containment system that provides a substantial level of accident mitigation. Due to limited experience in analysis of severe accidents significant uncertainty will exist in assessing a ccfp goal (which is based on experiences with LWRs) <u>ACRS position</u> The proposed ccfp value is reasonable, but .. this value is defined only with the context of a family of initiating events. It should be used to develop "requirements" and not just passed on .. The deterministic criterion proposed is not a simple alternative to the ccfp goal	<u>Proposed ANS acceptance criterion</u> For realistic bounding events derived from Level-1 PRA insights with frequency $> 10^{-6}$ /y containment overpressurization from quasi-static pressure buildup will be shown to meet ANS SAS deterministic containment integrity and ccfp goals. Similar to that planned for reactor containments like AP-600, MELCOR or CONTAIN-type estimates for containment pressurization buildup will be backed with COMMIX (or GOTHIC, COBRA/NC)-type evaluations for confirmation of the primary steel shell as an adequate passive heat-sink.

Industry position

- SP/90 will meet ccfp goal
- CE 80+ will meet ccfp goal when weighted over credible core damage sequences (viz., > 10⁻⁵/y excluding external events)
- GE ABWR will meet ccfp goal when weighted over credible accidents

ANS SAW Panel member comments / suggestions

C. Forsberg

Being a research reactor, containment pressurization from other sources like noncondensable gases such as from dewars of liquid nitrogen should be investigated.

Oscillating recriticality and potential for significant additional energy generation should be investigated

S. Hodge

Evaluate effectiveness of secondary containment if exhaust fans are not operable

Severe accident work should focus on dominant threats

Consider shifting to CONTAIN for ANS containment response analysis.

T. S. Kress

Steaming loads have not been adequately bounded, since chemical energy has not been factored into MCCI calculations.

ANS SP comments & suggestions

This point could be valid only for overpressurization of small enclosures and clearly not on a containment-wide basis. It will be planned for and addressed. Engineering department has been asked to provide input in this area.

Agree. This potential will be investigated as part of recriticality potential studies.

Agree. This important aspect will be investigated by severe accident group in conjunction with designers. Note that secondary containment fans are tied to safety-related diesel generators

Agree

This suggestion has merit, and will be considered strongly (at the very least to provide validation for MELCOR calculations). MELCOR was chosen due to its availability, its wide-spread acceptance and use, its support by the NRC and because our impressions are that it incorporates the same capabilities as CONTAIN does for transport evaluations.

This has been done indirectly, since debris temperatures were taken from CORCON calculations which include chemical energy input during MCCI. Again, hydrogen burning was permitted.

First round (4/20/1992)

1) Lack of containment spray system may be a certification issue. Vapor/gas bypass pathways may exist around water pools during severe accidents. Reliance on secondary containment and filter system will require defense.

2) Lack of containment heat removal system may be a certification issue. Bypass pathways may exist and detailed analysis will be required to determine the pressurization transient. A SBWR-type passive heat removal system should be considered to mitigate pool bypass.

3) Since containment pressurization will determine evacuation times, it is important that an accurate thermal-hydraulic analysis of containment response be performed. In particular, detailed 3-D CFD analysis should be performed to assess the accuracy of the somewhat simplistic MELCOR calculations that have been performed to date.

Second round (12/22/1993)

1) Based upon presentation of pressurization transient analysis work, the SP concurs that there is no need for an active safety-class containment heat removal system, because the ANS has an inherent passive containment heat-removal system.

2) The project has been responsive to the recommendation (made earlier) of the SP to undertake 3-D CFD analysis of the containment response to accidents.

This issue was evaluated to gauge its potential. Presently, based upon MELCOR evaluations, it is felt like sufficient margin will exist and that, active features will not be required. The issue of containment bypass is being taken seriously, and addressed in a systematic manner in Sect. 2.1.1.

These evaluations will be planned since we claim significant credit from passive cooling capability from the steel shell primary containment. Tools such as GOTHIC, COBRA/NC, PHOENICS, or COMMIX will be assessed. Simple, scaled experiments may be conducted to provide validation.

Agree.

Agree. As mentioned above, this work is planned.

Table A.1.4.1 Containment failure from combustion loads - regulatory positions, SAW and SRP member comments

Source / Comment or Guideline	Plans for Accommodation
<u>NRC SECY-93-087, 10 CFR 50.34 f(2)</u> TMI-related requirements requires a hydrogen control system that can safely accommodate hydrogen generated from 100% fuel-clad metal water reaction Because of uncertainties in phenomenological knowledge of hydrogen generation and combustion, new reactors should be designed as a minimum to the following criteria: - accommodate hydrogen generation equivalent to 100% metal-water reaction - limit containment hydrogen concentrations to no greater than 10 v/o - provide containment-wide hydrogen control (igniters, recombiners) for severe accidents <u>ACRS position</u> The ACRS recommends the evaluation of impact of combustion, and the possible detonation including stratification before establishing a fixed volume fraction limit. This is of particular importance for steel-shell containments.	<u>Proposed ANS acceptance criterion</u> "ANS containment-wide hydrogen concentrations will be kept below the detonability limit. Confirmatory work will be done to evaluate mixing and removal via use of ESFs and/or natural processes and demonstrate that ANS SAS containment integrity goals are met." <u>Important Note:</u> For ANS design, hydrogen generated from 100% clad water reaction results in a containment-wide concentration of less than 1 v/o, thereby, meeting current ALWR requirements by a wide margin.
<u>Industry position</u> - SP/90 will meet 10 v/o goal with 100% clad reaction - CE 80+ will meet 13 v/o goal with 75% clad reaction - GE ABWR utilizes a nitrogen-inerted containment coupled with recombiners to meet NRC requirements	
<u>ANS SAW Panel member comments / suggestions</u> <u>F. Chen</u> ANS SAP should consider potential for detonation due to hydrogen stratification to one side of containment. Containment load bearing capability may be much lower for such conditions than the ones considered by ANS SAP for ANS HSS	This possibility shall be investigated if our estimates of stratification potential indicates that such a configuration is physically reasonable

C. Forsberg

Program should consider catalytic recombiners in the experiment rooms as an accident prevention measure.

T.S. Kress

Work done for addressing hydrogen combustion, including detonation effects is appropriate

M. Merilo

Overtemperature protection must be considered from combustion events

GOTHIC or COBRA/NC codes which have been validated against experiments should be used to address hydrogen stratification issue.

M. Podowski

Hydrogen transport evaluations may not need experimentation since ANS may be able to learn from experiments conducted for power reactor scenarios

T. Theofanous

Hydrogen explosion analysis provides a good start towards closure. Future work should first consider potential for release and mixing.

This suggestion will be discussed with containment design task team and given due consideration for inclusion.

Future work will consider evolution and stratification potential prior to conducting detonation calculations

Agree

Agree. Past scrutiny of available tools has indicated GOTHIC to be a strong candidate for use. The acquisition of this code and its future use will be planned for.

Agree

Agree. As determined to be required by systems analysis or specific regulatory guidance, the issue of stratification potential will be addressed using industry codes like GOTHIC.

Table A 1.5.1 Containment failure from MCCI loads - regulatory positions, SAW and SRP member comments

Source / Comment or Guideline	Plans for Accommodation
<u>NRC SECY-93-087</u> The staff has not pronounced a quantitative guideline in this area since the EPRI criterion for debris coolability and quenching is under evaluation via research. The staff proposes a qualitative guidelines as follows for new reactors: - provide sufficient reactor cavity floor space to enhance debris spreading - provide for quenching debris in reactor cavity - protect the containment liner and other structural members with concrete, if necessary; and, ensure that the best estimate environmental conditions (pressure and temperature) resulting from MCCIs do not exceed Service Level C for steel containments for 24 h. Ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions for MCCI. <u>ACRS position</u> Qualification of what constitutes sufficient floor space is an open question. Otherwise ACRS agrees with the NRC staff position.	<u>Proposed ANS acceptance criterion</u> "Sufficient cavity floor space will be provided in conjunction with an appropriate basemat floor material and flooding capability to prevent significant ablation and combustible gas generation from realistically bounding core debris states. A long-term, safe stable state should be achieved."
<u>Industry position</u> - EPRI ALWR criterion requires cavity floor size to provide 0.02 m²/MWt, with a flooding capability. Steel shell or liners will be protected from core debris attack by at least 1 m (3 ft) of concrete - SP/90 will meet EPRI criteria and will provide for automatic flooding of lower cavity - CE 80+ will meet EPRI criteria, with the flooding water provided from the RWST - GE ABWR will meet EPRI criteria. Lower drywell will be flooded based upon melting of temperature-sensitive fusible plugs which allow suppression pool water to enter the drywell cavity due to high temperature in the lower drywell from presence of core debris	The EPRI proposed value for spreading is based upon the Lipinski porous debris bed configuration, and is inappropriate for ANS conditions. Similar capability for ANS may be possible, such that water from reactor pool is available to quench the debris in subpile room

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ANS SAW Panel member comments / suggestions

E. Chen

The modeled MCCI scenario in subpile room does not sound realistic (especially the case of MCCI without flooding), and should be revisited. Core melt progression evaluations should be conducted to derive realistic states of core debris on to the subpile room floor.

C. Forsberg

Carefully investigate CORCON's viability as a tool for ANS. Look into the assumptions with the CORCON code to justify applicability.

ANS SAP should consider the possibility of a recriticality event in the subpile room. If potential is found, one fix might constitute boring the subpile room floor.

Program should undertake some investigation of alternate materials (e.g., COMSORS) as a backup option to alumina concrete.

R. M. Harrington

Work towards ensuring water availability in subpile room.

Address applicability of CORCON code for ANS conditions

S. A. Hodge

Obtain CORCON.MOD3 and perform supporting hand calculations to show retention of core debris, independent of spreading.

Regulatory attitudes shaped as a result of "China Syndrome" should be sufficient motivation for choosing alumina concrete for basemat

The modeled MCCI scenarios were postulated conservatively to evaluate need for flooding and for use of a decisive deterrent to uncontrolled ablation and combustible gas generation. Melt progression evaluations are in progress to determine propensity for various configurations in subpile room. The case of MCCI without flooding was considered to accommodate the postulated situation where the breach from which core debris drains to the subpile room is closed off due to freezing of metallic melt, which then prevents water egress downwards. Due to ANS Project decision to assure flooding, no further study is needed for dry MCCI.

This work was initiated in the past. Preliminary findings have been documented in ORNL/TM-11761. Incremental assessments will be planned as necessary.

This is a good suggestion and will be pursued with ANS containment design task team to investigate cost and feasibility of boring alumina cement.

This suggestion will be discussed with containment design task team and given due consideration for inclusion, in conjunction with a severe accident scoping study to investigate relative merits of two options

This suggestion will be discussed with containment design task team and given due consideration for inclusion. This has been done and documented in ORNL/TM-11761. Incremental assessments will be planned as necessary.

The acquisition and use of CORCON.MOD3 will be looked into. It should be realized that any version of CORCON without the right chemistry models to account for ANS fuel will always be suspect. From this view, separate-effect heat transfer calculations will be performed to assure retention on alumina concrete (where chemical interactions will be largely absent).

The use of alumina concrete is the preferred choice at the present for the ANS subpile room. Practical aspects of ensuring flooding and robustness during attack will need to be ironed out with designers

T. S. Kress

Approach towards addressing MCCI and coolability based on Nuclear Regulatory Commission (NRC) guidelines is inappropriate. An alternate to using CORCON should be pursued, such as using HEATING.

M. Merilo

Switch to alumina cement based on CORCON analyses is not justified.

M. Z. Podowski

The need for debris-structure interaction experiments should be tied also to scoping evaluations which investigate the effects of clumping of debris. For example, uneven distribution of fuel debris may cause uneven ablation of concrete.

Debris coolability may not be of much concern if sufficient spreading is ensured

T. Theofanous

For addressing MCCI, alumina cement may work, but CORCON's viability as a modeling tool is definitely questionable.

This will be investigated further. For alumina cement case (with flooding), CORCON was used effectively as a heat transfer code only (viz., like HEATING).

For alumina cement with flooding CORCON was used conservatively as a straightforward heat transfer code. We believe the results to be reasonable, but further validations with a more sophisticated heat transfer code such as HEATING will be conducted.

Agree. This will be taken into account when discussing need for confirmatory testing not only for ablation potential, but for realistic evaluation of spreading behavior. Limiting configurations will be analyzed for overheating potential before determining need for experimentation.

Agree. However, the proposed effect of gas generation and cooling of debris particles over conventional concrete is hard to quantify in the absence of knowledge of how much structural material accompanies debris. Even without structural material, the proposed effect of gas generation creating a fluidized bed is qualitative and may require experimental qualification under ANS conditions. This issue will be investigated further in conjunction with analysis for effects of fuel clumping

As mentioned earlier, for case of alumina cement with flooding, CORCON was used in effect as a heat transfer tool. The other applications of CORCON for other cases is defensible only on basis of conservatism. It is agreed that important modeling features may be missing when applying CORCON for thin metal layers over conventional concrete. Such aspects have already been documented in ORNL/TM-1171.

ANS SP Comments

The Panel believes that the project should be careful not to let non-mechanistic scenarios drive the design of containment or other severe-accident mitigation features. It is hard to imagine how the core could attack the concrete in a dry cavity. If there is no mechanistic way to reach an unflooded condition, that branch should not be included in the event tree nor should it be analyzed.

Agree. Past work for dry MCCI was done based upon very rudimentary calculations on freezing potential of aluminum, and without ANS Project commitment to assure flooding. No further work will be done for dry MCCI. Future work will be driven from results of mechanistic core melt progression analysis. A similar mechanistic, best-estimate treatment will be utilized for setting up and quantifying the Level-2 PRA containment event tree.

Table A.1.6.1 Containment failure from recriticality loads - regulatory positions, SAW and SP member comments

Source / Comment or Guideline	Plans for Accommodation
<u>NRC</u> No available information on regulatory position is available. Recriticality potential for BWRs during severe accidents has been considered and found to be a low probability event.	<u>Proposed ANS acceptance criterion .</u> "Recriticality event for the ANS under severe accident conditions will be prevented by design to be well beyond design basis (viz., $< 10^{-7}/y$). Inadvertent recriticality, in any event will be shown to not threaten containment integrity such that the ANS SAS goals are met."
<u>Industry position</u> - None	
<u>ANS SAW Panel member comments / suggestions</u> <u>C. Forsberg</u> Oscillating recriticality and potential for significant additional energy generation should be investigated. ANS SAP should consider the possibility of a recriticality event in the subpile room. If potential is found, one fix might constitute boring the subpile room floor. <u>S. Hodge</u> It is not clear why dispersed configurations will result in a fully-blocked core condition. This aspect should be considered up front before further investigations. <u>M. Merilo</u> Recriticality issue may not be a significant problem and may be possible to analyze away. <u>M. Z. Podowski</u> Damage propagation and melt dispersion experiments are necessary to provide a sufficient modeling basis. Such experiments should be separate effect-type, preceded by a thorough scaling analysis.	Agree. This potential will be investigated as part of recriticality potential studies. This is a good suggestion and will be pursued with ANS containment design task team to determine cost and feasibility of boring alumina cement. This is a good point. However, it should be realized that core-flow blockage conditions leading to melting does not necessarily have to be a fully-blocked condition. This issue is to be revisited following guidance from several workshop participants. One reviewer, C. Forsberg, for example has indicated that recriticality may provide a pulsating energy source and hence, potentially increase the overall containment pressure loads. Analyzing away this problem may be far more difficult than finding a simple design fix. Agree. The precise nature of such experiments will be proposed and tied into the issue of "need" by association with closure and safety goals.

T. G. Theofanous

Recriticality issue should consider realistic scenarios accounting for axial dispersion and finite rates of core degradation. Some cases described are not realistic.

The front-end of severe accident scenarios will require improved definition. This work is planned for during this fiscal year.

ANS SRP comments

Recriticality potential should be considered for cases in which core material may have melted and relocated within the core

Agree. This potential will be assessed.

Table A 1.7.1 Containment failure from liner meltthrough - regulatory positions, SAW and SRP member comments

Source / Comment or Guideline	Plans for Accommodation
<u>NRC SECY-93-087</u> Containment failure and bypass due to meltthrough of liner at vulnerable locations shall be avoided as required for Mark-I containments. Heat exchanger liners which may be compromised by core debris attack shall be assessed.	<u>Proposed ANS acceptance criterion</u> "Liner meltthrough as a means for containment bypass or violation of ANS SAS goals for ANS will be shown to be an improbable event (viz., $< 10^{-7}$ /y). The ccfp for such events shall be shown to be less than 0.1." For ANS design, we do not expect to have a problem similar to the BWR Mark-I liner attack. The possibility of heat exchanger tube failure and MCCI-induced basemat meltthrough are addressed separately in Sects. 2.1.1 and 2.1.5, respectively.
<u>Industry position</u> - Liner meltthrough for BWR Mark-I containments is considered an incredible event provided water is available for the core debris in the drywell.	
<u>ANS SAW Panel member comments / suggestions</u> <u>M. Merilo</u> Liner melt-through may provide a viable containment failure mode.	Such features will be given increasing attention as the design matures. At the current design stage, we do not feel like liner melt-through (as applicable for some power reactors such as BWRs with Mark-I containments) is a credible path for relocation outside of containment.

Table A.1.8.1 Containment failure from Direct Containment Heating (DCH) - regulatory positions, SAW and SRP member comments

Source / Comment or Guideline	Plans for Accommodation
<u>NRC SECY-93-087</u> Containment designs for new reactors should include a depressurization system and cavity design features to contain ejected core debris. This issue is currently being resolved for the NRC by use of ROAAM approach. <u>ACRS position</u> DCH is an extremely improbable event, such that there is no need to require two modes of coping with this possibility. Either depressurization or cavity design provisions alone should be adequate. Because of possible safety benefits for other events, reliable depressurization is the preferred approach.	<u>Proposed ANS acceptance criterion</u> "DCH for ANS will be a highly unlikely event (viz., < 10⁻⁷/y). This will be achieved by assuring sufficiently low system pressure prior to release into the subpile room and demonstration that ANS SAS goals are met." DCH issue is not a certification area for ANS. This issue is closed for ANS.
<u>Industry position</u> - EPRI ALWR - cavity / pedestal/drywell should be designed to preclude entrainment of core debris into the containment atmosphere. A safety-grade RCS depressurization and vent system will be provided - SP/90 cavity configuration will prevent core debris from entering upper containment - CE 80+ provides for an indirect cavity vent path, including a debris collection chamber to de-entrain core debris, and a large floor area to enhance debris coolability. A safety grade depressurization of the RCS is included - GE ABWR provides a safety grade depressurization system and a suppression pool that surrounds the lower drywell cavity to reduce risk of high-pressure core ejection into the upper containment regions	It is expected that ANS core debris, upon melting through seals separating RCS from the reactor pool may naturally cause depressurization. Further, the relatively low operating pressure of ANS provides for a lesser potential for atomizing core debris if ejected under pressure into the subpile room. Containment overpressurization to failure or exceedance of ANS SAS containment integrity goals is not considered likely. Ancillary effects related to in-containment worker exposure are dealt with separately via. closure of containment bypass issue.
<u>ANS SAW Panel member comments / suggestions</u> <u>T. S. Kress</u> You have appropriately ruled out direct containment heating (DCH) for the ANS.	Agree. There may not be enough stored energy in primary coolant system and sufficient pressure gradient to cause core debris to be uniformly ejected into the containment atmosphere.

M. Merilo

DCH can be eliminated from consideration due to ANS being a low pressure operating system.

Agree. There may not be enough stored energy in primary coolant system and sufficient pressure gradient to cause core debris to be uniformly ejected into the containment atmosphere.

Table A.2.1 Best-Estimate Severe Accident Analyses - regulatory positions, SAW and SRP member comments

Source / Comment or Guideline	Plans for Accommodation
<u>DRAFT PSER FOR PRISM</u> Analyses for severe events are to be conducted with best-estimate methods. Source terms should be determined mechanistically. Must ensure sufficient data exists (through an R&D program) to allow sufficient confidence in mechanistic analyses methods <u>NUREG-1150</u> Work done for this area was largely best-estimate type using well-developed systems codes such as MELCOR, CONTAIN, SCDAP/RELAP5, with quantification of uncertainties <u>NUREG-1465</u> Best-estimate source terms derived mechanistically shall be used for site-suitability evaluations coupled with conservative meteorology	Best-estimate analyses will be conducted for Level-2 & 3 PRA, as well as for determination of accident management strategies, operator training and intervention. A Level-2 PRA is planned for ANS. Past work done for ANS site-suitability has been based on TID-14844 source terms. Mechanistic source terms will be developed for ANS in this area
<u>ANS SAW Panel member comments / suggestions</u> <u>T. Fujishiro</u> Licensing approach based upon past experiences with Fast Flux Test Facility (FFTF), coupled with satisfying NRC, Advisory Committee for Reactor Safeguards (ACRS) and Department of Energy (DOE) requirements is appropriate. ANS should, however, use a more realistic approach (viz., best-estimate methodology coupled with creative application of current regulatory guidelines). <u>S. A. Hodge</u> Begin developing ANS-specific core melt progression code module which can then be coupled with a containment transport code for systems analysis. Directly modifying the huge, ungainly code MELCOR may not be the right approach. Consider shifting to CONTAIN for ANS containment response analysis.	Agree. Best estimate analyses are planned for in PRA development. We also agree with suggestion to creatively implement severe accident accommodation guidelines for ANS specific application. Model development has begun in a small way towards evaluating realistic core melting fractions and damage propagation from core flow blockage events. The precise nature of coupling of this code with containment transport codes is being studied. This suggestion has merit, and will be considered strongly (at the very least to provide validation for MELCOR calculations). MELCOR was chosen due to its availability,

Spent core refueling accident needs to be evaluated. <u>T. S. Kress</u> Severe accident program should be used to demonstrate containment robustness, to support Level-2 and -3 PRA using best-estimate techniques, and to support accident management / recovery. MACCS may not be an appropriate code to use under certain conditions (especially within site boundary). <u>M. Merilo</u> ANS should conduct best-estimate analyses to accommodate potential regulatory changes. <u>M. Z. Podowski</u> Damage propagation and melt dispersion experiments are necessary to provide a sufficient modeling basis for best-estimate analyses and support of PRA, etc. Such experiments should be separate effect-type, preceded by a thorough scaling analysis.	its wide-spread acceptance and use, its support by the NRC and because our impressions are that it incorporates the same capabilities as CONTAIN does for transport evaluations Agree. This will be done in conjunction with systems analyses to see whether ANS SAS goals are met. Agree. This is precisely the aim of the program. Agree. The need and capability for addressing this issue will be factored into the overall ANS Level-3 PRA work Best estimate analyses similar to NUREG-1150 will be done in conjunction with PRA work using expert elicitation where uncertainties are large due to lack of adequate database. Agree.
<u>ANS SRP comments / suggestions</u> ANS Project should develop its own source terms mechanistically rather than rely on TID-14844 source terms. Evaluations should consider use of TRENDS-type models for aqueous chemistry	Agree. This is planned.

Table A.3.1 SAW & SP comments and suggestions on proposed experiments

Source / Comment or Guideline	Plans for Accommodation
ANS SP member comments / suggestions Vapor explosion experimental work is important. Be sure to include in these investigations substantial superheat (~500 C) and the mechanisms of melt dispersal.	Agree. Two sets of experiments have been proposed to investigate ignition, triggerability and thermal-to-mechanical energy conversion.
ANS SAW Panel member comments / suggestions F. Chen We should not blindly rush into doing experiments based on artificial scheduling or resources related requirements. Efforts should be made, up front, to define what assumptions or parameters are important for ANS. T. Fujishiro NSRR experiments can provide prototypic conditions for ANS only to an extent. Fundamental data from tests in well-controlled, simple and instrumented configurations should be used to derive necessary information analytically for safety studies. S. A. Hodge It is important to pursue small-scale experiments focussed upon the unique ANS fuel configuration as far as budgets will allow. .... Furthermore, without the results of an experiment program such as the one proposed, how could future inquiries with respect to the response of the fuel under severe accident conditions possibly be answered? If alumina cement is used as a basemat with flooding, the MCCI experiments may not be necessary. T. S. Kress Proposed experiments should be conducted to provide adequate basis for best-estimate work required for a PRA. Kress states, "I believe this (i.e., FCI) issue presents you with your biggest SA headache. You cannot rule it out on low probability grounds (as was done for LWRs) and, I suspect the bounding energetics will be too much for the containment." Relegate power excursion driven (Type 1) FCIs to a low-enough probability and use existing collaboration with Japan Atomic Energy Research Institute (JAERI) to the fullest, and strive for	Agree. Experimentation needs will be tied to satisfaction of SAS goals and specific needs identified during issue closure, FSAR and PRA work. Agree. We recognize potential limitations, but will continuously work with our collaborators in JAERI to come up with creative ways to achieve prototypic conditions to the extent possible. Agree. If alumina concrete integrity is not compromised due to thermal stresses from interface temperatures above 2000 K and debris accumulation / clumping is avoided, we agree with this assertion. This recommendation is being followed for Type 1 FCIs in the Nuclear Safety Research Reactor (NSRR) experiments. Tests are being conducted to simulate temperature excursions during flow blockage events in confined geometries to strive for prototypic conditions to the extent feasible.

prototypic testing conditions in order to use the data to directly justify any limitations placed on energetics. For conventional (Type 2) FCIs involving molten fuel/cladding (including liquid D₂) contacting water, resort to a completely empirical approach using geometries close to the suggested shock tube. For Type 2 experiments evaluate several contact modes statistically.

M. Merilo

Checklist prepared for experimentation is excellent and should be followed before next workshop presentation suggesting needs for individual experiments.

M. Podowski

Experiments should be tied to meeting safety goals and should be combined with careful scaling and separate effects analyses especially for FCI experiments.

The need for debris-structure interaction experiments should be tied to scoping evaluations which investigate the effects of clumping of debris. For example, uneven distribution of fuel debris may cause uneven ablation of debris.

Damage propagation and melt dispersion experiments are necessary to provide a sufficient modeling basis. Such experiments should be separate-effects type, preceded by a thorough scaling analysis.

Some safety experiments may have already been conducted for power reactors. For example, hydrogen transport, combustion and aerosol transport. Review of past experiments would be useful.

T. G. Theofanous

Proposed experiments have not been advocated in a strong-enough fashion. Experiments should be clearly tied into satisfying issue closure needs. It is not certain how the proposed shock tube and small-scale testing would be used.

Experiments to address Type 2 FCIs will plan to accommodate advice given to assure statistical fidelity. This will be decided based on results of scoping study looking at FCI loads from steam explosion events driven by flow blockages.

Agree. The prepared checklist will be extensively used in conjunction with issue closure process and best-estimate work needs.

We agree with this assertion. Experiment conduct will be tied into demonstrating that ANS SAS goals are met. The important aspect of scaling to simulate physics of the FCI process relevant to ANS conditions will be carefully studied.

Agree. This will be taken into account when discussing need for confirmatory testing not only for ablation potential, but for realistic evaluation of spreading behavior. Limiting configurations will be analyzed for overheating potential before conduct of experiments.

Agree. The precise nature of such experiments will be proposed and tied into the issue of "need" by association with closure and meeting of ANS SAS goals.

Agree. Every effort is being made to avoid duplication.

We agree that articulation of specific needs and payoffs was not possible to make as coherently as needed to support issue closure and satisfaction of ANS SAS goals. The issue closure process outlined in Section 2, along with work for FSAR and PRA will rectify this shortcoming.

Fig. A. 1.1.1 Reactor Containment Potential Release Paths

APPENDIX B

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October 2, 1995

Dr. Rusi P. Taleyarkhan, Manager
Severe Accident Methods &
Technology Development Group
Martin Marietta Energy Systems, Inc.
Oak Ridge National Laboratory
Bldg. 9104-1, MS-8057
Oak Ridge, TN 37831-8057

Re: ANS Severe Accident Program Overview and Planning Document

Dear Rusi:

Attached are my final comments on subject document. I appreciate your cooperation during this review process, and I am in good and general agreement with the final product. The points made here are either for emphasis or concerning aspects that are, I suspect, beyond your control.

It has been a pleasure working with you and your colleagues on the ANS project.

Best Regards.

Sincerely,

T.G. Theofanous
T.G. Theofanous, President

GENERAL COMMENTS

The document evolved under the influence of various review committees, and it shows. Were the ANS project allowed to proceed this would be a good time to sort everything out, adopt a coherent safety philosophy, a necessary and sufficient set of goals, and a sound demonstration methodology, and articulate them clearly in a from-scratch new document. Any important issues identified and debated during this "developmental phase," could be brought up and discussed, to any extent deemed appropriate for a lessons-learned sort of purpose (to avoid future relapses), in appendices.

I think that for this reactor, containment failure can, and should, be made "physically unreasonable," under any severe accident scenario with frequency greater than a screening frequency level. Typically, this level can be chosen as two or three orders of magnitude below the core melt frequency, and can be easily seen to fulfill the defense-in-depth concept of safety. For inherently safe reactors such as the ANS, there may be a problem in that the core melt frequency is already small enough to render the severe accident "window" thus defined beyond clear mechanistic (from an initiator standpoint) quantification. This sort of "difficulty" can be overcome within the ROAAM approach, and I certainly agree with the plan to address the severe accident window using ROAAM.

DETAILED COMMENTS

1. I do not agree with the approach regarding DOE order 548023. Accidents that belong in the "negligible" category should be left neglected. If DOE or anyone else want you to address a 10^{-8} accident, they should first explain to you what such numbers mean.
2. It is important to be clear that all risk goals regarding outside consequences can be subsumed by the ROAAM-related goal; that is, ensuring that containment failure is physically unreasonable. I would caution that pursuing these risk goals during implementation does not divert from the main focus on containment integrity. In the same vein, Table 1.3.1 would appear too grandiose and unnecessary.

3. It appears that you are "forced" to perform a level 2 PRA, and you are properly approaching this task by trying to make sure you have at hand all the tools normally employed for such a purpose. Now, I don't think you need to bother about such "traditions." Once ROAAM is complete there would be no need for a level 2 PRA, and even if it had to be done, it would be a rather trivial exercise. Recognizing this now could save significant resources later on and help avoid the "diversion" mentioned in the previous point.
4. I think using ROAAM as proposed could free ANS from mood changes in licensing. The report could be more forceful on this important aspect. Moreover, the report could express a more decisive point of view regarding the "treatment" of the some 200 workers and its effect on licensing. Even if we assume acute fatalities, the risk is much lower than that routinely taken by similar in size groups of people (i.e., air travel), and can be adequately addressed by the inherent safety of the reactor (i.e., CMF goal of 10^{-5}). This means we need no special risk goals, which, by the way, could be very hard to meet anyway.
5. By contrast to the above, I think the containment bypass goals are not strict enough. Bypass constitutes, in effect, containment failure, and I prefer a goal of 10^{-2} .
6. There is proper emphasis on ignition concerning FCIs, and I support it. On the hydrogen, at this point I am not very clear on how difficult it is going to be. I would advise extreme caution on using existing "mixing" codes, and it may be worth considering whether the subpile room could be steam-inerted under the conditions of interest.

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