Attachment 1

VOLUME 1 - TECHNICAL AND PROJECT MANAGEMENT PROPOSAL

SEVERE ACCIDENT INDEPENDENT ASSESSMENT (PHENOMENOLOGY)

A Proposal for:

U. S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Rockville, Maryland

In Response to:

RFP No. RS-RES-92-046

November 1991

By:

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Rockville, Maryland

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1. INTRODUCTION

The physical and chemical processes governing the progression of severe accidents and radiological releases in Light Water Reactors (LWRs) are very complex and subject to large uncertainties.

As part of the development of criteria for reactor siting, the first attempt to estimate the fission product source terms from a severe accident was made in connection with a formula developed in the late 1940's by the Reactor Safety Committee in the United States. In this prescription, it was assumed that 100% of noble gases, and 50% of the other fission products would be released to the environment by a severe accident that destroyed the reactor core. This formula, which was based on fallout patterns of atmospheric weapons tests, did not credit containment, an engineered safety feature adopted later.

Subsequently, the U. S. Atomic Energy Commission developed and proposed the concept of Design Basis Accidents (DBAs). In this concept, the most severe accident was defined as a Loss of Coolant Accident (LOCA) in the largest pipe, at the same time, although contradictory in principle, a source term was also defined to help establish the criteria for licensing plant sites. Estimates of fission product release were based on data that had been obtained by burning uranium metal in air; the estimates were also consistent with observations from the accident at Windscale in England. This is the approach used for reactor siting and licensing, as embodied in the current reactor siting criteria of 10 CFR Part 100.

The Reactor Safety Study (RSS), WASH-1400, was the first systematic and comprehensive evaluation of the risk associated with the operation of light water nuclear power plants. RSS analysts considered a wide scope of accidents, including those beyond the design basis accident envelope (severe accidents).

The accident at Three Mile Island in 1979 refocused attention on accidents that could lead to severe core damage. Both the NRC and nuclear industry undertook major programs on severe accidents, containment performance, and fission product release and transport behavior. These studies together with several plant-specific Probabilistic Risk Assessments (PRAs) provided many insights, including the determination that the best way to address the importance of severe accident issues was on a plant-by-plant basis rather than a generic approach through rule-making. This was articulated in the Commission's Severe Accident Policy Statement in August, 1985 (50 FR 32138).

This policy directive has led to the requirement for Individual Plant Examinations (IPEs), which are intended to identify plant-specific vulnerabilities to severe accidents and determine whether they can be fixed with low-cost procedural or hardware improvements. Along with this policy action, a significant amount of relevant information has become available from the NRC-sponsored research on severe accidents, containments, and source terms, and that research has been integrated into NUREG-1150 [1] for five LWR containment types.

As part of the integration plan for closure of severe accident issues, the NRC has devised a program [2,3] consisting of the major elements shown in Figure 1.1 and discussed below:

- Individual Plant Examination for vulnerabilities to severe accidents initiated from internal events (IPE), and external events (IPEEE),
- A Severe Accident Research Program (SARP) to support the other main elements of the integration plan, to continue developing the information in the areas of most significance to reactor risk and to continue confirmatory research, including phenomena resolution, code assessment and model validation,
- A program to ensure that licensees develop and implement Severe Accident Management programs at their plants,
- Upgrading of NRC and industry programs to improve plant operations,
- Implementation of recommendations resulting from the Containment Performance Improvement (CPI) Program.

A revised Severe Accident Research Plan (SARP) has been developed which addresses (1) issues pertinent to implementation of the IPE and CPI parts of the Integration Plan (near-term issues), and (2) conduct of confirmatory research needed to guide accident management programs and Commission's regulatory decision on severe accident issues (long-term issues). Table 1.1 provides a summary of the short and long-term severe accident issues.

Uncertainties associated with the outstanding degraded core issues, including core melt progression, fuel coolant interactions, fission product behavior, core concrete interactions, direct containment heating, and hydrogen behavior could impact the potential risk for both existing LWRs and future generation Advanced LWRs (ALWRs). The concern for these uncertainties is reflected in NUREG-1365 (the Revised Severe Accident Research Plan), NUREG-0956, NUREG-1150,

and several other recent publications. A team consisting of Energy Research, Inc. and several university subcontractors and consultants propose to support the NRC in its integration of experimental and analytical results of severe accident programs with NRC's severe accident research applicable to the new and unique features of ALWRs. In addition, this team proposes to provide general technical support to NRC in its evaluation of other severe accident and source term related issues.

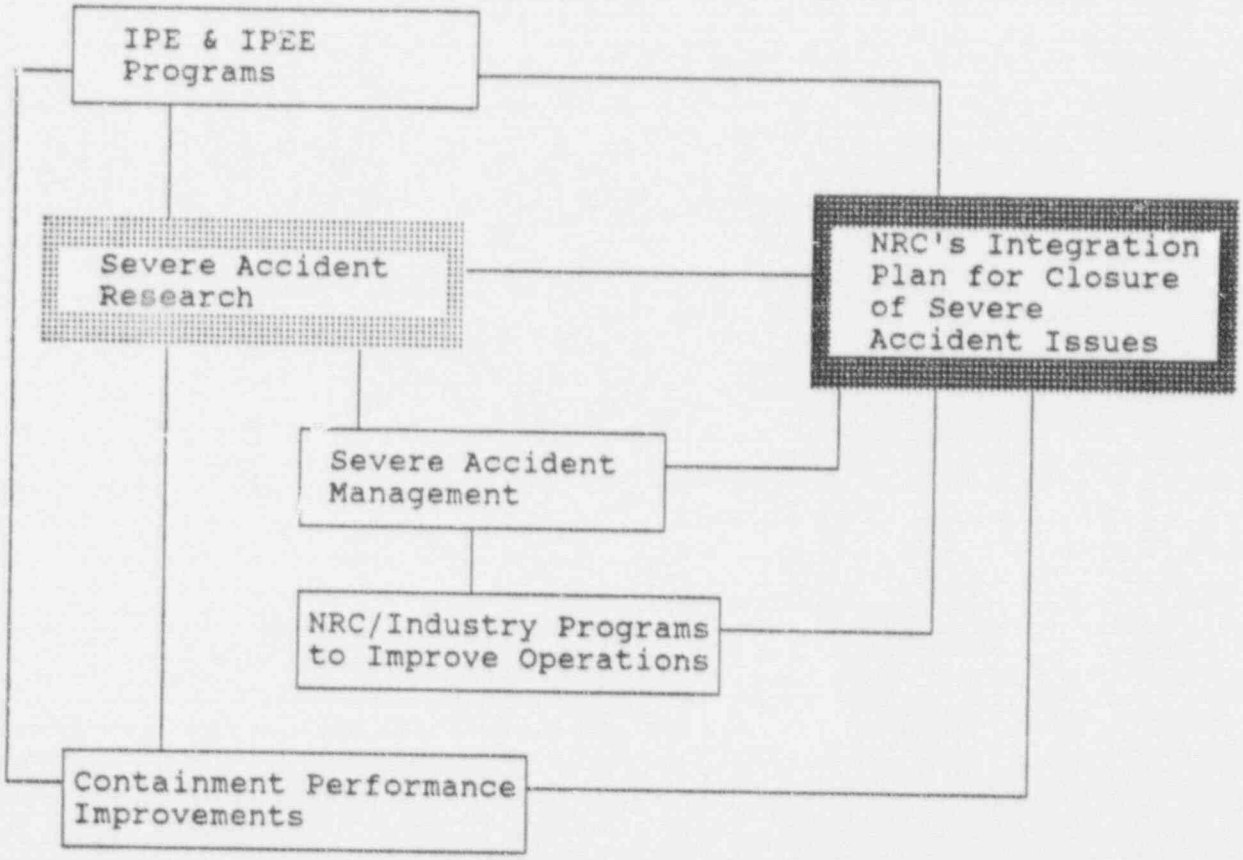

Figure 1.1 Relationship of the NRC's Integration Plan to Severe Accident Research

Table 1.1 Summary of NRC's Short and Long Term Severe Accident Issues

Issue	Description
Short Term	DCH & RCS Depressurization BWR/Mark-I Shell Attack Adding Water to Degraded Cores Scaling analysis
Long Term	In-Vessel Core Melt Progression Modeling Severe Accidents Hydrogen Transport and Combustion Molten Core-Concrete Interaction Fission Product Behavior and Transport Data Needs

2. REVIEW OF SEVERE ACCIDENT PHENOMENA

This chapter provides a brief review of the dominant severe accident and physical phenomena that affect severe accident progression for both Boiling Water Reactors (BWRs) and Pressurized Water Reactors (PWRs). The dominant and uncertain phenomena which impact on early containment failure and accident management actions are presented.

2.1 Cladding Ballooning

Fission gas produced during normal reactor operation fills the gap between the fuel and the cladding. The rise in temperature during a severe accident causes the pressure of the fission gas and the fill gas within the rod to exceed the system pressure outside the rod, and at the same time causes the strength of the cladding to decrease. The cladding balloons away from the fuel pellet stack and causes rod failure by cladding rupture. Cladding rupture usually occurs at 1050-1400 K [4]. Cladding ballooning can impact natural circulation flows between the core and the upper plenum and thus produce an enhancement in the heat-up of the core [5]. Ballooning and rupture allow steam to enter the fuel rods and cause oxidation of the inside surface of the cladding. Fuel pellets crack as a result of power changes during normal operation and hence the cladding/oxide and the fuel pellets can come into close contact. The contact between the fuel pellets and cladding leads to the formation of a continuous solid solution of UO_2 and ZrO_2 later in the course of accident. The influence of clad ballooning and rupture on melt propagation in the early stage of the accident is a source of uncertainty in severe accident analysis.

2.2 Metal Oxidation

The major impacts of Zircalloy oxidation on severe accident progression include; (a) enhancement of the heat-up rate, (b) molten cladding hold-up by the oxide shell, (c) embrittlement of cladding and (d) increase in the debris melting temperature [6]. The enhancement of the heat-up rate is the result of exothermic nature of the reaction (6.5 KJ/Kg of Zr oxidized, [7]). The heat-up rate due to decay heat is calculated to be 0.4 to 0.7 K/s for high and intermediate pressure transients [5]. But the heat-up rate was shown to increase to 1 to 2 K/s at 1300 K and 10 to 20 K/s for the LOFT experiments [4]. Oxidation of Zircalloy increases its melting temperature from 2030 K for Zircalloy to 2250 K for Oxygen saturated Alpha Zircalloy and then to 2970 K for ZrO_2 [7]. Also UO_2 and ZrO_2 form a continuous solid solution that has a melting temperature of 2800 K which is 300 K below the melting point of UO_2

[4].The shell of ZrO_2 on the surface of the cladding was found to holdup molten Zircalloy in PBF tests [4], thus preventing melt relocation. The breakup of the Zircalloy through the shell is being studied in the CORA test series.

Thermal shock failure of free standing embrittled fuel rods upon reflood leads to the formation of debris beds. Criteria for embrittlement of Zircalloy by oxidation have been reported in the literature [8]. The formation of debris beds due to thermal shock failure was confirmed in the PBF scoping experiment and the LOFT FP-2 test. The potential for significant enhancement in heat-up rate following reflood of a partially damaged core was observed as part of the LOFT-FP2 experiments. Accident management strategies being considered always involved water addition to a degraded core. Uncertainties remain as to the potential consequences of water injection on coolability, hydrogen production, shattering and energetic interactions.

2.3 Hydrogen Generation

During the early stages of the accident, Zircalloy oxidation is well understood, with oxygen diffusion through a ZrO_2 surface layer controlling parabolic reaction kinetics. Once Zircalloy melting starts and fuel dissolution commences, destruction of the protective ZrO_2 layer tends to accelerate the reaction. On the other hand, melt formation (formation of $(ZR,U)O_2$) leads to reduced Zircalloy surface-to-volume ratio, which would tend to decrease the hydrogen generation. The following conclusions are made from the results of the PBF-SFD, the NRU-FLHT and the ACRR-DF tests [7]. Zircalloy exhibits three melting points depending on its oxidation state and lattice structure, namely 2150 K for beta-Zircalloy, 2250 K for alpha-Zr(O) and 2970 K for ZrO_2. Thus if good fuel-cladding contact occurs, fuel liquefaction will commence at 2170 K. All of the above mentioned tests show that most of the hydrogen generated occurred after 2170 K with the fraction oxidized varying from 85-95%. Hence, it can be concluded that oxidation occurs primarily during and after melting of fuel. More detailed assessment of the available experimental data should provide additional insights into the hydrogen generation potential especially during reflood. This issue is expected to have a significant impact on potential accident management strategies for existing and advanced LWRs.

2.4 Eutectic Interactions at Low Temperatures

Formation of liquid phases by the eutectic interactions of core materials can lead to the initiation of core melt progression at temperatures well below the melting temperatures of the fuel and

the cladding. The first liquids to form and relocate are those formed by the reaction of Inconel spacer grids and the fuel rod Zircalloy cladding. Eutectic reaction of iron and nickel with zirconium leads to melt formation. The oxide layer on the surface of Zircalloy prevents the eutectic reaction initially, but this oxide dissolves rapidly in Zircalloy at temperatures above 1500 K. Only small amounts of iron and nickel are required to dissolve large quantities of zirconium.

PWRs use stainless steel clad Ag-In-Cd control rods housed in Zircalloy tubes. The control rod melts at 1050 K [6]. Cadmium exerts a vapor pressure that causes the stainless steel clad to balloon and contact Zircalloy. The eutectic reaction between stainless steel and Zircalloy is similar to the one between Inconel and Zircalloy, and liquids are formed by eutectic interaction at 1500 K. However, cadmium ballooning occurs only for low pressure scenarios; for higher pressures control rod failure occurs only after stainless steel melts and contacts the Zircalloy at 1700 K [9].

BWRs utilize control blades consisting of B_4C powder in stainless steel tubes which are located in the spaces between the Zircalloy channel boxes that surround the fuel assemblies. B_4C reacts rapidly with stainless steel at 1500 K to cause blade failure and liquid formation [6]. CORA tests [10] have shown that the liquid so formed penetrates the Zircalloy channel boxes and attacks the Zircalloy cladding of the fuel rods. In some PWRs, B_4C is used as a control material and it is enclosed in Al_2O_3 and contained in Zircalloy cladding. The Zircalloy cladding fails due to chemical interaction with Al_2O_3 at 1620 K [11]. A number of integral experiments have shown that spacer grids, control rods and control blades interact with Zircalloy cladding and initiate melting at temperatures as low as 1500 K [10,11].

Eutectic interactions at low temperatures have the potential for reduced releases of fission products during the in-vessel phase of the accident (most notably more refractory species of Sr and Ba) and early relocation of core material with potential influence on the subsequent progression of events.

2.5 Melting/Freezing and Core Blockage Formation

Metallic melts formed by eutectic interactions of spacer grids, fuel rod cladding and control materials move downward in the core. The tight, open, PWR core lattice and the steep decrease of power level (and hence decay heat levels) at the axial and radial peripheries of the core cause the melts to solidify, forming blockages in the flow channels between fuel rods. The freezing

temperature of the melts vary from 1070 K for Ag-In-Cd alloys to 1420 K for Fe-C eutectic [4]. Metallic blockages are often formed at spacer grids and have been observed in integral tests such as PBF SFD 1-4 tests, LOFT FP-2 test and the CORA experiments [6]. Analysis indicates that if a spacer grid is at a temperature below the freezing point of the relocating melt, it can strongly influence the trapping of the debris [12]. The lower blockage formed in TMI-2 was found to be 9 cm thick and almost completely filled the flow channels between the fuel rods [6]. The metallic blockages were also found to have UO_2 crystals suggesting that some fuel was melted and relocated even in this low-temperature melt.

Core blockage formation has implications for natural circulation flows during high pressure scenarios (PWRs), flow diversion, zirconium oxidations, material relocation and in-vessel fuel coolant interaction processes.

2.6 Fuel Liquefaction, Melt Pool and Crust Formation

At temperatures well below its melting point, UO_2 can be liquefied by dissolution in molten Zircalloy [6]. Zircalloy-rich eutectic formed by clad-spacer grid interactions can dissolve some UO_2 even at 1500 K [13], and experiments have shown complete dissolution at 2525 K [14]. The liquefied fuel accumulates on the top of the metallic blockage and refreezes. The ceramic material on the metallic crucible heats up because of the decay heat and a molten pool is formed within the ceramic crust. A large pool involving 45% of the core was formed in the TMI-2 accident. Similar, but smaller accumulations of ceramic melts representing the early stage of pool formation were observed in PBF SFD and LOFT FP-2 tests [4]. This melt pool was primarily composed of $(U,Zr)O_2$ with transition oxides such as Cr_2O_3 and Fe_3O_4 [6].

A very important observation concerning the early stage of core melt progression is that the process is not coherent. This is due to temperature variation in the core. Thus, as the ceramic pool is being formed atop a metallic blockage, other regions of the core can melt and relocate into the melt pool. The ceramic pool formation process is dynamic. The pool increases in size as more and more material melts into it and the crusts are thermally attacked by the naturally occurring melt pool and by the heat-up of their fuel constituents. The former mechanism causes heat to be transferred laterally and the latter mechanism is dominant for pool growth in the downward direction. The lower crust is thicker than the upper crust due to the higher thermal conductivity of the metallic constituents of the blockage. Melt propagation in a debris bed and crust formation have been studied by a number of

authors[15,16] but there is a need for further model development and assessment of experimental results.

2.7 Molten Pool Relocation Behavior and Crust Stability

The molten pool can break through the crust and attack the lower core plate and the lower plenum. In the TMI-2 accident, the crust failed in the side of the molten pool and approximately 20 tons of melt flowed into the lower plenum. Integral effects tests have not attained long enough times at high temperatures to produce information about this late phase phenomenon. A qualitative understanding of the thermal-hydraulic characteristics of the melt pool with natural circulation is needed in order to model the system behavior over a wide range of accident conditions so that the threshold and the location of the failure of the metallic crust can be determined.

The maximum pool superheat and the crust heat flux distribution were calculated by Epstein and Fauske [17]. This analysis used correlations from data that have Rayleigh numbers of about 10^{10} -- typical of laminar flow (near transition to turbulence) in horizontal layers. The pool superheat was determined to be 250 K. However, severe accident conditions have much higher Rayleigh numbers with pool turbulence. An analysis for turbulent natural convection in a molten pool was performed by O'Brien and Hawkes [18]. Their analysis used a finite element model of pool natural circulation using a K-e model for turbulence. Their results showed that the pool superheat predicted using the turbulence model was about 12 K, suggesting that the pool was well mixed by turbulence. The highest sideward local heat flux was found to occur in lateral location at an azimuthal angle of about 50 degrees from the bottom. However, the applicability of the K-e model to natural convection in a molten pool is questionable, and recent experimental results extending well into the turbulent regime have revealed good correlation to the previous data in the laminar regime.

This solution cannot be readily extended to the corium pool crust break-through problem, since the solution is only valid for the pools surrounded by stainless steel side wall with external nucleate boiling. Still, it can be inferred that natural convection in the molten pool preferentially transfers heat to the lateral boundaries of the molten pool. If the failure of the crust surrounding the molten pool is assumed to be determined by thermal stability criterion, then a balance between the heat transfer from the molten pool to the inner surface of the lower crust and the heat rejection properties of the porous bed surrounding the lower crust determines thermal stability. If the heat transfer from the molten pool exceeds the heat that can be rejected by the

surrounding porous bed, then the crust melts, relocates and refreezes. The crust melt/freeze cycle continues until the breakthrough occurs at the core barrel or the core support plate. Once again, applying the results of O'Brien and Hawkes literally, it is noted that heat flux at the bottom of the pool is negligible, but the maximum lateral heat flux is about 40 W/cm^2 for decay power of the order of 1 MW/M^3 [18]. This value can be compared to the boiling or natural convection heat flux in the porous bed outside the crust to determine the location of crust failure. The location of crust failure is important to determine what fraction of the molten pool flows into the lower plenum.

2.8 Relocation and Slump in BWRs

BWRs are fitted with an Automatic Depressurization System (ADS) that upon actuation causes rapid opening of several Safety/Relief Valves (SRVs). The BWR Emergency Procedures Guidelines direct the operators, under severe accident conditions, to manually actuate the ADS when the core becomes partially uncovered (but before any serious core damage has occurred). Rapid depressurization of the reactor pressure vessel leads to flashing and loss of inventory with subsequent core plate dryout. Without restoration of coolant injection to the reactor vessel, the subsequent core heat-up and degradation occurs in a dry environment with low steam flow rates. Under these conditions, the contribution of Zirconium oxidation to the core heat up is small. Large fractions of the core would melt simultaneously and relocate downwards by candling process and by the rapid relocation of the molten control blade and the channel box structural material (stainless steel and zirconium, respectively). With current information, it is not possible to determine whether the relocated metallic melt would fall on the lower core plate or forms a frozen crust above the plate. If the metallic melts were to form a blockage above the plate, debris bed formation and ceramic pool growth would lead to a scenario described earlier. However, if the downward motion of the molten core leads to accumulation of debris on the core plate, heat transfer from the relocated core material increases the temperature of the core plate and leads to the rapid failure of the core plate. Subsequently, the failed core plate and accumulated metallic melts drain into the lower plenum. It is expected that the fuel pellet columns, encased in ZrO_2 sheaths would be still standing since the weight of the fuel is supported by the control rod guide tubes and not by the core plate [19]. After the failure of the core plate, the additional relocating material falls directly into the lower plenum.

It should be noted here that there are few models available in the literature that predict retention and buildup of core debris above

the core plate for cases in which the core plate is sufficiently cooled by the reactor vessel injection to forestall dryout, heat-up and structural failure. This scenario is likely for all but the prolonged accident sequences such as station blackout with the total loss of injection.

A major uncertainty in our understanding of severe accident phenomenology is the question of metallic melt drainage or core blockage. This branch point has a large effect upon the characteristics of the melt released from the core into the lower plenum. The accident sequence envisioned above can occur not only in BWRs but also during PWR accident sequences involving dry conditions (pump seal LOCA)[19]. In the case of metallic melt drainage, layers of quenched melts are formed in the lower plenum water in the order of their time of melting and drainage. The control rod alloys are expected to be found at the bottom. Oxidized Zr-Fe and Zr-Ni, with some control material and the ceramics such $(U\text{-}ZR)O_2$, would be at the top interspersed with Zr-Fe and Zr-Ni.

2.9 FCI Inside Lower Plenum

The ceramic melt that pours into the lower plenum causes relatively little melting of the core support structures through which it flows because of the self insulating nature of the ceramic crusts [4]. The lower plenum in TMI-2 was filled with water, and because of the interactions with water, the relocating material broke up into a coolable debris geometry.

The coolability of the melt by water in the lower plenum can be subdivided into three stages: (1) melt jet breakup (with or without explosions), (2) freezing of melt droplets, and (3) coolablity of the debris that collects under water in the lower plenum.

The interaction of the melt jet that drains from the ceramic pool into the lower plenum water is of interest in determining the potential for steam explosion induced containment failure (Alpha-mode). This subject is controversial and has been studied extensively [20,21]. Alpha mode explosions are believed to have a very low probability and are not found to contribute significantly to the early containment failure in the NUREG-1150 study [1]. Instead, the quenching behavior of small side-pours by coolant in the lower plenum is of higher importance to both accident phenomenology and accident management issues.

Breakup of melt jets by coolant has been experimentally studied by Saito, et al. [22], and is currently being studied at the University of Illinois, under Department of Energy sponsorship. However there exists a need to extrapolate these results to severe

accident scenarios. The melt jet-coolant interaction problem is a three-layer (melt/steam/water) Kelvin Helmholtz stability problem. The leading edge layer was shown to play an important role in the breakup of the melt jet by Wang, et al [23]. The droplet diameter formed by melt jet breakup which is obtained by stability analysis is of importance for assessing debris bed coolability. The coolability of the debris beds has been studied widely [47,48].

Theofanous and co-workers [20], through the use of a detailed probabilistic analysis have concluded that the Alpha-mode containment failure has a probability of occurrence of 10^{-4} per core melt. Some of the mitigative factors include natural circulation in the vessel which causes heating of the upper bolts. These bolts fail upon impact from a slug from below at low energies. Other factors include high system pressure. Above 10 bars, an explosion would need an external trigger. Secondly the efficiency of the detonation would be substantially reduced by the intervening structure and by unmelted core fractions. There is also additional concern about potentially explosive melt-coolant interaction in the lower plenum where a layer of less dense coolant overlies a pool of more dense hot melt separated by a thin vapor film. In the stratified mode of contact, spontaneous explosions have been reported in the literature. Triggered explosions have been studied by Bang and Corradini [24]. Stratified explosions are not energetically significant.

2.10 Heat Transfer to Lower Head and Lower Head Failure

The location and mode of lower vessel head failure has been the subject of several studies. Four mechanisms have been identified by Rempe, et al., [25] as having the potential to cause the lower vessel head failure. These include (1) melt penetration induced failure of a lower head penetration, (2) ejection of a lower head penetration by heating or by pressure, (3) global failure of the lower head, and (4) jet impingement induced lower head failure.

BWR vessels have up to 200 penetrations; PWR vessels have anywhere from no penetrations to 50 penetrations. The penetrations can fail by thermal attack of the melt or debris in the lower plenum. Lower vessel failure is generally believed to be governed by penetration failure (except for some PWRs where there are no lower head penetrations). Calculations performed at SNL and EPRI using finite element analysis codes have calculated vessel rupture solely based on vessel temperature distributions and the ultimate strength of the head material, and only when the system pressure exceeded 120 bars [25].

Conduction heat transfer models for vessel thermal response have shown that the thermal fronts due to debris heating will penetrate both PWR and BWR vessels, but the BWR vessel response time will be much longer (thicker walls). Calculations for the in-vessel tube heat-up and failure have shown that the ratio of tube mass to debris mass governs the in-vessel tube failure. Since the BWR vessels have a larger number of penetrations, higher temperature debris is needed to cause in-vessel failure. However, tubes in either pressure vessel are expected to fail if subjected to debris temperatures in excess of 2200 K [25].

Melt penetration distances have been calculated for ceramic and metallic melts in the penetrations using a 'bulk freezing' model and a 'conduction' model. The bulk freezing model [26] assumes that turbulence prevents formation of a stable crust at the walls. The energy loss required to freeze the melt is equated to the turbulent convective heat transfer from the melt to the wall in order to arrive at the penetration length. The conduction model assumes that the crust formation at the wall governs the heat transfer. Once the frozen layer closes at the channel center, flow ceases and the remaining melt inside the channel freezes. Results from the conduction model [27] indicate that the melt penetrates beyond the vessel for both BWR and PWR. But the bulk freezing model calculations show that the melt will penetrate only through the larger penetrations such as BWR drain nozzles. Neither of the above models have been verified experimentally using typical LWR core materials and vessel/tube materials.

For depressurized conditions, thermal equilibrium calculations indicate that for penetrations with large diameters (such as the BWR nozzles), failure occurs after the melt penetrates beyond the vessel wall.

To summarize, the failure mode of the reactor pressure vessel is the most important problem in the field of accident phenomenology and accident management, since the vessel failure mode has a considerable impact on subsequent progression of events prior to containment failure.

2.11 Natural Circulation and Induced Ruptures

During the later stages of the accident in a PWR, the coolant inventory is depleted. Due to the axial and radial variations of power (and hence decay heat) the heat transfer to steam results in less dense fluid near the core center. The buoyancy driven flow pattern thus established has the hot fluid rising at the middle of the core and being replaced by the cooler fluid from the periphery. There are three possible and important natural circulation loops,

namely; (1) core and upper plenum natural circulation, (2) counter-current flows in the hot leg and steam generators, and (3) flow through natural recirculation. The net result of these natural circulation flows is transportation and redistribution of thermal energy and fission product aerosols and vapors to cooler RCS structures, such as the upper plenum internals, hot leg piping, pressurizer surge line and steam generator tubes. Ultimately the hot leg or some other piping can fail due to creep rupture or even melting prior to lower vessel head failure.

The RCS boundary failure leads to the depressurization of the RCS and ejection of corium at lower pressures which mitigates the effect of DCH. On the other hand, having the RPV at low pressure increases the potential for energetic fuel coolant interactions. In addition, natural circulation-induced steam generator tube rupture could lead to fission product releases involving containment bypass. Other implications include, the extent of cladding oxidation, fission product release, and retention on RCS structural surfaces.

The temperature-induced steam generator tube rupture has not been investigated widely. It is believed that vigorous natural circulation would cause failure above the tube sheet in the steam generator, whereas slow natural circulation in the hot legs would cause creep rupture in the hot leg. The stratified flow in the hot leg piping would lead to a combination of pressure and thermal gradients causing both longitudinal and circumferential bending. This would lead to creep at the top of the pipe while the bottom remains essentially elastic. But detailed analyses of the creep rupture phenomena require detailed knowledge of temperature distribution in the hot leg and the steam generators. In the absence of such detail, it is currently believed that the actual creep rupture induced failure point would be in the heat affected zone of the pipe to nozzle or pipe to safe end weldments [28].

It should be pointed out here that all of our knowledge of natural circulation is based on computer codes [29,30,31] and from the results of one scale experiment using simulant fluid [32]. No integral experiments have been performed (a new effort is also underway at the University of Maryland) with the use of steam-hydrogen mixtures which are the gases found in severe accidents. Condensation of steam in hot leg, upper plenum and possibly in the steam generators and formation of a stable stratified layer of hydrogen might prevent any natural circulation to the upper plenum and to the hot leg.

2.12 High Pressure Melt Ejection / Direct Containment Heating

In certain accidents such as those initiated by station blackout or a small break LOCA, degradation of the core can occur while the reactor coolant system is still at a high pressure. When the bottom head of the reactor vessel is breached in such accidents, the core melt will be ejected under pressure. The ejected materials will be dispersed out of the cavity at high pressure into the surrounding control volumes as particles, transferring heat to the atmosphere. The metallic components such as zirconium and steel, can react with oxygen and steam in the atmosphere and the resulting energy generation can pressurize the containment further. A large amount of aerosols can be generated in a high pressure melt ejection. If containment failure were to occur, a large release of radioactive materials would result. Dispersing core debris could also serve as an ignition source to promote reaction of H_2 with O_2 even if the hydrogen concentration is below the conventional flammability limit. Hydrogen burning would generate more energy and raise the pressure and the temperature inside the containment. Models and supporting experimental data are not currently available to provide insights into these potential release modes.

A number of experimental studies have been conducted with a view to obtaining data concerning the DCH phenomena. The Corium/Water Thermal Interaction (CWTI)[33] experiments at ANL have investigated phenomena involved in corium dispersal and direct containment heating. Other experiments (examples include SNL/SURTSEY-DCH, and BNL tests) have also been performed. Questions have been raised regarding the prototypic nature of these experiments as well as whether a geometric scaling is sufficient to simulate nuclear plant conditions.

A number of analyses have been performed using both systemic codes such as CONTAIN [34] and parametric codes developed in ANL [35] and BNL to analyze these experiments and to simulate DCH scenario in the severe accidents. However, none of these codes have properly accounted for radiative transfer from radiatively participating gases (primarily water vapor) and aerosols. Radiative transfer from clouds of aerosol particles in the containment may mitigate DCH by radiating heat to the containment walls and other structures and thereby cool the gases and reduce the pressure in the system. The extinction coefficient and single scattering albedo of gas-aerosol mixtures are important model parameters [35] and need to be determined for realistic modeling of radiation transfer in the containment atmosphere, particularly from corium aerosols as they are suddenly injected into the containment. In current codes, the presence of aerosols in the atmosphere is effectively ignored by assuming the atmosphere containing corium aerosols is opaque. This

is unrealistic and conservative. There exists a need for a systematic comparison of the experimental data and for improved modeling, including accounting for radiative transfer, to provide closure for the DCH problem.

2.13 Ex-Vessel Fuel-Coolant Interactions

The time period of fuel-coolant interaction is anywhere from the time of lower head failure to the time when a stable coolable ex-vessel configuration is achieved. The contact mode for ex-vessel FCIs depends on the particular mode of lower head failure. As discussed earlier, penetration failures are found to be more likely except when the system pressure is high at time of vessel failure. At low system pressure, assuming penetration failures because of the small breach area and release rates, only a small portion of the melt can be mixed with any available water in the cavity. A fully flooded cavity offers two extreme outcomes. If the melt jet is broken up by the water, the break-up leads to fine scale particles of millimeter size which are coolable by water. At the other extreme, if an energetic FCI occurs, the water will be expelled by the interaction leaving a water depleted cavity. Thus, the melt will be found on the concrete with an overlying coolant layer, with core-concrete interactions occurring below and fuel-coolant interactions above. The coolablity of the stratified melt-coolant regime is an area of uncertainty. It is believed that the melt-coolant interface is covered by a thin crust which prevents coolant ingress. For both the stratified melt layer and the debris bed configurations, the lack of coolablity leads to core-concrete interactions.

2.14 Core-Concrete Interactions

When the reactor vessel fails in low pressure scenarios, high temperature core debris may fall into the reactor cavity and interact with structural concrete. The heat flux from the molten pool is sufficient to decompose the concrete and melt the residual oxides. Water vapor is released from the conrete and from hydroxides. The concrete erosion rates are in the order of centimeters per hour [37]. The molten oxides and the molten steel from the reinforcing bars present in the concrete are added to the ceramic oxides and metallic Zirconium and steel already present in the core debris. The gases evolved by decomposing concrete oxidize at the temperature of the molten pool and are be reduced to hydrogen and carbon monoxide. The oxidizing reactions are also exothermic, adding to the decay heat from the pool. The gases released from the bottom of the pool rise through the pool as bubbles. The process is usually modelled as an analogue to film boiling, using a Taylor instability model [38]. Gases released at

the side of the molten pool are assumed to form a vapor film between the melt and concrete. The presence of the gas bubbles swells the pool, increasing the pool depth and the surface area of interaction with concrete. The rising gas bubbles also result in the production of aerosols containing fission products stripped from the debris. The fission products thus evolved may have only a minor effect upon depletion of total pool inventories. However, some of the species ma' be completely removed.

The model in CORCON-Mod2 [40], the state-of-the-art code, assumes that the various oxidic phases in the melt are highly miscible as are the metallic species; but the the oxide and metallic phases are assumed to be separated into two distinct layers. Bouyancy forces are sufficient to separate the two phases, even in the presence of vigorous gas bubbling due to evolving gases by concrete decomposition. If the oxidic phases are denser than the metallic phase, tho fuel oxide becomes mixed with the concrete oxide until all the oxides form a single layer above the metal. If water is present in the cavity, it will form an additional layer on the top of the pool. It is suggested that the water does not react violently with the molten layer but acts as a heat sink to cool the melt to a temperature below that of the solidification point of the melt. SURC tests show that the presence of water over the melt does not significantly alter the attack of concrete by the melt [41]. As time progresses, the pool grows, surface area increases, and decay heating decreases. Pool temperature and heat fluxes decrease, and possibility of refreezing arises. Substantial refreezing of the melt phase may occur. The question of the permeability and stability of the crusts to gases has not yet been resolved. Model deficiency areas have been identified as part of a recent review of the MELCOR computer code [52]. Limited data is also available to assess the fission product aerosol removal capability of shallow water pools overlying molten or partially molten debris pools.

2.15 Combustion Processes

Combustible gases produced by in-vessel and ex-vessel metal oxidation will likely result in combustion events in the containment or in the secondary building. Combustion can lead to containment or secondary building failure by over-pressurization due to deflagration or detonation, missile generation, and equipment failure due to thermal or pressure effects. Combustion can also affect the source term by altering the chemical state of fission products and mechanical resuspension of fission products from containment and structural surfaces. Uncertainties associated with hydrogen combustion involve (1) hydrogen transport and mixing, (2) deflagration, (3) flame acceleration and transition from deflagration to detonation, (4) detonations and (5) diffusion

flames [41]. Radiative transfer is an effective mode of cooling combustion products as the gases have strong infrared absorption bands and can radiate heat to the cold walls of the containment [39]. One of the deficiencies of the current conbustion models in integrated computer codes such as MELCOR is that they do not treat heat loss by radiation from the combustion products and, therefore, do not take credit for decrease in temperature of the gases in the containment.

2.15.1 Hydrogen Transport and Mixing

The hydrogen transport and mixing in an LWR Containment are sensitive to the hydrogen injection rate, the availability of forced circulation or turbulence, the plant geometry and the initial temperature distribution in the containment. If fans are operating, there will be good and rapid mixing in the containment. Containment sprays also promote turbulence. Analyses and test data suggest that for conditions similar to large break LOCAs, fairly good atmospheric mixing is expected. However, under certain accident conditions potential for hydrogen stratification and improper mixing exist.

Available computer codes are either based on lumped-parameter, one dimensional, control volume/flow path approach (HECTR, CONTAIN, MELCOR, etc.), or multi-dimensional models (HMS, COMMIX, etc.). Simulation of hydrogen/steam mixing under various accident conditions using these computer codes are unreliable (lack adequate models for condensation in the presence of non-condensables, numerical diffusion, etc.), thus requiring special attention. Future ALWRs will rely on natural circulation for decay heat rejection from the containment boundaries. Therefore, effects of hydrogen stratification and mixing need to be evaluated carefully, using either regional mixing models or models based on more advanced numerical solution techniques.

One very serious limitation of the one- and multi-dimensional models is that they do not account for the presence of a noncondensable gas such as hydrogen in the primary coolant system. For example, the COMMIX code is not capable of considering a two-component coolant such as a steam-hydrogen mixture in the RCS. This inability to account for hydrogen production, introduction, and diffusive and advective transport in the primary collant system greatly impacts on the ability of the code to simulate the thermal-hydraulics behavior, including thermal stratification and segregation of hydrogen in the reactor vessel, coolant-to-structure heat transfer in the reactor vessel and steam generator of a PWR plant as well as natural circulation in the primary coolant system.

2.15.2 Deflagration

Deflagration is a form of combustion in which the flame travels at subsonic speed relative to the unburned gas. Unburned gas is heated to reaction temperature by thermal conduction and mass diffusion from the hot burned gases. Deflagration is the most likely combustion mode in degraded core accidents. A general understanding of parameters that affect hydrogen deflagration has existed for several years. The lower flammability limits for hydrogen in air saturated with water vapor at room temperatures have been determined experimentally [42] and were found to reduce with increasing temperature. The effect of steam is to inert the mixture. The volume concentration of steam required to inert the mixture increases with temperature. Much less information has been obtained concerning the effects of carbon monoxide and aerosols on the combustion behavior.

Deflagration results only if a weak source serves to ignite the mixture. However, there exists several accident scenarios such as direct containment heating where the two gases need not be mixed sufficiently in a compartment to lead to flammable mixtures. Oxidation can occur volumetrically, The increasing temperature due to volumetric oxidation results in attaining auto-ignition temperatures. Auto-ignition is a volumetric oxidation process and does not require a flame or spark to occur [42].

2.15.3 Detonation

Flame acceleration occurs during the propagation of flame around obstacles. The flame front is distorted by the obstacle, establishing eddies or vortices that broaden the flame front and yield a high volumetric rate of heat generation. More air is entrained by the flame, and the burning rate and the flame front velocity increase. Flame speeds can be increased from a few meters per second to a few hundred meters per second depending on the distance travelled and the number of obstacles encountered [42]. Turbulence induced by fans can lead to flame acceleration if the requisite hydrogen concentration is available. Hydrogen detonations involve the reaction of hydrogen through the supersonic propagation of a burning zone or combustion wave. Detonation of hydrogen can potentially threaten containment integrity and equipment survivability. The detonation limits for premixed hydrogen-air mixtures widen with temperature, similar to conventional flammability limits.

2.15.4 Diffusion Flame

It is possible that hydrogen may burn simply as a diffusion flame.

A diffusion flame is one in which the burning rate is controlled by the rate of mixing of oxygen and fuel [42]. For the hydrogen to burn in this manner, it is necessary that at some locations, the mixture be within flammability limits. The diffusion flame can begin because of flashback from an outside ignition source or because the mixture temperature is above the spontaneous ignition temperature. Diffusion flames may occur in a number of scenarios of interest in reactor safety such as a small break LOCA, during HPME/DCH, or in-cavity oxidation of gases produced by core concrete interactions.

2.16 Fission Product Release

Small quantities of fission products created inside the fuel matrix are released from fuel pellets during normal operation. These fission products will reside in the gas between the fuel pellets and cladding. Typical gap inventories include 3-10% of noble gases and 2-5% of volatile fission products such as cesium iodide [43].

The cladding temperature at which the fission products are released from the gap is 1200 K [4]. But the gap release is not a significant contributor to severe accident source terms. Fission products are also released from the fuel as it heats up in a core melt accident due to vaporization and other thermally activated process resulting from the heat-up of the fuel. The relevant diffusive processes inside the fuel are atomic diffusion within the fuel grains, surface and molecular diffusion within the open porosities and the molecular diffusion within the gap and cladding breaches [44]. The noble gases and volatile fission products are released by diffusion to grain boundaries in the fuel where they collect as bubbles. In high burn-up fuel, the accumulation of bubbles during normal operation causes tunnels of interconnected porosity to form that greatly enhance the release rate of volatile fission products and the noble gases from the fuel during accidents. The noble gases and fission products are released in large quantities as the fuel grain structure is destroyed by fracture or liquefaction. A number of models have been developed to predict the fission product release rate from the fuel. The most important factors that govern the release rates are the maximum temperature of the fuel and the composition of steam/ hydrogen mixture that passes through the core region.

The actual release rate for each fission product is equal to the product of the fractional release multiplied by the initial local inventory of each radionuclide. The time dependent fractional release rate is correlated with a temperature dependent diffusion constant that represents penetration of oxidant into UO_2 and the geometric data for the fuel pellet. The current fission product

release model uses an experimentally based first order kinetics expression with the rate constant assumed to take the form of a simple exponential fit (CORSOR) or an Arrhenius expression (CORSOR-M) [45]. The important shortcomings of the CORSOR-M model are that they do not take into account any effect of fuel cladding interaction effects, and they do not take into consideration the effect of equiaxed grain growth during heat-up. Due to the neglect of these effects, the CORSOR-M model was found to overpredict the release rate of PBF SFD tests by a factor of about four [44].

The diffusive release of fission products holds only for fuel in the solid state. In a core melt accident, the first liquids are formed at 1500 K. Changing the fuel to a liquid state enhances volatile fission product release since bubble migration in a liquid is much more rapid than in the solid phase [4]. The enhancement of fission product release by fuel dissolution in molten Zirconium has been measured in several out-of-pile tests [6], especially above the melting point of Zircalloy (2100 to 2250 K). The noble gas fractional release rates measured have shown a rapid increase at fuel temperatures of 1570 K and then a levelling-off at about 2200 K [4]. An analysis of large bubbles in a molten pool has shown that the volatile fission products should decrease very rapidly. But the behavior of medium and low volatility fission products in ceramic pools can be understood in terms of the oxygen potential of the melt [6]. Strontium and cerium are stable as low volatility oxides and present in the ceramic melt at full inventory level. Other medium and low volatility products such as antimony and ruthenium are found in metallic melts at full inventory. Cesium is primarily present as CsOH and iodine is found principally as CsI aerosol[28].

The chemical form of iodine within the fuel matrix is not known with confidence and some experimental evidence has cast doubt on the basic thermodynamic calculations that predict stabilization as CsI [41]. Reaction with boric acid and metal oxide surfaces present in high temperatures can generate volatile species such as HI and CH_3I which can decompose to I_2. More recent experimental data of Oak Ridge National Laboratory appears to confirm the I propensity to form CsI.

2.17 Fission Product Transport and Deposition

Fission products are transported inside the reactor coolant system along with the flow of steam and hydrogen both as vapors and aerosols. Fission product vapors can condense on colder RCS structures as well as on aerosol particles during their transport. Fission particles can agglomerate, settle on, chemically react with, and heat the RCS structures. This in turn can lead to revaporization of previously settled or condensed fission products

on the surfaces. The mass transfer models are used to simulate the fission product vapor condensation and evaporation from surfaces, but considerable uncertainties exist in the selection of diffusion constants. Temperature changes throughout the RCS, the temperature-time profile at any location, and identity and vapor pressure of the chemical species deposited at the walls are all of importance. The uncertainty in our knowledge of RCS structural temperatures also leads to uncertainties in fission product evaporation/condensation calculations. Uncertainties also exist in the understanding of fission product chemistry during transport within the RCS.

2.18 Ex-Vessel Fission Product Release

In addition to fission product release from the RCS, fission products can also be released from core debris that collects on the basemat following vessel breach. The core debris reacts with the concrete basemat leading to evolution of gases which can in turn react in their passage through the corium pool. Fission product vapors and aerosols are transported to the containment atmosphere via the sparging gases. Nonradioactive material are also released by vaporization from the melt as well as reaction of concrete so that the net release is a significant ex-vessel production of aerosols that include fission products.

The most widely used model is that based on the approach developed by Powers [46] and is incorporated in the VANESA computer code. The chemical reactions are based on an ideal, equilibrium model (unit chemical activity, equilibrium thermodynamic constants). Vapor species in the gas bubbles formed due to vaporization will exist in proportion to their vapor pressures. When the gas bubbles exit the melt, many of the vapor species will immediately condense to form aerosols. Aerosols can also be directly formed from the melt as a result of mechanical action when the bubbles break through the melt surface. The remainder of the volatile species (not released in-vessel) and additional quantities of Te and some refractory materials such as lanthanides and actinides are generally expected to be released during this phase of the accident. If an overlying layer of water is present, the gas bubbles will be scrubbed by random diffusion and inertial impaction. The temperature of the molten pool, the chemical activity of the various species and compounds, the flow rate and void fraction of gases and the subcooling of the overlying liquid layer (if present) will all play important roles in determining the radiological releases to containment atmosphere.

2.19 Fission Product Transport Within Containment

The fission products which escape retention inside the RCS (or in pressure suppression pools in BWRs) will be transported into the containment. Natural processes of agglomeration and deposition will lead to the retention of aerosols inside the containment. In addition spray action will dominate fission product and aerosol removal processes. The physical phenomena governing fission product transport inside the containment are (1) agglomeration (random motion, gravity and turbulence), (2) removal (random motion, gravity, movement to surfaces by steam condensation, thermophoresis, diffusiophoresis, ice beds and sprays), (3) steam condensation on to aerosols and (4) homogenous nucleation of water droplets in the atmosphere. The transport and deposition of volatile species may be important as the chemical form of iodine is not usually well known. (Recent experimental studies at Oak Ridge National Laboratory indicate more than 95% of iodine is expected to react and form CsI.) Hydrogen burning and direct containment heating can revaporize and transport aerosols in the containment. In addition, the direct coupling between the containment thermal hydraulics and fission product transport due to, for instance, condensation of steam on aerosols is important in assessing radiological releases.

2.20 Severe Accident Issues for ALWRs

For the most part, severe accident research results are expected to be applicable to ALWRs. However, improvements are needed in analytical capabilities in predicting natural convection flows under new and unique geometrical conditions. Issues of interest to ALWRs include (1) performance of steam injectors for ECCS injection into the reactor vessel in SBWRs under severe accident conditions, (2) containment cooling effectiveness (by natural circulation) and the effectiveness of natural circulation and mixing in any annulus water pools, (3) stability issues resulting from gravity driven water supplies for SBWRs during degraded core conditions, and (4) ex-containment cooling effectiveness of free falling water films. Other relevant issues may surface as the designs continue to mature and NRC's review of these advanced reactors continues. Differences in severe accident response and characteristics exist between the General Electric SBWR and the Westinghouse AP-600 reactors. Innovative design features may introduce unique conditions for which a detailed assessment of the current state-of-the-art methods and research results must be performed.

3. PROPOSAL SCOPE, OBJECTIVES AND APPROACH

3.1 Proposal Scope and Objectives

The objectives of the proposed activities listed in Section C of the RFP No. RS-RES-92-046 are summarized below.

TASK AREA A: SEVERE ACCIDENT PHENOMENA

Technical support services are proposed to help reduce the uncertainty in key severe accident and source term issues, especially for issues with a large impact on severe accident risk for both existing and advanced light water reactors. Specifically, the ERI team proposes to participate in

(1) Review of the analysis of available experimental results and comparison and evaluation of different experiments that appear to be at variance,

(2) Evaluation of severe accident phenomena in comparison to more mature understanding of similar phenomena in other related fields,

(3) Evaluation of scaling issues, experimental techniques, instrumentation, ranges and measurement methods,

(4) Modification and/or development of phenomenological models, computer codes and performance of separate effects experiments, particularly in support of new technologies for ALWRs.

The subtasks will include technical support on (a) Core Melt Progression, (b) Fuel-Coolant Interactions, (c) Fission Product Behavior, (d) Core-Concrete Interactions, (e) Direct Containment Heating, and (f) Hydrogen Behavior.

TASK AREA B: GENERAL TECHNICAL SUPPORT

General technical services are also proposed in support of potential NRC-directed peer reviews of SARP issues, experiments, analytical models and computer codes. As directed by NRC, selected ERI team members will

(1) Attend topical and research information exchange meetings,

(2) Participate in peer reviews,

(3) Perform scoping evaluations of uncertainties (particularly related to advanced reactor safety claims or to design differences), and

(4) Evaluate and retrieve data or information supplements from non-nuclear sources and experience.

3.2 Proposed Approach for Selected Issues

Research has been performed on several aspects of accident phenomenology for a number of years. There is a need to evaluate the performed research, including both experiments and analytical models to assess the applicability, validity and importance of the on-going model development and experimental effort to achieve a reduction in severe accident risks and uncertainty and to determine their impact on accident managment strategies.

Consideration of uncertainties can take many forms. Bounding analyses may be acceptable, provided the results are not seriously distorted. In other instances, uncertainties can be circumvented through design and operational changes and procedures to eliminate uncertainties (i.e., RCS Depressurization).

Methods have been developed for quantification, aggregation, and propagation of uncertainties [37] for a variety of physical processes and models; however, nonlinearities, strong covariances, and model complexities all require a detailed evaluation before they can be used for particular studies.

The objective of this proposed effort is to support the Commission in developing a comprehensive framework for resolution and closure of risk significant severe accident issues for both existing and advanced light water reactors. This obviously necessitates focusing attention on the "bottom-up" phenomenological issues, while maintaining an overall perspective on the key risk attributes. Therefore, integration of research results for use in regulatory decision-making is an important element of the present proposal.

3.2.1. Scaling Analysis and Methodology

Since it is not practical to perform full-scale severe accident tests, meaningful theoretical and experimental simulations should be made. The first step in studying complex phenomena is to determine the relevant dimensionless parameters that characterize the problem (i.e., perform a scaling analysis). The global

parameters are best obtained from the most general equations and boundary conditions that describe the process. These parameters can be interpreted in terms of force, energy, and time ratios, etc. These are then evaluated for the prototype situations. In this way the dominant physical aspects of the problem are directly identified and the negligible aspects are identified so that simplifications in the theoretical analyses can be made with quantitative criteria to indicate the order of the approximations. In this way, it is possible to relate explicitly any theoretical and experimental models to reality. The dimensionless parameters also indicate the required resolution for thermophysical properties. They are essential for experimental simulation of prototype situations (similarity modeling), help minimize the number of experiments required for a range of variables, and indicate optimum correlations of data and scaling relations. The qualitative nature of the flow and heat transfer can be identified by the use of the parameters which help identify the regions in which the experimental and numerical resolution must be greater. Further discrimination among various coupled phenomena can be made from time-scale analyses. Each of the phenomena possible in an accident has a time scale associated with it. By identifying all the various time scales and focusing on disparate time intervals from the shortest on, the essential couplings are identified. The problem becomes more easily tractable and avoids such problems as numerical matrix stiffening. In summary, the dimensionless parameters and time scales are essential for meaningful (numerical and experimental) model studies and yield considerable insight even before solutions are obtained.

An example where scaling methodolgy can be taken to good advantage is the in-vessel and ex-vessel natural circulation loops in a PWR. Under severe accident conditions, buoyancy driven natural circulation of steam and/or hydrogen form recirculating loops. A good start at analyzing such flows and identifying relevant dimensionless parameters has been made [49]. The work has identified the key system scaling, as well as important in-vessel, steam generator and hot-leg parameters. However that work did not account for the presence of hydrogen in the coolant mixture and did not account for the radiation heat exchange between the participating gas (steam or steam-hydrogen mixture) and the RCS structures. At high temperatures, radiation was shown to be as important or more important than convection in the core [50]. Scaling analysis can be used effectively in identifying phenomena during a severe accident and permits simplification of the analysis even if the phenomena occurring are complex. In addition, results of in-pile/out-of-pile experiments and analysis can also be assessed in light of geometric scaling of systems and dynamic scaling of phenomena. Very often controversies and large

uncertainties in final results arise because of lack of attention that is given to scaling. Uncertainties arising out of extrapolation of experimental and analytical results to larger scales will be quantified.

3.2.2. Core Melt Progression

Long-term needs in in-vessel core melt progression includes a careful review of the full-scale experiments performed in several laboratories including the PBF, ACRR, NRU-FLHT and CORA experiments. Needs exist for the development of simplified models for the candling phenomena and for careful modeling of melting caused by materials interactions. More importantly, there exists a need for integral models for late phase core melt phenomena. Existing models for late phase modeling such as DEBRIS module [15] have to be reviewed and benchmarked against results from ACRR and other experiments. The late phase melting and freezing models should be integrated with other phenomena such as heat transfer from the molten pool and crust erosion. Late phase melt progression models should also be modified to model the effect of water addition, natural circulation, etc. to be able to evaluate success criteria for PRAs and for accident management information needs.

3.2.3. Fuel-Coolant Interactions

In the last few years, the modeling capabilities for fuel-coolant interactions have improved vastly and several small and large scale experiments have also been completed. A number of models have also been developed. There exists a need to review the available models and their experimental bases. An independent assessment of both small-scale simulant experiments and large-scale experiments can also be performed.

It is becoming increasingly evident that the likelihood of FCI-induced early containment failure is relatively small. Therefore, the FCI research should be focused on other outstanding issues. These include, development of models for melt jet breakup in water, dispersion and freezing of molten droplets, potential for fuel-coolant explosion, and steam generation pressure pulses. These models and the codes should be used to assess the effect of reflooding degraded cores and for modeling the ex-vessel fuel-coolant interactions processes. Phenomenological models supported by small-scale experiments are also needed to assess the potential for aerosolization and enhanced fission product releases (oxidation mode) due to non-energetic FCIs.

The range of hydrogen and steam generated obtained from small scale studies for multi-dimensional quenching of debris beds composed of

a large fraction of Zircalloy particles (such as Wang and Dhir, [51]) and from large scale studies should be scaled to obtain the loads on reactor containment upon quenching. Conditions for the failure of melt supporting crusts upon addition of water can be resolved through the use of heat transfer models in turbulent, buoyant melt pools and the heat rejection properties of the crusts and the surrounding coolant.

3.2.4. Molten Core-Concrete Interactions

Specific topics of interest include melt spreading on the containment floor including the effect of water in the cavity, improvement in the models of quasi-steady state behavior of molten pool attack of concrete, erosion of concrete structures, the volatilization of fission products, and production of aerosols. Existing computer codes such as CORCON-MOD3 have to be reviewed and if needed, alternative phenomenological models have to be developed and assessed against the results of small-scale and large-scale experiments. Several shortcomings of the CORCON-MOD2 code were pointed out in the MELCOR peer review report [49]. Phenomena of crust formation, freezing, breakup, water penetration, fragmentation, quench, debris dryout, remelt, and solid debris-concrete interaction will need additional assessment. This is an important issue for both existing and advanced LWRs. Corium spreading and ex-vessel flooding is often credited for achieving stable and coolable configurations, especially for ALWRs.

3.2.5 Direct Containment Heating

Several experiments such as SURTSEY and HIPS experiments in SNL and CWTI experiments at ANL have been conducted. Results of the on-going severe accident scaling methodology should be used as a basis for any further activity. It is important to establish the validity of the linear scaling approach of the available experiments performed by both NRC and industry contractors. Parametric DCH models have been incorporated into codes such as CONTAIN, which need to be reviewed for technical adequacy. Simulant experiments performed in BNL have provided valuable data for important phenomena such as entrainment. Results from these experiments have not been incorporated into the codes. In addition, important phenomena such as heat transfer from a cloud of debris to atmosphere have been modelled parametrically. There is a need to develop models supported by separate effects experiments for prediction of mechanical aerosolization and fission product release during high pressure melt-ejection process.

3.2.6 Hydrogen Behavior

Experiments conducted in large-scale facilities and detailed computer codes developed to simulate hydrogen mixing and combustion need to be reviewed and compared to each other. In addition, a large amount of fundamental experimental data have been obtained in small-scale experiments. The knowledge obtained from such empirical data, for instance, the effect of increased temperature upon flammability limits, has not been incorporated into the codes. Effects of mixing, stratification and transport are difficult to assess using existing multi-dimensional codes (COMMIX and HMS). These codes suffer from numerical diffusion problems, making it difficult to distinguish between real physical behavior and numerical irregularities. Special purpose models, based on advanced numerical methods, have been developed in other fields and can be used to investigate these processes. Regional mixing models may also be applicable under certain conditions. This is an important issue for ALWRs with passive containment heat removal systems.

3.2.7 Fission Product Behavior

In-vessel releases of fission products have been modelled by empirical models such as the CORSOR-M code. The newer data obtained in the PBD SFD tests and the ORNL tests needs to be evaluated for possible incorporation into MELCOR and other severe accident codes. Fission product release models also do not take into account fuel melting and liquefaction. The volatile fission products release observations of PBF-FP2 are not predictable by current models (releases were observed to peak at 0.1% in several tests [3].) This is of some interest to accident management and strategies for degraded core reflood. Dissolution of medium and low volatility fission products in ceramic pools with metallic intrusions noted in experiments have to be correctly modelled. Fission product release behavior from molten pool (in-vessel) are not currently mechanistically based. Here again scoping analyses may provide an adequate basis for regulatory decision-making. Fission product removal characteristics of shallow overlying pool can be assessed via small scale experiments. Fission product scrubbing effectiveness of water appears to hold the key on ex-vessel accident management strategies. An analytical and experimental demonstration of fission product removal efficiency of water pools could by itself lead to closure of several important containment performance issues, namely, MARK-I shell melt-through, MARK-II downcomer pipe melt-through, and other containment bypass failure modes.

3.2.8. Specific Issues for ALWRs

Experiments are being conducted by Westinghouse and several universities that model some of the phenomena unique to ALWRs. These experiments have to be assessed, especially regarding the validity of the scaling approach. A large database of experimental results unrelated to ALWR designs exist which can provide information for code and model assessment. Specific small scale experiment and modeling needs for ALWRs include modeling of stability criteria for SBWRs, modeling of natural circulation of primary coolant and modeling of natural circulation in the containment and containment performance.

4. STAFFING TO MEET THE NEED OF THE PROPOSED CONTRACT

This section specifically addresses Evaluation Criterion 1 of RFP. No. RS-RES-92-046, "Qualification and Experience". Energy Research, Inc. has put together a team of recognized experts in severe accident phenomenology, containment performance, heat transfer, fluid dynamics, nuclear systems engineering, and risk assessment. This team will be available as needed to fulfill the objectives of the proposed contract over a period of 3 years. There is considerable flexibility in structuring the technical support responsibilities consistent with the specific needs of Tasks A and B of the RFP. In putting together this team, Energy Research, Inc. considered the following important attributes for a knowledgeable and diverse technical team:

- Broad and thorough knowledge of physical sciences
- Recognized expertise in respective fields
- Detailed knowledge and understanding of the state of the art in the areas of:
  - severe accident phenomenolgy
  - scaling methodology
  - numerical methods
  - accident management
  - severe accident and source term issues
  - severe accident computer codes
  - uncertainties and quantification techniques
- Knowledge of the IPE process and those areas of severe accident closure that will affect IPE and IPEEE such as the CPI Program, IDCOR/NRC issues, accident management, and regulatory decision-making

In this section the above attributes are addressed by presenting the team's technical qualifications and experience in the technical subject areas important to this proposed effort. The areas presented are taken directly from the RFP. No. RS-RES-92-046 requirements.

The team is composed of individuals from Energy Research, Inc., Cornell University, Massachusetts Institute of Technology, Purdue University, the University of California at Los Angeles, and Stanford University. This team is capable of performing a wide range of technical evaluations, experimentation, analysis, model development, computer simulation and peer review, depending on the task order.

4.1 Proposed Contractual Options

In order to meet the needs identified in RFP. No. RS-RES-92-046, Energy Research, Inc. proposes two possible options, depending on the NRC's requirements, as discussed below:

OPTION 1: Evaluations Requiring Quick and Short Term Response

Those task orders requiring quick and short term response will be treated in this category. Energy Research, Inc. in its capacity as the Prime will assemble a group of technical experts consisting of ERI technical staff members augmented by individual consultants from the six participating universities. The team will work in close cooperation with the NRC project manager to fulfill its obligations effectively and efficiently. Task areas that could be addressed under this option include:

- Evaluation of scaling issues and governing phenomena as related to existing, on-going and planned experiments,
- Reviews of analysis of experimental work already completed,
- Comparison and evaluation of different experiments,
- Evaluation of experimental techniques, instrumentation, ranges and methods,
- Participation in expert/peer review efforts,
- Review and assessment of mathematical models, computer codes, and benchmark analyses,
- Evaluation of uncertainties, methods, and interpretation of results,
- Evaluation of severe accident issues relevant to ALWRs, and
- Review of data and/or information supplement from non-nuclear experience.

OPTION 2: Evaluations Requiring Model Development and Experiments

Those task orders requiring model development and performance of confirmatory or new separate effects experiments which would usually require, in addition to ERI staff members, use of university laboratory and computational facilities and/or graduate research assistants will be undertaken under this option.

Energy Research, Inc. in close cooperation with NRC project manager will issue a subcontract to the university with the available expertise and laboratories to fulfill the task objectives. The universities also participating in this option include (1) Cornell, (2) MIT, (3) Purdue, and (4) UCLA.

Task areas addressed under this option could include:

- Performance and conduct of separate effects experiments, in support of existing or advanced LWRs,
- Model development for core melt progression, fission product release and transport,
- Code development and/or modification,
- Development of special purpose scoping models, and
- Experimental benchmarking of phenomenological models.

4.2 PROJECT PERSONNEL

Identification of Key Personnel

Personnel considered key to this proposal are listed in the Tables 4.1 and 4.2 below. These tables display the percentage of time each person is currently committed to other projects, including teaching responsibilities over the course of the proposed contract period.

[redacted]

Key Personnel
M. Khatib-Rahbar
R. Vijaykumar
E. Cazzoli
A. Kuritzky
K. Cady
V. Dhir
M. Kazimi
R. Viskanta

B/1

Key Personnel
M. Golay
P. Griffith
J. Meyer
P. Moin

Support Personnel

Energy Research, Inc. has a core of support personnel available to assist with the contract work. This capability includes technical editing, secretarial support, word processing, and document research, and project management.

Qualifications

Table 4.3 summarizes the expertise of the Energy Research, Inc. team being proposed for the present contract. Detailed resume of each team member is provided in the Appendix. A brief summary of team members' qualifications follows.

Dr. Mohsen Khatib-Rahbar will be the overall Project Manager for this proposed contract. He has extensive research experience in severe accident phenomenology and probablistic risk assessment. He has published more than 90 papers or reports on severe accidents, source terms, methods for uncertainty analysis, consequence assessment, thermal hydraulics, PRA and numerical methods. He has also developed computer models for the simulation of thermal hydraulics and neuronic transients in LMFBRs and LWRs and codes for the prediction of forced and natural convection transients in LMFBR assemblies. Prior to founding Energy Research Inc., he was a staff scientist at Brookhaven National Laboratory where he managed programs dealing with PRA review of several nuclear plants, verification and benchmarking of STCP and MELCOR, quantification of uncertainties in source terms, fission product release consequences of severe accidents, the NUREG-1150/Zion draft report, and regulatory implications of new source terms. In 1988, he spent a year as a visiting scientist with the NRC Office of Research, where he participated in the Containment Performance Improvement program (BWRs and PWRs). He also assisted there in development of severe accident research criteria for evolutionary and advanced light water reactors. Dr. Khatib-Rahbar's past and present related activities include:

- Consultant to Pacific Northwest National Laboratory (severe accidents, source terms and risk assessment for DOE Defense Production Reactors).

- Consultant to Los Alamos National Laboratory for peer review of MELCOR and SCDAP/RELAP-5 severe accident codes.

- Consultant to the Swiss Federal Nuclear Safety Inspectorate (PRA and severe accident research).

- Consultant to the International Atomic Energy Agency (PRA, severe accidents and source terms).

- Consultant to U. S. Nuclear Regulatory Commission (severe accident issues for ALWRs).

- Chairman of DOE/Defense Programs' Independent Review Panel for Savannah River PRA, severe accident and risk applications programs.

- Member of a team performing the IPE reviews (Back-end) for U. S. Nuclear Regulatory Commission.

- Member of Draft NUREG-1150 Expert Group (Source Terms and Containment Loads).

- Contributor to Kouts Panels on Source Terms/Severe Accident Uncertainties.

- Contributor to Containment Loads Working Group.

- Independent Reviewer of the National Research Council Study of N-Reactor.

- Participation in Organization for Economic Cooperation & Development (OECD) Study on Corium Concrete Interactions.

- Technical co-chairman of an ANS-sponsored international meeting on Decay Heat Removal and Natural Convection in LMFBRs (1985).

- Member of the Technical Program Committee for the OECD-sponsored 1986 conference on Core-Concrete-Interactions.

- Invited Lecturer at the Power Reactor and Nuclear Fuel Development Corporation (PNC) (sponsored by PNC-Japan).

- Invited Lecturer at nuclear research centers and regulatory organizations in Taiwan and South Korea (sponsored by the U. S. Department of State).

- Invited lecturer at Tsinghua University in the Peoples Republic of China.

- Invited lecturer at several research organizations and universities in U.S., Italy, Germany, United Kingdom, Finland, Sweden, Switzerland and France.

- Organizer and chairman of a special session on "Uncertainty Analysis in Reactor Safety" for the American Nuclear Society 1988 Annual Meeting.

- Co-instructor for a course on Severe Accident Studies in Nuclear Power Plants, University of Maryland (Spring 1990).

Dr. Khatib-Rahbar holds a Ph.D. degree in Nuclear Science and Engineering from Cornell University.

Dr. Erik G. Cazzoli is a Senior Scientist at Energy Research Inc. He has more than twenty years of experience in model development, numerical simulation, severe accident phenomenology, source term, containment response, severe accident uncertainties, and severe accident code assessment. He has also developed computer models for the simulation of neuronic transients in LMFBRs and LWRs and codes for the prediction of forced and natural convection transients. Prior to joining Energy Research Inc., he was a staff scientist at Brookhaven National Laboratory where he participated in PRA review of Advanced Pressurized and Boiling water Reactors, Zion NUREG-1150, quantification of uncertainties in source terms, verification of STCP, CONTAIN and MELCOR computer codes, fission product release consequences of severe accidents, and development of GENSOR fission product transport code. At Energy Research, Inc. he is involved in development of computer codes for modelling severe accident behavior, application of severe accident codes (STCP, MELCOR and SCDAP/RELAP5) for containment performance and source term assessment. Dr. Cazzoli holds a Ph.D. degree in physics from the University of Bologna.

Dr. Rajgopal Vijaykumar is a Consulting Engineer with Energy Research Inc. He is currently involved in severe accident model development and analysis in support of several PRA and IPE reviews, including modeling of severe accidents using MELCOR and SCDAP/RELAP-5. He has more than 6 years of experimental and theoretical research experience in fluid dynamics and heat

transfer. This includes research in turbomachinery, film boiling heat transfer using holography and hydrogen bubble flow visualization, and development of computer models for film boiling heat transfer. Dr. Vijaykumar participated in the UCLA accident management program, where he developed a model for debris lower head thermal interactions. He holds a Ph.D. degree in Mechanical Engineering from the University of California at Los Angeles (UCLA).

Mr. Alan S. Kuritzky is a Senior Consulting Engineer with Energy Research, Inc. He is involved in PRA and IPE reviews. His work as part of the IPE and PRA Back-end review team includes review of issues pertaining to severe accident phenomenology, containment performance improvements, and accident management. He is also involved in the SRS PRA Level 2 review, including review of the development and quantification of the accident progression event tree. He has over ten years of experience in PRA and related fields. Mr. Kuritzky holds a B.S. degree in Nuclear Engineering from the University of Maryland.

Dr. K. Bingham Cady is a professor of Nuclear Science & Engineering and Associate Dean, College of Engineering, Cornell University. Professor Cady has extensive research experience in development and application of thermal-hydraulics models under design basis and severe accident conditions. He has trained graduate students, and participated in university and industrial research in severe accident, containment performance, and accident management. He was responsible for the development of the EPRI-CURL series of thermal-hydraulics systems simulation codes for the Electric Power Research Institute (EPRI). Dr. Cady has been a major contributor in the development of radiation heat transfer models for use in in-vessel accident management studies for EPRI. He was one of the key contributors for the development of IDCOR/EPRI MAAP computer code models and architecture. Therefore, his past involvement in the MAAP development effort is very valuable to this program. Dr. Cady hold a Ph.D. degree in Nuclear Engineering from MIT.

Dr. Parviz Moin is the Franklin P. and Caroline M. Johnson Professor of Mechanical Engineering at Stanford University. Dr. Moin is an internationally recognized authority in fluid dynamics and advanced numerical methods. His contributions include Direct Numerical Simulation (DNS) and Large Eddy Simulation (LES) techniques. These methods can be applied to solve natural convection problems in large enclosures where distinction between numerical and real "diffusion" have been problematic (natural circulation in ALWR containments in presence of non-condensable gases.) Under NASA and Air Force sponsorship, Professor Moin's group has developed DNS and LES codes for flows in complex

geometries. These codes are highly accurate for compressible and incompressible flows with heat transfer. Potential applications include turbulent natural convection in molten pools and passive heat removal in ALWR containments. Professor Moin holds an endowed chaired professorship of engineering at Stanford and is the Director of the Center for Turbulence Research. Dr. Moin has published extensively in fluid dynamics and heat transfer, and hold a Ph.D. in Mechanical Engineering from Stanford University.

Dr. Vijay K. Dhir is a Professor of Engineering and Applied Science in the Mechanical, Aerospace and Nuclear Engineering Department of the School of Engineering and Applied Science at UCLA. During the past seventeen years he has done both basic and applied research in the thermal sciences and in energy conversion systems. His basic research is on the phenomenological studies of phase change heat and mass transfer. This includes experimental and analytical investigation of pool and forced flow boiling under saturated and subcooled conditions, two phase flow in porous media, film condensation, simultaneous melting and condensation under steady state and transient conditions, and evaporation. In the applied areas he has worked on safety and thermal hydraulics of fission and fusion nuclear power reactors. The studies have included reflood heat transfer, degraded core heat transfer and fluid flow phenomenology, core-concrete interactions, natural convection and stratification in liquid metal cooled reactors, melting, freezing and plugging of coolant channels in transient overpower accidents. In the past he has served on various DOE review panels and has been a consultant to Atomics International, Canoga Park, in support of the design efforts for a pool type fast breeder reactor with inherent safety characteristics. He has also acted as a consultant to NBS, SAI, Battelle Northwest Laboratories, E.G. & G. Idaho, General Electric, EPRI, and Pickard, Lowe and Garrick. Dr. V.K. Dhir obtained his Ph.D.in Mechanical Engineering from the University of Kentucky, Lexington, and has published more than 100 papers in various national and international journals and proceedings of conferences. At UCLA he teaches courses in Heat Transfer, Thermodynamics, and Nuclear Reactor Thermal Hydraulic and Design.

Dr. Raymond Viskanta is the W.F.M Goss Distinguished Professor of Engineering at Purdue University. His Research focuses on heat transfer in buoyancy driven flows, solid-liquid phase change, radiative transfer in participating media, and combined conduction-radiation heat transfer and convection-radiation heat transfer. Dr.Viskanta has been a Springer and visiting Professor at the University of California at Berkeley and Guest Professor at the Technical University of Munich and Tokyo Institute of Technology. Prior to accepting employment with Purdue University, he was a

mechanical engineer at Argonne National Laboratory working in the Reactor Engineering Division. Dr. Viskanta has served as a consultant on heat transfer and thermal hydraulics to a number of national laboratories and industries and was a member of the peer review panel on the Draft Reactor Risk Reference Document, NUREG-1150. He is a consultant to the PRA subcommittee of the US Department of Energy Advisory Committee on Nuclear Facility Safety. Professor Viskanta is a technical editor of the ASME Journal of Heat Transfer and he serves on the editorial boards of several journals. He is an author of over three hundred journal and proceedings publications. He holds a Ph.D degree in Mechanical Engineering from Purdue University and is a member of the U.S. National Academy of Engineering.

Dr. Mujid Kazimi is Professor and Chairman of the Department of Nuclear Engineering at MIT. He has been active in severe accident research for 17 years. While at Brookhaven National Laboratory (1974-76) Dr. Kazimi was responsible for the post-accident heat removal program in Fast Breeder Safety Group of the Department of Nuclear Energy. This program was centered on dynamics of fuel melt relocation and resolidification. Since joining MIT in 1976, he has been directing research programs on thermal phenomena of severe accidents. This includes core melt concrete interaction, melt spreading and non-condensible effects on steam condensation. Dr. Kazimi has published extensively in all aspects of nuclear reactor thermal-hydraulics and safety (including co-authorship of a two-volume book). Professor Kazimi's other relevant professional experience include:

- Visiting Scientist, Thermal Hydraulic Division, Joint European Research Center, Ispra (Italy), summer, 1985.
- Member, Senior Committee on Economics, Safety and Environmental Aspects of Magnetic Fusion Energy, reporting to U.S. DOE, 1985-1988.
- Member, Planning Group for U.S. NRC Reactor Safety Research on Severe Accidents, 1988-1989.
- Member, ERICE Seminar on Fusion Safety and Economics, Italy, 1989.
- Chairman, Severe Accident Working Group, Heavy Water Reactor NPR, U.S. DOE, 1990-1991.
- Member, Thermal Hydraulic Working Group, Heavy Water Reactor, NPR, U.S. DOE 1990-1991.

- Member, Safety Subcommittee, Integrated Fast Reactor Program, University of Chicago, 1990-.

- Chairman, High-Level Waste Tank Advisory Panel, U.S. DOE, 1990-.

- Member, Technical Oversight Board for U.S. DOE/EM Independent Engineering Review, of Major Projects, 1991-.

- Member, Panel on Separations Technology and Transmutation Systems (STATS), National Research Council, U.S. DOE, 1991-1994.

- Member, Fusion Program Visiting Committee, Argonne National Laboratory, 1991-.

Dr. Kazimi hold a Ph.D. degree in Nuclear Engineering from MIT.

Dr. John Meyer is a Professor of Nuclear Engineering at MIT. He is an expert in thermal-hydraulics model development with applications to reactor accident conditions. Before joining MIT, he spent 20 years with the Westinghouse Bettis Atomic Power Laboratory in engineering and technical management positions in thermal-hydraulics design, nuclear design, thermal-hydraulic analysis development, reactor fuel element analysis, and design methods development for advanced fuel elements. He has also lectured at the University of California, Berkeley, and at the Carnegie Institute of Technology. Dr. Meyer has served as consultant to numerous industrial and governmental organizations, including: Brookhaven National Laboratory, Idaho National Engineering Laboratory, Los Alamos National Laboratory, Northeast Utilities, Savannah River Laboratory, and the Yankee Atomic Electric Company. He has published extensively and holds a Ph.D. degree in Mechanical Engineering from Carnegie Institute of Technology.

Dr. Michael Golay is a Professor of Nuclear Engineering at MIT. He has performed research on advanced reactor and nuclear power design innovation, nuclear reactor thermal hydraulics, severe accidents, nuclear reactor physics, power station waste heat disposal, evaporative and dry cooling towers, energy systems, energy-policy, and nuclear reactor regulations. Dr. Golay has been a consultant to more than 20 private, industrial and governmental organizations. He has published more than 140 technical papers and research reports. Dr. Golay holds a Ph.D. in Nuclear Science and Engineering from Cornell University.

Dr. Peter Griffith is a Professor of Mechanical Engineering at MIT. He is an international authority in heat transfer, two-phase flow, and power/energy conversion. He has served as a consultant to numerous industrial and governmental organizations for the last 30 years. Dr. Griffith has participated in the development of Code Scaling and Uncertainty (CSAU) and the severe accident scaling methodology development and implementation programs. He has received numerous honors and awards including the Heat Transfer Memorial and the Melville awards of the American Society of Mechanical Engineering (ASME). Dr. Griffith holds a Sc.D. in Mechanical Engineering from MIT.

Table 4.3 Summary of Technical Expertise of the Energy Research, Inc. Team Members

Name	Heat Transfer	Fluid Dynamics	Computational Methods	Experimental Techniques	Code Assessment & Benchmarking
M.Khatib-Rahbar	x	x	x		x
E. G. Cazzoli	x	x	x		x
R. Vijaykumar	x	x	x	x	x
A. S. Kuritzky					x
K. B. Cady	x	x	x		x
V. K. Dhir	x	x	x	x	x
M. Golay	x	x	x	x	x
P. Griffith	x	x		x	x
M.S. Kazimi	x	x	x	x	x
J. Meyer	x	x	x		x
P. Moin	x	x	x		x
R. Viskanta	x	x		x	x

Table 4.3 Summary of Technical Expertise of the Energy Research, Inc. Team Members (Cont.)

Name	Core Melt Progression	Fuel Coolant Interactions	Fission Product Release & Transport	Core-Concrete Interactions	Scaling Methods
M.Khatib-Rahbar	x		x	x	
E. G. Cazzoli	x		x	x	
R. Vijaykumar	x	x	x	x	
A. S. Kuritzky			x		
K. B. Cady	x		x	x	
V. K. Dhir	x	x		x	x
M. Golay			x		
P. Griffith		x			x
M.S. Kazimi	x	x	x	x	
J. Meyer					x
P. Moin					x
R. Viskanta	x	x		x	x

Table 4.3 Summary of Technical Expertise of the Energy Research, Inc. Team Members (Cont.)

Name	HPME/ DCH	Hydrogen Mixing	MELCOR	SCDAP/ RELAP-5	VICTORIA	COMMIX	HMS	ALWRs	IPE IPEEE	CPI	Accid. Manag.
M.Khatib-Rahbar	x	x	x	x	x	x	x	x	x	x	x
E. G. Cazzoli	x	x	x	x	x				x	x	
R. Vijaykumar	x	x	x	x					x	x	x
A. S. Kuritzky								x	x	x	x
K. B. Cady		x						x	x	x	x
V. K. Dhir	x	x	x	x							x
M. Golay		x				x	x	x			
P. Griffith	x	x						x			
M.S. Kazimi	x	x		x		x		x			
J. Meyer		x		x		x					
P. Moin		x				x	x				
R. Viskanta	x	x	x	x		x	x				

5. EXPERIMENTAL AND COMPUTATIONAL FACILITIES

This section provides a summary of the experimental and computational facilities available for use in the proposed effort. Experimental facilities are discussed in relation to the potential needs of severe accident phenomena assessment.

5.1 Energy Research, Inc.

Extensive computational capabilities exist at Energy Research, Inc. to exercise existing severe accident computer codes and to develop additional simulation models as needed. These computers include, desk-top personal computers, powerful workstations, and access to commercial mainframe computers.

Energy Research, Inc. staff have been involved in the modeling and application of severe accidents for several Swiss nuclear power plants using MELCOR, the Source Term Code Package (STCP), SCDAP/RELAP5, and ERPRA (an Energy Research, Inc. developed parametric severe accident and source term code) computer codes.

Through the existing team effort, the extensive computational facilities at the various participating universities, including the Cornell University Super Computer Center are also accessible for use in the proposed effort.

5.2 Purdue University

The Heat Transfer Laboratory of the School of Mechanical Engineering is equipped with much of the instrumentation required for the research. A Mach-Zehnder interferometer, with 250 mm diameter optics, is housed in the Laboratory, along with three complete shadowgraph systems, two hydrogen bubble systems, and assorted lasers, cameras, lenses, mirrors, optical benches and micrometer devices. The laboratory is also equipped with two Lindberg furnaces capable of operation to 1500 degrees Centigrade.

In addition to the specific apparatus, much of the equipment and instrumentation necessary for research is available in the Heat Transfer Laboratory or within the School of Mechanical Engineering. Three HP85F microcomputers with 50 channel data loggers and numerous others are available for automatic data acquisition. Processing is done by up-loading the data to a VAX 11/780 computer housed in the School of Mechanical Engineering. For flow visualization studies assorted helium-neon and argon-ion lasers, cameras, lenses, mirrors, optical benches, and micrometer devices are available. In addition, the laboratory is equipeed with an INSTAR IV high-speed video system. The video system has two

cameras, each with a sampling rate of 120 pictures/s, and high intensity stroboscopes which provide for 10^s exposure. A hot-film anemometer and LDV system are also available.

The Purdue University Computing Center (PUCC) is one of the largest and most varied academic computing centers in the nation. With systems ranging from super computers to microcomputers, it provides general-purpose computing services for instruction and research to the University's students, faculty and staff.

Large-scale systems operated by the Computing Center include three vector computers: an ETA-10P running UNIX, a CDC CYBER 205 with the VSOS operating system, and an IBM 3090-180E with a vector facility running VM/CMS. Two Sequent multi-processor systems, a Symmetry S81 and a Balance B21000, support UNIX for large numbers of instructional users. A DEC VAX 8800 and five dual-process DEC VAX-11/780 systems also run UNIX. The Computer Center operates a few DEC PDP-11/70 RSTS/E minicomputer systems and several microcomputer labs.

The host systems are interconnected with the Computing Center, with other computers at Purdue, and with several off-campus networks. Purdue is a node on the Defense Adanced Research Project Agency Network (ARPANET). Through its IBM system, the Center is connected to BITNET, an internaitonal network linking more than 350 educational institutions and research centers.

The School of Mechanical Engineering has its own machine shop (3 full-time machinists) and electronics shop (6 full-time technicians) for technical support.

5.3 University of California

The Boiling Heat Transfer Laboratory at UCLA has a variety of experimental facilities. These facilities include a 25 kW induction heater, a high pressure flow loop with a rating of 300 GPM at 400 psia, a low pressure flow loop rated at 1000 GPM and 80 psia,a 100 kW DC power supply, a 25 kW DC power supply, and a 100 kW steam generator. Additionally, a loop with a pressurizer is also available. The instrumentation and diagnostics facilities include a holographic set-up, 3-D hot wire anemometry, a gamma densitometer, a gas chromatograph, and several A/D and D/A converters. The laboratory has extensive capability for computerized acquisition of temperature, flow rate and pressure data. Recently a micro-wave facility to study natural convection in volumetrically heated layers has also been added. Considerable expertise exists in the laboratory to conduct experiments at high temperatures. In the past the equipment has been used to study

dryout heat fluxes, quenching of simulated fuel rods and core debris, and phenomena related to core-concrete interactions. The laboratory is connected to the School of Engineering and Applied Science Computer Network (SEASnet) and to the UCLA mainframe (IBM 3178) computer.

5.4 Massachussetts Institute of Technology

The Massachussetts Institute of Technology has extensive laboratory facilities for use in heat transfer and fluids experiments in the proposed effort. The Departments of Nuclear and Mechanical Engineering laboratories are both accessible for use in the proposed effort. Extensive computational facilities are also available for use in the proposed activities.

6. PROJECT MANAGEMENT

Energy Research, Inc. is a small engineering services company with an experienced and technically qualified staff and an established track record for excellent management of NRC and other government projects.

Dr. Mohsen Khatib-Rahbar of Energy Research, Inc. will be the overall Project Manager for the effort. He will ensure proper coordination of tasks and schedules with the NRC technical contact and proper execution of the contract with the Contractor Officer. Dr. Khatib-Rahbar will be assisted as needed by Mr. E. Hicky, an experienced technical and contract administrator. The project organizational structure is displayed in Figure 6.1

Depending on the nature of task order request(s), Energy Research, Inc. staff members will be augmented as needed by an appropriate cadre of technical experts from the participating universities, either as individual consultants (short term and quick response tasks) to Energy Research, Inc., or university subcontractors (longer term tasks requiring university research facilities).

At the conclusion of each task, Energy Research, Inc. will conduct a project review to evaluate organization, product quality, cost efficiency and control. A detailed analysis of project costs and status will be made monthly, as needed for government contracts. Project cost and status will be reviewed monthly by the Project Manager to preclude contract cost growth.

Energy Research, Inc. has participated in several task order contracts involving integration of ERI staff, university and other independent consultant activities for Department of Energy (DOE) funded programs. Similar activities are also on-going for the Swiss Federal Nuclear Safety Inspectorate.

Prior to starting Energy Research, Inc., Dr. Khatib-Rahbar managed one of the largest groups (with more than 16 scientists) at the Department of Nuclear Energy, Brookhaven National Laboratory. He was responsible for more than $2m in annual funding from the U. S. Nuclear Regulatory Commission, Offices of Research and Nuclear Reactor Regulations. He has also organized and managed several peer review programs involving up to 30 different scientists and professionals for both NRC (QUASAR Expert Review Groups in 1986, and Kouts Panels on Source Term Uncertainties in 1987) and DOE (DOE/DP Independent Review Panel in 1991). His management and technical track record is well established.

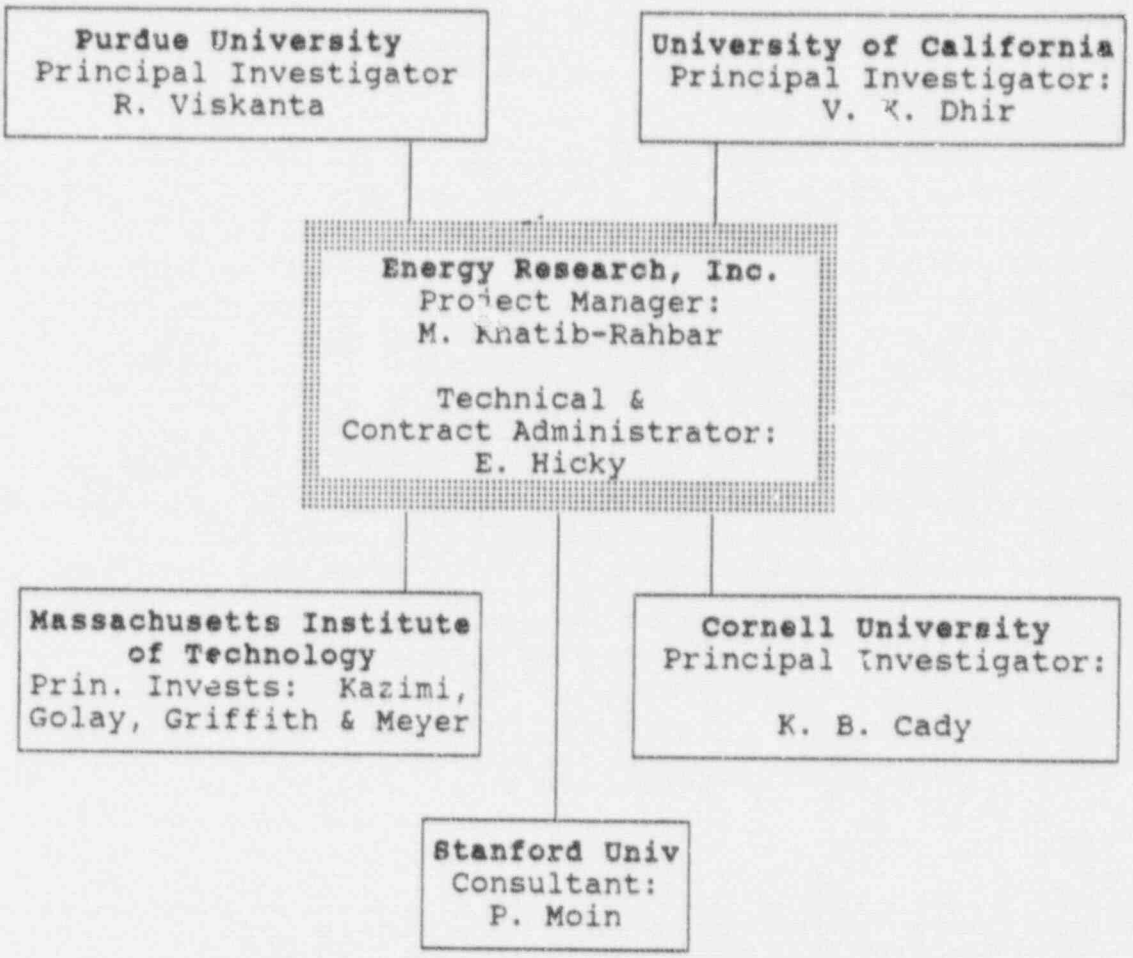

Figure 6.1 Organizational Structure

The consultants that Energy Research, Inc. has selected for this proposal provide the scientific depth and breadth of talent needed to carry on multiple task orders effectively and efficiently. The participating consultants are all recognized authorities in their respective fields and have participated in other research programs of this nature. The Energy Research, Inc. consultants for this proposal are: K. B. Cady, V. K. Dhir, M. Golay, P. Griffith, M. Kazimi, J. Meyer, P. Moin, and R. Viskanta. The requirements of the individual task orders will dictate the work the consultants will perform. Team composition will be determined by the NRC and Energy Research, Inc. Project Managers. None of the proposed consultants are employees of Energy Research, Inc. or the U. S. Government.

7. REFERENCES

1. "Severe Accident Risks: An Assessment For Five U.S. Nuclear Power Plants," NUREG-1150, U.S. Nuclear Regulatory Commission, (1990).

2. "Revised Severe Accident Research Program Plan," NUREG-1365, U. S. Nuclear Regulatory Commission (1989).

3. "Integration Plan For Closure of Severe Accident Issues,", SECY-88-147 (May 25, 1988).

4. R. R. Hobbins, D. A. Petti, D. J. Osetek and D. L. Hagerman, "Review of Experimental Results on Light Water Core Melt Progression," *Nuclear Technology*, 95, 287-307, 1991.

5. P. D. Bayless, C. A. Dobbe and R. Chambers, "Feedwater Transient and Small Break Loss of Coolant Accidents for a Nuclear Power Plant," NUREG/CR-4741, U.S. Nuclear Regulatory Commission.

6. R. R. Hobbins, D. A. Petti, D. J. Osetek and D. L. Hagerman, "Review of Experimental Results on Light Water Core Melt Progression," *Proceedings of the Eighteenth Water Reactor Safety Meeting*, Vol 2, 1-25 (1990).

7. A. W. Cronenberg, "In-Vessel Zircalloy Oxidation/Hydrogen Generation Behavior During Severe Accidents," NUREG/CR-5597, U.S. Nuclear Regulatory Commission.

8. H. M. Chung and T. F. Kassner, "Embrittlement Criteria for Zircalloy Cladding Applicable to Accident Conditions in Light Water Reactors," NUREG/CR-1344, U.S. Nuclear Regulatory Commission (1980).

9. D. A. Petti,"Silver-Indium-Cadmium Control Rod Behavior in Severe Reactor Accidents," *Nuclear Technology*, 87, 128 (1989).

10. S. Hagen, P. Hoffman, G. Schanz and L. Sepold, "Results of CORA Experiments on Severe Fuel Damage," *Proc. 26th Natl. Heat Transfer Conf.*, Philadelphia, Pennsylvania, August 6-9 (1989).

11. Z. R. Martinson, D. A. Petti, and B. A. Cook, "PBF Severe Fuel Damage Test Results Report," NUREG/CR-5163, U.S. Nuclear Regulatory Commission (April 1989).

12. R. O. Gauntt and R. D. Gasser,"Results of The DF-4 in-Pile Fuel Damage Test," *Proceedings of the Eighteenth Water Reactor Safety Meeting*, Vol 2, 25-39 (1990).

13. P. Hoffman, S. Hagen, G. Schanz and A. Skokan, "Reactor Core Materials Interactions at Very High Temperatures," *Nucl. Technology,"*, 146 (1989).

14. K. T. Kim and D. R. Olander, "Dissolution of UO_2 by Molten Zircalloy," *Journal of Nuclear Materials*, **154**, 85 (1988).

15. S. S. Dosanjh, "Melt Propagation in Dry Core Debris Beds," *Nuclear Technology*, **88**, 30 (1989).

16. D. W. Golden and P. Kuan, "Thermal Behavior of Cohesive Debris Beds in a Degraded Nuclear Reactor," *ANS Proceedings of the National Heat Transfer Conference*, Philadelphia, Pennsylvania, 4, 42, (1989).

17. M. Epstein and H. K. Fauske, "Thermal Behavior of Molten Corium During The Three Mile Island Unit 2 Core Relocation Event, "*Nucl. Technology*, 87, 1021 (1989).

18. J. E. O'Brien and G. L. Hawkes, "Thermal Analysis of a Reactor Lower Head with Core Relocation and External Boiling Heat Transfer," to be published in the AIChE Symposium Series, *National Heat Transfer Conference* (July, 1991).

19. S. A. Hodge, "Debris Bed Behavior in the BWR Lower Plenum," ORNL/NRC/LTR-90/10 (1990).

20. T. G. Theofanous, W. H. Amarasooriya, B. Najafi, M. A. Abolfadl, G. E. Lucas and E. Rumble, "An Assesment of Steam Explosion Induced Containment Failure, "NUREG/CR-5030 (1989).

21. S. Medhekar, M. Abolfadl, and T. G. Theofanous, "Triggering and Propagation of Steam Explosions," *ANS Proceedings of the 1988 National Heat Transfer Conference*, Houston, Texas, Vol 3, 244 July 24-27 (1988).

22. M. Saito, et. al. "Experimental Study on Penetration Behavior of Water Jet into Freon and Liquid Nitrogen," *ANS Proceedings of the National Heat Transfer Conference*, Houston, Texas, Vol. 3, p 173, July 24-27 (1988).

23. S. K. Wang, C. A. Blomquist, and B. W. Spencer, "Modeling of Thermal and Hydrodynamic Aspects of Molten Jet/Water Interactions," *ANS Proceedings of the National Heat Transfer Conference,* Philadelphia, Pennsylvania, Vol. 4, 225, August 6-9 (1989).

24. K. H. Bang and M. L. Corradini, "Stratified Vapor Explosion Experiments, *ANS Proceedings of the National Heat Transfer Conference*, Houston, Texas, Vol. 3, p 228, July 24-27 (1988).

25. J. Rempe, G. Thinnes, C. Allison and A. Cronenberg, "Lower Head Failure Analysis," *Proceedings of the Eighteenth Water Reactor Safety Meeting*, Vol. 2, 129-157 (1990).

26. R. W. Ostensen, et. al, "Fuel Flow and Freezing in the Upper Subassembly Structure Following an LMFBR Disassembly," *Trans. American Nuclear Society*, **18**, 214 (1974).

27. M. Epstein, et. al., "Transient Freezing of Flowing Ceramic Fuel in a Steel Channel," *Nuclear Science and Engineering*, **61**, 310 (1976).

28. H. Kouts, "Review of Research on Uncertainties in Estimates of Source Terms from Severe Accidents in Nuclear Power Plants, NUREG/CR-4883, U.S. Nuclear Regulatory Commission (1986).

29. H. P. Nourbakhsh, S. B. Kim, and M. Khatib-Rahbar, "Modeling of Natural Convection Processes During Degraded Core Accidents", *ASME Proceedings of the National Heat Transfer Conference*, HTD-96, Volume 2, 253-258, H. R. Jacobs (Editor), Houston, Texas, July 24-27 (1988).

30. R. Viskanta, S. Chelliah, and M. K. Moallemi, "Thermal Analysis of Core Barrel Heating and Coolant Recirculation During Core Uncovery in PWR Accidents," EPRI NP-5586, Electric Power Research Institute (1987).

31. M. K. Moallemi and R. Viskanta, " Coolant Recirculation in a Pressurized Water Reactor Under Loss-of-Coolant Accident Conditions," *Nuclear Science and Engineering*, **98**, 209 (1988).

32. W. A. Stewart, A. T. Pieczynski, and V. Srinivas, "Experiments on Natural Circulation Flow in a Scale Model PWR System During Postulated Degraded Core Accidents," *Proceedings of the Third International Topical Meeting on Reactor Thermal Hydraulics*, Newport, Rhode Island, American Nuclear Society, Vol. 1, 10 (1985).

33. B. W. Spencer, J. J. Sienicki and L. M. Mcumber, "Hydrodynamics and Heat Transfer Aspects of Corium/Water Interactions," EPRI-NP-5127, Electric Power Research Institute (1987).

34. D. C. Williams, K. D. Bergeron, D. E. Carrol, R. D. Gasser, J. L. Tills, and K. E. Washington, "Containment Loads due to Direct Containment Heating and Hydrogen Behavior: Analysis and Calculations With CONTAIN Code," NUREG/CR-4896, U.S. Nuclear Regulatory Commission (1987).

35. J. J. Sienecki and B. W. Spencer, "The PARSEC Computer Code for Analysis of Direct Containment Heating by Dispersed Debris," *ANS Proceedings of the National Heat Transfer Conference*, Houston, Texas, Vol. 3, p 355, July 24-27 (1988).

36 M.P. Menguc and R. Viskanta, "Radiation Transfer in a Cylindrical Vessel Containing High Temperature Corium Aerosols," *Nuclear Science and Engineering*, **92**, 570 (1986).

37. M. Khatib-Rahbar, et al. "A Probabilistic Approach to Quantifying Uncertainties in the Progression of Severe Accidents," *Nuclear Science & Engineering*, **102**, 219 (1989).

38. V. K. Dhir, I. Catton and J. Castle, "Role of Taylor Instability on Sublimation of a Horizontal Slab of Dry Ice," *Journal of Heat Transfer*, **99**, No. 3, 411 (1978).

39 R. Viskanta and M.P. Menguc, "Radiation Heat Transfer in Combustion Systems," *Progress in Energy and Combustion Science*, **13**, 97 (1987).

40. R. K. Cole, D. P. Kelly, M. A. Ellis, "CORCON-Mod2: A Computer Program for Analysis of Molten Core-Concrete Interactions," NUREG/CR-3920, U.S. Nuclear Regulatory Commission.

41. M. Silberberg, J. A. Mitchell, R. O. Meyer and C. P. Ryder, "Reassessment of the Technical Bases for Estimating Source Terms," NUREG-0956, U.S. Nuclear Regulatory Commission (1986).

42. W. Douglas, D. Stamps and M. Berman, "High Temperature Hydrogen Combustion in Reactor Safety Applications," *Nuclear Science and Engineering*, **109**, 39 (1991).

43. H. P. Nourbakhsh, M. Khatib-Rahbar, and R. E. Davis, "Fission Product Release Characteristics into Containment Under Design Basis and Severe Accident Conditions," NUREG/CR-4881 (1988).

44. K. Y. Suh and R. J. Hammersley, "Modeling of Fission Product Release and Transport of Severe Fuel Damage Analyses," *Nuclear Science and Engineering*, **109**, 26 (1991).

45. M. R. Kuhlman, D. J. Lehmicke, and R. O. Meyer, "CORSOR Users Manual," Battelle Columbus Laboratories, NUREG/CR-4173 (1985).

46. D. A. Powers, J. E. Brockman and A. W. Shiver, "VANESA: A Mechanistic Model of Radionuclide Release and Aerosol Generation During Core Debris Interactions with Concrete," NUREG/CR-4308, U.S. Nuclear Regulatory Commission (1985).

47. V. K. Dhir and I. Catton, "Dryout Heat Fluxes of Inductively Heated Particle Beds," *J. Heat Transfer*, **99**, 250 (1977).

48. R. Lipinski, "A Particle Bed Drout Model with Upward and Downward Boiling," *ANS Transactions*, **35**, 358 (1980).

49. A.T. Wassel, S.M. Ghiaasiaan, V.E. Denny and R.M. Traci, "Modeling of Natural Circulation in Reactor Coolant Systems," Fluid Physics Industries Report FPI-R88-05-04 (Work done under NRC Contract No. NRC-04-87-369)(1988).

50. R. Viskanta and D.M. Kim,"Heat Transfer from a Partially Uncovered Core," *Nuclear Engineering and Design*, **85**, 71 (1985).

51. C.H. Wang and V.K. Dhir,"An Experimental Investigation of Multidimensional Quenching of a Simulated Core Debris Bed," *Nuclear Engineering and Design*, **110**, 61 (1988).

52. B. Boyack et al. "The MELCOR Peer Review, Process, and Findings," *Proceedings of the Nineteenth Water Reactor Safety Meeting*, NUREG/CP-0118, 1-3 (1991).

APPENDIX

RESUMES OF KEY PERSONNEL:

Mohsen Khatib-Rahbar, Page 56
Erik G. Cazzoli, Page 68
Alan S. Kuritzky, Page 73
Rajgopal Vijaykumar, Page 77
K. Bingham Cady, Page 80
Vijay K. Dhir, Page 89
Michael W. Golay, Page 104
Peter Griffith, Page 120
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John E. Meyer, Page 135
Parviz Moin, Page 142
Raymond Viskanta, Page 152

Mohsen Khatib-Rahbar
Energy Research, Inc.

[redacted]

EDUCATION:

1978 Ph.D. Nuclear Science & Engineering, Cornell University

1974 BChE Chemical Engineering, University of Minnesota

1971 Dipl. High School, Fergus Falls Senior High School, Fergus Falls, Minnesota

PROFESSIONAL EXPERIENCE:

5/89 - Present President, Energy Research, Inc.

Providing technical support services in areas of; Probabilistic Risk Assessment (PRA), Severe Accident and Source Term Analysis, Hazard Analysis, Review and Evaluation of PRAs for Commercial LWRs and DOE Nuclear and Non-Nuclear Facilities.

10/88 - 4/89 Brookhaven National Laboratory (BNL) Technical Representative to DOE

Provided direct technical support to DOE in activities related to the safety of the DOE research and defense production facilities.

3/88 - 4/89 Technical Advisor, Office of Nuclear Regulatory Research (USNRC)

As BNL technical representative, provided direct technical support to Division of Safety Issue Resolution, Office of Nuclear Regulatory Research. Participated in the Containment Performance Improvement (CPI) program and activities related to the severe accident policy implementation for evolutionary and advanced reactors.

10/85 - 2/88 Group Leader, Accident Analysis Group, Brookhaven National Laboratory

Manager of the largest group (responsible for up to 16 PhD-level scientific and professional staff members) in the Department of Nuclear Energy. In this capacity

directed research and technical support programs in areas of; severe accident analysis, PRA methods development and review, accident consequence and source term analysis, methods in uncertainty analysis, and regulatory and policy implementation. Key contributor to the following NRC programs:

(1) NUREG-1150, (2) NUREG-0956, (3) Verification & Benchmarking of STCP and MELCOR, (4) QUASAR Severe Accident Uncertainty Study, (5) Assessment of IPE Methods, (6) Regulatory Implications of New Source Terms, (7) Evaluation of Protective Action Strategies, (8) Reviews of Level II & III PRAs for existing and advanced LWRs.

12/83 - 9/85 **Principal Investigator & Task Leader, Brookhaven**

Responsible for review of several PRAs including; Zion, Millstone-3 (EIS & PRA), Seabrook, Oconee, South Texas (EIS), and Westinghouse APWR (PRA & SER).

10/82 - 11/83 **Task Leader, LMFBR Code Development, Brookhaven**

Responsible for development of TWIST, a transient two-dimensional rod bundle code for study of forced, mixed, and natural convection transients in wire-wrapped LMFBR assemblies.

4/82 - 8/82 **Invited Visiting Scientist, Gesellschaft für Reactorsicherheit (GRS), Cologne, Germany.**

Developed digital computer models for simulation of operational transients in the SNR-300. This included models to study the behavior of the Plant Control and Plant Protection Systems.

1/78 - 3/82 **Task Leader for SSC-L Code Development Activities, Brookhaven**

Developed mathematical models, numerical methods and computer codes for simulation of single phase and multi-phase thermal-hydraulics transients in LMFBRs and LWRs. Participated in CRBRP licensing review. Also developed plant protection and control system models for the RAMONA-3B code (A BWR code with multidimensional neutronics).

9/74 - 12/77 **Cornell University - Ward Laboratory of Nuclear Engineering**

Graduate studies and university research in reactor dynamics, thermal hydraulics model development, numerical methods and system simulation (sponsored by the Electric Power Research Institute). Developed the EPRI-CURL systems code for LMFBRs.

PERSONAL:

Born on February 21, 1954
Married, 1 Child
U. S. Citizen

Publications:

Refereed Journals and Proceedings

1. M. Khatib-Rahbar, A. K. Agrawal, and E. S. Srinivasan, "Feedback Control Systems for Non-Linear Simulation of Operational Transients in LMFBRs", Proceeding of the International Meeting on Fast Reactor Safety Technology, Volume III, 1316, Seattle, Washington, American Nuclear Society, Illinois (1979).

2. A. K. Agrawal, L. G. Epel, J. G. Guppy, M. Khatib-Rahbar, and I. K. Madni, "An Analysis of the Steady-Power Natural Circulation Experiments in LMFBR Plants", ASME Paper No. 80-WA/HT-41 (1980).

3. A. K. Agrawal and M. Khatib-Rahbar, "Dynamic Simulation of LMFBR Systems," Journal of Atomic Energy Review", Volume 18, No.2,329-552, Invited (1980).

4. M. Khatib-Rahbar, I. K. Madni, and A. K. Agrawal, "Impact of Thermal Buoyancy in LMFBR Piping Systems", Proceedings of Specialists Meeting on Decay Heat Removal and Natural Convection in FBRs, pp. 69-80, Hemisphere, New York (1981).

5. M. Khatib-Rahbar, J. G. Guppy, and A. K. Agrawal,"Hypothetical Loss-of Heat-Sink and In-Vessel Natural Convection: Homogeneous and Heterogeneous Core Designs", Proceedings of the Specialists Meeting on Decay Heat Removal and Natural Convection in FBRs, 329-342, Hemisphere, New York (1981).

6. M. Khatib-Rahbar and K. B. Cady, "Dynamical Models and Numerical Simulation of System-Wide Transients in Loop-Type LMFBRs", Nucl. Eng. & Design, Volume 64, 259-281 (1981).

7. A. K. Agrawal, M. Khatib-Rahbar, and I. K. Madni, "Prediction of Decay Heat Removal Capabilities for LMFBRs and Comparison with Experiments", Nucl. Eng. & Design, Volume 66, 437-446 (1981)

8. M. Khatib-Rahbar, J. G. Guppy, and R. J. Cerbone, "LMFBR System-Wide Transient Analysis: The State of the Art and U.S. Validation Needs", Proceedings of the International Meeting on LMFBR safety and Related Design and Operations, Lyon-Ecually, France,Volume II, 67 (1982).

9. A. K. Agrawal, M. Khatib-Rahbar, R. T. Curtis, and D. L. Hetrick, "Thermohydraulic Modeling and Simulation of Breeder Reactors", Proceedings of the 10th IMACS World Congress, Canada, Volume 4, 123, Invited (1982).

10. A. K. Agrawal, M. Khatib-Rahbar, R. T. Curtis, and D. L. Hetrick, "Thermohydraulic Modeling and Simulation of Breeder Reactors", in Mathematics and Computers in Simulation, Transactions of IMACS (1983).

11. M. Khatib-Rahbar and E. G. Cazzoli, "Flow Excursion Induced Dryout at Low Heat Flux Natural Convection Boiling",ASME-JSME Thermal Engineering Joint Conference Proceedings, Volume 1, 43-53 (1983).

12. Co-author, "Super System Code, An Advanced Thermohydraulic Simulation Code for Transients in LMFBRs", Proceedings of Advances in Reactor Computations Meeting, Salt Lake City, Utah (March 28-31, 1983).

13. M. Khatib-Rahbar and E. G. Cazzoli, " Intra-Assembly Flow Redistribution in LMFBRs: A Simple Computational Approach", First Proceedings of Nuclear Thermal Hydraulics, 13-28 (1983).

14. M. Khatib-Rahbar and E. G. Cazzoli, "TWIST: A Transient Two-Dimensional Intra-Subassembly Thermal Hydraulics Model for LMFBRs", Second Proceedings of Nuclear Thermal Hydraulics, 106-113 (1984).

15. W. T. Pratt, M. Khatib-Rahbar, R. A. Bari, and J. Rosenthal, "Insights Gained from In-depth Reviews of Several Industry Generated PRAs", Proceeding of the International ANS/ENS

Topical Meeting on Probabilistic Safety Methods and Applications, Volume 1, 23-32, San Francisco (1985).

16. I. K. Madni, J. G. Guppy, and M. Khatib-Rahbar, "Thermal Hydraulic Modeling for the Primary System of Experimental Breeder Reactor-II", Proceedings of the 5th International Conference on Mathematical Modeling, Berkeley, California (July 1985).

17. M. Khatib-Rahbar and E. G. Cazzoli, "Flow Excursion Induced Dryout at Low Heat Flux Natural Convection", ASME Transaction, Journal of Heat Transfer, Volume 108, 425-432 (May 1986).

18. M. Khatib-Rahbar, et al, "QUASAR Uncertainty Study", Transaction of the 14th Water Reactor Safety Information Meeting, 14-1,NUREG/CR-0081 (October 1986).

19. M. Lee, R. E. Davis, and M. Khatib-Rahbar, "Impact of Zr Metal and Coking Reactions on the Fission Product Aerosol Release During Molten Core Concrete Interactions", AIChE Symposium Series No.257, Volume 83, 391 (1987).

20. M. Khatib-Rahbar, et al, "Reanalysis of Zion Risk", Transaction of the 15th Water Reactor Safety Information Meeting, 7-11,NUREG/CR-0090 (October 1987).

21. M. Khatib-Rahbar, et al, "Quantification of Uncertainties in Source Term Estimates for a BWR with MARK I Containment," Transaction of the 15th Water Reactor Safety Information Meeting, 20-9, NUREG/CR-0090 (October 1987).

22. H. P. Nourbakhsh, M. Khatib-Rahbar, R. Davis, and J. Read, "Formulation of Radiological Releases to Containment for Design Basis and Beyond the Design Basis Accidents", Paper Presented at the Nuclear Reactor Severe Accident Chemistry Symposium at the American Chemical Society Meeting, Toronto, Canada (June 5-11, 1988).

23. H. P. Nourbakhsh, S. B. Kim, and M. Khatib-Rahbar, "Modeling of Natural Convection Processes During Degraded Core Accidents", ASME Proceedings of the National Heat Transfer Conference, HTD-96, Volume 2, 253-258, H. R. Jacobs (Editor), Houston, Texas (July 24-27, 1988).

24. T. Ishigami, E. Cazzoli, M. Khatib-Rahbar, and S. D. Unwin, "Techniques to Quantify the Sensitivity of Deterministic Model Uncertainties", Nuclear Science & Engineering, Volume 101, 371-383 (1989).

25. M. Khatib-Rahbar, et al, "A Probabilistic Approach to Quantifying Uncertainties in the Progression of Severe Accidents", Nuclear Science & Engineering, Volume 102, 219-259 (1989).

26. J. W. Yang, E. Schmidt, E. Cazzoli, and M. Khatib-Rahbar, "Analysis of Fission Product Revaporization in a BWR Reactor Coolant System During a Station Blackout Accident", Fourth Proceeding of Nuclear Thermal Hydraulics, 139-147, ANS Meeting, Washington, D. C. (1988).

27. S. D. Unwin, E. G. Cazzoli, R. E. Davis, M. Khatib-Rahbar, M. Lee, H. Nourbakhsh, C. K. Park, and E. Schmidt, "An Information-Theoretic Basis for Uncertainty Analysis: Application to the QUASAR Severe Accident Study", Journal of Reliability Engineering and System Safety, Volume 26, 143-162 (1989).

28. I. K. Madni, E. G. Cazzoli, and M. Khatib-Rahbar, "Assessment of Uncertainties in Early Offsite Consequences from Nuclear Reactor Accidents", Nuclear Technology, Volume 90, 63-77 (1990).

29. M. Khatib-Rahbar and K. Murphy, "Probabilistic Safety Measures for Assessment of DOE Nuclear Reactor Facilities," Proceedings of the Probabilistic Safety Assessment and Management (PSAM) Conference, Beverly Hills, CA, February 4 - 7 (1991).

30. M. Khatib-Rahbar, E. G. Cazzoli, I. K. Madni, H. P. Isaak, S. L. Chan, and U. Schmocker, "Characterization of Fission Product Releases Resulting from Severe Reactor Accidents in LWRs," Proceeding of the International Symposium on the Use of Probabilistic Safety Assessment for Operational Safety, IAEA-SM-321/66, Vienna, Austria, June 3- 7, 1991

31. U. Schmocker, S. Chakraborty, H. Deutschmann, R. Fenske, H. P. Isaak, M. Khatib-Rahbar, E. G. Cazzoli, N. Hanan, "Approach to Regulatory Review of Swiss PSAs," Proceeding of the International Symposium on the Use of Probabilistic Safety Assessment for Operational Safety, IAEA-SM-321/10, Vienna, Austria, June 3- 7, 1991

32. M. Khatib-Rahbar and U. Schmocker, "A Perspective Comparison of Swiss PSAs and NUREG-1150," Invited Paper at the Schweizerische Vereinigung Für Atomenergie (SVA) Kommission Für Ausbildungsfragen, Fortgeschrittene Sicherheitsanalyse, Winterthur, Switzerland November 4 - 5 (1991).

33. B. E. Boyack, V. K. Dhir, J. A. Gieseke, T. J. Haste, M. A. Kenton, M. Khatib-Rahbar, M. T. Leonard, and R. Viskanta, "MELCOR Peer Review," Proceedings of the 19th Water Reactor Safety Research Meeting, Bethesda, Maryland, October 28 - 30 (1991).

Refereed Transactions and Abstracts

34. K. B. Cady, M. Khatib-Rahbar, G. F. Pavlenco, and J. R. Raber, "CURL: A Transient Simulation Code for LMFBRs," Trans. Am. Nucl. Soc. Volume 27, 543 (1977).

35. M. Khatib-Rahbar and K. B. Cady, "Establishment of Buoyancy-Induced Natural Circulation in Loop-Type LMFBRs," Trans. Am. Nucl. Soc. Volume 28, 432 (1978).

36. I. K. Madni, M. Khatib-Rahbar, and A. K. Agrawal, "Effect of Pump-Stored Kinetic Energy on Natural-Circulation Decay-Heat Removal in Loop-Type LMFBRs," Trans. Am. Nucl. Soc. Volume 32, 502 (1979).

37. M. Khatib-Rahbar, I. K. Madni, A. K. Agrawal, and E. G. Cazzoli, "Transition to Natural Convection from Low-Temperature, Low-Flow Conditions in Loop-Type LMFBRs," Trans. Am. Nucl. Soc. Volume 32, 503 (1979).

38. J. G. Guppy, W. C. Horak, M. Khatib-Rahbar, and G. J. Van Tuyle, "Analysis of Natural Circulation Events in B & W Plants Using SSC-W," Trans. Am. Nucl. Soc. Volume 35, 315 (1980).

39. M. Khatib-Rahbar, A. K. Agrawal, and I. K. Madni, " Analysis of Steady-State Natural Circulation Experiments in the EBR-II," Trans. Am. Nucl. Soc. Volume 38, 410 (1981).

40. M. Khatib-Rahbar and J. G. Guppy, "Digital Simulation of LMFBR Plant Protection and Control Systems," Trans. Am. Nucl. Soc. Volume 38, 511 (1981).

41. M. Khatib-Rahbar, "Core-Coolability Following Loss-of-Heat Sink Accidents," Trans. Am. Nucl. Soc. Volume 45, 365, Invited (1983).

42. M. Khatib-Rahbar and E. G. Cazzoli, "Intra-Assembly Flow Redistribution in LMFBRs: A Simple Computational Approach," Trans. Am. Nucl. Soc. Volume 45, 816 (1983).

43. M. Khatib-Rahbar, E. G. Cazzoli, and J. G. Guppy, "Analysis of Loss-of-Heat Sink Accidents in LMFBRs," Paper Presented at the Specialist Meeting on Liquid Metal Thermal Hydraulics in Plena and Pipes, Sunnyvale, CA, Invited (January 17-19, 1983).

44. M. Khatib-Rahbar and E. G. Cazzoli, "TWIST: A Transient Two-Dimensional Intra-Subassembly Thermal Hydraulics Model for LMFBRs," Trans. Am. Nucl. Soc. Volume 45, 818 (1984).

45. M. Khatib-Rahbar, C. Park, W. T. Pratt, R. A. Bari, G. Marino, and C. Ryder, "QUASAR: A Methodology for Quantification of Uncertainties in Severe Accident Source Terms," Trans. Am. Nucl. Soc. Volume 53, 354 (1986).

46. M. Lee and M. Khatib-Rahbar, "Sensitivity of In-Vessel Hydrogen Generation and Fission Product Release to Parameter Variation in a Melt Progression Model," Trans. Am. Nucl. Soc. Volume 54, 228 (1987).

47. T. Ishigami, E. G. Cazzoli, M. Khatib-Rahbar, and S. Unwin, "A Sensitivity Analysis Technique for Application to Deterministic Models," Trans. Am. Nucl. Soc. Volume 55, 313 (1987).

48. J. W. Yang, M. Khatib-Rahbar, "A Comparative Study of STCP and SCDAP Simulation of PBF SFD Test 1-1," Trans. Am. Nucl. Soc. Volume 55, 441 (1987).

49. H. P. Nourbakhsh, S. B. Kim, and M. Khatib-Rahbar, "In-Vessel Flow Characterization Under Severe Accident Conditions," Trans. Am. Nucl. Soc. Volume 55, 755 (1987).

50. M. Lee, E. Cazzoli, Y. Liu, H. Nourbakhsh, E. Schmidt, S. Unwin, and M. Khatib-Rahbar, "Uncertainties in Source Term Estimates for a Station Blackout Accident in a BWR with MARK I Containment," Trans. Am. Nucl. Soc. Volume 56, 365 (1988).

51. I. K. Madni, E. Cazzoli, M. Khatib-Rahbar, and A. Tingle, "An Integral Approach for Calculating Uncertainties in Consequences from Nuclear Reactor Accidents," Trans. Am. Nucl. Soc. Volume 56, 354 (1988).

52. J. W. Yang, E. Schmidt, E. Cazzoli, and M. Khatib-Rahbar, "Analysis of Fission Product Revaporization in a BWR Reactor Cooling System During a Station Blackout Accident", Trans. Am. Nucl. Soc. Volume 57, 370 (1988).

53. M. Khatib-Rahbar, et al.,"Characterization of Fission Product Releases Resulting from Severe Accidents in LWRs," Int. Symposium on Use of PSA for Operational Safety, IAEA, Vienna, Austria, IAEA-SM-321/66 (1991)

54. U. Schmocker, S. Chakraborty, H. Deutchmann, R. Fenske, H. Isaak, M. Khatib-Rahbar, E. Cazzoli, and N. Hanan, "Approach to Regulatory Review of Swiss PSAs," Int. Symposium on Use of PSA for Operational Safety, IAEA, Vienna, Austria, IAEA-SM-321/10 (1991)

55. S. Chan, H. Isaak, U. Schmocker and M. khatib-Rahbar, "Characterization of Severe Accidents Using MELCOR: A Perspective," Paper presented at Cooperative Severe Accident Research Program (CSARP) Semi-Annual Meeting, Bethesda, MD (May 1991).

Significant Reports

56. K. B. Cady, G. F. Pavlenco, and M. Khatib-Rahbar, "CURL-A Liquid-Metal-Cooled Fast Breeder Reactor Simulation," Cornell University Reactor Laboratory Report, CURL-50 (July 1976).

57. K. B. Cady, M. Khatib-Rahbar, G. F. Pavlenco, and J. R. Raber, "CURL-L: A Dynamic Analysis of Loop-Type LMFBRs," Cornell University Reactor Laboratory Report, CURL-52 (Nov. 1977).

58. M. Khatib-Rahbar, "System Modeling for Transient Analysis of Loop-Type Liquid Metal-Cooled Fast Breeder Reactors," Cornell University Reactor Laboratory Report, CURL-53 (May 1978).

59. Co-Author, "Users Manual for the SSC-L Code," Topical Report, NUREG/CR-0452, BNL-NUREG-50914 (1978).

60. K. B. Cady, M. Khatib-Rahbar, G. F. Pavlenco, J. R. Raber,"EPRI-CURL: A Dynamic Analysis of Loop-Type LMFBRs," Electric Power Research Institute, Inc., EPRI-NP-1001 (May 1979).

61. M. Khatib-Rahbar, "Modeling of Plant Protection and Control Systems for SSC," Brookhaven National Laboratory, Topical Report, NUREG/CR-1605, BNL-NUREG-51241 (June 1980).

62. W. Wulff, S. Cheng, D. J. Diamond, and M. Khatib-Rahbar, "A Description and Assessment of RAMONA-3B: A Computer Code with Three-Dimensional Neutron Kinetics for BWR System Transients,"

Brookhaven National Laboratory Topical Report, NUREG/CR-3664 (October 1981).

63. M. Khatib-Rahbar and E. G. Cazzoli, "Modeling and Analysis of Low Heat Flux Natural Convection Sodium Boiling in LMFBRs," Brookhaven National Laboratory, Topical Report, NUREG/CR-3498 (1982).

64. M. Khatib-Rahbar, "Digital Simulation of Plant Protection and Operational Control Systems in the SNR-300," GRS report, Koeln (July 1982).

65. Co-Author, "Super System Code (SSC, Rev.2): An Advanced Thermo-hydraulic Simulation Code for Transients in LMFBRs," NUREG/CR-3169 (1983).

66. M. Khatib-Rahbar, E. G. Cazzoli, "Two Dimensional Modeling of Intra-Subassembly Heat Transfer and Buoyancy-Induced Flow Redistribution in LMFBRs," Brookhaven National Laboratory, NUREG/CR-3498 (1984).

67. M. Khatib-Rahbar, W. Pratt, H. Ludewig, R. Barrett, and P. Easely, "Review and Evaluation of Millstone Unit 3 Probabilistic Safety Study: Containment Failure Modes, Radiological Source Terms and Offsite Consequences," Topical Report, NUREG/CR-4143 (1985).

68. M. Khatib-Rahbar et al., "Containment Failure Modes and Radiological Source Terms for the South Texas Project," BNL Technical Report A-3819, 8-23-85 (August 1985).

69. M. Khatib-Rahbar, et al, "Estimates of Steam Spike and Pressurization Loads for a Subatmospheric Containment," BNL Technical Report A-3804, 1-31-86 (January 1986).

70. M. Khatib-Rahbar, et al, "A Review of the Seabrook Station Probabilistic Safety Assessment. Containment Failure Modes and Radiological Source Terms," Topical Report, NUREG/CR-4540 (February 1986).

71. C. Park, A. K. Agrawal, M. Khatib-Rahbar, H. Ludewig, and W. T. Pratt, "A Review and Evaluation of the Oconee Unit 3 PRA," Topical Report, NUREG/CR-4374, Volume 3 (April 1986).

72. M. Khatib-Rahbar, "QUASAR: Quantification and Uncertainty Analysis of Source Terms for Severe Accidents in LWRs, Part I - Methodology and Program Plan," Topical Report, NUREG/CR-4688, BNL-NUREG-52008, Volume 1 (June 1986).

73. M. Khatib-Rahbar, et al, "Independent Verification of Radionuclide Release Calculations for Selected Accident Scenarios," Topical Report, NUREG/CR-4629, BNL-NUREG-51998 (July 1986).

74. H. P. Nourbakhsh, R. E. Davis, and M. Khatib-Rahbar, "Effectiveness of BWR Pressure Suppression Pools in Retaining Fission Products". BNL Technical Report A-3788, 8-1-86 (August 1986).

75. R. E. Davis, H. P. Nourbakhsh, and M. Khatib-Rahbar, "Fission Product Removal Effectiveness of Chemical Additives in PWR Containment Sprays," BNL Technical Report A-3788, 8-12-86 (August 1986).

76. H. Ludewig, M. Khatib-Rahbar, and W. T. Pratt, "Radiological Release Estimates for Energetic Events in a Large Dry Pressurized Water Reactor Containment," BNL Technical Report A-3804, 12-31-86 (December 1986).

77. J. W. Yang and M. Khatib-Rahbar, "STCP Simulation of PBF Severe Fuel Damage Scoping and 1-1 Tests," BNL Tech Report A-3290, 12-31-87 (Dec. 1987).

78. M. Khatib-Rahbar, et al., "Evaluation of Severe Accident Risk and Potential for Risk Reduction: Zion Power Plant," NUREG/CR-4551, BNL-NUREG-52029, Volume 5 (February 1987).

79. H. P. Nourbakhsh, M. Khatib-Rahbar, and R. E. Davis, "Fission Product Release Characteristics into Containment Under Design Basis and Severe Accident Conditions," NUREG/CR-4881, BNL-NUREG-52059 (1988).

80. S. D. Unwin, E. Cazzoli, R. Davis, M. Lee, H. Nourbakhsh, E. Schmidt, and M. Khatib-Rahbar, "The Formulation of Probability Distributions for the QUASAR Program," BNL Technical Report, A-3286, 9-18-87 (September 1987).

81. T. Ishigami, E. Cazzoli, M. Khatib-Rahbar, and S. Unwin, "Quantification and Uncertainty Analysis of Source Terms for Severe Accidents in LWRs (QUASAR), Part II - Sensitivity Analysis Techniques," NUREG/CR-4688, BNL-NUREG-52008, Vol 2 (October 1987).

82. M. Khatib-Rahbar, et al, "On the Uncertainties in Core Melt Progression, Fission Product Release and Pressurization Loads for BWR with MARK I Containment," BNL Technical Report, A-3286 2-20-88 (February 1988).

83. I. K. Madni, E. Cazzoli and M. Khatib-Rahbar, "A Simplified Model for Calculating Early Off-Site Consequences from Nuclear Reactor Accidents," NUREG/CR-5164, BNL-NUREG-52153 (July 1988).

84. M. Khatib-Rahbar and W. T. Pratt, "A Technical Evaluation Report for Westinghouse Advanced Pressurized Water Reactor PRA - Containment Response Analysis," BNL Technical Report A-3705 7-28-88 (July 1988).

85. M. Lee, E. Cazzoli, E. Schmidt, Y. Z. Liu, S. D. Unwin, and M. Khatib-Rahbar,"QUASAR, Part III - Propagation of Uncertainties and Output Sensitivity and Importance Analysis," NUREG/CR-4688, BNL-NUREG-52008, Volume 3 (1988).

86. M. Khatib-Rahbar, "A Probabilistic Risk Assessment Methods Guidance for DOE Nuclear Facilities," Prepared for the Department of Energy, ERI-89-5-31, Draft (May 1989).

87. M. Khatib-Rahbar, K. G. Murphy, and F. H. Rowsome, "A Preliminary Approach to Implementation of the DOE Nuclear Safety Goal Policy," ERI-89-08-18, Draft (November 1989).

88. Co-author, "DOE Facility Hazard Audit Procedures Guide", Pacific Northwest National Laboratory Report, Department of Energy, Draft (December 1989)

89. M. Khatib-Rahbar, et al, "Probabilistic Risk Assessment Review Guidance for Swiss Nuclear Power Plants," ERI/HSK 90-7-01 (July 1990).

90. M. Khatib-Rahbar, et al, "A Quick Review of the Mühleberg Probabilistic Safety Assessment, Part II: Level 2," HSK 11/273, ERI/HSK 90-9-02 (September 1990).

91. M. Khatib-Rahbar, et al, "A Preliminary Review of the Beznau Risk Assessment (BERA), Part II: Level 2," ERI/HSK 91-1115 (November 1991).

Erik G. Cazzoli
Energy Research, Inc.

EDUCATION:

1973 Post-Doctorate School, Physics, University of Bologna

1972 Ph.D. Physics, University of Bologna (with honors)

PROFESSIONAL EXPERIENCE:

1990 - Present Senior Consulting Engineer, Energy Research, Inc.

Responsible for severe accident analysis and Probabilistic Risk Assessment (PRA) activities at Energy Research, Inc. Involved in application of MELCOR and SCDAP/RELAP-5 for source term and containment loading studies in support of PRA review activities. Developing models for severe accident source term, containment, and accident consequence analysis.

1972 - 1990 Brookhaven National Laboratory (BNL)

July 1987 - June 1990 Research Engineer in the Safety and Risk Evaluation Division of the Department of Nuclear Energy at BNL. Responsible for the calculation of consequences following nuclear reactor accidents through the use of the CRAC2 and MACCS codes; responsible for the implementation of a simplified PC-based model (SMART) which computes doses and health effects following a reactor accident. Involved in the quantification of uncertainties of releases following severe accidents at Light Water Reactors (LWRs) as part of the QUASAR program. Involved in the NUREG-1150 analysis of the Zion plant, in the proprietary Safety Evaluation of Core melt accidents for the W/SP-90 plants and in the review of the Advanced Boiling Water Reactor (ABWR) design and PRA. Responsible for the implementation of Level-2 Probabilistic Risk Assessment of the BNL research reactor HFBR.

January 1986 - July 1987 Responsible for fission product release and transport calculations using

the STCP and CONTAIN computer codes in the Accident Analysis Group at BNL. This work involved assessment of radiological releases for severe accidents in BWRs and PWRs. Key contributor to Zion Risk Rebaselining Activities as part of the NUREG-1150 study.

March 1985 - January 1986 Responsible for the implementation, management, and development of a series of PC-based programs which control the real-time analysis and data display of the BNL Nuclear Plant Analyzer HIPA, a faster-than-real-time simulator of BWR power plants. Studied a variety of accident sequences including ATWS, Station Blackout, and Turbine Trip events.

July 1976 - February 1982 Responsible for the modeling, implementation and application of the Loop and Pool versions of the Super System Code (SSC), sponsored by the U.S. Nuclear Regulatory Commission for the analysis of postulated accidents in LMFBRs. Responsible for the design, development and implementation of the TWIST computer code, sponsored by the NRC, a thermal hydraulic model for study of inter- and intra-assembly heat transfer and flow redistribution in LMFBR assemblies. Responsible for the analysis of ATWS-type accidents (Anticipated Transients Without Scram) in Boiling-Water Reactors using the RAMONA-3B code , with objective the evaluation of the Emergency Procedure Guidelines of the NRC.

November 1973 - June 1976 Research Associate and later Assistant Physicist, in the Bubble Chamber Group of the Department of Physics, headed by Nicholas Samios (Currently, BNL Director). Participant to experimental research on weak interactions of neutrinos with nuclei both at BNL and Fermi Laboratory in Chicago. Member of the group that first reported experimental evidence of the existence of Charmed Baryons (1975).

1972 - 1973 Assistant Professor and Physicist at the INFN (Italian Nuclear Physics Institute) in Bologna, ITALY

Involved in collaboration with the Universities of

Pisa, Glasgow, Oxford and Edinburgh on an experiment on Ko particles at CERN (Geneva,Switzerland) and Rutherford Laboratory (Chilton-Didcot, England), with objective the study of decay ratios and form factors of Strange Particles.

1970 - 1971 **Meteorological Station, CNR (Italian Research Committee) Verona.**

Involved in studies of hail formation and the possible ground control of hailstorms through the use of radar tracking and cloud insemination techniques.

PERSONAL:

U.S. Permanent Resident with Italian citizenship.
Married, one child (Both wife and son are U.S. Citizens).

PUBLICATIONS:

1. E. G. Cazzoli, et al. "Independent Verification of Radionuclide Release Calculations for selected Accident Scenarios," NUREG/CR-4629, (July 1986).

2. C. K. Park and E. G. Cazzoli, "QUASAR screening sensitivity analysis: application to the NAUA Aerosol transport code," BNL Technical Report A-3286 (February 1987).

3. E. G. Cazzoli (co-author), "Evaluation of Severe Accident Risks and Potential for Risk Reduction: Zion Power Plant," NUREG/CR-4551 (February 1987).

4. E. G. Cazzoli (co-author),"Independent verification of the Source Term Code Package (MOD1) : analysis of an ATWS scenario in a BWR Mark I design", BNL Technical Report A-3290 (March 1987).

5. E. G. Cazzoli (co-author), "Zion Risk Rebaselining study: Treatment of Uncertainties and Results", ANS Meeting, San Diego, California (1988).

6. E. G. Cazzoli (co-author), "A Sensitivity Analysis Technique for application to Deterministic Models," ANS Meeting, San Diego, BNL-NUREG-39934 (1988).

7. E. G. Cazzoli (co-author), "Quantification of uncertainties in Source Term Estimates for a BWR with MARK I Containment," ANS Winter Meeting, San Diego, BNL-NUREG-40116 (1988).

8. E. G. Cazzoli (co-author), "Quantification and Uncertainty analysis of Source Terms for Severe Accidents in Light Water Reactors (QUASAR) - Expert Review Groups II-IV," BNL Technical Report A-3286 (July 1986).

9. E. G. Cazzoli (co-author), "The Formulation of Probability Distribution for the QUASAR Program," BNL Technical Report A-3286 (September 1987).

10. H. P. Nourbakhsh, E. G. Cazzoli and M. Khatib-Rahbar,"QUASAR Screening Sensitivity Analysis: Application to the Suppression Pool Aerosol Removal Code (SPARC)," BNL Technical Report A-3286 (July 1987).

11. E. G. Cazzoli (co-author), "QUASAR Screening Sensitivity Analysis: Application to the CORCON and VANESA codes," BNL Technical Report A-3286 (June 1987).

12. E. G. Cazzoli (co-author), "Reanalysis of ZION Risk," 15th Water Reactor Safety Research Information Meeting, October 22 - 25, Gaithersburg, MD, BNL-NUREG-40117 (1987).

13. I. K. Madni, E. G. Cazzoli and M. Khatib-Rahbar, "A Simplified Model for Calculating Early Offsite Consequences from Nuclear Reactor Accidents," NUREG/CR-5164, BNL-NUREG-52153 (July 1988).

14. J. W. Yang, E. Schmidt, E. Cazzoli and M. Khatib-Rahbar, "Analysis of Fission Product Revaporization in a BWR Reactor Coolant System During a Station Blackout Accident," Proceedings of the ANS/ENS International Conference on Nuclear Reactor Thermal Hydraulics, Washington, D. C. (November 1988).

15. E. Cazzoli et al.,"Requantification of the Zion Risk Profile," BNL-NUREG-42262 (January 1989).

16. H. P. Nourbakhsh and E. Cazzoli, "A Simplified Approach for predicting Radionuclide Releases from Light Water Reactor Accidents," BNL Technical Report A-3788 (January 1989).

17. T. Ishigami, E. Cazzoli, M. Khatib-Rahbar, and S. D. Unwin, "Techniques to Quantify the Sensitivity of Deterministic Model Uncertainties", Nuclear Science & Engineering, Volume 101, 371-383 (1989).

18. E. G. Cazzoli (co-author), "A Probabilistic Approach to Quantifying Uncertainties in the Progression of Severe Accidents", Nuclear Science & Engineering, Volume 102, 219-259 (1989).

19. S. D. Unwin, E. G. Cazzoli, R. E. Davis, M. Khatib-Rahbar, M. Lee, H. Nourbakhsh, C. K. Park, and E. Schmidt, "An Information-Theoretic Basis for Uncertainty Analysis: Application to the QUASAR Severe Accident Study," Journal of Reliability Engineering and System Safety, Volume 26, 143-162 (1989).

20. I. K. Madni, E. G. Cazzoli, and M. Khatib-Rahbar, "Assessment of Uncertainties in Early Offsite Consequences from Nuclear Reactor Accidents", Nuclear Technology, Volume 90, 63-77 (1990).

Alan S. Kuritzky
Energy Research, Inc.

EDUCATION:

1981 B.S. Nuclear Engineering, University of Maryland.
1984 NRC PRA Technology Transfer Program.

PROFESSIONAL EXPERIENCE:

1991 - Present **Senior Nuclear Engineer, Energy Research, Inc.**

Task leader for Beznau (a Swiss PWR with large, dry containment) level I & II PRA review. In this capacity, responsible for review and assessment of the PRA logic models, construction of system fault trees, analysis of internal and external events, binning of accident sequences into Plant Damage States (PDSs), review of Containment Event Trees (CETs), and performance of sensitivity and importance analyses to identify key plant, systems and containment vulnerabilities. Also participating in the NRC/IPE back-end review activities. Other responsibilities include, evaluation of accident phenomenology for the Savannah River K-Reactor for DOE/Defense Programs.

1988 - 1991 **Consultant, Advanced Technology Engineering Systems, Inc.**

Provided technical consulting services to the maintenance and technical services directorate of Advanced Technology Engineering Systems Inc. Responsible for development of system unavailability modeling studies used as part of Maintenance Prcgram Optimization projects. These studies involved fault tree analysis and quantification using both generic industry data and plant-specific unavailability data. Also, directed an effort at Rochester Gas and Electric to develop a plant specific data base for the Ginna station individual plant evaluation (IPE).

1984 - 1988 **Consultant, Technology Research Corporation**

Involved in the Connecticut Yankee PRA effort for Northeast Utilities (NU). Responsible for system dependencies analysis and support state development. Performed system fault tree analysis of all onsite and offsite electrical power systems (i.e AC and DC), including time-dependent station blackout analysis. Involved in Connecticut Yankee Integrated Safety Assessment Program (ISAP). Responsible for evaluating and prioritizing the safety benefit of various NRC and utility-initiated concerns.

Also participated in Yankee Rowe Nuclear Power Station (YNPS) PRA effort for Yankee Atomic Power Company. Developed an electric power availability states model and performed system fault tree analysis for all AC electric power systems.

As part of Northeast Utilities Availability Improvement Program (AIP) for their Millstone 1, 2 and 3 and Connecticut Yankee nuclear power plants, co-authored reports identifying potential items for availability improvement, based on historical productivity loss data and current plant activities. Also as part of this program, performed in-depth RAM analysis of the refueling outages at Millstone 1, and the RCS valves and Turbine Protection System at Millstone 2. Served as project leader for the development of a RAM data base for the four NU nuclear units. Documented the RAM data base methodology and constructed generic and plant specific data bases.

1982 - 1984 **PRA Engineer, United States Nuclear Regulatory Commission.**

As a probablistic risk assessment engineer with the U.S. Nuclear Regulatory Commission, performed a generic probablistic analysis of the DC power system for the revision of NUREG-0666, "A Probablistic Safety Analysis of DC Power Supply Requirements for Nuclear Power

Plants," and also as part of a cost benefit analysis for the resolution of generic issue A-30, "Adequacy of Safety Related DC Power Supply." Analysis included modification of event trees, construction of fault trees, failure modes and effects analysis, and human reliability analysis.

Provided technical support for resolution of unresolved safety issue A-44, "Station Blackout." Major accomplishments included a state-of-the-art analysis of loss of offsite power data (including data review and classification, trend analysis and cluster analysis) and the development of a set of models to calculate time-dependent station blackout frequencies.

1981 - 1982 **Nuclear Engineer, Science Applications, Inc.**

As a nuclear engineer with Science Applications, Inc., served as a member of the Interim Reliability Evaluation Program (IREP) team that performed a probablistic risk assessment of Millstone 1. Was involved in the quantification phase using the SETS code.

PUBLICATIONS:

1. "Demonstration of Reliability-Centered Maintenance, Volume 3: First Annual Progress Report from the Ginna Nuclear Power Station," EPRI NP-6152, Interim Report, (coauthor), September 1989.

2. Kuritzky, A.S., Karimi, R., and Herzig, N. S., "Northeast Utilities Availability Improvement Program," 15th Inter-RAM Conference. Portland, Oregon, June 1988.

3. "Methodology for Reliability, Availability and Maintainability (RAM) Data Base," (coauthor), Northeast Utilities Service Company, March 1987.

4. "Availability Improvement Program for MP1, MP2, MP3 and CY-1987 update," (coauthor), Northeast Utilities Service Company, March 1988.

5. "Availability Improvement Program for MP1, MP2, MP3 and CY-1986 update," (coauthor), Northeast Utilities Service Company, December 1986.

6. "Availability Improvement Program for MP1, MP2, MP3 and CY-1985," (coauthor), Northeast Utilities Service Company, October 1985.

7. "Connecticut Yankee Probablistic Safety Study," NUSCC 149, (coauthor), February 1986.

8. "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to USI A-44," NUREG-1032, (coauthor), May 1985.

9. "Interim Reliability Evaluation Program, Analysis of the Millstone Point Unit 1 Nuclear Power Plant," NUREG/CR-3085, (coauthor), February 1983.

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EDUCATION:

1990 Ph.D. Mechanical Engineering, Univ. of Calif., Los Angeles.
1986 M.S. Mechanical Engineering, Univ. of Rochester, New York.
1984 B.S. Mechanical Engineering, I.I.T., Madras, India.

PROFESSIONAL EXPERIENCE:

4/91 - Present **Consulting Engineer, Energy Research Inc.**

Currently involved in analysis of severe accident issues for several Swiss safety studies, including Beznau and Goesgen reactors using MELCOR, SCDAP/RELAP-5 and ERPRA severe accident computer codes. Also developing severe accident melt progression and fission product release models for use in uncertainty quantification (e.g., Direct Containment Heating, Hydrogen Combustion, Ex-Vessel Coolability). Other responsibilities include, review and assessments of Containment Event Trees and containment loading issues for use in risk assessment studies for several Swiss nuclear power reactors.

12/90 - 2/91 **Post-Doctoral Fellow, University of California, Los Angeles.**

Research in application of thermal hydraulics methods to accident management issues. Developed a model to assess reactor pressure vessel lower head failure following severe reactor accidents. Studied the effect of cavity flooding as an accident management strategy. Also developed heat transfer models and finite element codes for application to manufacturing processes and simulation of heat transfer in laser welding and grinding.

1/86 - 1/90 **Graduate Research Engineer, University of California, Los Angeles.**

Doctoral thesis research involved the modeling of the thermal hydraulics of subcooled film boiling on a vertical plate using mathematical models and finite difference methods. Used the model to obtain heat transfer coefficients and cooling rates during quenching. Validated results from the model by experiments. Designed and constructed an experimental apparatus to study film boiling from a vertical plate. Liquid temperatures and flow velocities were measured by holographic interferometry and a hydrogen bubble flow visualisation system respectively.

9/84 - 12/85 **Research Assistant, University of Rochester, Rochester, New York.**

Designed and constructed an experimental apparatus to study flow field in a compressor rotor. Flow Velocities near the tip of a compressor rotor were measured using a Laser-Doppler system. Developed a finite element code based on NASTRAN to solve for the flow field analytically.

PUBLICATIONS:

1. Vijaykumar, R. and Dhir, V.K., "A Holographic study of subcooled film boiling on a vertical surface" , Multiphase flow,heat and mass transfer, HTD. , Vol 109, pp. 125-132, 1989.

2. Dhir, V.K., and Vijaykumar, R., "A Study of Subcooled film boiling on a vertical surface - Hydrodynamic aspects," accepted for publication by ASME Journal of heat transfer.

3. Dhir, V.K., and Vijaykumar, R., "Hydrodynamics and Wave-structure during Subcooled film boiling on a vertical surface", Proceedings of the ASME-JSME engineering joint conference , Volume 4, 55-64 (1991).

4. Vijaykumar, R., and Dhir, V.K., "A Study of Subcooled film boiling on a vertical surface - Thermal aspects," submitted for publication to ASME Journal of heat transfer.

5. Vijaykumar, R. and Gans, R.F., "Laser-Doppler velocimetric study of boundary layer development in the annulus region of a compressor rotor, Journal of turbomachinery, Vol 101, pp. 377-385, 1988.

K. Bingham Cady
CORNELL UNIVERSITY

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TITLE: Professor of Applied and Engineering Physics, Nuclear Science and Engineering.

EDUCATION:

Hyde Park Public High School, Chicago, Illinois.
Attended 1949-51, no degree.

University of Wisconsin, Madison, Wisconsin
College of Letters and Science 1951-53, no degree;
Ford Foundation Pre-induction Scholarship 1951-53,
Phi Eta Sigma.

Massachusetts Institute of Technology, Cambridge, Massachusetts
Naval Architecture and Marine Engineering, S.B. 1956;
Tau Beta Phi 1955.

Northeastern University, Boston, Massachusetts
Graduate evening division, Mathematics and Physics 1956-57, no degree.

Massachusetts Institute of Technology, Cambridge, Massachusetts
Nuclear Engineering, Ph.D. 1967, Sigma Xi 1960, Bethlehem Steel Industrial Fellowship, Woodrow Wilson Fellowship, Atomic Energy Fellowship 1960-62.

PROFESSIONAL EXPERIENCE:

Moore-McCormack Steamship Lines, New York City, New York,
Engineering Cadet, 1954.

Bethlehem Steel Company, Shipbuilding Div., Quincy, Massachusetts
Preliminary hull design, rotating machinery performance and testing, oil tanker economics, nuclear radiation shield design and testing, 1956-59; assignment to Westinghouse Naval Reactor Facility NRTS, Idaho Falls, 1959.

Jackson and Moreland, Inc., Boston, Massachusetts.
Nuclear systems engineering (BONUS), heat transfer and radiation shield design (PM-3A), power generation economics, 1960-62.

Knolls Atomic Power Laboratory, General Electric Company, Schenectady, New York.
On leave from Cornell, Spring 1966; consultant 1966-69; Reactor physics computational methods development, Monte Carlo sampling of scattering laws, thermal and epithermal flux mismatch, control absorbers, fast flux cross section averaging, group transfer scattering matrices, thermal scattering kernels, resonance cross section evaluation.

U.S.Atomic Energy Commission, Division of Reactor Licensing, Bethesda, Maryland.
On leave from Cornell, Fall 1969 and Spring 1970; evaluation of Safety Analysis Reports in support of operating license applications for nuclear power plants.

Brookhaven National Laboratory, Department of Nuclear Energy, Upton, Long Island, New York.
On leave from Cornell, Spring 1978; consultant 1978-82; reactor safety analysis, LMFBR system code development; modeling of thermal, fluid flow, and control systems for sodium-air heat exchangers.

Hanford Engineering Development Laboratory, Westinghouse Hanford Co., Richland, Washington
Consultant 1980-81; LMFBR containment response code development.

Fauske and Associates, Inc., Burr Ridge, Illinois
Consultant 1982-present; LWR degraded core cooling, Clinch River Breeder Reactor probabilistic risk assessment; on leave from Cornell 1983, Modular Accident Analysis Program.

Cornell University, College of Engineering
Assistant Professor (Engineering Physics) 1962-67
Associate Professor (Engineering Physics 1967-84, Nuclear Science and Engineering 1977-84)
Professor (Applied and Engineering Physics, Nuclear Science and Engineering) 1984-present
Associate Dean for Professional Programs (half time) 1984-85
Associate Dean for College Affairs (full time) 1985-present

TEACHING EXPERIENCE

Nuclear Science Seminar
Advanced Thermal Science
Nuclear Reactor Theory I
Nuclear Reactor Theory II

Nuclear Reactor Engineering
Classical Mechanics
Nuclear Reactor Design Seminar
Environmental Impact of Nuclear Energy
Environmental Effects of Electric Power Production
Thermal Pollution and Waste Heat Disposal
Introduction to Nuclear Science and Engineering
Nuclear Chemical Engineering
Fission, Fusion, and Radiation

SOCIETIES AND COMMITTEES

Society of Naval Architects and Marine Engineers, 1955-62
American Nuclear Society, 1960-present
Niagara Finger Lakes Section, ANS, 1960-present, Chairman 1966-67, Executive Committee 1962-69
College of Engineering design Committee, Secretary, 1964
Engineering Library Committee, 1965-68
College Program Committee, 1965-67
Academic Standards Committee of the Graduate School, 1965-67
University Radiation Safety Committee, 1968
Engineering Physics Academic Advisor, 1967-67, 1973-77, 87-present
Division of Basic Studies Academic Advisor, 1963-65, 1967-69, 1971-73, 1987-present
Faculty Advisor, Tau Beta Pi, 1967
Continuing Education Advisory Committee, 1968
Engineering Committee on Heated Discharges to Lakes, 1968
Cornell Energy Project, 1970-73
Faculty Council of Representatives, 1971-74
University Faculty Review and Procedures Committee, 1973-76
Core Curriculum Committee, 1973-74
Graduate Professional Programs Committee, 1965-88, Chairman 1984-88
Engineering Cooperative Education Program, Director, 1984-present
Engineering College Policy Committee, ex-officio 1984-present
Engineering Facilities Committee, Chairman, 1984-present
United Way Campaign, Engineering Chairman 1986-87, University Chairman 1987-88
United Way of Tompkins County, Member, Board of Directors, 1987-present
Elected Trustee, Village of Lansing, 1988-90
Southern Cayuga Lake Intermunicipal Water Commission, Commissioner, 1988-90

PH.D. RESEARCH SUPERVISION

Charles R. MacVean,"A Simplified Cell Theory Applied to the Calculation of Thermal Neutron Spectra in Light Water Lattices", CURL-10, September 1964.

William E. Schilling,"Neutron Transport Theory Analysis of Central Water Gaps Effects in the Cornell University Critical Assembly", CURL-14, September 1965.

Ehud Greenspan, "Theory and Measurement of Neutron Importance in Nuclear Reactors", CURL-15, September 1966

Christopher F. Masters, "A Procedure for Evaluating Modified Pulsed Source Experiments in Subcritical Nuclear Reactors", CURL-16, September 1966

Burt A. Zolotar, "Monte Carlo and Experimental Analyses of Nuclear Reactor Fluctuation Models", CURL-20, September 1967 (with M. S. Nelkin)

Tohru Haga, "The Physics of Water Gaps in Light Water Reactors", CURL-22, September 1967

David J. Osias, "Investigation of Space Dependent Fast Fission Ratios, and Measurements of Fission Fragment Recoil Ranges in Aluminum", CURL-23, September 1967

Michael W. Golay, "Neutron Pulse Propagation in Multiplying Assemblies", CURL-32, 1969

Paul Young, "Evaluation of Fissile Cross Sections from 10 keV to 10 MeV" (no degree)

Gad Shani, "Neutron Wavesin Reflected Reactors", CURL-378, September, 1970

Robert W. Schleicher, "Stochastic Decision Making Applied to Nuclear Reactor Safety", CEP-72-20, August 1972

Chaim Braun, "An Electric Power System Expansion Model" (no degree)

Grigore F. Pavlenco, "Transient Analysis of the Primary System of a Liquid Metal Cooled Fast Breeder Reactor Plant", CURL-49, January, 1976

Mohsen Khatib-Rahbar, "System Modeling for Transient Analysis of Loop-Type Fast Breeder Reactors", CURL-53, May 1978

Sati P. Mitra, "Stochastic Models and Reliability Parameter Estimation Applicable to Nuclear Power Plant Safety", CURL-56, 1979

John Raber, "System Modeling for Transient Analysis of Pool-Type Fast Breeder Reactors", CURL-57, 1980

Marc A. Kenton, "A General Theory for Simulations of Nonlinear Time-Dependent Systems", CURL-58, 1981

Randall J. Miller, "A Computer Implementation of Response Theory", August 1985.

M.S. RESEARCH SUPERVISION

Teruo Tamai, "Fast Effect Calculations in Light Water Reactors", 1964
David T. Austin, "Experimental Analysis of Rossi-Alpha and Noise Parameters in a Subcritical Reactor", CURL-21, September 1967
Dennis Bley, "Boiling Water Reactor Safety"
Doyle Mitchell, "Effects of Random Forced Outages on Power System Expansion Strategies"
Alex Borghi,"Control System Simulation of LMFBRs," CURL-49, 1976
Anthony Hui, "NUFUEL, A Computer Code for Calculating the Nuclear Fuel Cycle Cost of a Light Water Reactor", CURL-51, September 1977
Joseph Uzdzinski III, "Sensitivity Analysis of Multiple Ionization in an Electron Beam Ion Source", May 1986.

PUBLICATIONS

1) K.B. Cady, "Economic Comparison of Tankers: 18,000 Dwt to 1000,000Dwt", Journal of Commerce October 1956

2) K.B. Cady and C.R. Mac Vean, "A simplified Formulation of Space-Energy Cell Theory", Trans. Am. Nucl. Soc. 6, 214 (1963).

3) K.B. Cady and M. Clark Jr., "Neutron Transport in Cylindrical Rods", Nucl. Sci. Eng. 18, 491 (1964).

4) K.B. Cady, T. Tami, and W.E. Schilling, "Escape Probability formulation for Intensive Reactor Quantities", Trans. Am. Nucl. Soc. 7, 257 (1964).

5) K.B. Cady, C.F. Masters, and B.A. Zolotar, "Asymptotic Reactor Theory Analysis of the Garelis-Russell Experiment", Trans.Am. Nucl. Soc. 8 432 (1965).

6) E. Greenspan and K.B. Cady, "Some Applications of the Thermal-Neutron Importance Function to Unit-Cell Studies", Trans. Am. Nucl. Soc. 9, 243 (1966).

7) E. Greenspan and K.B. Cady, "Calculation of Thermal-Neutron Importance Functions in Heterogeneous Lattices", Trans. Am. Nucl. Soc. 9, 242 (1966).

8) E. Greenspan, K.B. Cady, and J.P. Howe, "Economic Potential of Variable Enrichment Fuel Elements for Power Reactors", Trans. Am. Nucl. Soc. 9, 295 (1966).

9) E. Greenspan, K.B. Cady, and H.C. Aderhold, "Measurement of the Fast Importance Function Distribution and of the Geometric Buckling Employing a Californium-252 Spontaneous Fission Source", Trans. Am. Nucl. Soc. 9, 491 (1966).

10) C.F. Masters and K.B. Cady, "A Procedure for Calculating the Parameters Measured by Modified Pulse Source Experiments", Trans. Am. Nucl. Soc. 9, 510 (1966).

11) C.F. Masters and K.B. Cady, "A Procedure for Evaluating Modified Pulsed Source Experiments in Subcritical Nuclear Reactors", Nucl. Sci. Eng. 29, 272 (1967).

12) K.B. Cady, G.J. Kirouac, and T.J. McInereney (KAPL), "The Mismatch of Thermal and Epithermal Fluxes", Nucl. Sci. Eng. 29, 299 (1967).

13) D.Osias and K.B. Cady, "Investigations of Space Dependent Values of Sigma-28 in Light Water UO-2 Cores", Trans. Am. Nucl. Soc. 10, 586 (1967).

14) M.W. Golay and K.B. Cady, "A Formulation of Neutron Dispersion Relations in Terms of Reactivity", Trans. Am. Nucl. Soc. 10, 588 (1967).

15) D.T. Austin, B.A. Zolotar, and K.B. Cady, "Comparison of the Waiting Time Alpha with the Rossi-Alpha", Trans. Am. Nucl. Soc. 10, 591 (1967).

16) K.B. Cady and F. Feiner, "University Research in Nuclear Power", Engineering Cornell Quarterly 3 No.3 (1968).

17) M.W. Golay and K.B. Cady, "Axial Neutron Pulse Propagation in Multiplying Cylinders", Trans. Am. Nucl. Soc. 12, 659 (1969).

18) M.W. Golay and K.B. Cady, "Transform Domain Analysis of Neutron Pulse Propagation Experiments", Trans. Am. Nucl. Soc. 12, 678 (1969).

19) E. Greenspan and K.B. Cady, "The Measurement of Neutron Importance Functions", J. Nucl. Energy 24, 529 (1970).

20) M.W. Golay and K.B. Cady, "Neutron Pulse Propagation in Multiplying Assemblies", Nucl. Sci. Eng. 43, 303 (1970).

21) G. Shani and K.B. Cady,"Neutron Waves in a Reflected Reactor", Trans. Am. Nucl. Soc. 14, 866 (1971).

22) K.B. Cady and J. Hass, "Electric Utility Optimum Mix Model", CEP Report 71-3, January 1971

23) J.A. Franco and K.B. Cady, "Review and Analysis of Radwaste Management for Light Water Reactors," CEP Report 71-10, August 1971.

24) J.A. Franco and K.B. Cady, "Radioactive Waste Management at Nuclear Fuel Reprocessing Plants", CEP Report 72-2 February 1972.

25) R.W. Schleicher and K.B. Cady, "An Estimate of the Risk of Accidental Radiological Exposure from a Boiling Water Reactor", CEP Report 72-12, April 1972.

26) R.W. Schleicher and K.B. Cady, "Stochastic Decision Making Applied to Nuclear Reactor Safety", Trans. Am. Nucl. Soc. 15, 827 (1972).

27) C. Braun and K.B. Cady, "An Electric Power System Expansion Model", CEP Report 73-1, February 1972.

28) K.B. Cady, M Khatib-Rahbar, G.F. Pavlenco and J.R. Raber, "A Transient Simulation Code for LMFBR's ", Trans. Am. Nucl. Soc.27, 543 (1977).

30) K.B. Cady, M Khatib-Rahbar, G.F. Pavlenco and J.R. Raber, "CURL-L, Dynamic Analysis of Loop-Type Liquid Metal Cooled Fast Breeder Reactors", CURL-52, November 1977.

31) K.B. Cady, G.F. Pavlenco and M. Khatib-Rahbar, "A Liquid Metal Cooled Fast Breeder Reactor Simulation", CURL-50, July 1976.

32) K.B. Cady and A. Hui, "NUFUEL, A Computer Code for Calculating the Nuclear Fuel Cycle Cost of a Light Water Reactor", CURL-51, August 1977.

33) K.B. Cady and M. Khatib-Rahbar, "Establishment of Buoyancy Induced Natural Circulation in Loop-Type LMFBR's " Trans. Am. Nucl. Soc.38, 432 (1978).

34) K.B. Cady"An Air-Blast Heat Exchanger System Model", BNL-NUREG-26237, June 1979.

35) S. Mitra and K.B. Cady, "Bayesian Prediction Model for Fire

Occurrences in Nuclear Power Plants", Trans. Am. Nucl. Soc. 31, (1979)

36) K.B. Cady, G.F. Pavlenco, M. Khatib-Rahbar, and J.R. Raber, "EPRI-CURL, Dynamic Analysis of Loop-Type LMFBR's", EPRI-NP-1001, 1979.

37) S.P. Mitra and K.B. Cady, "Estimation of Common-Mode Failures from Near-Miss Data", BNL-NUREG-26321 (1979).

38) M. Khatib-Rahbar and K.B. Cady, "Stability and Accuracy Criteria for Numerical Solution of Large Systems of Ordinary Differential Equations", Trans. Am. Nucl. Soc. 32, (1980).

39) K.B. Cady and J.R. Raber, "EPRI-CURL-P, Systems Modeling for a Pool-Type Liquid Metal Fast Breeder Reactor", EPRI-NP-2170 (1981).

40) M.G. Peipho, K.B. Cady, and M.A. Kenton, "An Importance Theory for Nonlinear Ordinary Differential Equations", HEDL-TME 81-24, UC-79T, Tp (1981).

41) M. Khatib-Rahbar and K.B. Cady, "Dynamic Models and Numerical Simulation of System-Wide Transients in Loop-Type LMFBR's", Nucl. Eng. and Design 64, 259-281 (1981).

42) K.B. Cady and M.A. Kenton, "An Importance Theory for Lumped Parameter Systems", Trans. Am. Nucl. Soc. 38, 331 (1981).

43) K.B. Cady and M.A. Kenton, "A General Theory of Response", CURL-60, 1982, revised 1983.

44) K.B. Cady, R.E. Henry, and J.T. Madell, "Steam Venting During a Loss-of-AC-Power Transient", FAI/82-18, August, 1982.

45) R.E. Henry, M.A. Kenton, J.R. Gabor, R.W. MacDonald, A. Sharon, G.M. Hauser, T.E. Ewing, D.R. MacFarlane, and K.B. Cady, "Modular Accident Analysis Program", Vol. I and II, Fauske and Associates, Inc. 1983.

46) K.B. Cady and R.J. Miller, "An Application of Response Theory to Steam Venting During a Loss-of-AC-Power Transient", Nucl. Eng. and Design 101 (1987).

47) K.B. Cady, C.D. Pettygrove, and D.K. Westby, "Quantifying Uncertainty in Investment Analysis", Real Estate Review 16, No. 1, Spring 1986.

48) K.B. Cady, "Educating Masters of the Engineering Profession", Engineering Cornell Quarterly 21, No. 4, Summer, 1987.

49) K.B. Cady, S.L. Phoenix, T.N. Rhodin, "Restructuring Graduate Engineering Education: The MEng.Program at Cornell", Engineering Education 87, No. 11, 1988.

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Associate Editor, ASME Symposium Volume, "Topics in Two-Phase Heat Transfer and Flow", 1978.
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January 1988 - June 1991:	Professor and Vice Chairman in Mechanical, Aerospace and Nuclear Engineering Department University of California, Los Angeles. Responsible for undergraduate affairs of the Department. Also, serve as Mechanical Engineering Undergraduate Student Advisor.
July 1982 - December 1987:	Professor in Mechanical, Aerospace and Nuclear Engineering Department University of California, Los Angeles
July 1978 - June 1982:	Associate Professor in Chemical, Nuclear, and Thermal Engineering Department, University of California, Los Angeles.
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JOURNAL PUBLICATIONS

Dhir, V.K., Lienhard, J.H., "Laminar Film Condensation on Plane and Axi-Symmetric Bodies in Nonuniform Gravity," J. Heat Transfer, Trans. ASME, Series C, 93-1:97-100, February, 1971.

Lienhard, J.H., Eichhorn, R., and Dhir, V., "Laminar Natural Convection Under Nonuniform Gravity", J. Heat Transfer, Trans. ASME, Series C., Vol. 94, 1:80-86, February, 1972.

Dhir, V.K., Lienhard, J.H. Technical Brief, discussion (and application to film boiling) of ASME paper No. 71-FE-7, \fIJ. Basic Engr., Trans. ASME\fR, Series D, 94-1:160-162, March 1972.

Lienhard, J.H., and Dhir, V., Technical Brief, "A Simple Analysis of Laminar Film Condensation with Suction", J. Heat Transfer, Trans. ASME, Series C, Vol. 94, 3:334-336, August, 1972.

Lienhard, J.H., and Dhir, V., "Hydrodynamic Prediction of Peak Pool Boiling Heat Fluxes from Finite Bodies", presented at the Winter Annual Meeting of ASME at New York, N.Y., November 26-30, 1972. J.

Heat Transfer, Trans. ASME, Series C., Vol. 95, 2:152-158, May, 1973.

Lienhard, J.H., Dhir, V.K. Technical Brief, Discussion of ASME paper No. 72-HT-R, J. Heat Transfer, Trans. ASME, Series C, 95-2:178, May 1973.

Sernas, V., Lienhard, J.H., Dhir, V.K. Technical Brief, "The Taylor Wave Configuration During Boiling from a Flat Plate," Int. J. Heat Mass Transfer, 16-9:1820-1822, September 1973.

Lienhard, J.H., Dhir, V.K., Riherd, D.M. "Peak Pool Boiling Heat Flux Measurements of Finite Horizontal Flat Plates." presented at the Winter Annual Meeting of ASME at Detroit, MI, November 11-15, J. Heat Transfer, Trans. ASME, Series C, 95-4:477-482, November 1973.

Dhir, V., and Lienhard, J.H., "Taylor Stability of Viscous Fluids with Application to Film Boiling", Int. J. Heat Mass Transfer, Vol. 16, 11:2097-2109, November, 1973.

Dhir, V.K., Lienhard, J.H. "Similar Solutions for Film Condensation with Variable Gravity or Body Shape," J. Heat Transfer, Trans. ASME, Series C, 95-4:483-486, November 1973.

Dhir, V.K. Lienhard, J.H. "Peak Boiling Heat Flux in Viscous Liquids," J. Heat Transfer, Trans. ASME, Series C, 96-1:71-78, February 1974.

Lienhard, J.H., Dhir, V.K. "Laminar Film Condensation on Non-Isothermal and Arbitrary Heat-Flux Surfaces and on Fins," J. Heat Transfer, Trans. ASME, Series C, 96-2:197-203, May, 1974.

Dhir, V.K., Lienhard, J.H. Technical Brief, "Laminar Film Condensation on Submerged Isothermal Bodies," J. Heat Transfer, Trans. ASME, Series C, 96-4:555-557, November 1974.

Gat, U., Dhir, V.K., Crewe, G.F. "Effect of Pumping on Caking Properties of Coal," Canadian Inst. of Mining and Metallurgy Bulletin, p. 1-4, December 1974.

Crewe, G.F., Gat, U., Dhir, V.K. "Alkaline Solution Decaking of Bituminous Coal," Fuel, 54-1:20-23, January 1975.

Dhir, V.K. "Quasi-Steady Laminar Film Condensation of Steam on Copper Spheres," J. Heat Transfer, Trans. ASME, Series C, 97-3:347-351, August 1975.

Dhir, V.K., Catton, I. "Dryout Heat Fluxes for Inductively Heated Particulate Beds," J. Heat Transfer, Trans. ASME, Series C, 99-2:250-256, May 1977.

Dhir, V.K., Wong, K., and Kastenberg, W.E., "Solidification of a Sphere with Integral Heat Generation and Convective Cooling at the Surface," Nuclear Sci. and Engr., Vol. 63, No. 2:350-356, July, 1977.

Dhir, V.K., Castle, J., Catton, I. "Role of Taylor Instability on Sublimation of a Horizontal Slab of Dry Ice," J. Heat Transfer, Trans. ASME, Series C, 99-3:411-418, August 1977.

Rhee, S.J., Dhir, V.K., Catton, I. "Natural Convection Heat Transfer in Inductively Heated Particles," Paper No. 77-WA/HT-2, presented at Winter Annual Meeting of ASME, Atlanta, Nov. 27-Dec. 2, 1977, J. Heat Transfer, Trans. ASME, Series C, 100-1:78-85, February 1978.

Dhir, V.K., Purohit, G.P., "Subcooled Film Boiling Heat Transfer from Spheres," Paper No. 77-HT-78 presented at AICHE-ASME Heat Transfer Conf., Salt Lake City, UT, August 15-17, 1977, Nuclear Engineering and Design, 47:49-66, May, 1978.

Wong, K., Dhir, V.K., and Kastenberg, W.E., "A Mechanisitic Study of Fuel Freezing, Channel Plugging and Continued Coolability During Fast Reactor Overpower Excursions," Nuclear Technology, Vol. 39, pp. 121-137, July, 1978.

Taghavi-Tafreshi, K., Dhir, V.K., Catton, I. "Thermal and Hydrodynamic Phenomena Associated with Melting a Horizontal Substrate," Paper No. 78-WA/HT-44. Presented at ASME Winter Annual Meeting, San Francisco, Dec. 1978, J. Heat Trnasfer, Trans. ASME, Series C, 101-102, 318-325, May 1979.

Dhir, V.K., Catton, I. "Dryout Heat Fluxes in Debris Beds Cooled at the Bottom and Having Subcooled Liquid at the top," presented at Int. Topical Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, October 1978, Nuclear Technology, Vol. 46:356-361, 1979.

Dhir, V.K., and Kastenberg, W.E., "Thermal Hydraulic Analysis of Fuel Blockages Formed During Transient Overpower Accidents in LMFBR's," Presented at the International Topical Meeting on Nuclear Power Reactor Safety, Brussels, Belgium, October, 1978, Nuclear Technology, Vol. 46:447-452, 1979.

Dhir, V.K., "On the Use of Minimum Film Boiling Temperature to Predict Coherent Molten Metal Coolant Interactions," Nuclear Science and Engineering, 70-2:119-126, May 1979.

Taghavi-Tafreshi, K., Dhir, V.K. "Taylor Instability in Boiling, Melting and Condensation or Evaporation," Int. J. of Heat and Mass Transfer, 23:1433-1445, 1980.

Varela, D.W., Kastenberg, W.E., Dhir, V.K. "Implications of In-Channel Plugging During Fast Reactor Overpower Excursions," Nuclear Technology, 49-102-114, June 1980.

Lienhard, J.H., Dhir, V.K. "Prediction of the Minimum Pool Boiling Heat Flux," J. of Heat Transfer, 102-3:457-460, August 1980.

Dhir, V.K., Technical Brief, "Sublimation of a Slab of Dry Ice--An Analog of Pool Boiling on a Flat Plate," J. Heat Transfer, Trans., ASME, Series C, 102, 2:380-382, May, 1980.

Dhir, V.K., Discussion of the paper, "On the Minimum Film Boiling Conditions for Spherical Geometries," J. Heat Transfer, 102: 790-791, November, 1980.

Dhir, V.K., Duffey, R.B., Catton, I. "Quenching Studies On a Zircaloy Rod Bundle," J. of Heat Transfer, 103-2:293-299, May 1981.

Taghavi-Tafreshi, K., Dhir, V.K. "Analytical and Experimental Investigation of Simultaneous Melting Condensation on a Vertical Wall," Heat Transfer, 104:25-33, 1982.

Dhir, V.K., Catton, I. "Boiling in a Porous Bed," Mechanics and Physics of Bubbles in Liquids, Vol. 38, 69-76, 1982.

Naik, A.S., Dhir, V.K. "Forced Flow Evaporative Cooling of a Volumetrically Heated Porous Layer," Intl. J. Heat and Mass Transfer, Vol. 25, No. 4, 541-552, 1982.

Chen, C.L., Dhir, V.K. "Hydrodynamics of a Bubble Formed at Vent Pipe Exit," Intl. J. Multiphase Flow, Vol. 8, 147-163, 1982.

Conn, R.W., Dhir, V., Ghoniem, N.M. et al. "Studies of the Physics and Engineering of Deuterium-Deuterium Barrier Tandem Mirror Reactors," Nuclear Technology/Fusion, 2-4:563-589, October 1982.

Dhir, V.K. "On the Coolability of Degraded LWR Cores," Nuclear Safety, Vol. 24, No. 3, May-June 1983.

Taghavi-Tafreshi, K., Dhir, V.K., "Shape Change of an Initially Vertical Surface Undergoing Condesnation-Driven Melting," J. Heat Transfer, 105:235-240, May 1983.

Weede, J., Dhir, V.K., "Critical Heat Flux Enhancement Using Local Tangential Flow Injection," Nuclear Technology, 4-2:483-488, Sept. 1983.

Brown, M.L., Keller, L., Deshpandey, C., Bunshah, R.F., Dhir, V.K., Taylor, R.J., "An Experimental Study of the Thermal Performance of Refractory Coatings Exposed to Tokamak Plasma," J. Nucl. Materials, 122 & 123:1140-1144, 1984.

Tsai, F.P., Jakobsen, J., Catton, I., Dhir, V.K., "Dryout of an Inductively Heated Bed of Steel Particles with Subcooled Flow from Underneath the Bed," Nucl. Tech., 65:10-15, April 1984.

Drucker, M.I., Dhir, V.K., Duffey, R.B. "Two Phase Heat Transfer in Tubes and over Rod Bundles and Blockages," Journal Heat Transfer, 106:856-864, November 1984.

Marshall, J., Dhir, V.K., "Effect of Overlying Liquid Layer on Dryout Heat Flux Measurements," Nucl. Sci. and Engr., Vol 91, 109-113, 1985.

Bui, T.D., Dhir, V.K., "Film Boing Heat Transfer on an Isothermal Vertical Surface," J. Heat Transfer, 107, 4:764-771, 1985.

Singh, S., Dhir, V.K., "Scaling of Thermal Oscillations in Tubes of Once Through Steam Generators," J. Solar Energy Engineering, Vol. 10, No. 4, 343-351, 1985.

Galamba, D., Dhir, V.K. "Transient Simultaneous Condensation and Melting of a Vertical Surface," J. Heat Transfer, 107, 4:812-818, 1985.

Bui, T.D., Dhir, V.K., "Transition Boling Heat Transfer on a Vertical Surface," J. Heat Transfer, Vol. 107, No. 4, 756-763, 1985.

Dhir, V.K., "Some Aspects of Two Phase Flow Through Porous Media," Nucl. Engr. & Design, Vol. 95, Nos. 1-3, 1986.

Galamba, D., Dhir, V.K., Taghavi, K., "Analytical and Experimental Investigation of Simultaneous Melting Condensation on a Vertical Wall in the Presence of a Non Condensible Gas," Int'l. J. Heat and Mass Transfer, Vol. 29, No. 12, 1869-1880, 1986.

Marshall, J., Dhir, V.K., "On the Predictin of Porosity of Beds Composeed of Mixtures of Spherical Particles," Chemical Engineering Communications, Vol. 45, 1986.

Tung, V.X., and Dhir, V.K., "Quenching of Debris Beds Having Variable Permeability in the Axial and Radial Directions," Nucl. Engr. and Design, Vol. 99, 1987.

Dhir, V.K., Scott, J.H., "On the Superposition of Injection Induced Swirl During Enhancement of Subcooled Critical Heat Flux," Int'l. J. Heat Mass Transfer, Vol. 30, No. 10, pp. 2013-2022, 1987.

Dhir, V.K., Amar, R. and Mills, J.C., "A One Dimensional Model for Prediction of Stratification in Horizontal Pipes Subjected to Fluid Temperature Transient at Inlet, Part I - Hydrodynamic Model," Nuclear Engineering and Design, Vol. 107, 1988.

Dhir, V.K., Tung, V.X., Amar, R. and Mills, J.C., "A One Dimensional Model for Prediction of Stratification in Horizontal Pipes Subjected to Fluid Temperature Transient at Inlet, Part II - Thermal Model," Nuclear Engineering and Design, Vol. 107, 1988.

Tung, V.X. and Dhir, V.K., "A One Dimensional Model for Two Phase Flow Through Porous Media," Int'l. J. Multiphase Flow, Vol. 3, 1988.

Wang, C.H.S. and Dhir, V.K., "Experimental Investigation of Multidimensional Quenching of a Simulated Core-Debris," Nucl. Engineering and Design, Vol. 107, 1988.

Galamba, D. and Dhir, V.K., "Transient Condensation - Melting of a Subcooled Vertical Surface," Numerical Heat Transfer, 1988.

Dhir, V.K. and Liaw, S.P., "Framework for a Unified Model for Nucleate and Transition Pool Boiling," Jour. Heat Transfer, Vol. 111, No. 3, 1989.

Liaw, S.P. and Dhir, V.K., "Void Fraction Measurement During Saturated Pool Boiling of Water on Partially Wetted Vertical Surfaces," Jour. Heat Transfer, Vol. 111, No. 3, 1989.

Guo, Z. and Dhir, V.K., "Single and Two Phase Heat Transfer in Tangential Injection Induced Swirl Flow," Int'l J. Heat and Fluid Flow, Vol. 10, No. 3, 1989.

Kianjah, H. and Dhir, V.K., "Experimental and Analytical Investigation of Disperse Flow Heat Transfer in Rod Bundles," Int'l. J. Experimental Thermal and Fluid Science, Vol. 2, 1989.

Brown, M.L., Dhir, V.K. and Taylor, R.J., "Poloidal and Toroidal Heat Flux Distribution in CCT," Nucl. Engrng. and Design/Fusion, 1990.

Tung, V.X., and Dhir, V.K., "Experimental Study of Boiling Heat Transfer from a Sphere Embedded in Liquid Saturated Porous Media," Journal of Heat Transfer, Vol. 112, No. 2, 1990.

Tung, V.X., and Dhir, V.K., "Finite Element Solution of Multi-Dimensional Two Phase Flow Through Porous Media with Arbitrary Heating Condtions," Int'l. J. Multiphase Flow, Vol. 16, No. 6, 1990.

Hung, T.C., and Dhir, V.K., "A Study of Decay Heat Removal in SAFR with the MDSNP Code," Nuclear Technology, Vol. 90, No. 3, 1990.

Hung, T.C., and Dhir, V.K., "The Investigation of Conjugate Heat Trnasfer Phenomena in Advanced Fast Reactors," Nuclear Technology, Vol. 92, No. 3, 1990.

PAPERS PUBLISHED IN PROCEEDING OR RECORDS OF CONFERENCES & SYMPOSIA

Lienhard, J.H., Dhir, V.K. "On Making Hyudrodynamic Predictions of the Peak and Minimum Boiling Heat Fluxes," Paper No. 248, IV All Union Heat and Mass Transfer Conf. held at Minsk, USSR, May 15-20, 1972.

Dhir, V.K., Catton, I. "Predictions of Dryout Heat Fluxes in Beds of Bolumetrically Heated Particles," Proc. Int. Meeting on Fast Reactor Safety and Related Physics, IV:2026-2035, October 1976.

Dhir, V.K., Castle, J., Catton, I., Kastenberg, W.E., Doshi, J. "Role of Wall Heat Transfer and Other System Variables on Core Compaction and Recriticality," Proc. Int. Meeting on Fast Reactor Safety and Related Physics, III:1172-1182, October 1976.

Wong, K., Dhir, V.K., Kastenberg, W.E. "A Study of Fuel Freezing and Channel Plugging During Hypothetical Overpower Transients," Proc. Int. Meeting on Fast Reactor Safety and Related Physics, p. 1757-1766, October 1976.

Chan, C.K., Dhir, V.K., Catton, I. "Water Pool Free Surface Response Under Transient Air Injection," Proc. of Two Phase Flow and Heat Transfer Symposium Workshop, held at Ft. Lauderdale, October 1976, p.1-7.

Dhir, V.K., Chan, C.K., Catton, I. "Air Bubble Behavior and Associated Hydrodynamic Forces During Vent Clearing in a BWR Suppression Pool," Proc. Int. Conf. on Thermal Reactor Safety, Sun Valley, ID, August 1977.

Dhir, V.K., Catton, I., Cho, D. "Heat Transfer from a Liquid Pool to a Decomposing Solid Surface," Proc. Third Post Accident Heat Removal Information Exchange, Argonne ANL 78-10, p. 141-142, Nov. 1977.

Dhir, V.K., Catton, I., "The Effect of Particle Size Distribution and Non-Heated Consitutents in Debris Bed Dryout," Proc. Third Post Accident Heat Removal Information Exchange, Argonne ANL 78-10, p. 62-73, November 1977.

Dhir, V.K., Catton, I., "Study of Transient Air Injection into a Pool of Water," Proc. of OECD Conf., Paris, France, June 11-14, 1978.

Dhir, V.K., "Study of Transient Transition Boiling Heat Fluxes from Spheres Subjected to Forced Vertical Flow," Proc. Int'l. Heat Transfer Conf., Toronto, Canada, 2:451-457, August 1978.

Chan, C.K., Dhir, V.K., Catton, I., "Vent Clearing Phenomena and Hydrodynamic Forces During Transient Injection of Air in a Pressure Suppression Pool," Proc. Int'l. Heat Transfer Conf., Toronto, 5:113-119, August, 1978.

Dhir, V.K., Catton, I., "Natural Convective Cooling of a Debris Bed," Proc. 4th Int'l. Post Accident Heat Removal Information Exchange Mtg., Ispra, Italy, October 1978. (Also publ. in European Appied Research Reports, 1-6:34-54, 1979.)

Brinsfeld, W.A., Toman, W.I., Dhir, V.K., Catton, I., "Heat Transfer and Penetration from a Heated Pool to a Substrate," Proc. Int'l. Mtg. on Fast Reactor Safety and Tech., Seattle, August 1979, IV:2140-2149.

Dhir, V.K., "Dryout Heat Fluxes in Debris Beds Containing Particles of Different Size Distributions," Proc. Int'l. Mtg. Fast Reactor Tech., Seattle, August 1979, II:770-780.

Dhir, V.K., Guarro, S., Lin, J.C., Motamed, M., Okrent, D., "Pressurized Water Reactor (PWR) System Simulation and Disturbance Analysis for Anomalous Transients and Degraded System Conditions," Proc. Winter Simulation Conf., San Diego, 1979, 1:135-143.

Dhir, K.V., Cheng, H.M., "Transient Pool Boiling in Spheres Subjected to an External Pressure Pulse," Paper No. HT-12, 19th Nat'l. Transference Conf., Orlando, FL, July 1980.

Grotz, S., Dhir, V.K. "Pebble Bed Blankets for D-D Fusion Reactors," AIChE Symposium Series, 77-208:39-43, August 1981.

Ghoniem, N.M., Conn, R.W., Dhir, V.K., Grotz, S. "Pressure Vessel Blanket Concept for D-D Tandem Mirror Fusion Reactors," AIChE Symposium Series, 77-208: 35-38, August 1981.

Dhir, V.K., Chu,W., Lee, H. and Catton, I., "On the Hydrodynamic Characteristics of Two Phase Flowthrough Porous Media," Proceedings of NRC Topical Meeting on Reactor Thermal Hydraulics, NUREG CP-0043, 295-317, Sept., 1982.

Swanson, D.G., Catton, I., Dhir, V.K. "A Thoria Rubble Bed for Post Accident Core Retention," Proceedings of Fifth Post Accident Heat Removal Conference, Karlsruhe, Germany, 307-312, Nov. 1982.

Tsai, F.P., Jakobsson, J., Catton, I., Dhir, V.K. "Dryout Heat Flux in Debris Beds with Flow from Below," Proceedings of the Fifth Post Accident Heat Removal Conference, Karlsruhe, Germany, pp. 165-168, Nov. 1982.

Wong, R.M., Catton, I., Dhir, V.K. "Solid Particle Penetration of a Melting Substate," Proceedings of Fifth Post Accident Heat Removal Conference, Karlsruhe, Germany, 248-250, Nov. 1982.

Marshall, J., Dhir, V.K. "On the Counter Current Flow Limitations in Porous Media," Proc. of Int'l Meeting on Light Water Reactor Severe Accident Evaluation, Boston, Massachusetts, Vol. 2, 18.5-1 - 18.5-7, 1983.

Ahmed, S.Z., Dhir, V.K., "On the Simulation of Penetration of Solidified Core Material into Concrete," Proc. Int'l. Mtg. on Light Water Reactor Severe Accident Evalution, Boston, Massachusetts, Vol. 2, 12.9-1 - 12.9-7, 1983.

Chu, W., Dhir, V.K., Marshall, J., "Study of Pressure Drop, Void Fraction and Relative Permeabilities of Two Phase Flow Through Porous Media," AIChE Symp. Series, No. 225, Vol. 79, 1983.

Tung, V.X., Dhir, V.K., Squarer, D. "Forced Flow Cooling Studies of Volumetrically Heated Porous Layers," Proc. Int'l. Conf. on Nuclear Reactor Thermal Hydraulics, Santa Barbara, Jan. 1983, p. 876-883.

Tung, V.X., Dhir, V.K. "Quenching of a Hot Particulate Bed by Bottom Flooding," Proc. Joint ASME-JSME Heat Transfer Conf., Honolulu, Hawaii, March 1983, pp. 489-495.

Weede, J., Dhir, V.K. "Experimental Study of the Enhancement of Critical Heat Flux Using Tangential Flow Injection," Fundamentals of Heat Transfer in Fusion Systems, HTD-Vol. 24, 1983.

Kianjah, H., Dhir, V.K., Singh, A. "Enhancement in Heat Transfer in Disperse Flow and Downstream of Blockages in Rod Bundles," Proc. Int'l. Mtg. on Thermal Nuclear Reactor Safety, Sept. 10-13, 1984, Karlsruhe, W. Germany.

Tung, V.X., Dhir, V.K. "On Fluidization of a Particulate Bed During Quenching by Flooding From the Bottom," Proc. 6th Information Exchange Mtg. on Debris Coolability, November 79, 1984, UCLA.

Tung, V.X., Dhir, V.K., Squarer, D. "Quenching by Top Flooding of a Heat Generating Particulate Bed with Gas Injection at the Bottom," Proc. 6th Int'l. Exchange Meeting on Debris Coolability, November 7-9, 1984, UCLA.

Dhir, V.K., Amar, R.C., Mills, J.C. "Stratification in Horizontal Pipes Subjected to Fluid Temperature Transient at Inlet," Winter Annual Meeting of the American Society of Mechanical Engineers, Paper # 84-WA/HT-4, December 1984.

Pafford, D., Dhir, V.K., Anderson, E.B., and Cohen, J. "A Model for Ground Surface Heating During a Prescribed Burn", Proc. 8th Conf. on Fire and Forest Meteorology, Society of American Foresters, pp. 24-32, 1985.

Dhir, V.K., Amar, R.C., Mills, J.C. "A Thermal-Hydrodynamic Model for Predicting Stratification in Horizontal Pipes Subjected to Fluid Temperature Transient at Inlet", Proc. 2nd Spec. Mtg. on Natural Circulation Phenomena, Brookhaven Nat'l. Lab, April 1985, pp. 1-63.

Dhir, V.K., Mills, J.C. "On the Design Options to Mitigate Stratification in LMFBR Piping", Proc. Int'l. Mtg. on Fast Reactor Safety, Knoxville, TN, April 21-25, 1985, pp. 451-456.

Dhir, V.K. "The Role and Limitations of Hydrodynamic Theory in Predicting Boiling Transitions", Review Paper given at 8th Nat'l. Heat Transfer Conf., Visakhapatnam, India, Dec. 1985.

Dhir, V.K. "Enhancement in Heat Transfer as a Result of Tangential Injection", Proc. 4th Symp. on Energy Engrg. Sciences, (Chemical

Process and Control; Nonlinear Systems) ANL, May 1986, pp. 45-54, Argonne National Lab, Argonne, Illinois.

Liaw, S.P. and Dhir, V.K., "Effect of Surface Wettability on Transition Boiling Heat Flux on Vertical Surface," Proc. Int'l. Heat Transfer Conf., pp. 2031-2036, August 1986.

Valentkovich, V. and Dhir, V.K., "Preliminary Theoretical and Experimental Investigation of a Swirl Flow Phase Separator for Microgravity Applications," Proc. 4th Symp. on Space Nucl. Power, Albuquerque, Jan. 1987.

Ayazi, F. and Dhir, V.K., "A Thermal and Hydrodynamic Limit for Minimum Heat Flux and Wall Superheat During Subcooled Film Boiling on a Horizontal Cylinder," Paper No. AIAA-87-1537, June 8-10, Honolulu, HI, 1987.

Hung, T.C., Dhir, V.K. and Kastenberg, W.E., "A Study of Decay Heat Removal in Sodium Advanced Reactor with the Modified DSNP Code," Proc. of Safety of Next Generation Power Reactors, Seattle, May 1988.

Kianjah, H. and Dhir, V.K., "Experimental and Analytical Investigation of Dispersed Flow Heat Transfer in Tubes or Rod Bundles," Proc. First World Conf. on Experimental Heat Transfer, Fluid Mechanics and Thermodynamics, Dubrovnik, Yugoslavia, Sept. 1988.

Dhir, V.K. and Tung, V.X., "A Thermal Model for Fully Developed Nucleate Boiling of Saturated Liquids." Winter Annual ASME Meeting in Chicago, December 1, 1988.

Vijaykumar, R. and Dhir, V.K., "A Holographic Study of Subcooled Film Boiling on a Vertical Surface," Multiphase Flow, Heat and Mass Transfer, HTD Vol. 109, 1989.

Guo, Z.X. and Dhir, V.K., "An Experimental and Analytical Study of Swirling Bubbly Flow,"Heat Transfer Measurements Analysis and Flow Visualization, HTD-Vol. 112, 1989.

Galamba, D. and Dhir, V.K., "Simultaneous Condensation and Melting of a Vertical Surface with Uniform Heat Generation in the Condensate," AIChE Symposium Series, 269, Vol. 85, 1989.

Mardini, J., Lavine, A.S., Dhir, V.K. and Anderson, E., "Heat and Mass Transfer in Live Fuel During the Process of Drying, Pyrolysis and Ignition," Heat Transfer in Radiation, Combustion and Fires, HTD Vol. 106, 1989.

Tung, V.X. and Dhir, V.K., "Two Dimensional Two-Phase Flow in Volumetrically Heat Porous Media," Proceedings of the International Nuclear Reactor Thermal Hydraulics Conference, Karlsruhe, West Germany, Oct. 1989.

Dhir, V.K., Tung, V.X., Chang, F. and Yu, J., "Enhancement of Forced Convection Heat Transfer Using Single and Multi-Stage Tangential Injection," Heat Transfer in High Energy Heat Flux Applications, HTD Vol. 119, 1989.

Dhir, V.K., "Nucleate and Transition Boiling Heat Transfer Under Pool and External Flow Conditions," Proceedings of 9th Int'l. Heat Transfer Conference, Jerusalem, 1990.

Lingineni, S., and Dhir, V.K., "Theoretical and Experimental Investigation of Vaport Extraction Systems for Remediation of Hazardous Waste Sites," Proceedings of American Society of Civil Engineers National Conference on Environmental Engineering, Arlington, Virginia, 1990.

Guo, Z., and Dhir, V.K., "Flow Reversal in Injection Induced Swirl Flow," Winter Annual Meeting of the American Society of Mechanical Engineers, Dallas, Texas, Nov. 1990.

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Department of Nuclear Engineering
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TITLE: Professor of Nuclear Engineering
Registered Professional Engineer (Massachusetts #28539)

PERSONAL:

DATE OF BIRTH: Sept. 5, 1942 MARITAL STATUS: Married,
CITIZENSHIP: U.S.A. NUMBER OF CHILDREN: Two
BIRTHPLACE: Pittsburgh, PA

PRINCIPAL FIELD OF INTEREST: Nuclear Power Technology Innovation, Fluid Mechanics and Heat Transfer

EDUCATION:

B.S. Mechanical Engineering, Georgia Institute of Technology/ University of Florida, April 1964

Ph.D. Nuclear Science & Engineering, Cornell University, June 1969.

EMPLOYMENT HISTORY:

Employer	**Position**
Dates	
Massachusetts Institute of Technology	Professor
1986-Present	
" "	Associate Professor
1975-1986	
" "	Assistant Professor
1971-1975	
Rensselaer Polytechnic Institute	Research Associate
1/69-9/71	
Dept. of Nuclear Science - Cornell	Teaching Assistant
9/65-6/67	

*On Sabbatical leave at Laboratoire Nationale D'Hydraulique of Electricite de France, Chatou, France during 1980

CONSULTING:

Stone & Webster Engineering Corp., Boston, MA 1971-Present
Thermal hydraulic modeling of LWR containment safety system performance, severe accident radionuclide releases, nuclear power economics

Thermo Electron Corp., Waltham, MA 1973-1974
Environmental effects of co-generating power plant operation

U.S. Atomic Energy Commission, Germantown, MD 1974
Implications of national breeder reactor strategy alternatives

Avco Magnetics, Inc., Cambridge, MA 1975-1976
Fluid mechanical modeling

Energy Resources Company, Inc., Cambridge, MA 1975
Nuclear safety studies

Walden Research Division of Abcor, Inc., Cambridge, MA 1976
LMFBR containment safety system performance analysis

The Analytic Sciences Corp., Reading, MA 1976
High level radioactive waste disposal

Department of Public Utilities; Commonwealth of Massachusetts 1976
Nuclear power price analysis

Kaplan, Latti and Flannery, Boston, MA 1976
Nuclear power station heat rejection

Department of Justice, State of New York 1976-1977
Environmental effects of nuclear fuel reprocessing

Commonwealth Edison Co., Chicago, IL 1977
Cold weather cooling tower performance

Contract Research Corp., Belmont, MA 1977
Nuclear safety regulation

Jason M. Cortell Assoc./Farbridge House, Waltham/Boston, MA 1977-1979
Nuclear power station environmental effects

Dynatech Research and Development, Cambridge, MA 1978-1979
Dry cooling tower economics

U.S. Nuclear Regulatory Commission, Washington, DC 1979-1980

Analysis of regulatory policies

U.S. Department of Energy, Washington, DC 1981-1981
Nuclear regulatory policy

Westinghouse Electric Co., Pittsburgh, PA 1982-1982
Fluid flow measurement

West Valley Agricultural Protection Alliance, Phoenix, AZ 1982-1983
Cooling tower drift releases

Electric Power Research Institute, Palo Alto, CA 1983-1988
Fission product transport in reactor containment buildings

Ontario Hydro, Toronto Canada 1985, 1989
Steam transport in reactor containment buildings

Los Alamos National Laboratory, Los Alamos NM 1991-Present
Non-Prescriptive nuclear safety regulation

Westinghouse Hanford Co., Richland WA 1991-Present
High level nuclear waste remediation

MIT SPONSORED RESEARCH PROJECTS:

Project Title Dates	**Sponsor**
A.D. Little Professorship in Environmental Science & Engineering* 1973-1974 Integration of teaching and research activities concerning environmental effects of electric power production	Arthur D. Little, Inc.
MEKIN: MIT-EPRI 1974-1975 Nuclear Reactor Core Kinetics Code Project Development of a computer program (MEKIN) for calculation of three-dimensional coupled thermal-hydraulic-neutronic nuclear reactor transient behavior	Electric Power Research Institute (EPRI)
LMFBR Outlet Plenum Mixing* 1974-1980 Experimental investigation and model	U.S. Dept. of Energy (DOE)

development for analysis of turbulent, buoyant recirculating LMFBR outlet plenum flows

Cooling Tower Drift Elimination* — Northeast Utilities, New England Electric System
1974-1977
Comparative experimental and theoretical analysis of industrial cooling tower drift eliminator performance

Large Military Installation* — U.S. Army Corps of Engineers
1974-1979
Total Energy Analysis
Analysis of optimal community total energy system configurations

HTGR Containment Atmosphere Mixing* — Philadelphia Electric Co.
1975-1975**
Investigation of air entrainment in helium jets within reactor containments

Mixed-Mode Waste Heat Disposal Systems — EPRI, U.S. Energy Research & Development Administration (ERDA)
1976-1977
Theoretical and experimental analysis of a coupled thermal storage reservoir-dry cooling tower system for power station cooling

International Comparison of Nuclear Power — ERDA-DOE
1976-1978
Regulatory Practices* (Task in ERDA-Light Water Technology Project)
Analysis of structural flaws in the U.S. system for nuclear safety regulations, and comparison to the systems of England, France, West Germany and Sweden

MIT Energy System Analysis* — MIT Physical Plant
1977-1978
Analysis of the economic and technical feasibility of conversion of the MIT district heating system to cogeneration operation

Modeling of Atmospheric Plumes* — Consolidated Edison Company, Northeast Utilities
1977-1981
Numerical simulation of turbulent, buoyant plumes in the atmosphere using

a two-equation turbulence model

Droplet Impact Behavior on Water Films* — DOE
1978-1980
Investigation of droplet-film interaction with application to cooling tower drift eliminators and sulfur scrubbers

* Principal Investigator
** Terminated when the General Atomic Co. defaulted on its HTGR power station contracts.

Comparative Evaluation of Cooling Tower Field Drift Measurement Methods* — EPRI
1979-1983
Comparative performance testing of alternative field drift measurement methods

Hydrogen Transport and mixing in Nuclear Containment Atmospheres* — Northeast Utilities, Duke Power Co.,
1982-1984

Modeling of containment atmosphere behavior during severe core damage accidents — Edison Co., Public Service Electric& Gas Corp. (NJ)

Decision Issues for LWR Plant Longevity Extension* — Northeast Utilities
1984-1985
Study to define the problems to be addressed in extending LWR operating life

Loss of Offsite AC Power as a Test Example of Nonprescriptive Safety Regulation* — U.S. Nuclear Regulatory Commission
1985-1987
Prescriptive nuclear plant safety licensing, using the loss of offsite AC power accident as an example

Requirements for Advanced LWR's* — Electric Power Research Institute
1985-Present
A group of separate projects contributing to a definition of design requirements for future LWR plants. Topics in design simplification, horizontal steam generator use, plant modularization, effects of system complexity upon human error

Conceptual Design of a Small BWR* — Electric Power Research Institute and U.S. DOE
1985-Present
A collaborative project with General Electric Co. and Bechtel Power Corp. to evolve a conceptual design for an advanced boiling water reactor

* Principal Investigator

International Comparison of Light Water Reactor Power Station Performance — U.S. DOE
1985-1987
Comparison of capacity factor records for LWR in France, Japan, Sweden, Switzerland, West Germany and the United States from 1974-1984

PUBLICATIONS:

NUCLEAR REACTOR TECHNOLOGY

Nuclear Power Station Design Innovation

V.P. Manno and M.W. Golay, "Nuclear Power Plant Design Innovation Through Simplification," Nuclear Engineering and Design 85 (1985): 315-325.

M.W. Golay, "An Agenda for Improving Present-Day Reactors," Technology Review 87 (1984): 49-50.

R.K. Lester, M.W. Golay, et al., "National Strategies for Nuclear Power Reactor Development," MIT Program on Nuclear Power Plant Innovation Report No. MITNPI-PA-002 (1985).

M.W. Golay, et al., Milestone Report No. 1, "Safety Regulation of Advanced Reactors," Report No. MITNPI-TR-003 (1985).

M.W. Golay, et al., Milestone Report No. 2, "Implementation of LWR Safety Improvement for Loss of Offsite Power Events in a Non-Prescriptive Regulatory Framework," Report No. MITNPI-TR-004 (1985).

M.W. Golay and J.H. Moinzadeh, "Extending the Life of Nuclear Power Plants: Technical and Institutional Issues," MIT Energy Lab and Nuclear Engineering Dept. Report No. MIT-EL 86-003 (1986).

M.W. Golay, "LWR Innovation Needs in the United States," Technical Workshop on 1990's Water Reactor Design, Korea Nuclear Society, Seoul, Korea (1986).

M.W. Golay, V.P. Manno and P.H. Seong, "A Methodology for Simplification of Light Water Reactor System Design," MIT Reactor Innovation Project Report No. MITNPI-TR-026 (1987).

Nuclear Power Station Design Innovation

E.F. Love and M.W. Golay, "The Role of Human Error in Reactor Coolant Pump Seal Maintenance," MIT Reactor Innovation Project Report No. MITNPI-TR-022 (1987).

M.W. Golay, V.P. Manno and C. Vlahoplus, Jr., "Implementation of LWR Safety Improvements for Loss of Offsite Power Events in a Non-Prescriptive Regulatory Framework," MIT Reactor Innovation Project Report No. MITNPI-TR-019 (1987).

M.W. Golay, V.P. Manno and C. Vlahoplus, Jr., "Non-Prescriptive Nuclear Safety Regulation: The Example of Loss of Offsite Power," Nuclear Safety 80, (1988): 42-60.

M.W. Golay, "Light Water Reactor Innovation Needs in the United States: The Massachusetts Institute of Technology LWR Innovation Project," Nuclear Technology 29, (1988): 6-20.

P.H. Seong, V.P. Manno and M.W. Golay, "Application of a Power Plant Simplification Methodology: The Example of the Condensate Feedwater System," Nuclear Engineering & Design 110 (1988): 33-46.

K. Hansen, et. al., "Making Nuclear Power Work: Lessons from Around the World," Technology Review 92 (Feb/Mar 1989): 30-40.

M.W. Golay and N.E. Todreas, "Advanced Light-Water Reactors," Scientific American, (April, 1990): 82-88.

M.W. Golay, P.H. Seong and V.P. Manno, "A Measure of the Difficulty of System Diagnosis and its Relationship to Complexity," Int'l J. Gen. Systems 16 (Jan-Mar, 1990): 1-23.

Nuclear Reactor Thermal Hydraulic Analysis

R.G. Bennett and M.W. Golay, "Development of an Optical Method for Measurement of Temperature Fluctuations in Turbulent Flows," Trans. Amer. Nucl. Soc. 22 (1975): 581.

A.E. Levin and M.W. Golay, "Simulation of Flow Behavior in BWR Coolant Channels During Transients," MIT-EPRI MEKIN Project Report No. MEK-43 (1975).

L. Guillebaud and M.W. Golay, "Investigation of Iterative Methods for Solution of Large, Sparse Bandwidth Matrix Equations," MIT Department of Nuclear Engineering Report No. MITNE-178, MIT-EPRI (1976).

Nuclear Reactor Thermal Hydraulic Analysis

R.G. Bennett and M.W. Golay, "Interferometric Investigation of Turbulently Fluctuating Temperature in an LMFBR Outlet Plenum Geometry," Paper 76-WA/HT-82, ASME Winter Annual Meeting, New York (1976); MIT-ERDA Report No. COO-2245-29TR (1977); later published in J. Heat Transfer 100 (1978): 334.

Y.B. Chen and M.W. Golay, "Validation of Turbulence Models for LMFBR Outlet Plenum Flows," National Heat Transfer Conference Paper No. 77-HT-34 (1977).

Y.B. Chen and M.W. Golay, "Coolant Mixing in the LMFBR Outlet Plenum," MIT-ERDA Report No. COO-2245-44TR (1977).

V.P. Manno and M.W. Golay, "Measurement of Heat and Momentum Eddy Diffusivities in Recirculating LMFBR Outlet Plenum Flows," MIT-ERDA Report No. COO-2245-61TR (1978).

D.R. Boyle and M.W. Golay, "Transient Effects in Turbulence Modeling," MIT-U.S. Dept. of Energy Report No. DOE/ET/37240-83TR (1980).

M.W. Golay, "Turbulent Flow Modeling in the LMFBR Outlet Plenum," IAHR Workshop on Refined Modeling of Flows, Chatou, France (1980).

D.R. Boyle and M.W. Golay, "Measurement of a Recirculating Two-Dimensional Flow and Comparison to Turbulent Model Predictions, II: Transient Case," J. Fluids Engineering 105 (1983): 439-445.

D.R. Boyle and M.W. Golay, "Measurement of a Recirculating Two-Dimensional Flow and Comparison to Turbulent Model Predictions, II: Transient Case," J. Fluids Engineering 105 (1983): 447-454.

M.W. Golay, "Comparison of LMFBR Outlet Plenum Turbulent Flow Measurements and Caculations," IAHR Workshop on LMFBR Outlet Plenum Flows, Sunnyvale, CA (1983).

V.P. Manno and M.W. Golay, "Analytical Modeling of Hydrogen Transport in Reactor Containments," MIT Energy Lab and Nuclear Engineering Dept. Report No. MIT-EL 83-009 (1983).

V.P. Manno and M.W. Golay, "User's Manual for the LIMIT Code - Hydrogen Analysis in Reactor Containments," MIT Energy Lab and Nuclear Engineering Dept. Report No. MIT-EL 83-010 (1983).

K.Y. Huh and M.W. Golay, "Treatment of Physical and Numerical Diffusion in Fluid Dynamic Simulations," MIT Energy Lab and Nuclear Engineering Dept. Report No. MIT-EL 83-011 (1983).

V.P. Manno and M.W. Golay, "Application of the LIMIT Code to the Analysis of Containment Hydrogen Transport," Nuclear Technology 67 (1984): 302-311.

V.P. Manno, M.W. Golay and K.Y. Huh, " Analytical Models for Simulating Hydrogen Transport in Reactor Containment Atmospheres," Nuclear Science and Engineering 87 (1986): 349-360.

K.Y. Huh, M.W. Golay and V.P. Manno, "A Method for Reduction of Numerical Diffusion in the Donor Cell Treatment of Convection," J. Computational Physics 63 (1986): 201-221.

V.P. Manno and M.W. Golay, "Hydrogen Transport in Containments: A Survey of Analytical Tools and Benchmark Experiments," Nuclear Safety 25 (1984): 797-814.

V.P. Manno and M.W. Golay, "Hydrogen Mixing and Stratification in Containment Atmospheres," Nuclear Technology 70 (1985): 124-132. Also presented at Third International Topical Meeting on Reactor Thermal Hydraulics, Newport, RI (1985).

B.K. Riggs, M.W. Golay and V.P. Manno, "A Correlation for Heat and Mass Transfer to Containment Sprays," Third International Topical Meeting on Reactor Thermal Hydraulics, Newport, RI (1985).

P.E. Roege, B. Day, E.S. Beckjord and M.W. Golay, "Application of Leak-Before-Break Criteria to Pressurized Water Reactors," Second International Topical Meeting on Nuclear Power Plant Thermal Hydraulics and Operation, Tokyo, Japan (1986).

M. Siddique, M.W. Golay and M.S. Kazimi, "The Effect of Hydrogen on Forced Convection Steam Condensation," AIChE Symposium Series, National Heat Transfer Conference, Philadelphia, PA, (August 6-9, 1990).

Nuclear Reactor Physics

M.W. Golay, Doctoral Thesis: "Neutron Pulse Propagation in Multiplying Assemblies," Cornell University (1969).

M.W. Golay and K.B. Cady, "A Fomulation of Neutron Dispersion Relations in Terms of Reactivity," Trans. Amer. Nucl. Soc. 10 (1967): 558.

M.W. Golay and K.B Cady, "Transform Domain Analysis of Neutron Pulse Propagation Experiments," Trans. Amer. Nucl. Soc. 12 (1969).

M.W. Golay and K.B. Cady, "Axial Neutron Pulse Propagation in Multiplying Cylinders," Trans. Amer. Nucl. Soc. 12 (1969).

M.N. Kaushal, B.K. Malaviya, M.W. Golay and E.R Gaertner, "Time of Flight Measurements of Neutron Spectra in Aluminum," Trans. Amer. Nucl. Soc. 12 (1969): 756.

M.W. Golay, et al., "Fast Neutron Spectra and Time Moments in a Bulk Aluminum Assembly," Trans. Amer. Nucl. Soc. 13 (1970): 689.

M.W. Golay, et al., "Fast-Neutron Spectra in Aluminum," Trans. Amer. Nucl. Soc. 14 (1971): 836.

D.C. Gibbs, et al., "Studies of Fast-Neutron Spectra Across a Material Interface," Trans. Amer. Nucl. Soc. 14 (1971): 383.

M.W. Golay and K.B Cady, "Neutron Pulse Propagation in Multiplying Assemblies," Nucl. Sci. Eng. 43 (1971): 303.

POWER STATION WASTE HEAT DISPOSAL

Atmospheric Plume Behavior

R.G. Bennett and M.W. Golay, "Numerical Modeling of Buoyant Flows in a Turbulent Stratified Atmosphere," MIT Energy Lab and Nuclear Engineering Dept. Report No. MIT-EL 79-002 (1979).

R.G. Bennett and M.W. Golay, "User's Guide for Numerical Modeling of Buoyant Plumes in a Turbulent Stratified Atmosphere," MIT Energy Lab and Nuclear Engineering Dept. Report No. MIT-EL 79-004 (1979).

R.G. Bennett and M.W. Golay, "Airborne Plume Dispersion from Two-Dimensional Computational Hydrodynamic Simulations," Trans. Amer. Nucl. Soc. 33 (1979): 179.

M.W. Golay, "Investigation of the Flow-Resulting from a Neutrally-Buoyant Vertical Jet Injected into a Crosswind," Report HE 041/81.10, Laboratoire Nationale d'Hydraulique, Electricite de France (1981).

R. Hamza and M.W.Golay, "Behavior of Buoyant Moist Plumes in Turbulent Atmospheres," MIT Energy Lab and Nuclear Engineering Dept. Report No. MIT-EL 81-020 (1981).

M.W. Golay, "Numerical Modeling of Buoyant Plumes in a Turbulent Stratified Atmosphere," Atmospheric Environment 16 (1982): 2373-2381.

R. Hamza and M.W. Golay, "Modeling of Moist Plumes in Turbulent Atmospheres," Atmospheric Environment 20 (1986): 9-19. Also presented at IAHR Cooling Tower Workship, Interlaken, Switzerland (1985).

Evaporative and Dry Cooling Towers

T.J. Flanagan and M.W. Golay, "Augmentation of Natural Draft Evaporation Cooling Tower Performance," Trans. Amer. Nucl. Soc. 16 (1973): 31.

E.C. Guyer and M.W. Golay, "A Model for Salt Drift Deposition from Spray Ponds," Environmental Effects of Cooling Systems at Nuclear Power Plants, Symp. Proc., IAEA (1974): 331.

E.C. Guyer and M.W. Golay, "Evaluation of Combined Thermal Storage Pond and Coolling Tower Systems," Trans. Amer. Nucl. Soc. 22 (1976): 509

J. Chan and M.W. Golay, "Numerical Simulation of Cooling Tower Drift Eliminator Performance,: Numerical/Laboratory Computer Methods in Fluid Mechanics, Symp. Proc., A.A. Pouring, ed., Amer. Soc. Mech. Engrs. (1976): 22[illegible].

E.C. Guyer and M.W. Golay, "An Engineering and Economic Evaluation of Some Mixed-Mode Waste Heat Rejection Systems," MIT Nuclear Engineering Dept. Report No. MITNE-191 (1976).

E.C. Guyer and M.W. Golay, "Mathematical Models for Predicting the Thermal Performance of Closed-Cycle Waste Heat Dissipation Systems," MIT Nuclear Engineering Dept. Report No. MITNE-192 (1976).

J. Chan and M.W. Golay, "Comparative Performance Evaluation of Current Design Evaporative Cooling Tower Drift Eliminators," Atmospheric Environment 11 (1977): 775.

E.C. Guyer and M.W. Golay, "Evaluation of Combined Thermal Storage Pond/Dry Cooling Tower Waste Heat Disposal Systems," Nat. Heat Transfer Conf., Paper No. 77-HT-57 (1977).

E.C Guyer and M.W. Golay, "Investigation of the Fluid Mechanical Behavior of a Thermal Storage Reservoir for Dry Cooled Central Power Stations, "Waste Heat Management and Utilization, S.S. Lee, ed., 7-C-209, U. of Miami (1977).

J. Chan and M.W. Golay, "Design Study of Cooling Tower Drift Eliminators," Trans. Amer. Nucl. Soc. 26 (1977): 113.

J.K. Chan and M.W. Golay, "Comparative Evaluation of Cooling Tower Drift Eliminator Performance," MIT Energy Lab Report No. MIT-EL 77-004 (1977).

J.K. Chan and M.W. Golay, "'DRIFT' - A Numerical Simulation Solution for Cooling Tower Drift Eliminator Performance," MIT Energy Lab Report No. MIT-EL 77-006 (1977).

J.K. Chan and M.W. Golay, "Design Studies of Current Design and Improved Cooling Tower Drift Eliminators," in Cooling Tower Environment - 1978, R.S. Nietubicz, ed., U of Maryland (1978).

M.W. Golay, "Comparative Evaluation of Cooling Tower Drift Eliminator Performance," IAHR Cooling Tower Workshop, Chatou, France (Nov. 1978).

M.W. Golay, S.H. Chen, F.R. Best and W.J. Glantschnig, "Comparison of Field Measurement Methods for Cooling Tower Drift," IAHR Cooling Tower Workshop, San Francisco, CA (Sept. 1980).

W.J. Glantschnig, M.W. Golay, S.H. Chen and F.R. Best, "Light Sacttering Device for Sizing and Velocimetry of Large Droplets Utilizing a Ring-Shaped Laser Beam," Applied Optics 21 (1982): 2456-2460.

E.C. Guyer and M.W. Golay, "Fluid Mechanical Design of a Dry Cooling System Thermal Storage Reservoir, or Stratificaton Minimization in Horizontal Channel Flow," J. Solar Energy Engineering, Trans. ASME 105 (1980): 174-180.

W.J. Glantschnig, M.W. Golay, S.H. Chen and F.R. Best, "Design and Operation of a Light-Scattering Device for Sizing and Velocimetry

of Large Droplets," Electric Power Research Institute Report No. EPRI CS-3098 (1983).

M.W. Golay, W.J. Glantschnig and F.R. Best, "Comparison of Methods for Measurement of Cooling Tower Drift," Electric Power Research Institute Report No. EPRI CS-3355 (1984).

B.K. Ching, M.W. Golay and T.F. Johnson, "Impacts of Droplets Upon a Liquid Surface at Oblique Angles," Science 226 (1984): 535-537. Also presented at IAHR Cooling Tower Workshop, Monterey, CA (1986).

M.W. Golay, W.J. Glantschnig and F.R. Best, "Comparison of Methods for Measurement of Cooling Tower Drift," Atmospheric Environment 20 (1986): 269-291. Also presented at IAHR Cooling Tower Workshop, Budpest, Hungary (1982), and American Power Conference (1983).

Total Energy System and Energy Conservation Analysis

A. Von Nida and M.W. Golay, "A Dual-Purpose HTGR for Military Installations," Trans. Amer. Nucl. Soc. 17 (1974): 245.

A. Von Nida and M.W. Golay, "An Analysis of Feasibility of a Brayton Cycle HTGR-Power Combined Thermal-Electric Utility System," Environmental Effects of Cooling Systems at Nuclear Power Plants, Symp. Proc., IAEA (9174): 703.

M.R. Doyle, M.W. Golay and N.C. Rasmussen, "Comparison of Coal and Nuclear Systems Millitary Utility Applications and Consequences of Reactor Accidents," U.S. Army Corps of Engineering Facilities Engineering Support Agency and MIT Nuclear Engineering Dept. Report No. FESA-RT-2002 (1974).

L.R. Metcalfe and M.W. Golay, "Final Report: Storage of Off-Peak Nuclear Power Through Hydrogen Generation," Report No. FESA-RT-2005 (1975).

J.W. Stetkar and M.W. Golay, "Simulation of the Behavior of a Brayton-Cycle HTGR-Powered Thermal Utility System for Total Energy Applications," Trans. Amer. Nucl. Soc. 22 (1975): 601.

J.W. Stetkar, F.R. Best and M.W. Golay, "Final Report: Design of a Nuclear Powered Total Energy System for Ft. Bragg, North Carolina," Report No. FESA-RT-2011 (1976).

J.W. Stetkar and M.W. Golay, "User's Manual: TDIST, A Program for Community Energy Demand Analysis and Total Energy System Response Simulation," Report No. FESA-RT-2024 (1976).

W. Boyd and M.W. Golay, "Final Report: Economic and Technical Aspects of Coal Gassification for Use in Gas Turbine Operations," Report No. FESA-RT-2012 (1976).

J. Kelly and M.W. Golay, "Final Report: Economic and Technical Aspects of Gas Turbine Power Stations in Total Energy Applications," Report No. FESA-RT-2013 (1976).

F.R. Best, S.B. Goldman and M.W. Golay, "Optimal System Configuration Total Energy Analysis for a Large Military Installation," in Heat Transfer in Energy Conservation, R.J. Goldstein, ed., Amer. Soc. Mech. Engrs. (1977): 113.

S.B. Goldman, F.R. Best and M.W. Golay, "Transmission and Distribution Cost Data for Use in Community Total Energy System Analysis with Ft. Knox, Kentucky as an Example." Report No. FESA-RT-2020 (1977).

S.B. Goldman, F.R. Best and M.W. Golay, "Energy Production by Solid Waste Incineration," Report No. FESA-RT-2022 (1977).

S.B. Goldman, F.R. Best and M.W. Golay, "User's Manual: TDIST2, A Computer Program for Community Energy Consumption Analysis and Total Energy System Design," Report No. FESA-RT-2047 (1977).

F.R. Best, S.B. Goldman and M.W Golay, "Final Report: Analysis of Nuclear and Coal Fueled Total Energy System Option for Ft. Knox, Kentucky," Report No. FESA-RT-2039 (1977).

F.R. Best, S.B. Goldman and M.W. Golay, "A Computer Program for Community Energy Consumption, Analysis and Total Energy System Design," Cogeneration District Heating Applications, I. Oliker, ed., Amer. Soc. Mech. Engrs. (1978): 30.

D.B. Ebiling-Koning and M.W. Golay, "TDIST3, A Computer Code for Community Energy Consumption Analysis and Total Energy System Design," Report No. FESA-TSD-2062 (1978).

G.S. Was and M.W. Golay, "Economic Feasibility Study of Total Energy System Options for the Massachusetts Insitute of Technology," MIT Nuclear Engineering Dept. Report No. MITNE-718 (1978).

D.B. Ebeling-Koning and M.W. Golay, "Performance Optimization of a Multi-Heat Source Thermal Utility System," Report No. FESA-TSD-2064 (1979).

S.B. Goldman and M.W. Golay, "Strategies for Scheduling Power Plants Producing Both Heat and Electric Energy," Energy 4 (1979): 1033.

G.S. Was and M.W. Golay, "Cogeneration - An Energy Alternative for the United States?" Energy 4 (1979): 1026.

Energy-Environmental Policy

M.W. Golay, Review of Rand Corp. Report: "California's Electricity Quandry," Ecology Law Quarterly, 3, U. Cal. (Berkeley) (1973): 434.

M.W. Golay, "Offshore Nuclear Power Stations," Oceanus, Woods Hole Oceanographic Institute XVIII (Summer, 1974).

M.W. Golay and L.R. Anderson, "The Role of Environmental Statements in Promoting Internalization of Environmental Costs and an Evaluation of Their Benefit-Cost Analysis Practices," Trans. Amer. Nucl. Soc. 17 (1974): 59.

G.C. Daley, M.W. Golay and M. Baughman, "Competition Among Electricl Energy Supply Sectors in the United States Energy Market," Trans. Amer. Nucl. Soc. 17 (1974): 221.

J.M. Willefert and M.W. Golay, "Comparative Analysis of United States and French Nuclear Power Plant Siting and Construction Regulatory Policies," Trans. Amer. Nucl. Soc. 27 (1977): 173.

M.W. Golay, I.I. Saragossi and J.M. Willefert, "Comparative Analysis of the American and French Nuclear Power Plant Siting and Construction Regulatory Policies and Their Economic Consequences," MIT Nuclear Engineering Dept. and Energy Lab Report No. MIT-EL 77-044-WP (1977).

M.W. Golay and I.I. Saragossi, "Effects of Environmental Protection and Public Regulatory Practics Upon Light Water Reactor Economics," MIT Nuclear Engineering Dept. and Energy Lab Report No. MIT-EL 78-009 (1978).

M.W. Golay, "How Prometheus Came to be Bound: Nuclear Regulation in American," Technology Review 83 (1980): 28.

E.Beckjord, M.W. Golay, E. Gyftopoulos, K. Hansen, R. Lester and K.D. Winje, "International Comparison of LWR Performance," MIT Energy Lab Report No. MIT-EL-87-004 (1987).

M.W. Golay, "Longer Life for Nuclear Plants," Technology Review, (May-June, 1990).

PROFESSIONAL ORGANIZATIONS:

American Nuclear Society
Member: Awards Committee, Standards Committee
Thermal Hydraulics Division - Program Committee
Environmental Sciences Div. - Various Offices including Chairman 1976-1980
Northeast (Boston) Local Section Chairman 1977-1980

American Society of Mechanical Engineers
Member: Environmental Heat Transfer Committee (K-19), Heat Transfer Division 1976-1986

American Association for the Advancement of Science

AWARDS RECEIVED:

Tau Beta Pi, Phi Kappa Phi and Sigma Tau Honorary Societies 1964
George M. Verity Scholarship (Armco Steel Corp.) 1960-1964
A.E.C. Fellowship in Nuclear Science & Engineering 1964-1967
Arthur D. Little Professorship in Environmental Sciences and Engineering, MIT 1972-1974

GOVERNMENT COMMITTEES, SERVICES, ETC.:

EPA - Review of Boston Air Quality Plan 1974-1975

NAS - Committee on Mineral Resources and the Environment Uranium Resource Estimate Review 1974

Office of Technology Assessment 1976 ERDA Program Review 1975
Engineering Council for Professional Development Student Development Committee 1973-1978

ANS - 1977 Annual Meeting: Chairman Environmental Impacts of Cooling Towers, Invited Session 1977- 1979 Annual Meeting: Chairman, Atmospheric Dispersion, Invited Session 1979

American Power Conference - Chairman, Electric Power Policy Issues, Invited Session 1978

U.S. DOE/NRC Safety Research Program Review 1983

Chernobyl Accident Review Committee, Atomic Industrial Forum 1986-1988

Peter Griffith
Department of Mechanical Engineering
Massachusetts Institute of Technology

[redacted]

PERSONAL:

Born: 9/23/27, London, England
Marital Status: Married to Kathleen Griffith, 2 children.

EDUCATION:

Public Schools, Huntington, L.I.
New York University, B.M.E., 1950
University of Michigan, M.S., 1952
Massachusetts Institute of Technology, SC.D. 1956.
Thesis: "Dynamics of Bubbles in Nucleate Boiling."

PRINCIPAL FIELDS OF INTEREST:

Heat Transfer, Two-Phase Flow, Power Energy

PROFESSIONAL POSITIONS AT MASSACHUSETTS INSTITUTE OF TECHNOLOGY:

Professor	1964 - present
Associate Professor	1960 - 1964
Assistant Professor	1956 - 1960
Instructor	1954 - 1956
Research Assistant	1952 - 1954

OTHER PROFESSIONAL EXPERIENCE:

University of Michigan: Teaching Fellow, 1950-52; Research Assistant, 1952, 53.
University of London: Research, 1955
Westinghouse Atomic Power, Research, 1957, 58, 59.
Thompson Ramo Woolridge: Research, 1960, 62
Institutt fur Atomenergi: Research, 1961
Argonne National Laboratory: Research, 1963
Northeastern University: Teaching, 1956
University of Cambridge, Teaching 1959- 1960
M.W. Kellogg Company: Engineering, 1965
American Electric
Power Service Corp. Engineering, 1966
General Electric
Atomic Power (KAPL), Engineering, 1969.
Birla Institute of Technology and Science, Teaching, 1969-1970.

INDUSTRIAL CONSULTING RECORD:

Thompson Ramo Woolridge	1961-1962
Institutt fur Atomenergi	1961-1962
Ingersoll-Rand Research, Inc.	1963-1964
Northern Research	1963-1964
Foster-Miller	1965 to present
M.W. Kellogg Co.	1965-1971
E G & G	1965-1967, 1977
Scott Paper	1966-1980
Idaho Nuclear corp., (Aerojet Nuclear), Idaho National Laboratory	1967 to present
Megapulse	1971-1976
Futures Group	1972-1973
A.D. Little	1973-1985
Innotech	1972-1980
United Technology (International Fuel Cells)	1972 to present
MITRE	1972-1980
Stone & Webster	1974-1980
Mobil Oil	1972-1973
Kennecott Copper	1974-1980
USAEC (ANL)	1975-1975
USAEC (LMFBR Project)	1974-1980
American Petroleum Institute	1974-1977
Teledyne Materials Research	1973-1983
Argonne National Lab	1974-1983
General Electric (Portland, ME)	1974-1977
Dynatech	1972-1985
Combustion Engineering	1975-1978
Jerome Serchuck Enterprises	1973-1979
ERDA	1974-1976
Brookhaven National Lab, NRC	1974 to present
Lincoln Lab	1975-1978
Bechtel	1975-1980
Dyna International	1975-1976
Westinghouse (Heat Exchanger Division)	1975-1976
Halcon International Co.	1975-176
Sperry	1976-1981
Oak Ridge National Lab	1975 to present
IBM	1976
Pratt and Whitney Aircraft	1976-1979
Applied Physics Lab	1975-1981
Los Alamos Scientific Lab	1976-1985
Creare	1976 to present
Consolidated Edison	1976 to present
Exxon Enterprises	1976-1979

Auburn Instruments	1976-1981
Union Carbide	1979-1981
Riley Stoker	1979 to present
Merix Corporation	1978-1986
Sperry	1978-1980
Bolton-Emerson	1979-1980
Oblon, Fisher, Spivak, McClelland & Maier	1979-1980
Institute De Investigaciones Electricas (Mexico)	1979-1980 1981-1987
Southwest Engineering	1983-1985
Modar	1983-1984
Avco-Lycoming	1983-1984
High Vacuume	1983
Simpson, Gampertz & Heger, Inc.	1983 to present
TTI Engineering	1985-1986
Burrnah Oil	1985-1986
International Fuel Cells	1985 to present
Novo Prignone	1985-1987
Lytron Inc.	1985 to present
Adams, Shoemaker and McSorley	1987 to present
Meditech	1987-1988
Thermo Electron Engineering	1987 to present
Pequod Associates	1988
Brigham, Dana, Gould	1988
Halometrix	1988-1989
Northeast Utilities	1988 to present
Small Break LOCA-Technical Program Group (NRC)	1989 to present
Severe Accident Scaling Methodology Group (NRC)	1989 to present
Natural Circulation Technical Program Group (NRC)	1990 to present
Dynagen	1990 to present
Rackman, Sawyer & Brewster	1990 to present
Altran	1990 to present

DEPARTMENT AND INSTITUTE COMMITTEES, ETC.

Student Advisor - Fourth Year	1973 to present
Senior Registration Officer	1962 to present
Sophomore Advisor	1962-1963
Curriculum Committee	1963 (Chairman, 1966, 67)
Junior Advisor	1963-1964
EPL Steering Committee	1978-1981
Undergraduate Officer	1980-1983, 1985 to present
Holt Committee Chairman	1972 to present
Reactor Safeguards Committee	1966 to present

Committee on Academic Performance	1962-1968; 1982-1985
M.E. Dept. Safety Committee, Chairman	1987 to present
M.E. Dept. Chem. Hygiene Officer	1990 to present

GOVERNMENT COMMITTEES, SERVICE, ETC.:

Advanced Code Advisory Committee	1975-1984
LMFBR Thermal & Hydraulic Task Review Committee (ERDA)	1974-1982
Associate Editor of Journal of Heat Transfer	1975-1978
NRC Source Term Committee	1986-1987
Code Applicability Scaling and Uncertainty Committee	1987-1989
Technical Program Groups (3)	1989 to present

AWARDS AND HONORS RECEIVED:

Invited Lecturer, Institution of Mechanical Engineers Meeting on Boiling Heat Transfer, Manchester, England, Sept. 1965.

Best Paper, 10th Annual AME-AIChE Heat Transfer Conference, 1968.

Principal Lecturer at Spring Heat Transfer Symposium, Stillwater, Oklahoma, March, 1968

Heat Transfer Memorial Award, 1975

Fellow, ASME, 1975.

Best Paper, ASME Heat Transfer Division in 1984, 1985.

Melville Medal, ASM (best paper award), 1986.

PROFESSIONAL AND ORGANIZATION MEMBERSHIPS:

Sigma Xi, Tau Beta Phi, Pi Tau Sigma, ASME, AIChE, ANS

Refereed Conferences

1. Rohsenow, W.M., Griffith P., "Correlation of Maximum Heat Flux Data for Boiling of Saturated Liquids", AIChE Symposium Series No. 18, 1956.

* 2. Griffith, P. "The Correlation of Nucleate Boiling Burnout Data", ASME Paper 57-HT-21.

* 3. Griffith, P., Wallis, J.D., "The Role of Surface Conditions in Nucleate Boiling", AIChE Symposium, Series No. 30, 1960.

* 4. Griffith, P., "Flooding and Burnout in Closed-End Vertical Tubes", Institution of M.E., Symposium on Two-Phase Flow, PU-25, 1962.

5. Griffith, P., "Geysering in Liquid-Filled Lines", ASME Paper, 62-HT-30.

6. Griffith, P., "Two-Phase Flow Regime Detection", ASME Paper 64-WA/HT-43.

* 7. Rose, S., Griffith, P., "Flow Properties of Bubbly Mixtures", ASME Paper 65-HT-58.

* 8. Haberstroh, R.E., Griffith, P. "The Slug Annular Flow Regime Transition", ASME Paper 65-HT-52, 1965.

* 9. Maulbetch, J.S., Griffith, P., "A Study of System Induced Instabilities in Forced Convection Flows with Sub-cooled Boiling", 3rd International Heat Transfer Conference, August 1966.

* 10. Yih, T.S., Griffith, P., "Unsteady Momentum Fluxes in Two-Phase Flow and the Vibration of Nuclear System Components", ANL-7685, May 1970.

* 11. Bhatt, B.L., Griffith, P., Meheshwari, B., "Multiphase Flow in Pipes", AIChE Symposium Series No. 48, February 1972, Dallas, Texas.

* 12. Gonzalez-Santalo, J.M., Griffith, P., "Two-Phase Mixing in Rod Bubble Sub-channels", ASME Paper 72-WA/VE-19.

* 13. Singh, G., Griffith, P., "An Analysis of Slug-Annular Transition in Two-Phase Flow in Down Sloping Pipelines", Paper No. HMT-72-71, 1st National Heat and Mass Transfer Conference, Madras India, December 1971.

14. Ybarrondo, L.J., Griffith, P., Fabic, S., McPherson, G.D, "Examination of LOFT Scaling", 74-WA/HT-53.

* 15. Goldstein, S.D., Griffith, P., "Spray Cooling of Laser Mirrors", ASME Paper 74-HT-17, Boston, 1974.

* 16. Griffith, P., Avadesian, C.T., Walkush, J.P., "Counter-Flow Critical Heat Flux", AIChE Symposium Series #131, Heat-Transfer, San Franciso,

* 17. Kirchner, W., Griffith, P., "Reflood Heat Transfer and Carry-over, Recommendations for Calculation", AIChE Symposium Series, Heat Transfer, Boston, 1975.

* 18. Iloeje, O.C., Rohsenow, W.M., Griffith, P., "Three Step Model of Disposed Flow Heat Transfer (Post CHF Vertical Flow)", 75-WA/HT-1.

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Refereed Conferences (continued)

* 19. Singh, G., Griffith, P., "Down Sloping Inclined Pipe Pressure Drop and Hold-up", ASME Paper 76-Pet-29.

* 20. Singh, G., Griffith, P., "Down Sloping Inclined Pipe Flow Regime Transitions", ASME Paper 76-Pet-30.

* 21. Smith, R.A., Griffith, P., "A Simple Model for Estimating Time to Critical Heat Flux in a Pressurized Water Reactor Loss of Coolant Accident", ASME Paper 76-HT-12.

* 22. Plummer, D.M., Griffith, P., Rohsenow, W.M., "Post Critical Heat Transfer to Flowing Liquid in a Vertical Tube", Paper No. 76-ASME/AIChE-13, August 1976.

* 23. Griffith, P., Pearson, J.F., Lepkowski, R.J., "Critical Heat Flux During a Loss of Coolant Accident", Nuclear Safety 18, No. 3, May-June 1977.

* 24 Cheung, Y.L., Pearson, J.F., Griffith, P., "Modeling Two-Phase Flow Transients", AIChE Paper 7th National Heat Transfer Conference, Salt Lake City, Utah, 1977.

* 25. Singh, B., Griffith, P., "Gravity Dominated Two-Phase Flows in Vertical Rod Bundles", AIChE Symposium Series 199, p. 126, 1980.

* 26. Levin, A.E., Griffith, P., "Analytical and Experimental Stimulation of Boiling Oscillations in Sodium with a Low-Pressure Water System", AIChE Symposium Series, Heat Transfer, Milwaukee, 1981.

27. Griffith, Peter, "Unsolved Problems in Reactor Thermal Hydraulics", Proceedings of 3rd International Meeting on Reactor Thermal Hydraulics, Newport, RI, Published by the ANS, October 1985.

* 28. Liao, Lih-Yih, Parlos, Alex and Griffith, Peter, "Heat Transfer, Carry-over, and Fall Back in Nuclear Steam Generators During Transients", 8th International Heat Transfer Conference, San Francisco, CA, August 1986.

29. Griffith, Peter, "Transient Critical Heat Flux", International Centre for Heat and Mass Transfer" Conference held at Dubrovnik, Yugoslavia. May 1987 Paper 7.3.

* 30. Keck, R.G. and Griffith, P., "Models and Equations for the Prediction of Erosive-Corrosive Wear in Steam Extraction Piping" ASME Paper 87-JPGC-35, Presented in Miami, Oct. 1987.

31. Griffith, Peter, "Characteristics of Bubbly Flow" Multiphase Science and Technology, Hemisphere Publishing Corporation 1987, p. 155.

32. Griffith, Peter, "Post Dryout Heat Transfer Quench Data", Multiphase Science and Technology Volume, Hemisphere Publishing Corporation, 1987 p. 395.

* 33. Chou, Yuanching, Griffith, Peter, "Admitting Cold Water Into Steam Filled Pipes Without Waterhammer Due to Steam Bubble Collapse" PVP Vol. 156, Transient Thermal Hydraulics in Vessel and Piping Systems, ASME Book No. H00471, 1989.

*outgrowth of supervised thesis

Publications in Refereed Journals

* 1. Griffith, P., "Bubble Growth Rates in Boiling", Trans ASME, Vol. 80, pp. 721-27, 1958. No. 208, Vol. 77, 1981.

2. Wallis, G.B., Griffith, P., "Two-Phase Slug Flow", Trans ASME, Journal of Heat Transfer, Vol. 83, pp. 307-321, 1961.

* 3. Moissis, R., Griffith, P., "Entrance Effects in a Two-Phase Slug Flow", Trans ASME, Journal of Heat Transfer, Vol. 84, pp. 29-39, 1962.

4. Griffith, P., "The Prediction of Low Quality Boiling Voids", Trans ASME, Journal of Heat Transfer, Vol. 83, pp. 327-333, 1964.

* 5. Lee, K.S., Griffith, P., "The Stability of an Annulus of Liquid in a Tube", Trans ASME, Journal of Basic Engineering, Vol. 86, pp. 1-5, 1964.

* 6. Suo, M., Griffith, P., "Two-Phase Flow in Capillary Tubes", Trans ASME, Journal of Heat Transfer, Vol. 86, Series D, pp. 576-582, 1964.

* 7. Han, C-Y, Griffith, P., "The Mechanism of Nucleate Pool Boiling", International Journal of Heat Transfer, Vol. 8, pp. 887-914, 1965.

* 8. Umur, A., Griffith, P., "The Mechanism of Dropwise Condensation" Trans ASME, Journal of Heat Transfer, Vol. 87, pp. 275-282, 1965.

* 9. Gerstman, J., Griffith, P., "Laminar Film Condensation on the Underside of Horizontal and Inclined Surfaces", International Journal of Heat Transfer, Vol. 10, pp. 567-580, 1967.

* 10. Lee, M.S., Griffith, P., "The Effect of Surface Thermal Properties and Finish on Dropwise Condensation", International Journal of Heat and Mass Transfer, Vol. 10, pp. 567-580, 1967.

* 11. Andeen, G., Griffith, P., "Momentum Flux in Two-Phase Flow", Trans ASME, Journal of Heat Transfer, Vol. 90, pp. 211-222, 1967.

12. Bergles, A.E., Griffith, P., "Projection Slides for Classroom Demonstrations of Heat Transfer with Boiling and Condensation", Bull of Mechanical Engineering Education, No. 6, pp. 79-83, 1967.

* 13. Shiralkar, B., Griffith, P., "Deterioration in Heat Transfer to Fluids at Supercritical Pressure and High Heat Fluxes", Trans ASME, Journal of Heat Transfer, No. 91, pp. 27-36, 1969.

* 14. Farmer, R., Griffith, P., "Liquid Droplet Deposition in Two-Phase Flow", Trans ASME Journal of Heat Transfer", Vol. 92, pp. 589-594, 1970.

* 15. Singh, G., Griffith, P., "Determination of the Pressure Drop Optimum Pipe Size for a Two-Phase Slug Flow", Trans ASME, Journal of Engineering Industry, No. 92, pp. 717-726, 1970.

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Publications in Refereed Journals (continued)

16. Mikic, B.B., Rohsenow, W.M., Griffith, P., "On Bubble Growth Rates", Intl Journal of Heat and Mass Transfer, No. 13, pp. 657-666, 1970.

* 17. Shiralkar, B., Griffith, P., "The Effect of Swirl, Inlet Conditions, Flow Direction, and Tube Diameter on the Heat Transfer to Fluids at Supercritical Pressure", Trans ASME, Journal of Heat Transfer, No. 92, pp. 465-474, 1970.

* 18. Graham, C., Griffith, P., "Drop Size Distribution and Heat Transfer in Dropwise Condensation", International Journal of Heat and Mass Transfer, No. 16, pp. 23-35, 1973.

* 19. Iloeje, O.C., Plummer, D., Rohsenow, W.M., "An Investigation of the Collapse of the Surface Rewet in Film Boiling in Forced Vertical Flow", Trans ASME, Journal of Heat Transfer, No. 97, pp. 166-175, 1975.

* 20. Smith, R.A., Price, F.A., Griffith, P., "An Analysis of Critical Heat Flux in Flow Reversals", Trans ASME. Journal of Heat Transfer, 98, May 1976.

* 21. Kirchner, W.L., Griffith, P., "Reflood Heat Transfer in a Light Water Reactor", AIChE Symposium Series-Heat Transfer, St. Louis, Vol. 62, 1976.

* 22. Tan, H.E., Griffith, P., "A Model for Quenching with Vertical Flow Boiling", AIChE Symposium Series-Heat Transfer, St. Louis, No. 169, August 1976

* 23. Cluss, E.M., Griffith P., "The Effects of Grid Spacers on Post-Critical Heat Flux Heat Transfer", Trans ANS, Vol. 32, p. 822, 1979.

* 24. Griffith, Peter, Pearson, J.F., Lepkowski, R.J., "Critical Heat Flux During Loss-of-Coolant Accident", Nuclear Safety Vol. 18, No. 3, May-June 1977.

25. Griffith, P., "Boiling Heat Transfer in Nuclear Reactor Technology", Trans ANS, Vol. 32, p. 808, 1979.

* 26. Hsia, D.Y., Griffith, P., "Steam Generator Flow Instability Modeling During the Reflood Stage of a Loss-of-Coolant Accident", Nuclear Science and Engineering, 78, pp. 431-437, 1981.

27. Sudo, Y., Griffith, P., "Upper Plenum Dump During Reflood in PWR Loss-of-Coolant Accident", Journal of Nuclear Science and Technology, Vol. 18, No. 7, p. 487, 1981.

* 28. Robershotte, P., Griffith, P., "Downflow Post Critical Heat Flux to Low Pressure Water Nuclear Technology", Vol. 56, p. 134, 1982.

* 29. Benchaita, M.T., Griffith, P., Rabinowicz, E., "Erosion of Metallic Plate by Solid Particles Entrained in a Liquid Jet", Trans ASME, Journal of Engineering for Industry, Vol. 105, No. 3, pp. 215-222, 1983.

* 30. Calia, C., Griffith, P., "Modes of Circulation in an Inverted U-Tube Array with Condensation", Trans ASME, 104, p. 769, 1982.

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Publications in Refereed Journals (continued)

* 31. Bar-Cohen, A., Ruder, Z., Griffith, P., "Circumferential Wall Temperature Variations in Horizontal Boiler Tubes", International Journal Multiphase Flow, Vol. 9, No. 1, pp. 1-12, 1983.

* 32. Bjorge, R.W., Griffith, P., "Initiation of Water Hammer in Horizontal and Nearly Horizontal Pipes Containing Steam and Subcooled Water", Trans ASME Journal of Heat Transfer, Vol. 105, Nov. 1984.

* 33. Blanchard, D.J., Griffith, P., Rabinowicz, E., "Erosion of a Pipe Bend by Solid Particles Entrained in Water", Trans ASME Journal of Engineering for Industry, Aug. 1984, Vol. 106, pp 213-217, 1983.

* 34. Wu, Shang Z., Wormley, D., Rowell, D., and Griffith, P., "Boiler Implosion Control in Fossil Fuel Power Plants".

* 35. Bar-Cohen, A., Ruder, Z., and Griffith, P., "Development and Validation of Boundaries for Circumferential Isothermality in Horizontal Boiler Tubes", International Journal of Multiphase Flow, Vol. 12 No. 1, 63-77, 1986.

* 36. Ruder, Z., Bar-Cohen, A., Griffith, P., "An Example of Application of the Bounding Dryout Criteria in Horizontal Steam Generating Tubes for the Case of Circumferentially Non-Uniform Heat Flux", International Jnl of Multiphase Flow, Vol. 12, No. 5, pp 845-852, 1986.

* 37. Bowman, J. and Griffith, P., "Determining the Noncondensable Gas Fractions at Elevated Temperatures and Pressures Using Wet and Dry Bulb Temperature Measurements", Trans ASME, Journal of Heat and Transfer, 109 pp. 271-272, February 1987.

* 38. Kim, S.N. and Griffith, Peter, "PWR Pressurizer Modeling" Nuclear Engineering and Design, vol. 102 (1987) 199-209.

* 39. Ruder, Zvi, Bar-Cohen A., Griffith P., "Major Parametric Effects on Isothermality in Horizontal Steam Generating Tube at Low and Moderate-Steam Qualities" International of Heat and Fluid Flow, Vol. 8, No. 3 1987 218-227.

* 40. Sanchez-Caldera, LE, P Griffith, E Rabinowicz. "The Mechanism of Corrosion-Erosion in Steam Extraction Lines of Power Stations," ASME Trans., Jnl of Engineering for Gas Turbines and Power, vol. 110 (April 1988) p 180-185.

* 41. Paik, CY, G Mullen, C Knoess, P Griffith, "Steam Separator Modelling for Various Nuclear Reactor Transients," Nuclear Engineering and Design, Vol. 108 (1988) p 83-97.

* 42. Griffith, Peter, deFritas, G., "Simulating Corrosion-Erosion Mass Transfer Coefficients Using Plaster Models," Proceedings of ANSA Annual Meeting, June 1989.

* 43. Basser, Peter, McMahan, T., Griffith, P., "The Mechanism of Mucus Clearance by Cough," Jnl. of Biomechanical Engineering, Trans. ASME, Nov. 1989.

*outgrowth of supervised thesis

* 44. Griffith, P., Sununu, J., "Wear in Check Valves Due to Flow Induced Motion", Nuclear Engineering and Design, Vol. 115, p 323-330, No's 2 and 3, July 1989.

Publications in Refereed Journals (continued)

45. Swierzewski, T.J., Griffith, P., "Presenting Water Hammer in Large Horizontal Pipes Passing Steam and Water," Jnl. of Heat Transfer, Trans. ASME, May 1990.

46. Boyack, B.E., Catton, I., Duffey, R.B., Griffith, P., Katsma, K.R., Lellouche, G.S., Levy, S., Rohatgi, U.S., Wilson, G.E., Wulff, W., and N. Zuber, "An overview of the code scaling, applicability, and uncertainty evaluation methodology," Nuclear Eng. and Design, Vol. 119, p. 1, No. 1, May (I), 1990

47. Wilson, G.E., Boyack, B.E., Catton, I, Duffey, R.B., Griffith, P., Katsma, K.R., Lellouche, G.S., Levy, S., Rohatgi, U.S., Wulff, W. and N. Zuber, "Characterization of important contributors to uncertainty," Vol. 119, p. 17, No. 1, May (I), 1990

48. Griffith, P., Katsma, K.R., Lellouche, G.S., Levy, S., Rohatgi, U.S., Wilson, G.E. and N. Zuber, "Assessment and ranging of parameters," Vol. 119, p. 33, No. 1, May (I), 1990

49. Lellouche, G.S., Levy, S., Boyack, B.E., Catton, I., Duffey, R.B., Griffith, P., Katsma, K.R., May, R., Rohatgi, U.S., Wilson, G., Wulff, W. and N. Zuber, "Uncertainty evaluation of LBLOCA analysis based on TRAC-PF1/MOD 1," Vol. 119, p. 67, No. 1, May (I), 1990

50. Zuber, N., Wilson, G.E., Boyack, B.E., Catton, I., Duffey, R.B., Griffith, P., Katsma, K.R., Lellouche, G.S., Levy, S., Rohatgi, U.S. and W. Wulff, "Evaluation of scale-up capabilities of best estimate codes," Vol. 119, p. 97, No. 1, May (I), 1990.

51. Catton, I., Duffey, R.B., Shaw, R.A., Boyack, B.E., Griffith, P., Katsma, K.R., Lellouche, S., Levy, Rohatgi, U.S., Wilson, G.E., Wulff, and N. Zuber, "A physically based method of estimating PWR large break loss of coolant accidents," Vol. 119, p. 109, No. 1, May (I), 1990.

* 52. Li, Shipeng, Backer, S., and P. Griffith, "Texturing with steam cond ion heating" Part XI in Texturing of Thermoplastic Yarns," Textile Research Jo 619-626, 1990.

* 53. Li, Shipen, Backer, S., and P. Griffith "Texturing of Thermoplastic Yarns Part XII: The Heat Transfer Process in Steam Condensation Heating" Textile Research Journal, Vol. 60, No. 12, Dec. 1990.

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Other Major Publications

1. Griffith, P., "Dropwise Condensation", "Two-Phase Flow", chapters in Handbook of Heat Transfer edited by W.M. Rohsenow and J. Hartnett, McGraw-Hill, 1973 and 1985.

2. Griffith, P., "Condensation" chapter in Handbook of Multiphase Systems edited by G. Hetsroni, Hemisphere Publishing Co., 1982.

3. Griffith, P., "Dropwise Condensation" section in Heat Exchanger Design Handbook edited by D.B. Spalding and J. Taborek, Hemisphere Publishing Co., 1983.

4. Griffith, P., "The Slug-Annular Flow Regime Transition at Elevated Pressure", ANL-6796, 1963.

5. Griffith, P., "Nucleation and Bubble Formation in Boiling", Proceedings of the 1956-66 Institution of Mechanical Engineers, 180 Part 1.

* 6. Bjornard, T., Griffith, P., "PWR Blowdown Heat Transfer", Chap. 1 in "Thermal-Hydraulic Aspects of Nuclear Reactor Safety", vol. 1, ASME 1977.

7. Griffith, P., "National Circulation" Paper in EPRI WS-81-212 conference on Simulation Methods for Nuclear Power Systems, May 1981, Tucson, Arizona 1981.

8. Griffith, Peter, "Unsolved Problems in Reactor Thermal Hydraulics", Proceedings of 3rd International Meeting on Reactor Thermal Hydraulics, Newport, RI, Published by the ANS, October 1985.

9. Liao, Lih-Yih, Parlos, Alex and Griffith, Peter, "Heat Transfer, Carry-over, and Fall Back in Nuclear Steam Generators During Transients", 8th International Heat Transfer Conference, San Francisco, CA, August 1986.

10. Griffith, Peter, "Transient Critical Heat Flux", International Centre for Heat and Mass Transfer" Conference held at Dubrovnik, Yugoslavia. May 1987 Paper 7.3.

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R.D. Gasser and M.S. Kazimi, "Onset of Convection in a Porous Medium with Internal Heat Generation," J. Heat Transfer, 98, February 1976: 24-30.

M.H. Chun, R.D. Gasser and M.S. Kazimi, "Dynamics of Solidification of Flowing Fluids with Application to LMFBR Post-Accident Fuel Relocation," Proc. Int.l Meeting in Fast Reactor Safety and Related Physics, Chicago, IL, October 1976.

W. Gustavson, M.S. Kazimi and J.C. Chen, "Heat Transfer and Fluid Dynamics in a Volume-Heated Boiling Pool, " Proc. Int. Meeting on Fast Reactor Safety and Related Physics, Chicago, IL, October 1976.

S.S. Tsai and M.S. Kazimi, "The Potential for Penetration of a Hot Vapor Jet into a Subcooled Liquid," Presented at ASM Winter Meeting, New York, 76-WA/HT-78, December 1976.

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M.S. Kazimi and J.C. Chen, "A Condensed Review of the Technology of Post-Accident Heat Removal for the LMFBR," A Critical Review sponsored by the American Nuclear Society, Nuclear Technology, 38, May 1978: 339-367.

R.A. Rothrock and M.S. Kazimi, "On the Potential for Plugged Core Pressurization following a Hypothetical Accident in an LMFBR," Nucl. Eng. and Design, 53, 1979: 197-205.

R.C. Smith, W.M. Rohsenow and M.S. Kazimi, Volumetric Heat Transfer Coefficients for Direct Contact Evaporation," J. Heat Transfer, 104 (2), May 1982: 264-270.

S.M. Mohammed and M.S. Kazimi, "Hydrodynamics of Degraded Core Cooling," Nucl. Eng. and Design, 71, 1982: 33-43.

R.A. Rothrock, M.S. Kazimi and W.M. Rohsenow, "Hydrodynamic Characteristics of a Gas Blowdown into a Two-Dimensional Liquid Pool, " Nucl. Eng. and Design, 71, 1982: 241-250.

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M. Lee and M.S. Kazimi, "A Sensitivity Analysis of the Impact of Corium Heat Transfer to Concrete on Containment Pressure, " Proc. of Int'l Mtg. on LWR Severe Accident Evaluation, Cambridge, MA, August 1983.

M. Lee, G. Brown and M.S. Kazimi, "A Heat Transfer Model for the Corium Concrete/Interface, Proc. of Int'l Mtg. on LWR Severe Accident Evaluation, Cambridge, MA, August, 1983.

M.Lee and M.S. Kazimi, "Interfacial Heat Transfer Between Bubble Agitated Immiscible Liquid Layers," Proc. of Sixth Info. Exchange Mtg. on Debris Coolability, Los Angeles, CA, November 7-9, 1984.

L. Kao and M.S. Kazimi, "Containment Pressure Response to a Meltdown Condition of the Light Water Reactor," Proc. of Sixth Info. Exchange Mtg. on Debris Coolability, Los Angeles, CA, November 7-9, 1984.

L.S. Kao, M. Lee and M.S. Kazimi, "Assessment of Heat Transfer Models for Corium-Concrete Interaction," Proc. of Int'l Conf. on Reactor Thermal Hydraulics, Newport, RI, October 1985.

B.R. Sehgal, M.S. Kazimi, M.L. Corradini, "Current Status of Molten Core-Concrete Interaction," Specialist Mtg. on Core Debris/Concrete Interactions, Palo Alto, CA, September 2-4, 1986.

L.S. Kao and M.S. Kazimi, "The Impact of Heat Transfer Models for Core/Concrete Interaction," Specialist Mtg. on Core Debris/Concrete Interactions, Palo Alto, CA, September 2-4, 1986.

L.S. Kao and M.S. Kazimi, "A Heat Transfer Model for Core/Concrete

Interaction," National Heat Transfer Conference, Pittsburg, PA AIChE symposium Series #257, 83, 1987.

L.S. Kao and M.S. Kazimi, "The Impact of Heat Transfer Models on Core/Concrete Interaction," Nuclear Technology, 78, August 1987: 170-184.

M.S. Kazimi, "On the Liner Failure Potential in MARK-1 BWR's," Nuclear Science & Engineering, Vol. 103, 1989: 59-69.

M. Siddique, M.W Golay and M.S. Kazimi, "The Effect of Hydrogen on Forced Convection Steam Condensation, " AIChE Symposium Series, National Heat Transfer Conference, Philadelphia, PA, August 6-9, 1989.

A Dehbi, M. Golay and M.S. Kazimi, "The Effect of Air in Steam Condensation Under Turbulent Natural Convection," presented at the 1990 ANS Winter Meeting, Washington, D.C. November 11-16, 1990.

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PUBLICATIONS:

1. J. E. Meyer, "The Negative Pressure Problem in Hydrodynamic Lubrication," Ph.D. Dissertation, Carnegie Institute of Technology, June 1955.

2. J. E. Meyer, "Study of the Force-Deflection Characteristics of a Belleville Spring," WAPD-198, April 1958.

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4. J. E. Meyer, "Dimensionless Form for Complete-Loss-of Flow Parameters,"in "Bettis Technical Review," WAPD-BT-9, August 1958, pp. 51-59.

5. J. C. Fairey, J. E. Meyer, J. B. Callaghan, S. H. Meanor, A. V. Pace, and R. B. Smith, "PROP and JET - A Program for the Synthesis and Survey of Three-dimensional Power Shapes on the IBM-704," WAPD-TM-116, September 1958.

6. J. E. Meyer, "The ART Program for the Digital Solution of Reactor Thermal Transients," Trans. Am. Nucl. Soc. 1, 135-136 (1958).

7. J. E. Meyer and R. B. Smith, "The ART Program for the Digital Solution of Reactor Thermal Transients," in "Bettis Technical Review," WAPD-BT-13, May 1959, pp. 80-86.

8. O. J. Marlowe and J. E. Meyer, "TURBO, A Program for Two-Dimensional Depletion Calculations," in "Bettis Technical Review," WAPD-BT-13, May 1959, pp. 89-92.

9. J. E. Meyer and J. B. Callaghan, "PROP and JET - A Program for the Synthesis and Survey of Three-Dimensional Power Shapes," in "Bettis Technical Review," WAPD-BT-13, May 1959, pp. 99-101.

10. J. E. Meyer and J. N. Grace, "Selection of Burnable Poisons and Zone Loading for Optimum Nuclear Design," Trans. Am. Nucl. Soc. 2, 139-140 (1959).

11. J. E. Meyer, R. B. Smith, H. G. Gelbard, D. E. George and W. D. Peterson, "ART - A Program for the Treatment of Reactor Thermal Transients on the IBM-704," WAPD-TM-156, November 1959.

12. J. E. Meyer and W. D. Peterson, "ART-04 - A Modification of the ART Program for the Treatment of Reactor Thermal Transients on the IBM-704," WAPD-TM-202, July 1960.

13. J. E. Meyer, "Conservation Laws in One-Dimensional Hydrodynamics," in "Bettis Technical Review," WAPD-BT-20, September 1960, pp. 61-72.

14. J. E. Meyer, "Hydrodynamic Models for the Treatment of Reactor Thermal Transients," Nucl. Sci. Eng. 10, 269-277 (1961).

15. J. E. Meyer and W. D. Long, "A Channel Integral Model for the Treatment of Transient Fluid Flow," in "Bettis Technical Review," WAPD-BT-23, August 1961, pp. 1-16.

16. J. E. Meyer and J. S. Williams, Jr., "A Momentum Integral Model for the Treatment of Transient Fluid Flow," in "Bettis Technical Review," WAPD-BT-25, May 1962, pp. 47-72.

17. J. E. Meyer and E. A. Reinhard, "A Small Perturbation Approach to the Study of Parallel Channel Boiling Flow Oscillations," WAPD-TM-342, January 1963.

18. J. E. Meyer and R. P. Rose, Application of a Momentum Integral Model to the Study of Parallel Channel Boiling Flow Oscillations," Trans. Am. Soc. Mech. Engrs. 85, Series C, 1-9 (1963).

19. L. L. Lynn and J. E. Meyer, "A Numerical Comparison of the Implicit and Explicit Techniques for the Convective Boundary Condition," Trans. Am. Soc. Mech. Engrs. 85, Series C, 280-281 (1963).

20. J. E. Meyer and J. V. Reihing, Jr., "Boiling Induced Reactivity Feedback," in Naval Reactor Physics Handbook, Vol. I, Selected Basic Techniques," A. Radkowsky, Ed., pp. 915-933, United States Atomic Energy Commission, Washington 1964.

21. J. E. Meyer and E. A. Reinhard, "Numerical Techniques for Boiling Flow Stability Analysis," Trans. Am. Soc. [illegible] 87, Series C, 311-312 (1965).

22. E. Duncombe, J. E. Meyer and W. A. Coffman, "Comparison with Experiment of Calculated Dimensional Changes and Failure Analysis of Irradiated Bulk Oxide Fuel Test Rods Using the CYGRO-1 Computer Program," WAPD-TM-583, September 1966.

23. F. T. Dunckhorst, L. L. Lynn, W. A. Coffman and J. E. Meyer, "GLUB-1 - A FORTRAN-IV Digital Computer Program for Waterlogged Fuel Element Analysis," WAPD-TM-569, November 1966.

24. W. A. Coffman, F. T. Dunckhorst, L. L. Lynn and J. E. Meyer, "Analysis of a Waterlogged Fuel Element Subjected to a Power Excursion," Nucl. Appl. 4, 372-381 (1968).

25. A. Golpalakrishnan, V. E. Schrock and J. E. Meyer, "Transient Non-Boiling Heat Transfer in a Reactor Coolant Channel During Accidental Power Excursions," ASME Publication 72-WA/HT-50, 1972.

26. J. E. Meyer, A. S. Hanson, M. Pitek, and S. P. Schultz, "Utility Needs for Fuel Rod Modeling Capability," Trans. Am. Nucl. Soc. 26, 219-220 (1977).

27. M. N. Fardis, C. A. Cornell, and J. E. Meyer, "A Probabilistic Seismic Analysis of Containment Liner Integrity," Trans. 4th Intl. Conf., Str. Mech. in Reactor Tech. K(b), Paper K4/16 (1977).

28. S. P. Schultz, A. L. Bement, and J. E. Meyer, "Using Fuel Performance Prediction in Light-Water Reactor Fuel Management," Trans. Am. Nucl Soc 27, 730-731 (1977).

29. J. E. Meyer, "Structural Advantages of Steady State Fusion Power Reactors," MIT Plasma Fusion Center Report PFC/TR-78-3, March 1978.

30. W. L. Weaver III, J. E. Meyer, and A. K. Agraval, "A Few-Pressure Model for Transient Two-Phase Flows," Trans. Am. Nucl. Soc. 28, 273-274 (1978).

31. G. G. de Andrade and J. E. Meyer, "Load-Following Operation of a Pressurized Water Nuclear Power Plant," Trans. Am. Nucl. Soc. 28, 597-598 (1978).

32. Y. Y. Liu, J. E. Meyer, and A. S. Argon, "Theoretical Analysis of Pellet-Cladding Mechanical Interaction," Trans. Am. Nucl. Soc. 30, 184-185 (1978).

33. J. E. Meyer, "Some Physical and Numerical Considerations for the SSC-S Code," Brookhaven National Laboratory Report BNL-NUREG-50913, September 1978.

34. J. T. Maki and J. E. Meyer, "LWR Fuel Performance Analysis, Fuel Cracking and Relocation," MIT Energy Laboratory Report MIT-EL78-038, October 1978.

35. M. N. Fardis, C. A. Cornell, and J. E. Meyer, "Accident and Seismic Containment Reliability," J. Structural Div. ASCE 105, No. ST1, Proc. Paper 14305, 67-83 (1979).

36. Y. Y. Liu, J. E. Meyer, and A. S. Argon, "A Formulation for the Analysis of Pellet-Cladding Mechanical Interaction," Trans. 5th Intl. Conf., Str. Mech. in Reactor Tech. D, Paper D3/5 (1979).

37. J. E. Meyer and A. O. Adegbulugbe, "Functional Requirements for Fusion Reactor First Walls - A Status Report," MIT Nuclear Reactor Laboratory Report MITNRL-002, March 1980.

38. J. E. Rivera and J. E. Meyer, "Stainless Steel Clad for Light Water Reactor Fuels," MIT Energy Laboratory Report MIT-EL80-021, July 1980.

39. D. P. Griggs, R. B. Vilim, C. H. Wang, and J. E. Meyer, "Analysis of Forces on Core Structures During a Loss-of-Coolant Accident," MIT Energy Laboratory Report MIT-EL80-029, August 1980.

40. O. K. Harling, G. P. Yu, N. J. Grant and J. E. Meyer, "Application of High Strength Copper Alloys for a Fusion Reactor First Wall," J. Nucl. Matls. 103 and 104, 127-132 (1981).

41. A. O. Adegbulugbe and J. E. Meyer, "Failure Criteria for Fusion Reactor First Wall Structural Design," J. Nucl. Matls. 103 and 104, 161-166 (1981).

42. G. P. Yu and J. E. Meyer, "Relationship of Material Properties to the Design of a Fusion Reactor First Wall," Trans. Am. Nucl. Soc. 39, 259-260 (1981).

43. E. Bobrov, L. Bromberg, D. R. Cohn, N. Diatchenko, R. J. LeClaire, J. E. Meyer and J. E. C. Williams, "High Field Tokamaks with DD-DT Operation and Reduced Tritium Breeding Requirements," MIT Plasma Fusion Center Report PFC/RR-83-5, February 1983.

44. L. Bromberg, D. R. Cohn, E. Bobrov, N. Diatchenko, R. J. LeClaire, J. E. Meyer and J. E. C. Williams, "High Field Tokamaks with DD-DT Operation and Reduced Tritium Breeding Requirements," MIT Plasma Fusion Center Report PFC/CP-83-7, April 1983.

45. S. Harriague and J. E. Meyer, "A Study of Friction and Axial Effects in Pellet-clad Mechanical Interaction," in "Water Reactor Fuel Element Performance Computer Modeling," J. Gittus, ed., pp. 187-217, Applied Science Publishers, England, 1983.

46. S. V. Guerreiro Ribeiro and J. E. Meyer, "Mechanisms of Flow-induced Vibrations of Circular Cylinders in Crossflow," Trans. 7th Intl Conf. Str. Mech. in Reactor Tech. B, Paper B5/7 (1983).

47. R. J. LeClaire, J. E. Meyer, L. Bromberg, D. R. Cohn and J. E. C. Williams, "Aspects of Long Pulse Commercial Tokamak Reactor Design," MIT Plasma Fusion Center Report PFC/CP-83-15, Dec. 1983.

48. G. Garner, R. S. Ornedo, O. L. Deutsch, J. E. Meyer, and S. M. Divakaruni, "Robust Technique for Failure Detection and Vessel Level Validation in Boiling Water Reactors," EPRI/ISA Symp. New Technologies in Nucl. Power Plant Instr. and Contr., Nov. 1984.

49. R. J. Witt and J. E. Meyer, "Computer Techniques for Sensor Validation During EBR-II Natural Circulation," MIT Nuclear Engineering Dept. Report MITNE-264, Nov. 1984.

50. P. M. Blanch and J. E. Meyer, "Northeast Utilities Signal Validation Applications for Millstone-3 and Connecticut Yankee Plants," EPRI Seminar: Nuclear Power Plant Safety Control Technology, Palo Alto, CA, Feb. 1985.

51. G. Garner, S. M. Divakaruni, and J. E. Meyer, "BWR Level Estimation Using Kalman Filtering Approach," EPRI Seminar: Nuclear Power Plant Safety Control Technology, Palo Alto, CA, Feb., 1985.

52. M. M. Blancaflor, R. J. Lord, and J. E. Meyer, "The Implementation of Signal Validation on the Millstone Point Unit 3 SPDS," Trans. Am. Nucl. Soc. 49, 375 (1985).

53. S. P. Kao and J. E. Meyer, "A Two-Fluid Pressurizer Digital Model," Trans. Am. Nucl. Soc. 49, 469-470 (1985).

54. P. M. Blanch and J. E. Meyer, "Northeast Utilities Signal Validation Applications for Safety Parameter Display Systems," Intl. Topical Meeting on Computer Applications for Nuclear Power Plant Operations and Control, Pasco, WA, Sept. 1985.

55. S. P. Kao and J. E. Meyer, "A Plant-Computer-Based Pressurized Water Reactor Primary System Thermal Hydraulic Model," Trans. Am. Nucl. Soc. 50, 637-638 (1985).

56. Y. Nakagawa and J. E. Meyer, "Effects of Pulsed Op[illegible]ion on the Lifetime of Turbine Rotors," Fusion Tech. 8, 1783-1788 (1985).

57. R. J. LeClaire and J. E. Meyer, "The Impact of Thermal Fatigue and Other Design Limits on Pulsed Commercial Tokamak Reactor Design," J. Fusion Energy, 4, 353-364 (1985).

58. R. J. Witt and J. E. Meyer, "Demonstration of Methods for Analytic Measurement of Natural Circulation Flow in EBR-II," MIT Nucl. Engr. Dept. Report MITNE-270, February 1986.

59. M. M. Blancaflor and J. E. Meyer, "Signal Validation for Millstone-3 SPDS; Special Problems and Solutions," SPDS Implementation and Emergency Response Facilities, EPRI Seminar, Boston, MA, May 1986.

60. J. A. Bernard, K. S. Kwok, D. D. Lanning, A. F. Henry, and J. E. Meyer, "Applications of the Reactivity Constraint Approach to the Transient Control of Spacecraft Reactors," Proc. 4th Symp. Space Nucl. Power Systems, Albuquerque, NM, January, 1987.

61. E. V. Depiante and J. E. Meyer, "Parity Simulation of Two-Phase Thermal Hydraulics in Plant Components," Trans. Am. Nucl. Soc., 54, 179-180 (1987).

62. J. A. Bernard, K. S. Kwok, D. D. Lanning, A. F. Henry, and J. E. Meyer, "Transient Control of Reactor Power Generation for Rapid Maneuvering," in "Space Nuclear Power Systems 1987," M. S. El-Genk and M. D. Hoover, eds., pp. 299-309, Orbit Book Co., Malabar, FL, 1988.

63. J. I. Choi, J. E. Meyer, and D. D. Lanning, "Automatic Controller for Steam Generator Water Level during Low Power Operation," Nucl. Eng. Des. 117, 263-274 (1989).

64. E. V. Depiante and J. E. Meyer, "Parity Simulation of Single-Phase Thermal Hydraulics," Nucl. Sci. Eng. 104, 153-168 (1990).

65. E. V. Depiante and J. E. Meyer, "Two-Phase Thermal-Hydraulic Analysis Using Parity Simulation," Nucl. Sci. Eng. 104, 169-182 (1990).

66. E. V. Depiante and J. E. Meyer, "Parity Simulation of a Boiling Water Reactor System," Nucl. Eng. Des. 118, 163-178 (1990).

67. L. D. Simmons and others (incl J. E. Meyer), "Naval Propulsion Systems - Phase I: Survey of Alternate Technologies," IDA Paper P-2532, Institute for Defense Analyses, Alexandria, VA, February, 1991.

68. R.J.Lord and J.E.Meyer, "Instrument Calibration Reduction Program," EPRI NP-7207-CCML Vols 1-3, Computer Code Manual, Electric Power Research Institute, Palo Alto, CA, March 1991.

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EMPLOYMENT:

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Associate Professor, Stanford University, 1986-1989
Research Scientist, NASA Ames Research Center, 1982- 1986.
Acting Assistant Professor, Stanford University, 1980-1982
National Research Council Fellow, NASA Ames, 1978-1980
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AWARDS:

NASA's Exceptional Scientific Achievement Medal, 1985
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Awarded endowed chair, Franklin P. and Caroline M. Johnson Professor of Mechanical Engineering, Dec. 1990.

EXTERNAL ACTIVITIES:

Editorial Board of the Journal of Scientific Computing
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PUBLICATIONS:

Review Articles

"Numerical Simulation of Turbulent Flows", by Rogallo, R. S. and Moin, P., Annual Rev. of Fluid Mech., 1984, Vol. 116, 99-137.

Book Reviews

Turbulence in Fluids: Stochastic and Numerical Modeling, by M. Lesieur, SIAM Review, 1989, 31, 510-511.
Turbulence and Random Processes in Fluid Mechanics, by M.T. Landahl and E. Mollo-Christensen, SIAM Review, 1989, 31, 511-512.

Publications in Recognized Journals

Moin, P. and Kim, J. 1980 On the numerical solution of incompressible flows involving solid boundaries. J. Comp. Physics 35, 381-392.

Moin, P. and Kim, J. 1982 Numerical investigation of turbulent channel flow. J. Fluid Mech. 118, 341-377.

Moser, R.D., Moin, P. and Leonard, A. 1983 A spectral numerical method for the Navier-Stokes equations with applications to Taylor-Couette flow. J. Comp. Physics 52, 524-544.

Kim, J. and Moin, P. 1985 Application of a fractional step method to incompressible Navier Stokes equations. J. Comp. Physics 59, 308-323.

Moin, P. and Kim, J. 1985 The structure of vorticity field in turbulent channel flow. Part I: Analysis of instantaneous fields and statistical correlations. J. Fluid Mech. 155, 441-464.

Kim, J. and Moin, P. 1986 The structure of vorticity field in turbulent channel flow. Part II: Study of ensemble averaged fields. J. Fluid Mech. 162, 339.

Moin, P., Leonard, A. and Kim, J. 1986 Evolution of a curved vortex filament into a vortex ring. Phys. Fluids 29, No. 4, pp 955-963.

Moser, R.D. and Moin, P. 1987 On the effects of curvature in wall-bounded turbulent flows. J. Fluid Mech. 175, 479-510.

Rogers, M.M. and Moin, P. 1987 The structure of the vorticity field in homogeneous turbulent flows. J. Fluid Mech. 176, 33-66.

Kim, J., Moin, P. and Moser, R.D. 1987 Turbulent statistics in fully developed turbulent channel flow at low Reynolds number. J. Fluid Mech. 177, 133-166.

Rogers, M.M. and Moin, P. 1987 Helicity fluctuations in incompressible turbulence. Phys. of Fluids 30, 2662-2671.

Adrian, R.J. and Moin, P. 1988 Stochastic estimation of conditional eddies: homogeneous turbulence. J. Fluid Mech. 190, 531-554.

Jimenez, J.J., Moin, P., Moser, R.D. and Keefe, L. 1988 Ejection mechanisms in the sublayer of a turbulent channel. Phys. Fluids 31, 1311.

Mansour, N.N., Kim, J. and Moin, P. 1988 Reynolds stress and dissipation rate budgets in turbulent channel flow. J. Fluid Mech. 194, 15-44.

Piomelli, U., Moin, P. and Ferziger, J. 1988 Model consistency in large eddy simulation of turbulent channel flows. Phys. Fluids 31, 1884-1891.

Chambers, D.H., Adrian, R.J., Moin, P., Stewart, D.S. and Sung, H.J. 1988 Karhunen Loeve expansion of Burger's model of turbulence. Phys. Fluids 31, 2573-2582.

Piomelli, U., Moin, P., Kim, J. and Ferziger, J.H. 1989 New approximate boundary conditions for large eddy simulations of wall-bounded flows. Phys. Fluids A, 1, 1061-1068.

Mansour, N., Kim, J. and Moin, P. 1989 Near wall k-e turbulence modeling. AIAA Journal, 27, No. 8, 1068-1074.

Moin, P. and Moser, R.D. 1989 Characteristic eddy decomposition of turbulence in a channel. J. Fluid Mech. 200, 471-509.

Piomelli, U., Moin, P. and Ferziger, J.H. 1989 Near wall boundary conditions for large eddy simulation. Phys. Fluids A, 1, 1061-1068.

Choi, H. and Moin, P. 1990 On the space-time characteristics of wall pressure fluctuations. Phys. of Fluids A, 2 (8), 1450-1460.

Moin, P., Shih, T., Driver, D. and Mansour, N. 1990 Direct numerical simulation of a three-dimensional turbulent boundary layer. Phys. of Fluids A, 2 (10), 1846-1853.

Pauley, L.L., Moin, P. and Reynolds, W.C. 1990 The structure of two-dimensional separation. J. Fluid Mech., 220, 397-412.

Lee, M., Kim, J. and Moin, P. 1990 Structure of turbulence at high shear rate. J. Fluid Mech., 216, 561-583.

Jimenez, J. and Moin, P. 1991 The minimal flow unit in near wall turbulence. J. Fluid Mech., 225, 213-240.

Le, H. and Moin, P. 1991 An improvement of fractional step methods for the incompressible Navier-Stokes equations. J. Comp. Phys., 92, 2.

Lee, S., Lele, S.K., and Moin, P. 1991 Eddy shocklets in decaying compressible turbulence. Phys. Fluids A, 3 (4), 657-664.

Rai, M. and Moin, P. 1991 Direct simulations of turbulent flow using finite-difference schemes. J. Comp. Phys., 96 (1), 15-53.

Piomelli, U., Moin, P. and Ferziger, J.H. 1991 Large eddy simulation of the flow in a transpired channel. AIAA J. of Thermophysics and Heat Transfer, 5 (1), 124-128.

Germano, M., Piomelli, U., Moin, P., and Cabot, W.H. 1991 A dynamic subgrid-scale eddy viscosity model. Phys. Fluids A, 3 (7), 1760-1765.

Piomelli, U., Cabot, W.H., Moin, P., and Lee, S. 1991 Subgrid-scale backscatter in turbulent and transitional flows. Phys. Fluids A, 3 (7), 1766-1771.

Moin, P. 1991 Towards large eddy and direct simulation of complex turbulent flows. Computer Methods in Applied Mechanics and Engineering 87, 329-334.

Choi, H., Moin, P., and Kim, J. 1991 On the effect of riblets in fully developed laminar channel flows. Phys. Fluids A, 3 (8), 1892-1896.

Colonius, T., Lele, S.K., and Moin, P. 1991 The free compressible viscous vortex. J.Fluid Mech., 230, 45-73.

Publications in books and in invited or refereed symposium proceedings

Mansour, N.N., Moin, P., Reynolds, W.C. and Ferziger, J.H. 1977 Improved methods for large eddy simulation of turbulence. In Turbulent Shear Flows I (ed. F. Durst, B.E. Launder and F. W. Schmidt). pp. 379-385. Springer.

Moin, P., Mansour, N.N., Reynolds, W.C. and Ferziger, J.H. 1978 Large eddy simulation of turbulent shear flows. In Topics in Applied Physics, Vol. 90, pp. 400-409. Springer.

Kim, J. and Moin, P. 1979 Large eddy simulation of turbulent channel flow-ILLIAC IV calculation. In Turbulent Boundary Layers-Experiments, Theory and Modeling, The Hague Netherlands. AGARD Conf. Proc. no. 271.

Moin, P. and Kim, J. 1981 Large-scale numerical simulation of wall-bounded turbulent shear flows. In Proceedings of the 1980-81 AFOSR-HTTM-Stanford Conference on Complex Turbulent Flows. Vol. II, pp. 865-868.

Moin, P. 1982 Numerical simulation of wall-bounded turbulent shear flows. Lecture Notes in Physics, Vol. 170, 55-76. Springer.

Mansour, N.N., Kim, J. and Moin, P. 1983 Computation of turbulent flows over a backward facing step, Proc. of the 4th Turbulent Shear Flow Conf., Karlsruhe, W. Germany, Sept. 1983.

Moin, P. 1984 Probing turbulence via large eddy simulation. AIAA 84-0174.

Kuhn, G.D., Moin, P., Kim, J. and Ferziger, J.H. 1984 Turbulent flow in a channel with a wall with progressive waves. Proc. of ASME Symposium on Laminar/Turbulent Boundary Layers: Control, Modification and Marine Applications; New Orleans, Feb 1984.

Moin, P., Rogers, M.M. and Moser, R.D. 1985 Structure of turbulence in the presence of uniform shear. Proc. of the 5th Symp. on Turb. Shear Flows, Cornell Univ., Ithaca, NY, Aug 7-9, 1985.

Moser, R.D. and Moin, P. 1985 Effects of wall curvature on turbulence statistics. Proc. of the Fifth Symposium on turbulent Shear Flows. Cornell Univ., Ithaca, NY, Aug 7-9, 1985.

Moin, P. 1985 Recent studies of the structure of turbulence using simulation databases. In Direct and Large Eddy Simulation of Turbulence, (Schumann and Friedrich eds.) Vieweg Braunschweig, pp 181-187.

Moin, P. 1985 A note on the structure of turbulent shear flows. In theoretical approaches to turbulence (Dwoyer, Hussaini and Voigt eds.) Vol 58, pp 365-368. Springer.

Moin, P. 1987 Analysis of turbulence data generated by numerical simulations. AIAA 87-0194.

Piomelli, U., Moin, P. and Ferziger, J.H. 1987 Model consistency in the large eddy simulation of turbulent channel flows. AIAA paper 87-1446.

Moin, P., Adrian, R.J. and Kim, J. 1987 Stochastic estimation of organized structures in turbulent channel flow. Proc. of Sixth Symposium on Turbulent Shear flows. Sept. 7-9, 1987, Toulouse, France.

Mansour, N.N., Kim, J. and Moin, P. 1987 Near wall k-e modeling. Proc. of Sixth Symposium on Turbulent Shear flows. Sept. 7-9, 1987, Toulouse, France.

Lee, M.J., Kim, J. and Moin, P. 1987 Turbulence Structures at high shear rate. Proc. of Sixth Symposium on Turbulent Shear flows. Sept. 7-9,1987, Toulouse, France.

Moin, P. and Spalart, P.R. 1987 Contributions of numerical simulation databases to the physics, modeling, and measurement of turbulence. In Recent Advances in Turbulence, edited by R. J. Arnt and W. K. George, Hemisphere, 1987.

Moin, P. 1988 Similarity of organized structures in turbulent shear flows. In Near Wall Turbulence, eds., S.J. Kline and N.H. Afgan, Hemisphere Pub. Corp., pp. 2-6, 1988.

Hunt, J.C.R., Moin, P., Lee, M., Moser, R.D., Spalart, P., Mansour, N.N., Kaimal, J.C. and Gaynor, E. 1989 Cross Correlation and Length Scales in Turbulent Flows near Surfaces, in Advances in Turbulence 2, editors, H.-H. Fernholz and H.E. Fiedler, Springer-Verlag Berlin.

Moin, P., Shih, S., Sendstad, O. and Driver, D. Direct numerical simulation of a three-dimensional turbulent boundary layer, in Recent Developments in Computational Fluid Dynamics, edited by T. E. Tezduyar and T. J. R. Hughes, AMD-Vol. 95, 139-146.

Kim, J. and Moin, P. 1989 Transport of passive scalar in turbulent channel flow. In Turbulent Shear Flows 6, editors, J.C. Andre et al, 86-96, Springer-Verlag.

Rai, M. and Moin, P. 1989 Direct simulations of turbulent flow using finite-difference schemes, AIAA-89-0369.

Moin, P., Shih, T., Driver, D., and Mansour, N. 1989 Numerical simulation of a three-dimensional turbulent boundary layer, AIAA-89-0373.

Piomelli, U., Moin, P., and Ferziger, J. 1989 Large eddy simulation of the flow in a transpired channel, AIAA-89-0375.

Pauley, L., Moin, P., and Reynolds, W.C. 1989 On the structure of two- and three-dimensional unsteady separation, AIAA-89-0287.

Moin, P., Kim, J., and Choi, H. 1989 On the active control of wall-bounded turbulent flows, AIAA-89-0960.

Keffe, L., Moin, P., and Kim, J. 1990 Applications of chaos theory to shear turbulence. The Ubiquity of Chaos, American Association for Advancement of Science, editor: S.Krasner, 56-63.

Colonius, T., Lele, S.K., and Moin, P. 1991 Scattering of sound waves by a compressible vortex, AIAA-91-0494.

Lee, S., Lele, S.K., and Moin, P. 1991 Direct numerical simulation and analysis of shock turbulence interaction, AIAA-91-0523.

Rai, M. and Moin, P. 1991 Direct numerical simulation of transition and turbulence in a spatially evolving boundary layer, AIAA-91-1607.

Neves, J.C., Moin, P., and Moser, R.D. 1991 Numerical study of axial turbulent flow over long cylinders, 8th Turbulent Shear Flows Conference, Munich, Germany.

Sendstad, O., and Moin, P. 1991 On the mechanics of 3-D turbulent boundary layers, 8th Turbulent Shear Flows Conference, Munich, Germany.

Published Abstracts

Moin, P., Reynolds, W.C., and Ferziger, J.H. 1978 Large eddy simulation of turbulent channel flow. Bull. of the Am. Physical Soc., Vol. 23, 8, 989.

Moin, P. and Kim, J. 1981 Three-dimensional time dependent calculation of turbulent channel flow. Bull. of the Am. Physical Soc., Vol. 25, 9, 1083.

Kim, J. and Moin, P. 1981 Investigation of flow structure in the wall region of turbulert boundary layers via a numerical simulation. Bull. of the Am. Physical Soc., Vol. 25, 9, 1103.

Moin, P. 1981 Effect of transpiration on the structure of a wall-bounded turbulent flow. Bull. of the Am. Physical Soc., Vol. 26, 9, 1245.

Moin, P. 1982 On the structure of the vorticity vector field in turbulent channel flow. Bull. of the Am. Physical Soc., Vol. 27, 9, 1198.

Moin, P. 1983 Proper orthogonal decomposition of turbulence in a channel. Bull. of the Am. Physical Soc., Vol. 28, 9, 1396.

Moser, R.D. and Moin, P. 1983 Numerical simulation of modulated wavy vortices in Taylor-Couette flow. Bull. of the Am. Physical Soc., Vol. 28, 9, 1398.

Kim, J. and Moin, P. 1983 Effect of wall motion on turbulence structure. Bull. of the Am. Physical Soc., Vol. 28, 9, 1396.

Moser, R.D. and Moin, P. 1984 Near wall structures in curved turbulent channel flow. Bull. of the Am. Physical Soc., Vol. 29, 9, 1528.

Moin, P. 1984 Hairpin vortices in wall-bounded turbulent flows. Bull. of the Am. Physical Soc., Vol. 29, 9, 1520.

Mansour, N.N., Kim, J. and Moin, P. 1985 Turbulence statistics from a channel flow simulation. Bull. of the Am. Physical Soc., Vol. 30, 10, 1733.

Rogers, M.M. and Moin, P. 1985 The structure of homogeneous turbulent flows. Bull. of the Am. Physical Soc., Vol. 30, 10, 1733.

Moin, P. 1985 High Reynolds shear stress motions in the wall region. Bull. of the Am. Physical Soc., Vol. 30, 10, 1723.

Kim, J. and Moin, P. 1986 Flow structures responsible for the bursting process. Bull. of the Am. Physical Soc., Vol. 31, 10, 1716.

Lee, M.J., Kim, J. and Moin, P. 1986 Effect of high shear rate on the structure of turbulence. Bull. of the Am. Physical Soc., Vol. 31, 10, 1725.

Mansour, N.N., Kim, J. and Moin, P. 1986 The dissipation rate of the Reynolds stress tensor. Bull. of the Am. Physical Soc., Vol. 31, 10, 1725.

Piomelli, U., Ferziger, J.H., Moin, P. and Kim, J. 1986 New models for the large eddy simulation of turbulent channel flows. Bull. of the Am. Physical Soc., Vol. 31, 10, 1725.

Moin, P. and Adrian, R.J. 1986 Stochastic estimation of conditional eddies in homogeneous turbulent shear flow. Bull. of the Am. Physical Soc., Vol. 31, 10, 1717.

Keefe, L., Moin, P. and Kim, J. 1987 The dimension of an attractor in turbulent Poiseuille flow. Bull. of the Am. Physical Soc., Vol. 32, 10, 2026.

Moser, R.D. and Moin, P. 1987 Characteristic eddy decomposition of turbulence in a channel. Bull. of the Am. Physical Soc., Vol. 32, 10, 2029.

Moin, P., Jimenez, J., Keefe, L. and Moser, R.D. 1987 On the mechanics of intense shear layers in the wall region. Bull. of the Am. Physical Soc., Vol. 32, 10, 2030.

Adrian, R.J., Chambers, D.H., Stewart, D.S., Sung, H. and Moin, P. 1987 Application of the proper orthogonal decomposition to solutions of the forced Burgers' equation. Bull. of the Am. Physical Soc., Vol. 32, 10, 2070.

Rogers, M. M. and Moin, P. 1987 Helicity fluctuations in incompressible turbulent flows. Bull. of the Am. Physical Soc., Vol. 32, 10, 2080.

Adrian, R.J., Chambers, D., Moin, P., Stewart, D. and Sung, H. 1987 Structure of Karhunen-Loeve expansion for randomly forced Burgers equation. Bull. of the Am. Physical Soc., Vol. 32, 10, 2099.

Piomelli, U., Moin, P. and Ferziger, J. 1987 Wall layer models for large eddy simulations. Bull. of the Am. Physical Soc., Vol. 32, 10, 2041.

Lee, M.J., Colonius, T., and Moin, P. 1988 The vorticity field associated with the streaks in turbulent shear flows. Bull. of the Am. Physical Soc., Vol. 33, 10, 2229.

Pauley, L.L., Moin, P. and Reynolds, W.C. 1988 A study of unsteady laminar boundary layer separation. Bull. of the Am. Physical Soc., Vol. 33, 10, 2250.

Colonius, T., Moin, P., Lele, S., Rogers, M., and Moser, R. 1988 Determination of vortical structures in turbulent shear flows using the eigen structure of the deformation tensor. Bull. of the Am. Physical Soc., Vol. 33, 10, 2254.

Moser, R., Colonius, T., Hussain, A.K.M.F., Rogers, M., and Moin, P. 1988 Vorticity dynamics in a time-developing mixing layer. Bull. of the Am. Physical Soc., Vol. 33, 10, 2255.

Moin, P. and Colonius, T. 1988 Vorticity equation and organized structures in turbulent channel flow. Bull. of the Am. Physical Soc., Vol. 33, 10, 2287.

Choi, H., Kim, J. and Moin, P. 1989 Active turbulence control in wall-bounded flows. Bull. of the Am. Physical Soc., Vol. 34, 10, 2270.

Colonius, T., Lele, S.K. and Moin, P. 1989 Acoustic scattering by a compressible vortex. Bull. of the Am. Physical Soc., Vol. 34, 10, 2318.

Jimenez, J. and Moin, P. 1989 A minimal turbulent channel. Bull. of the Am. Physical Soc., Vol. 34, 10, 2280.

Lee, S., Lele, S.K. and Moin, P. 1989 Simulation of spatially evolving turbulence. Bull. of the Am. Physical Soc., Vol. 34, 10, 2252.

Le, H., Moin, P. and Kim, J. 1990 Direct numerical simulation of turbulence over a backward-facing step. Bull. of the Am. Physical Soc., Vol. 35, 10, 2228.

Choi, H., Moin, P. and Kim, J. 1990 Turbulence control for drag reduction in wall-bounded flows. Bull. of the Am. Physical Soc., Vol. 35, 10, 2232.

Sendstad, O. and Moin, P. 1990 On the mechanics of 3D turbulent boundary layers. Bull. of the Am. Physical Soc., Vol. 35, 10, 2266.

Piomelli, U., Moin, P., Cabot, W.H. and Germano, M. 1990 A dynamic subgrid scale stress model for large-eddy simulations. Bull. of the Am. Physical Soc., Vol. 35, 10, 2268.

Cabot, W.H., Piomelli, U., Moin, P., Lee, M.J. and Lee, S. 1990 Subgrid-scale backscatter in turbulent flows. Bull. of the Am. Physical Soc., Vol. 35, 10, 2281.

Moin, P., Lee, S. and Lele, S.K. 1990 Eddy shocklets in decaying compressible turbulence. Bull. of the Am. Physical Soc., Vol. 35, 10, 2315.

Lee, S., Lele, S.K. and Moin, P. 1990 Interaction of turbulence with a shock wave. Bull. of the Am. Physical Soc., Vol. 35, 10, 2315.

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Member, ASME Heat Transfer Division Committee K-12 on Aeronautics and Astronautics Heat Transfer, 1971-1974

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General Chairman, Second AIAA/ASME Thermophysics and Heat Transfer Conference, Palo Alto, California, 1978

Technical Program Chairman, 18th ASME/AIChE National Heat Transfer Conference, San Diego, California 1979

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Technical Editor ASME Journal of Heat Transfer, 1981-1987

Member, Organizing Committee of the 5th NATO Advanced Study Institute on Natural Convection: Fundamental and Applications, Izmir, Turkey, 1984

Member, Editorial Advisory Board, Annual Review in Numerical Fluid Mechanics and Heat Transfer, 19867-1989

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Member, Advisory Subcommittee, Mechanical Engineering and Applied Mechanics Division, National Science Foundation, 1983-1985.
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Member, The Scientific Council of the International Centre of Heat and Mass Transfer, 1986-

Member, Editorial Advisory Board, AIAA Journal of Thermophysics and Heat Transfer, 1986-

Editor, Experimental Heat Transfer, 1987-1990

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Member, National Research Council's Associateship Program, Panel on Engineering, Applied Science and Mathematics, 1987-

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CONSULTING ACTIVITIES (Last Five Years:)

Argonne National Laboratory, Energy and Environmental Systems Division, 1977-present

Ball Corporation, 1981-1988

Brookhaven National Laboratory, Nuclear Energy Division, 1986-1987

Burns and Roe Service Corporation, 1989-present

Eclipse Inc., Eclipse Combustion, 1989-present

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HONORS AND AWARDS (Partial Listing):

Senior U.S. Scientist Award, Alexander von Humboldt Foundation, Bonn, Federal Republic of Germany, 1975

Fellow, American Society of Mechanical Engineers, 1975

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AIAA Thermophysics Award, American Institute of Aeronautics and Astronautics, 1979

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K. Sasaguchi and R. Viskanta, "Phase Change Heat Transfer During Melting and Resolidification of Melt Around Cylindrical Heat Source(s)/Sink(s)," J. Energy Resources Eng., 111, 43-49 (1989).

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C. Beckermann and R. Viskanta, "An Experimental Study of Solidification of Binary Mixture and Double-Diffusive Convection in the Liquid," Chem.Eng.Commun. 85, 135-156 (1989).

S. Chellaiah and R. Viskanta, "Melting of Ice-Aluminum Balls System," Exp. Thermal Fluid Sci. 3, 179-192 (1990).

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