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ALWR SEVERE ACCIDENT ISSUE RESOLUTION IN SUPPORT OF UPDATED EMERGENCY PLANNING

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I. INTRODUCTION

The Advanced Light Water Reactor (ALWR) Program in the U.S. is a cooperative, cost-sharing undertaking between the U.S. government, industry, and a number of international participants, with the objective of developing the next generation of nuclear power plants. The ALWR designs emphasize improvements in safety and operational reliability through simplification, improved safety margins, innovative passive safety systems, enhanced man-machine interfaces, and incorporation of the lessons learned from the operation of existing LWR plants. An important component of the improved safety characteristics of ALWRs is the consideration of severe accidents in the plant design. The U.S. Department of Energy (DOE) initiated the Advanced Reactor Severe Accident Program (ARSAP) to assist in the transfer of severe accident technology from the U.S. national laboratories to the industry to implement this approach.

The basic design requirements for this new generation of nuclear power plants were developed, under the management of the Electric Power Research Institute (EPRI) by the utilities and documented in the Utility Requirements Document (URD).[1] The URD safety policy is based on the traditional "defense-in-depth" approach, which emphasizes prevention through safety systems which prevent accidents from progressing to core damage, and mitigation to ensure that accidents are mitigated and contained. In a major departure from previous practice, severe accidents, including postulated core melt events, are specifically included in the defense-in-depth design considerations for ALWRs.

As a result of this approach, the emergency planning assumptions and criteria warrant a review and reevaluation for ALWR designs. ALWRs present a risk profile that is significantly different than that which served as the basis for the emergency planning requirements for operating plants. The determination of this profile necessarily requires the characterization of the severe accident response of ALWRs.

The severe accident risk characteristics of the ALWRs reflect an emphasis on accident prevention, which is quantified in the URD as a maximum permissible core damage frequency of less than one occurrence in 100,000 reactor years. For severe accident sequences of a frequency lower than this criterion, the URD safety policy requires provisions to arrest, mitigate, and contain the accident and, accordingly, opportunities to terminate a core melt sequence are provided whenever practical at every stage of core degradation. This includes design provisions to maximize the chances of success for reflooding the reactor by depressurizing the primary system, provisions to ensure retention of core debris in the reactor vessel by cooling the outside of the reactor vessel, and provisions for a more favorable geometry for core debris cooling in the reactor cavity in order to slow and then terminate a core-concrete interaction. For all risk-significant branches of the containment event tree, it must be demonstrated that early containment failure is avoided.

This paper addresses the severe accident issue resolution tasks which were undertaken by the U.S. ALWR Program and ARSAP to ensure that the capability of passive ALWRs to arrest, mitigate and contain severe accidents would be sufficient to justify a significant change in the appropriate emergency planning requirements. The next section summarizes all of the issue resolution activities that will culminate in the issuance by the U.S. Nuclear Regulatory Commission (NRC) of a Final Safety Evaluation Report for the passive ALWR URD, scheduled for January 1994. The following section addresses more recent activities undertaken by ARSAP to enhance the issue resolution basis and to provide additional confirmatory evidence supporting the URD criteria. Included are the ongoing activities to establish a technical case, if possible, for in-vessel retention for the passive PWR and for the accommodation of ex-vessel steam explosions in the passive BWR. Finally, conclusions regarding the significance of these efforts for ALWR emergency planning are provided.

II. UTILITY REQUIREMENTS DOCUMENT CONTAINMENT PERFORMANCE DESIGN CRITERION FOR EMERGENCY PLANNING

As noted above, the basic design requirements for the new generation of plants are contained in the ALWR URD. The requirements include technical design criteria, and associated methodology, for ALWR emergency planning in the areas of containment performance and offsite dose. This paper focuses on containment performance. A summary of the emergency planning containment performance design criterion is as follows:

> Plant design characteristics and features shall be provided to preclude core damage sequences which could bypass containment and to establish capability for withstanding core damage sequence loads. Con-

tainment loads representing those associated with low pressure core damage sequences shall not exceed ASME Service Level C/Unity Factored Load limits. Accident sequences will be shown not to result in loads exceeding those limits for about 24 hours or longer; beyond about 24 hours, there shall be no uncontrolled release.

The URD severe accident requirements in support of the containment performance design criterion were developed from a deterministic perspective. First, a set of design characteristics and features was defined to address severe accident containment challenges. A comprehensive set of twenty-three potential severe accident challenges was identified based on systematic consideration of past PRAs, operating experience, severe accident research, and unique design aspects of the ALWR. The first 13 challenges represented events which could occur independent of or precede core damage, such as bypass accidents. The remaining 10 challenges could occur as a result of a severe accident, such as containment pressure loads from a core damage event.

A systematic evaluation of the URD was performed to assess the degree to which each of the 23 potential challenges was addressed in the requirements.[2] This systematic evaluation included a challenge by challenge assessment of the requirements for both the passive PWR and the passive BWR. The adequacy of the integrated set of URD requirements to prevent and mitigate each containment challenge was reviewed in the context of the risk posed by the challenge (considering experience with current plants and its applicability to passive ALWRs) and those passive plant features which were being provided to ensure improved performance in addressing the challenge. The overall adequacy of the URD requirements in each instance was confirmed by engineering judgment based on available risk studies, the nature of the systems available to address the challenge (e.g., redundance, independence, passive nature), and the time available for possible backup operator response. This evaluation concluded that potential challenges, regardless of the low likelihood of severe accidents for these designs, have been explicitly and effectively addressed in the URD.

Second, the results of this systematic evaluation were applied to establish the types of severe accident sequences for which containment response should be evaluated against the Service Level C/Unity Factored Load limits as specified in the containment performance design criterion. This is necessary since a number of accident sequence types (i.e., major classes of sequences) are potentially precluded or otherwise impacted by design. The conclusion from this work is that the one sequence type most worthy of evaluation for these designs is a low RCS pressure core melt into an intact containment with containment systems functioning as designed.[2]

To supplement this deterministic perspective, the URD also requires application of PRA to confirm that the appropriate severe accident sequence characteristics are being considered in the evaluation of containment response against the Service Level C/Unity Factored Load limits. From a probabilistic perspective, the URD requires that functional sequence types with frequency greater than approximately 10^{-7} per year be evaluated for containment response. This 10^{-7} per year frequency threshold for sequence types to be evaluated for containment response is consistent with the NUREG 1420[3] limit for insignificant risks and with previous regulatory guidance (e.g., Standard Review Plan guidance to evaluate potential accidents from hazards in the vicinity of the plant site which exceed approximately 10^{-7} per year). Also, consideration of functional sequence types greater than approximately 10^{-7} per year helps provide assurance that the cumulative effects of such sequence types will not exceed the 10^{-6}, 1 rem PRA goal for offsite consequences.

The probabilistic evaluation serves an overview or checking role in part because the frequencies of interest are near the practical limits of reliable quantification. The 10^{-7} criterion is applied as a screening tool to focus particular attention on those sequence types whose best estimate frequency is higher, but sequence types below the threshold are also subject to review to ensure that the reasons why they were predicted to be of low frequency are identified and confirmed to be consistent with both the capabilities of the design and the current understanding of severe accident phenomenology and the relevant uncertainties.

Preliminary PRA results indicate that the only sequence types, if any, with frequency greater than approximately 10^{-7} per year are low pressure core melts into an intact containment with containment systems functioning as designed, similar to the sequence type defined from the deterministic evaluation above.

The preceding discussion focussed on the utility requirements for ALWRs which were developed in collaboration with the reactor vendors, architect engineers, and international participants. The NRC has formally reviewed the passive URD and has developed their own policy positions[4] on ALWR severe accident requirements for both evolutionary and passive plants. These positions provide a basis for resolution of all issues for the passive plants and, in our judgment, these resolutions should suffice to support ALWR emergency planning in the following way: while the NRC has yet to take any position on emergency planning, ALWR designs which satisfy the NRC policy positions should qualify for consideration as members of the class of reactor designs eligible to utilize updated emergency planning requirements.

The NRC criteria most pertinent to emergency planning require assurance of containment integrity for about 24 hours, given credible severe accident challenges, and no uncontrolled release thereafter. As this example illustrates the emphasis is placed on preventing early containment failure and, accordingly, substantial time would be available for any necessary offsite protective actions. To illustrate further, the NRC criterion allows for the possibility that ex-vessel core debris might accumulate in an uncoolable configuration, but that design features would nevertheless prevent containment failure for about 24 hours and mitigate the impacts of any subsequent releases. This approach is consistent with the results to date from the MACE testing program,[5] which is continuing in order to further reduce the uncertainty regarding the prospects for core debris cooling, assuming relocation of debris to the floor of the reactor cavity/lower drywell and flooding per ALWR requirements.

Evolutionary ALWRs are proceeding with design certification utilizing these NRC policy positions. The evolutionary plant designers have supplied the NRC-required analyses to demonstrate containment integrity for approximately 24 hours and, further, that there will be no uncontrolled releases beyond this time. The U.S. Department of Energy's Advanced Reactor Severe Accident Program (ARSAP) has evaluated both the uncertainties affecting the ALWR containment challenge conclusions and the potential for enhancing the issue resolutions proffered by the NRC for passive ALWRs and has concluded that incremental efforts on issue resolution are warranted.

ARSAP has elected to employ the Risk-Oriented Accident Analysis Methodology (ROAAM) to probe the effectiveness of design features that are now seen to afford the greatest promise for terminating severe accidents safely (but earlier in the accident progression) and to strengthen the deterministic basis for concluding that specific related challenges to containment integrity have been effectively addressed by passive ALWR designs. The ROAAM methodology provides a framework to discriminate between significant and unimportant uncertainties and guides the design of supplementary investigations which were seen to be needed to complete or support enhanced issue resolution on a sound basis. The strength of the technique in promoting issue resolution is evident in the record of its prior applications.[6,7] The emphasis on decomposing the issue and identifying the controlling physics was seen to afford the appropriate elements in the right sequence to guide the process of creating a new resolution strategy effectively.

The ongoing efforts employing the ROAAM methodology focus on the capability of the passive ALWRs to retain a damaged core within the reactor vessel, to accommodate potential ex-vessel steam explosion loads, and to ensure effective cooling of ex-vessel debris. These tasks were selected based on their significance to the assured severe accident capability of passive ALWR designs.

III. SUMMARY OF CURRENT ISSUE RESOLUTION ACTIVITIES

A. Application of ROAAM for Passive ALWR In-Vessel Retention

While the ALWR Requirements had been developed with an emphasis on promoting ex-vessel debris coolability, evidence has begun to accumulate that retention of core debris in the reactor vessel may be expected given the particularly favorable features of the passive PWR design, in particular. The concept of in-vessel retention has been advanced for the Loviisa plant in Finland[8] and, more generally, as an accident management strategic objective to be considered in Individual Plant Examinations for existing reactors in the US.[9] The capability for demonstrating in-vessel retention (IVR) in the passive PWR is based on the following particularly favorable design features of the one available design, the AP600: a comparatively low power density which limits the local heat flux in a debris pool; a captive, large inventory of coolant in the containment which passively initiates flooding of the reactor lower head for any sequence in which core damage might occur; provision to operate manual valves ensuring complete cavity flooding for all sequences; stand-off insulation supported from the cavity wall ensuring an ample flow channel for vessel cooling; no penetrations in the lower head which would otherwise raise additional issues in demonstrating lower head integrity; internal design features which govern the mechanism for debris relocation to the lower head and thereby help to ensure no dynamic overloading; and a passive containment cooling system with reflux of the condensate to the reactor cavity.

The ROAAM effort for these design features requires an assessment of the heat transfer characteristics of the submerged reactor pressure vessel lower head in core damage events to determine whether failure of the reactor vessel and release of the core debris into the reactor cavity can be prevented. A ROAAM framework was constructed for the principal, quasi-steady evaluation which emphasizes the determination of local heat fluxes imposed on the lower head, comparison with the heat flux that can be accommodated by the surrounding water, prediction of possible wall thinning considering material interactions and melting, and demonstration that the residual wall thickness will accommodate all imposed loads. The framework quantification is in process and is drawing upon available information[10,11] and new supporting experimental and analytical investigations.

The ARSAP activities in support of IVR begin with analyses being performed by Argonne National Laboratory to establish and document conditions for melt relocation within the reactor vessel, including masses, flow rates, composition, and superheat. These evaluations provide the boundary conditions for in-vessel retention and for evaluation of the magnitude of potential in-vessel explosions (discussed further below) which might jeopardize the integrity of the lower head during early, transient conditions. Much larger debris quantities, up to 100%, are evaluated under later, quasi-steady conditions.

ARSAP is also conducting and interpreting experiments in two facilities to establish the capability to transfer heat from a reactor vessel lower head to surrounding water. The Cylindrical Boiling (CYBL) facility at Sandia National Laboratory affords a unique capability to demonstrate and directly observe external boiling at heat flux levels above expected values, but below CHF on the lowermost region of the lower head in 3-D at nearly full scale. The ULPU-2000 experimental loop at UCSB is also unique in affording the capability to achieve CHF and observe the mechanism leading to CHF in an approximately full-scale 2-D slice of a passive PWR lower head and vessel wall.

The CYBL facility had been designed and constructed to support the New Production Reactor program and thus the vessel is torispherical while the passive PWR uses a hemispherical shape. Given the availability of the facility, however, tests were run to observe the three dimensional boiling process near the bottom of the lower head where the lesser degree of curvature in CYBL was judged to represent the hemispherical lower head in a passive PWR in a conservative manner. Two CYBL tests have been run to demonstrate heat removal from the bottom of the lower head at heat flux levels somewhat above the heat fluxes predicted to be imposed there for a passive ALWR. CYBL tests were run with a uniform and an edge-skewed power distribution with the average heat flux as high as 200 kw/m^2 on the downward facing surface. The observed boiling in CYBL is particularly dramatic when viewed from directly below the vessel as symmetric waves of vapor spread from the center and condense with a fast, regular period. The boiling process in CYBL appears to be both axisymmetric with intermittent bubble formation and collapse and fully consistent with the boiling process observed in ULPU.

ULPU tests performed to date with a hemispherical lower head have provided CHF data for a range of reactor conditions and insights into the mechanisms involved. The two dimensional facility is designed to represent the vapor flow path along the lower head at full scale and to permit power shaping to reproduce local conditions at any location of interest on the three dimensional lower head. A thick copper block is used to represent the steel lower head; the heat flux in the block can be as high as 2000 kw/m^2; the facility height of ~7 m can represent any degree of gravity-induced subcooling; and both pool boiling and natural circulation can be modeled with variable inlet and exit flow restrictions. Typical values of measured CHF include ~450 kw/m^2 at the bottom center of the lower head and ~1 MW/m^2 along the upper 30° angle of the lower head even under pool boiling conditions (thus prototypic reactor values would be higher). The flow regimes and apparent mechanism of critical heat flux have been observed to change from the lowermost region where the buoyancy forces are nearly perpendicular to the flow path and the upper edge of the lower head where they are nearly aligned. Even a small amount of subcooling was found to dramatically increase CHF, while a polished surface appears to be detrimental.

While the preceding experiments focus on the heat removal capability from the outside of the lower head, other efforts focus on predicting the heat flux that may be imposed from the debris pool on the inside. ARSAP and the Electric Power Research Institute are jointly sponsoring ongoing experiments at UCLA to characterize corium pool behavior at high Rayleigh numbers using simulant materials at small scale in hemispherical geometry.[10] These tests employ Freon-113 as the debris simulant in Pyrex bell jars at increasing scales to achieve the target Rayleigh numbers. Freon-113 was selected considering appropriate similarity laws and because early tests demonstrated that microwave heating produced uniform volumetric power deposition in this particular fluid. Experiments initially used water cooling and an adiabatic pool top surface, but the first top crust simulation was run recently and liquid nitrogen cooling experiments are also planned. The UCLA experimental results have been represented in terms of the relative heat flux imposed on the inside surface as a function of the angular deviation along the hemispherical bottom. These results have been compared with available results from the two-dimensional COPO experiments conducted in Finland[11] and were found to be quite consistent when the weighing of the COPO data is recalculated for the hemispherical shape. Further data are being collected in both experiments and will be evaluated as they become available. Meanwhile, the consistency observed to date affords reasonable confidence that the relationships which are being used for the current evaluation, are reasonable.

To evaluate the implications of the heat flux data collected both for the inside corium pool and the outside water, an integral thermal hydraulic model was constructed. The model represents the interior corium pool based on its total heat generation and the heat flux distribution from the UCLA/COPO experiments. The local thickness of the oxidic material crust is calculated in the model, including its heat generation and thermal resistance. Conductivity through the lower head is modelled as is the heat removal capability of the surrounding water based on the ULPU data. Wall thinning is modeled if the predicted inside surface temperature would otherwise exceed either the steel melting temperature or the effective melting temperature of the steel considering possible internal eutectic formations, whichever is lower. The model can include a separated metal layer atop the molten pool and only with a metal layer are material interactions (i.e. eutectics affecting effective wall melting temperature) important. The model is suitable for conducting sensitivity studies to assess the relative importance of the various parameters.

Preliminary evaluations with the model predicted no wall thinning at most locations. Thinning was predicted to be possible near the top of the debris pool or along the edge of a metal layer. The extent of possible thinning was bounded based on a minimum eutectic temperature of ~1335°C at about half of the initial wall thickness (initial thickness ~15 cm or ~6 in.) even when parameters were varied over broad ranges. This residual thickness would ensure a substantial thickness at temperatures below the creep range and thereby preclude vessel failure for the low pressure conditions that will exist in a passive PWR at this stage of an accident sequence. While the preliminary evaluation indicated large margins to failure, efforts are underway to develop applicable structural response distributions that would represent best-estimate failure frequencies as a function of applied load and residual wall thickness.

Finally, ARSAP is analyzing in-vessel steam explosions to confirm that the expected relocation of debris to the lower head can not trigger an explosion of sufficient magnitude to cause lower head failure and thereby defeat the retention strategy. For the debris relocation conditions presently estimated, there are large margins to failure in the initial calculations. Overall, the planned effort on in-vessel retention is more than 50% complete and the results to date are encouraging, indicating that lower head failure would not occur in the passive PWR.

With in-vessel retention and passive containment cooling, containment integrity would be maintained indefinitely, notwithstanding a severe accident, and the need for even long term emergency planning protective actions is correspondingly reduced. With this effort, ARSAP is focusing on key technical issues of generic interest to the industry in the context of the specific application to a passive PWR, a particularly favorable design.

B. Application of ROAAM for Passive ALWR Ex-Vessel Explosions

Assuming that a severe accident could progress to vessel failure and debris discharge, the aspect of ex-vessel debris-water interactions must be considered since at least partial preflooding of the lower drywell/reactor cavity can occur for current passive ALWR designs. ARSAP elected to focus the evaluation of ex-vessel explosions on the passive BWR design, since this design is less ideally suited for in-vessel retention of core debris as there is a substantial volume below the reactor that makes assured deep flooding more difficult. Assuming failure of the lower head caused by core debris that relocated within the vessel, hot, possibly superheated, debris exiting the vessel may fall unimpeded for sufficient distance to acquire a significant velocity and fragment as it mixes with the water which may be somewhat subcooled. For these conditions steam explosions are not precluded and thus may challenge the adjacent structures. Recognizing this potential hazard, the current passive BWR design, the Simplified BWR or SBWR, affords a substantial corium shield structure to absorb any loads and protect the containment boundary.

The evaluation of ex-vessel steam explosions was undertaken to determine whether the available technological base was or could be made sufficient to permit reasonable quantification with the ROAAM methodology. If such a quantification could be performed, it was expected to confirm the shield design (and thus demonstrate that early containment failure had been prevented as it must be to support ALWR emergency planning). The ROAAM framework developed to guide the evaluation of ex-vessel steam explosions requires evaluation of the range of initial conditions for both the core melt and the water in the lower drywell. The causal relationships are then to be quantified and linked for the premixed melt mass (and corresponding void fractions) versus time as a function of the pour area, the explosion peak pressure and structural impulse as a function of the premixed mass, and the fragility of the surrounding structures in terms of failure frequency as a function of impulse magnitude. Uncertainty regarding trigger magnitude may be incorporated as a splinter (parallel) scenario for later evaluation. Further, there is some evidence that the debris-water impact will control the system response under those conditions least favorable for void formation and this hypothesis will be explored and incorporated as appropriate. This effort is drawing upon ongoing experimental and analytical investigations.

The ARSAP activities in support of the evaluation of ex-vessel explosions build on work performed previously at the University of California in Santa Barbara (UCSB). First, the NRC had sponsored work to enhance the demonstration that alpha mode failure of reactor containments was not a significant safety concern.[12] That effort included experiments and related analyses to demonstrate that premixing considerations placed a fundamental limit on the potential in-vessel explosion magnitude. This was true because the experiments had been designed to demonstrate that more extensive premixing of fuel debris and saturated coolant in the reactor lower plenum necessarily entailed the creation of additional voids in the mixture. Thus, as the quantity of potential reactants (i.e. premixed fragmented debris and water) increased with the potential to increase explosion magnitude, the accompanying formation of voids served to limit the work potential and became dominant. The net effect was that a bound could be placed on the effective work potential thus making alpha-mode failure too improbable to warrant further consideration based on known limitations in the controlling phenomenology.

Premixing phenomena were important for our purposes also since premixing would be a consideration in ex-vessel explosions. More importantly, however, the effort demonstrated an approach in which effects that had been hypothesized were investigated experimentally and incorporated into validated models in sufficient detail to provide a fundamental limitation on the potential explosion effects.

Prior to initiating ARSAP work, UCSB reported[13] on another significant development, the preliminary demonstration of the ESPROSE.m analytical model which incorporated for the first time a representation of the actual debris-water interactions in a propagating explosion. The mechanism, termed "microinteractions", had been observed during the alpha-mode research in single drop shock tube experiments. The shock tube facility, SIGMA, had been explicitly designed to allow observation of the kinetics of fragmentation in a valid simulation of the steam explosion environment, including the pressure level. ESPROSE.m is a two-dimensional explosion analytical model, with the "m" standing for microinteractions. This model was initiated to capture the observed partial coupling of debris and water and succeeded in doing so in a manner which appeared to hold great promise for establishing a more phenomenologically correct calculational capability for explosive behavior than had ever been available before. The partial coupling was captured by using three fields in the model: the fuel debris particles; the water; and the microinteraction zone which is specified to permit fuel and water to interact in the proper proportions within the node. The two dimensional capability of ESPROSE.m was also judged to be important for ex-vessel steam explosions where the geometry involved, particularly for shallow water pools, is unambiguously two dimensional. This two-dimensional behavior was seen to afford a new fundamental limitation on the explosive damage potential as discussed further below.

In the final pre-ARSAP task, UCSB had performed experiments for the New Production Reactor program[14] in which they succeeded in getting single drops of aluminum to react vigorously in the shock tube apparatus, SIGMA. These vigorous reactions included substantial oxidation occurring at a threshold of 200-300 bar shock pressure and 1500-1600°C drop temperature. The extension of the shock tube temperature range above the previous ~1000°C limit and the evidence of exothermic oxidation as a contributor to the drop response were viewed for our purposes as constructive steps toward establishing the capability to confirm drop behavior at conditions closer to those of prototypic fuel debris.

ARSAP planned its activities to build on each of these developments. Early ARSAP efforts focussed on development and validation of the analytical tools to be used for ex-vessel explosions. The preliminary ESPROSE.m model from the OECD FCI meeting[13] was reviewed, tested, and revised until the first operational version was established in October 1993. The model is representing the microinteractions with an entrainment factor of 3-4, values which are conservative based on the SIGMA experiments. Demonstration calculations verified that this version discriminated reasonably between KROTOS-21 (with 1000°C tin), which was at most mildly explosive, and KROTOS-28 (with ~2400°C aluminum oxide) which yielded a supercritical detonation.

Our interest in the ESPROSE.m calculations, however, focussed on the effects of two-dimensional behavior and of the free surface. These effects, which are termed "explosion venting", are a newly identified key behavior and the next fundamental limitation on the explosion potential.[13] Explosion venting applies to open pool geometries with a large aspect ratio. Calculations were performed for pools of one and three meter depths, assuming subcooled water, that illustrated substantial relieving of explosive energy at the pool free surface resulting in significant reductions in the impulse loading on the surrounding structures. The explosion venting refers to both the direct venting in the explosion zone of the driving component of the load and the reflections of pressure waves off the free surface of the pool which mitigate the transmitted component of the load. Note that the possibility of water subcooling is a factor which increases the potential ex-vessel explosion loads because it reduces the rapid void formation during premixing that was credited in the alpha-mode evaluation[12]. The increased initial pressure that occurs near the center with lessened void formation can be offset by the free surface effects for shallow pools, however.

The continuing ARSAP effort is planned to include an evaluation of debris pour conditions by Argonne National Laboratory extending through vessel failure and discharge. A separate evaluation will establish the possible water level and water temperature range in the lower drywell prior to vessel failure. The evaluations are just beginning, however.

Two experimental efforts have been planned, principally to confirm the present models. The first is an upgrade of the SIGMA shock tube facility which has been designated as "SIGMA-2000". These tests, as discussed above, are designed to determine how an individual droplet of premixed debris would respond to an arriving shock wave in a simulated explosion environment. The way that the droplet fragments and intermixes with the surrounding coolant is the key to determining the explosive propagation potential. The observed interaction during these tests can be used to calibrate the microinteractions model which is then able to predict the rapid heat transfer needed to develop a true explosion. The upgrade to the SIGMA facility involves a new droplet heater and release mechanism capable of raising the initial droplet temperature to 2000°C. Improvements have also been made in the imaging of the fragmenting droplet following arrival of the imposed shock pressure. Planned tests will be conducted for a range of coolant conditions ranging from subcooling to partial voiding. An initial test has been performed using tin at 1800°C and a much greater degree of fragmentation was observed after the test than had been seen in earlier tests at lower temperatures.

The second planned experiment involves an upgrade to the MAGICO premixing facility which has been designated as "MAGICO-2000". These tests involve the release of preheated spheres into a water pool to observe their dispersion in the water (premixing) and the accompanying void formation. The upgrade of this facility involves a new heater and release design affording sphere temperatures up to 2000°C and release of the spheres at greater heights above the water pool than in previous experiments. Planned tests with aluminum oxide and zirconium oxide spheres are being prepredicted with the PM-ALPHA premixing model and the results are expected to confirm its extrapolation to higher temperatures from prior experiments performed at ~1000°C. Predicted displacement of water at the point of pool entry, given greater initial velocity due to increased drop height, will also be verified.

A recent hypothesis[15] involves the final fundamental mechanism that may be important in bounding the explosion potential for debris entering subcooled pools at relatively high velocity. The hypothesis focusses on the effect of the impact of the debris at the free surface on the formation of a vapor blanket around individual debris particles. The coherent collapse of such blankets is generally understood to be the key mechanism driving an explosion. The current hypothesis is that impact velocity and subcooling can combine to limit the formation of these vapor blankets, leading to small scale but intense interactions which serve to fragment and quench the debris without permitting the formation of the larger coarse debris premixtures necessary for a significant explosion. Tests in MAGICO-2000 will be designed to provide evidence needed to further evaluate this hypothesis. Its possible effect in limiting explosion magnitudes has been termed the "penetration cutoff" since debris-water interaction during the initial pool penetration is expected to limit ("cutoff") the explosive potential.

While previous efforts on explosions have characterized the impact on structures in terms of integral work potential, the present effort yields time dependent pressure loadings (impulses) amenable to dynamic structural analysis. Corresponding criteria for structural capability are being developed.

The results are expected to confirm that the corium shield in the SBWR lower drywell effectively protects the containment boundary from failure due to ex-vessel steam explosions which may occur if hot debris is discharged from the reactor vessel and the lower drywell is partially flooded. This evidence, if obtained as expected, will confirm that the passive BWR containment will withstand credible early challenges and thus also confirm the case for ALWR emergency planning. With this effort, ARSAP is focusing on key technical issues of generic interest to the industry in the context of the specific application to the passive BWR where the potential interaction of hot debris and water under depressurized conditions must be addressed in the design certification review underway.

C. Application of ROAAM for Passive ALWR Ex-Vessel Cooling

Notwithstanding the above described effort on in-vessel retention, ARSAP remains interested in developing a better understanding of the potential for effective ex-vessel debris cooling. An analysis of ex-vessel debris cooling in the current passive BWR (SBWR) was recently performed and reported by the vendor and ARSAP.[16] That evaluation drew upon available evidence and included judgments necessary to complete preliminary quantification of a decomposition event tree which captured the importance of multiple stages of debris release on the cooling phenomenology. The analysis illustrates the expectation that the MACE testing configuration, involving a liquid debris layer with water on top, applies to only a fraction of the core debris in any realistic evaluation (a "high" initial discharge of 20-60% of the core debris was assigned only a 10% probability). The planned next MACE test (designated "M3") will involve 20 cm of debris at much larger scale than previous tests (120x120 cm vs. prior 50x50 cm) and may reveal the phenomenology which controls the coolability of such a layer for prototypic reactor conditions. Without such evidence, any strategy for assuring debris cooling that could involve such a debris layer cannot be evaluated with sufficient certainty to draw firm conclusions about debris cooling. Thus, ARSAP's effort directly addressing ex-vessel debris cooling has focussed on supporting this test, to be conducted in 1994. Some effort is being expended on alternative concepts for ensuring that debris would be coolable.

IV. CONCLUSIONS

Prior efforts, which have been previously reported, establish the severe accident containment performance capabilities in ALWRs that ensure mitigation and thus sufficient time for any emergency planning response during an event. This capability is the foundation for significant changes in emergency planning requirements. Recently issued NRC policy documents establish their criteria for evaluating the containment performance of these designs on a consistent basis. These criteria, to be shown to be satisfied by ALWRs during the design certification process, should in our judgment qualify these designs for inclusion in a class of reactors eligible to utilize simplified emergency planning, commensurate with the demonstrated reduction in severe accident risk.

In key areas, the ALWR Program, the reactor vendors, and the DOE-sponsored Advanced Reactor Severe Accident Program are continuing efforts to strengthen the issue resolution basis for passive ALWRS. Two specific efforts discussed in this paper address the expected capability to retain a damaged core in-vessel given deep-flooding of the reactor cavity (the investigation is based on a passive PWR design like the AP600) and efforts to complete and validate new calculational tools to demonstrate that passive ALWR designs can accommodate potential ex-vessel steam explosion loads without failure (the investigation is based on a passive BWR design like the SBWR). If successful, these efforts will facilitate the case for ALWR emergency planning and reduce the likelihood that offsite protective actions would ever be needed for these designs even in the event of a severe accident.

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