INTERNATIONAL NUCLEAR ENERGY RESEARCH INITIATIVE

In-Vessel Retention Technology Development and Use for Advanced PWR Designs in the USA and Korea

Principal Investigator (U.S): T.G. Theofanous
University of California, Santa Barbara

Project Number: : 2002-022-K (I)

Project Start Date: : January 15, 2002

Collaborators: S.J. Oh (KHNP, Korea)
J.H. Scobel (Westinghouse, US)

Project End Date: January 15, 2004

Reporting Period: Jan 15 2002 - Jan 15, 2004

Research Objective:

In-Vessel Retention (IVR) of molten core debris by means of external reactor vessel flooding is a cornerstone of severe accident management for Westinghouse's AP600 (advanced passive light water reactor) design. The case for its effectiveness (made in previous work by the PI) has been thoroughly documented, reviewed as a part of the licensing certification, and accepted by the US Nuclear Regulatory Commission. A successful IVR would terminate a severe accident, passively, with the core in a stable, coolable configuration (within the lower head), thus avoiding the largely uncertain accident evolution with the molten debris on the containment floor. This passive plant design has been upgraded by Westinghouse to the AP1000, a 1000 MWe plant very similar to the AP600. The severe accident management approach is very similar too, including In-Vessel Retention as the cornerstone feature, and initial evaluations indicated that this would be feasible at the higher power as well. A similar strategy is adopted in Korea for the APR1400 plant.

The overall goal of this project is to provide experimental data and develop the necessary basic understanding so as to allow the robust extension of the AP600 In-Vessel Retention strategy for severe accident management to higher power reactors, and in particular, to the AP1000 advanced passive design.

The project is organized in terms of two base-technology tasks, and two implementation tasks. The purpose of the base-technology tasks, carried out by UCSB, is to improve the thermal margins; that is the difference between the thermal loading (by the melt natural convection) on the inside and the limits of coolability (due to boiling crisis) on the outside. The purpose of the two implementation tasks is to apply the results of the base-technology tasks, together with timing, as deduced from core degradation/meltdown considerations, to assess IVR performance for the AP1000 (Westinghouse) and the APR1400 (KHNP).

Research Accomplishment:

UCSB is in charge of Base-Technology tasks and the overall coordination of the project, while Westinghouse and KHNP are in charge of Implementation tasks for the AP1000 and APR1400 respectively.

On the Base-Technology tasks, UCSB had been occupied principally with the Coolability (Critical Heat Flux, or CHF) issue, and in particular searching for mechanisms of CHF enhancement under IVR prototypic conditions. The Thermal Loading (Natural Convection Heat Transfer) issue was also addressed in the work in the project's second year.

On the Coolability task, testing in the modified (for AP1000 geometric features) full-scale ULPU-2400 Configuration V facility (Figure 1), and in the small-scale BETA facility, produced a number of important results that were used by Westinghouse to support their IVR case in the AP1000 certification by the US Nuclear Regulatory Commission. These included:

- We demonstrated that coatings normally-used (and left on) for protection during shipping of the Reactor Pressure Vessel (RPV), can degrade burnout performance and recommended that they be removed. This was accepted by Westinghouse and it defines the base material for the burnout experiments as RPV grade steel.
- We defined a thermal insulation design that enhances (via streamlining the flow) the limits to coolability. The streamlined geometry such as employed in ULPU Configuration V is shown to significantly increase the coolability margins previously defined in ULPU ($\sim 1.5 \text{ MW/m}^2$) in connection with the AP600 certification. In the upper region, a critical heat flux of 1.8 to 2.0 MW/m^2 (an increase by $\sim 20\%$) is an appropriate estimate of the expected AP1000 performance (including plant-specific inlet and exit geometries). (Figure 2).
- We obtained important basic understanding on the relative role of hydrodynamics and heater surface nanostructure (including the effects of water chemistry). Thus we are able to connect the ULPU (copper material) performance to reactor (RPV grade steel) and bolsters the reliability of the empirical results from ULPU. In particular, we found that a totally clean (de-ionized) coolant is able to promote such a degree of "cleansing" of the heater surface as to have a rather significant deleterious effect on the CHF. At the other extreme, the presence of TSP (tri-sodium-phosphate), a typical dissolved substance in reactor cavity water (an alkaline solution) has an outstanding beneficial effect on CHF. The presence of Boric Acid, also a normal ingredient here, diminished somewhat this enhancement,

Figure 1. AP1000-Related ULPU-2400 Configuration V Facility

but still the CHF is considerably higher than that with pure water. We use the term “aging” to describe the aggregate of these not-yet-fully-understood, molecular scale phenomena. An aged copper surface such as that employed in ULPU exhibits a similar coolability performance as the bare external surface of the RPV steel.

- We defined boiling/ condensation loads for use by Westinghouse in the structural design of the reflecting thermal insulation. Under representative AP1000 exit geometry at the RPV nozzle gallery, the natural circulation flow is dominantly subcooled, it is modulated by periodic flashing and sweepout events, and associated pressure loss phenomena, at the exit.

The main deliverable on this Coolability work, Report CRSS-03/06, was produced on schedule, and delivered to the US NRC in Westinghouse’s response to the Draft SER on AP1000.

Figure 2. Limits of Coolability in AP1000-Related ULPU-V.

At the basic understanding level, our discovery of the enhancement of resistance to burnout via coolant chemistry is far-reaching. We call this phenomenon *Molecular Wicking*. While not observed previously, these trends are consistent with the basic origin of burnout, as is developing from our understanding in a co-current NASA-funded work carried out in BETA with nanofilm heaters, under controls much stricter than is possible with a large-scale facility such as ULPU.

On the Thermal Loading task, Computational Fluid Dynamics (CFD) simulations of the natural convection startup process in a fluid layer upon a sudden application of a thermal boundary condition were performed. Numerical results revealed a regular pattern of instability that evolves into natural convection loops. Most interestingly, natural convection experiments, BETA-NC, were performed on nanofilm heaters that allows transient pattern identification through IR thermometry. In the BETA-NC experiments, natural convection develops in response to the application of a constant heat-flux boundary condition on the fluid-layer’s bottom surface. Temperature of the bottom surface is measured by a high-speed, high-resolution infrared camera (Figure 3). Results of the 2D and 3D numerical simulations were compared with first-of-a-kind thermal patterns of natural convection

Figure 3. Infrared Thermometry of Nano-Film Heater under Natural Convection.

heat transfer from the infrared thermometry. The BETA-NC data are essential in evaluating capability of existing numerical schemes to capture such pattern formation in natural convection. The numerical simulation capability was also applied to examine natural convection heat transfer in metal-rich spherical-segment fluid layer stratified in the bottom layer of the pressure vessel lower head. The effects of Prandtl number (metallic layer separated at the bottom) and segment geometry (as compared to a fluid layer at the top of the oxidic pool) were delineated.

On the Implementation task, Westinghouse produced a document on the assessment of in-vessel retention for the AP1000 design that synthesized information submitted to US NRC during the license application review. This includes core degradation, melt relocation to the reactor vessel lower head and formation of melt pool in it, assessment of thermal loading, and thermal margin for IVR in AP1000. Most notably, the analysis presents bounding cases for challenges to the vessel integrity from two potential failure modes postulated from phenomena associated with lower plenum material interactions.

The first potential failure mode is from high heat fluxes generated by a bottom metal layer with a significant fraction of the fission products partitioned into the metal. The lower bound critical heat flux at the bottom point of the reactor vessel lower head for the AP1000 geometry is greater than 800 kW/m^2 . The upper bound of the heat flux from the bottom metal layer to the vessel wall is predicted to be less than 500 kW/m^2 . Therefore, the heat flux from a bottom metal pool is not expected to exceed the critical heat flux at the bottom of the reactor vessel lower head. The ratio of the maximum heat flux to the minimum critical heat flux, q/q_{CHF} is 0.625. The second potential failure mode is from a thinned metal layer producing high heat fluxes via the focusing effect at the top. For a bounding case of top metal layer thinning due to material interactions, the peak heat flux to the RPV wall is 1720 kW/m^2 . The lower bound critical heat flux in this region (angular position $\sim 90^\circ$) is 1890 kW/m^2 . The q/q_{CHF} is 0.91, demonstrating that margin-to-failure is maintained for this bounding case. The assumptions in this analysis are conservative with respect to increasing the heat loading to the vessel.

The main effort by KHNP was the construction of a natural circulation loop based on APR1400 design and conducting the tests. Both air-water and steam-water tests were conducted. Oscillatory flow behavior was observed on some steam-water test runs. The regime map depicting the oscillatory flow was derived. In collaboration with KHNP, researchers at KAIST performed forced flow CHF tests and developed CHF correlations. To perform an integrated IVR performance evaluation for APR1400, four representative scenarios for APR1400 were examined using MAAP. Evaluation was also made using SCDAP code. The integrated IVR performance was evaluated for APR1400 in support of the implementation of in-vessel retention in APR1400 plant.

Planned Activities:

All the tasks of the project were completed in January 2004.

Publications of Project Related Research Results:

1. T.N. Dinh, J.P. Tu, T. Salmassi, T.G. Theofanous, "Limits of Coolability in AP1000-Related ULPU-2400 Configuration V Facility", ANS Proceedings of the 10th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, NURETH-10, Seoul, Korea, October 5-9, 2003
2. J.P., Tu, T.N. Dinh, T.G. Theofanous, "Enhancing Resistance to Burnout via Coolant Chemistry", ANS Proceedings of the 10th Intern. Topical Meeting on Nuclear Reactor Thermal Hydraulics, Seoul, Korea, October 5-9, 2003.
3. T.N. Dinh, R.R. Nourgaliev, and T.G. Theofanous, 2003. "Understanding the ill-Posed Two-fluid Model", ANS Proceedings of the 10th Intern. Topical Meeting on Nuclear Reactor Thermal Hydraulics, Seoul, Korea, October 5-9, 2003.
4. B.R. Sehgal, A. Theerthan, A. Giri, A. Karbojian, H.G. Willschutz, O. Kymalainen, S. Vandroux, J.M. Bonnet, J.M. Seiler, K. Ikkonen, R. Sairanen, S. Bhandari, M. Burger, M. Buck, W. Widmann, J. Dienstbier, Z. Techy, P. Kostka, R. Taubner, T. Theofanous, T.N. Dinh, "Assessment of Reactor Vessel Integrity (ARVI)", *International Journal of Nuclear Engineering and Design*, pp 23-53, Vol 221 Iss 1-3, 2003.
5. T.N. Dinh, J.P. Tu, and T.G. Theofanous, "Burnout in High Heat Flux Boiling: The Hydrodynamic and Physico-Chemical Factors", 42st Aerospace Sciences Meeting (Thermophysics), Reno, Nevada, Jan 2004. AIAA-2004-0165.
6. T.N. Dinh, J.P. Tu and T.G. Theofanous, "Hydrodynamic and Physico-Chemical Nature of Burnout in Pool Boiling", International Conference on Multiphase Flow, Yokohama, Japan, May 2004. Paper 296. 14p.
7. J.P. Tu, T.N. Dinh and T.G. Theofanous, "An Experimental Study of Boiling Heat Transfer with Nanoparticles", 6th International Symposium on Heat Transfer, Beijing, June 2004.
8. R.R.Nourgaliev, T.N.Dinh, and T.G.Theofanous, 2004, "A pseudo-compressibility method for the numerical simulation of incompressible multifluid flows", *International Journal of Multiphase Flow*.
9. R.R. Nourgaliev, T.N. Dinh and T.G. Theofanous, 2003, "A characteristics-based approach to the numerical solution of the two-fluid model". Proceedings of FEDSM03, FEDSM-2003-45551, ASME/JSME Joint Fluids Engineering Conference, July 6-11, Honolulu, Hawaii, 2003.
10. T.N. Dinh, J.P. Tu and T.G. Theofanous, "Two-Phase Natural Circulation Flow in AP1000 In-Vessel Retention-Related ULPU-V Facility Experiments", 2004 International Congress on Advances in Nuclear Power Plants, Pittsburgh, PA, June 13-17, 2004.
11. T.N. Dinh, Y.Z. Yang, J.P. Tu, R.R. Nourgaliev and T.G. Theofanous, "Rayleigh-Bénard Natural Convection Heat Transfer: Pattern Formation, Complexity and Predictability", 2004 International Congress on Advances in Nuclear Power Plants, Pittsburgh, PA, June 13-17, 2004.

US/ROK I-NERI Projects: 2002-022-K(1)
In-Vessel Retention Investigations (PI: T.G. Theofanous, UC-Santa Barbara)
Final Report

MONTH	PLANNED (k)	ACTUAL (k)	Cumulative planned	Cumulative Actual
Jan-02	26.5	2	26.5	2
Feb-02	26.5	57	53	59
Mar-02	26.5	39	79.5	98
Apr-02	26.5	32	106	130
May-02	26.5	38	132.5	168
Jun-02	26.5	24	159	192
Jul-02	26.5	34	185.5	226
Aug-02	26.5	47	212	273
Sep-02	26.5	7	238.5	280
Oct-02	26.5	15	265	295
Nov-02	26.5	18	291.5	313
Dec-02	26.5	9	318	322
Jan-03	22.46	9	340.46	331
Feb-03	22.46	21	362.92	352
Mar-03	22.46	12	385.38	364
Apr-03	22.46	12	407.84	376
May-03	22.46	16	430.3	392
Jun-03	22.46	13	452.76	405
Jul-03	22.46	22	475.22	427
Aug-03	22.46	29	497.68	456
Sep-03	22.46	48	520.14	504
Oct-03	22.46	38	542.6	542
Nov-03	22.46	14	565.06	556
Dec-03	22.46	52	587.52	608
Jan-04	22.48	2	610	610
Total	\$ 610.00	\$ 610.00

I-NERI FINAL REPORT

Project Number and Title: Project 2002-022-K(1) . In-vessel Retention Technology Development and Use for Advanced PWR Designs in the USA and Korea

Lead US Investigating Organization: University of California, Santa Barbara (UCSB)
US Principal Investigator: T.G. Theofanous

Lead Collaborating Organization: Korea Hydro & Nuclear Power Co. (KHNP)
Lead Collaborating Principal Investigator: S.H. Oh

Other Collaborating Organization: Westinghouse Electric Company (WES)
Collaborating Investigator: J. H. Scobel

Reporting Period: January 15, 2002- January 15, 2004

Research and Development Area: Advanced Light Water Reactor Technology

Project Status Summary: Completed (see Executive Summary provided on separate pages)

Project organization:

OVERALL ORGANIZATION CHART

* - Special Plant-Specific Natural Circulation tests requested by KINS

** - Accommodation of AP1000 Design Specifics and Changes in ULPU Test Facility

Narrative:

This project aims to develop a basic understanding and create a foundation that allows assessment of the applicability of the In-Vessel Retention (IVR) concept to high-power reactors. The IVR technology was first developed by the PI of this project for Finland's Loviisa VVER-440 plant (1989-1993) and approved by the Finnish Radiation and Nuclear Safety Authority (STUK). The IVR technology was then adopted and further developed at UCSB during 1991-1996 as a Severe Accident Management Strategy in Westinghouse Advanced LWR design of AP600, which was certified with US Nuclear Regulatory Commission. In a nutshell, the IVR process involves flooding the reactor cavity in a core meltdown sequence to retain the molten core within the reactor pressure vessel (RPV) boundary (Fig.1).

Figure 1. In-Vessel Retention (IVR) phenomena.

Application of the IVR to higher power reactors is challenged by reduced margins to failure in these reactors as compared to medium-power reactor AP600. Specifically, the margin to failure is determined by the relation between limits to coolability (critical heat flux or CHF) by boiling on the external vessel surface and thermal loading from in-vessel naturally-convecting, nuclear decay-heated corium pool in the RPV lower head. Notably, based on knowledge and technologies developed for AP600, the margin appears small at the upper region, where a focusing effect of molten metal was previously identified (Fig.2).

Figure 2. AP1000 IVR thermal margin (estimates based on AP600 technology)

In the present project, we examine possibilities to enhance the coolability limit. A main hypothesis is that the CHF can be enhanced by streamlining the flow of coolant between the reactor vessel and thermal insulation. The hypothesis is examined under Task A.1.

Task A1: Carry out a comprehensive set of ULPU-2000 experiments to quantify limits to coolability in Configuration IV.

Lead organization: University of California, Santa Barbara

Task status: Completed.

The ULPU facility was designed to study critical heat flux from a full-scale slice of a hemispherical reactor pressure vessel under externally flooding conditions. While in ULPU Configuration III we simulated the reactor vessel reflecting thermal insulation as normally is (for the AP600 design, for example) the purpose of the Configuration IV experimental campaign is to optimize (coolability) performance by streamlining the flow path between the reactor vessel and reflecting insulation (Fig.3).

The specific idea came from scoping tests conducted in preparation for this work. In the present test program the flow path was created by a curved baffle (same curvature as the

vessel) that was hinged at the upper end ($\theta \sim 90^\circ$), while its lower end ($\theta \sim 0^\circ$) was allowed to take various positions at distances of 2½, 5, 7 and 9 inches from the “reactor vessel”.

We have completed a series of 28 determinations of simulated burnout heat fluxes in the ULPU-2000 Configuration IV facility (Table 1). The results confirm preliminary indications taken prior to commencing this work, that streamlining the flow path around the lower head, and enhancing convection is beneficial to performance. In fact, the improvement was so great that in many locations we reached the power capability of the facility without observing burnout. Still, we have sufficient data to observe key trends, and proceed intelligently towards further optimizing the design (see Task A4). As part of this work, we have also made the necessary changes in facility specifications, and we have completed significant part of the facility modifications that allowed us to reach $\sim 2.5 \text{ MW/m}^2$ (ULPU-2400 facility).

Note that in ten cases CHF could not be reached with the power capability of the facility — these not-yet-at-burnout fluxes are marked as “null” in Table 1. These data reveal some interesting trends, which were very helpful in judging our next moves as we sought to further optimize performance through modifications of the flow path (Task A4). The no-circulation data, on the other hand, are found to be generally comparable to the Configuration III results — the CHF values are probably too low to be of interest to AP1000, and will not be pursued further here. In the discussion that follows it is convenient to take up in turn the lower ($0\text{--}30^\circ$), middle ($30\text{--}60^\circ$), and upper ($60\text{--}90^\circ$) regions of the lower head.

Lower Region. Over this whole region the effect of increased convection in Configuration IV was to bring CHF up to the power capability limit of the lower block – 940 kW/m^2 to $\sim 850 \text{ kW/m}^2$, almost linearly, as we move from position 0° to 30° , which is gradually reduced to just a few percent at 30° for the large gap (9”). The smaller gaps, 2½” and 5” would be expected to show continuing enhancement, along the flow path, but the 940 kW/m^2 limit of the facility did not allow us at this time to distinguish trends. Nevertheless, based on these data we can put a clear lower bound of 900 kW/m^2 over this whole lower region, and this is very comfortable for the kinds of thermal loads expected in AP1000.

Middle Region. Here, we see a very significant increase of CHF due to the presence of the Configuration IV baffle as well. In this heat block (30° to 60°), the heat flux limit is 1410 kW/m^2 , and this was reached under the 2½”-gap configuration. We can also see that all other gap dimensions (5”, 7”, 9”) produced essentially the same CHF of $\sim 1200 \text{ kW/m}^2$. Note that in Configuration III the CHF in this region varies from ~ 900 to $\sim 1100 \text{ kW/m}^2$, so enhancement is relatively small. This is because convection in this region, in Configuration III, is already quite strong, so rather small gaps are required to significantly improve upon it.

Upper Region. The behavior in the upper region is characterized well by performance at two positions — 75° to 80° and $\sim 90^\circ$. In the 75° to 80° region, we reached the facility heat flux limit (third block) of 1880 kW/m^2 , and the data show an optimal gap of 5”, for which the limit was reached without observing burnout. Both a larger and a smaller gap produced burnouts at somewhat smaller fluxes — 1786 kW/m^2 for 9”, while with 2½” we could only reach 1786 kW/m^2 because the run was interrupted with a burnout occurring in the downstream position. Clearly with a very small gap, in the upper positions we may be starving the liquid flow, and

this suggests a design where side slots on the insulation (baffle) would allow water to enter along the path as needed by the pressure differential. In the 90° position, on the other hand, we observe a decrease of CHF to $\sim 1600 \text{ kW/m}^2$, which is essentially the Configuration III value. We explained this by an “exit” phenomenon, whereby local convection is affected by the absence of vapor generation beyond the 90° position. If so, it also would be very relevant to the reactor, as this is exactly the situation at the top of the metal layer. In a later task (A4), we overcame this by locally tailoring the flow path, which we could not do in the present series of tests with the baffle hinged at the top, at a fixed distance of 9”. With this local problem alleviated, we can look at this upper region as easily accommodating fluxes of $\sim 2 \text{ MW/m}^2$, which should provide a comfortable margin to the need of AP1000.

Figure 3. ULPU facility configurations III and IV.

Several key conclusions made on the basis of present data can be listed as follows:

- In the upper region (~ 75) we reached the limit of 1880 kW/m^2 without observing burnout. We believe there is a strong basis for expecting coolability to persist above 2 MW/m^2 (Fig.4).
- At the very top of the heated wall (90°) we found a sudden drop of performance to $\sim 1.6 \text{ MW/m}^2$, and this appears to be related to “exit” phenomena that affect (adversely) local two-phase flow structure. As a remedy, we proposed to locally tailor the flow path (see

Task A4).

- In the middle ($30^\circ - 60^\circ$) and lower ($0^\circ - 30^\circ$) regions we can easily exceed 1410 kW/m^2 and 940 kW/m^2 , respectively; that is, CHF levels that, as in the case of AP600, make these regions totally unimportant for IVR performance.

Table 1. Summary of ULPU-2000 runs (Configuration IV)

Run #	Power Name	Peak-Flux Angular Position °	CHF with Natural Convection (kW/m^2)	CHF with Open Loop CHF (kW/m^2)	Burnout Inception Position °	Baffle Position (Inch)
1	U0A	0~13	855		10~16	9
2	U8A	11~15	940		10~16	9
3				695	10~16	9
4			940/Null			$2\frac{1}{2}$
5			940/Null			5
6			940		10~16	7
7	U14A	23~27	940		22~30	9
8				789	19~26	9
9			940/Null			$2\frac{1}{2}$
10				940/Null		$2\frac{1}{2}$
11				705	19~26	5
12	U24A	42~46	1198		43~50	9
13				902	40~46	9
14			1410/Null			$2\frac{1}{2}$
15			1128		40~46	5
16			1198		40~46	7
17	U30A	53~57	1269		50~56	9
18				1057	50~56	9
19			1410/Null			$2\frac{1}{2}$
20	U40B	72~76	1786		67~73	9
21				1316/Null	80~83	9
22				1316/Null	83~86	$2\frac{1}{2}$
23			1786/Null		80	$2\frac{1}{2}$
24			1880/Null			5
25				1222	86~90	5
26	U48A	88~90	1598		83~90	9
27				1052	83~90	9
28		83~90	1598		83~90	9

/Null: means no burnout detected. The number shows maximum power heat flux reached.

It is realized however that accurate representation of geometries is important in determining the coolability limit since it affects parameters of two-phase natural circulation that eventually influences flow regimes in boiling. Also the facility needs to be modified to be able to accommodate, reliably, higher heat fluxes. Figures 5 and 6 show the facility in overall, and new heating blocks to upgrade power to 2400 kW/m^2 .

Figure 4a. ULPU-2000 Configurations IV scoping data – natural circulation data.

Figure 4b. ULPU-2000 Configurations IV scoping data – open loop results.

Figure 5. ULPU-2000 facility – overall view.

Figure 6. ULPU-2400 facility – upgraded power and heating systems.

Task A2: Two-phase flow analysis of ULPU experiments

Lead organization: University of California, Santa Barbara

Task status: Completed.

We use a one-dimensional model of two-phase flow to analyze natural circulation and to determine flow rate in the ULPU. Geometrical factors are represented through hydraulic loss factors. Results of the analysis show that the use of empirical correlations obtained for two-phase flow pressure drops in small tubes may be inadequate to represent flow in large, inclined rectangular channels in ULPU. For example, when experimental values of void fraction are used, an excellent agreement between calculated flow rates and flow rates measured in ULPU experiments in Configuration IV was achieved.

An important point in using ULPU for full-length simulation of two-phase flow under IVR conditions is the ability to adequately represent two-phase flow conditions at a particular angular position in a slice ULPU geometry versus that of hemispherical geometry of the reactor lower head (Fig.7). For this, a power shaping principle developed for AP600 is fully applicable. We use a reference thermal load (Fig. 8) from ACOPO experiments (also at UCSB), and a power profile that allows the flow enthalpy at the tested angular position be matched (Fig. 9). More detailed description of power shaping principle can be found in DOE reports for IVR AP600.

Figure 7. Heat flux shaping principle.

Data of natural circulation flow from ULPU-2400 Configuration V experiments were documented, analyzed and published as

T.N. Dinh, J.P. Tu and T.G. Theofanous, "Two-Phase Natural Circulation Flow in AP1000 In-Vessel Retention-Related ULPU-V Facility Experiments", 2004 International Congress on Advances in Nuclear Power Plants, Pittsburgh, PA, June 13-17, 2004.

Figure 8. Reference thermal load based on ACOPO data and correlation for heat transfer in volumetrically-heated liquid pools.

Figure 9. Example of heat flux shaping (for angle 70-75°).

Task A3: Carry out detailed, multidimensional numerical simulations of the two-phase boundary layer for both Configuration III and Configuration IV geometries, and attempt to understand the differences in CHF performance

Lead organization: University of California, Santa Barbara

Task status: Completed.

This is a complementary task, with a goal is to bring to bear advanced methods in CFD to the analysis of two-phase flow under conditions of interest to IVR. The work resulted in pushing the state of the art. On the one hand, we have developed an advanced solver coupled with a level-set method for front capturing that enables direct numerical simulation of two-fluid flow with a high density ratio.

R.R.Nourgaliev, T.N.Dinh, and T.G.Theofanous, 2004a, "A pseudo-compressibility method for the numerical simulation of incompressible multifluid flows", *International Journal of Multiphase Flow*.

On the other hand, a novel scheme for solving two-fluid equations was developed. Robust, stably convergent solutions were obtained for one-dimensional model of incompressible two-fluid air-water faucet problems. Significant progress in understanding the ill-posed two-fluid model was achieved.

T.N. Dinh, R.R. Nourgaliev, and T.G. Theofanous, 2003. "Understanding the ill-Posed Two-fluid Model", ANS Proceedings of the 10th Topical Meeting on Nuclear Reactor Thermal Hydraulics NURETH-10. 2003.

R.R. Nourgaliev, N Dinh and T. Theofanous, 2003, "A characteristics-based approach to the numerical solution of the two-fluid model". Proceedings of FEDSM03, FEDSM-2003-45551, ASME/JSME Joint Fluids Engineering Conference, July 6-11, Honolulu, Hawaii, 2003.

Task A4: Flowpath optimization experiments

Lead organization: University of California, Santa Barbara

Task status: Completed.

In order to achieve an accurate representation of flowpath in the AP1000 design (Fig. 10), two major modifications were made for the water inlet (Fig.11) and exit from riser (Fig.12)

Significant modifications were made to the flow inlet and exit configuration to represent specifics of AP-1000 design. A new baffle design was made and implemented that allows the flow path be regulated in ULPU-2400 Configuration V tests. This is a crucial step towards establishing flow conditions that enhance the limit of coolability in ULPU-2000.

Figure 10. Representation of AP1000 design geometry.

Figure 11. AP1000 water inlet geometry represented in ULPU-2400 Configuration V.

ULPU Configuration V

Figure 12. AP1000 exit from riser geometry represented in ULPU-2400 Configuration V.

Two series of experiments were conducted on the new Configuration V of the ULPU-2400 facility. 16 runs were made to obtain critical heat fluxes (CHF): 3 runs with a baffle position of 6" and 13 runs with 3" baffle position. The focus of these tests was placed on a higher region where the IVR margin-to-failure was predicted to be relatively small for AP1000 design. The experimental result confirms a trend of CHF enhancement due to flowpath organization first quantitatively reported in ULPU-2000 Configuration IV tests of Task A.1 of this project. Notably, significant enhancement was found in the ULPU-2400 bottom and middle heater blocks. At the higher region, where the flow's vapor quality becomes high, enhancement of the critical heat flux was measured to be less pronounceable. Such effect was previously (see Task A.1) attributed to an "exit" phenomenon (Fig.13). Furthermore, it appears that coolability limits in the upper region is more sensitive to surface aging conditions.

We have examined two-phase flow behaviors that may be responsible for the exit phenomena. We believe that a basic understanding of mechanisms of burnout in the ULPU upper region is critical to enable suggesting a viable strategy for design modification and flowpath and surface optimization. This objective is addressed with Tasks in area C.

Figure 13. ULPU-2400 Configuration V results with natural circulation.

Task B1: Dynamic loading measurements

Lead organization: University of California, Santa Barbara

Task status: Completed.

A scheme for measuring dynamic loadings in ULPU-2400 was devised, that uses a number of pressure transducers to be installed at designed locations to provide information on dynamic system behavior in the ULPU-2400 two-phase natural circulation loop. Measurements of pressures were performed as part of ULPU-2400 experiments.

Data were documented in 500-page report for INERI project partners and for Westinghouse engineers to use in their structural analysis for AP1000 thermal insulation.

T.N. Dinh, J.P. Tu, T. Salmassi, T.G. Theofanous, "Limits of Coolability in AP1000-Related ULPU-2400 Configuration V Facility", *CRSS-Research Report 03/07*, prepared for DOE I-NERI Program, Santa Barbara, 547p., July, 2003.

Task C1: Mini-ULPU coolability visualization tests

Lead organization: University of California, Santa Barbara

Task status: Completed.

The objective of these experiments is to obtain visual images of what happens at the heater surface under IVR conditions. Previously, the focus is placed on the lowermost region where the coolability limit is lowest. Although it is so, the lowermost region was found to not pose a significant threat to the vessel failure under IVR conditions due to reduced heat load from molten corium pool to its lower boundary. In the present project, we therefore focused on visualization of flow patterns in the middle-to-high region where the need to understand boiling coolability limit is most crucial for IVR in AP1000. We have implemented a number of visualization windows into the upper region that allows us to use a laser lighting and a high-speed (up to 20,000 fps) and high-resolution digital video camera to observe two-phase flow phenomena. The windows allow us to see two-phase mixing regions as well as vapor-rich regions next to the heater surface. We obtained information on slug-like flow behaviors, slug characteristics and frequency.

Task C2: Surface coating tests on BETA

Lead organization: University of California, Santa Barbara

Task status: Completed.

Experiments were performed using samples of vessel coated with different paints that were proposed for AP1000. Also two different baking protocols were employed.

Paint name and painting method applied

BT: Brush painted Thermoline 4900 (light gray 0705)
(a replacement of Black Carboline 4674)
SC: Sprayed Carbocoat 818 (white 0800)
ST: Sprayed Thermoline 4900 (light gray 0705)
SCz: Sprayed Carbozinc 11 HS (gray 0700)

Heating protocol

A. Sample baked 8 h in oven at 545°F
B. Sample baked 24 h in oven warmed to and kept at 545°F

Figure 14. Sample painted surfaces tested in BETA for AP1000.

In addition, the heater surfaces were carefully examined using optical microscopy (Fig. 14). Most importantly, the experimental results show that the paint layer may be peeled off, causing early burnout (Fig.15). We found one type of paint that performs well in distill water boiling experiments.

SAMPLES	HEATING PROTOCOL	
	A (8 HOURS)	B (24 HOURS)
BT
SC		
ST		
SCz Sprayed Carbozinc	1400 kW/m²	1360 kW/m²

SCz: 2-3 mil, inorganic (35% zinc) silicate-base coating

Figure 15. BT, SC and ST samples show peeling-off and early burnout.

Results of this Task were documented in a CRSS Technical Report distributed to project partners and DOE.

Task C3: Confirmatory ULPU coolability tests

Lead organization: University of California, Santa Barbara

Task status: Completed.

After experiments were performed on ULPU, it became clear that the potential for CHF enhancement is the most critical aspect for IVR performance. With respect to Burnout, results of ULPU work shows that the presence of a baffle promotes much improved resistance to burnout, but it also introduces some sensitivities that needed better understanding. This is in contrast to the previous (AP600-related) work, where the baffle simulating the thermal insulation was found to have no significant effect. In addition, experiments in BETA, also at UCSB, with chemical additives to match the chemical and pH environment of the containment water, show significant effects which deserved much closer examination.

The scope of confirmatory testing (Task C3) was extended to include optimization and detail testing that quantify the effect of thermal insulation designs and water chemistry.

Task C3-A. *Thermal insulation design optimization experiments on the ULPU-2400 facility and examination of containment water chemistry effects.*

The ULPU-2400 Configuration V facility allows changing the baffle position to represent variants of the RPV thermal insulation design. The current facility also represents the AP1000 reactor geometry, particularly the water inlet into insulation and exit from the raiser.

We performed detailed analysis of experimental results obtained on ULPU-2400 Configuration V facility (Figs 16-17), and produced a Technical Report CRSS-03/06 “Limits of Coolability in AP1000-Related ULPU-2400 Configuration V Facility” on schedule, and delivered it to the US NRC in Westinghouse’s response to the Draft SER on AP1000.

Besides the temperature/power histories, and CHF determination for each run (Figs.18-19), we report natural circulation flow rates, pressures and pressure differentials around the loop, and provide spectral information on these time-series data. The three series of tests addressed flow path dimensions around the lower head (baffle positioning), power shape, and water chemistry respectively.

A total of thirty-six experiments in three test series were carried out and analyzed. Experiments were carried out in three test series, as summarized in Tables 2 and 3. The main distinction is that between series M and C we explore the effect of heater surface properties (especially as affected by water chemistry), while between series C and P we consider sensitivity to power shapes. As expected, the height of the flow loop turned out to be very important. Even at the highest power levels, the bulk flow at the exit of the heated section remained subcooled (it is ~13 K subcooled at the inlet due to the gravity head), and boiling occurred by flashing at some higher elevation, depending on the power level.

Fig.16. ULPU-2400 Configuration V Facility.

Figure 17. AP1000 thermal insulation representation in ULPU-2400 Configuration V. Three baffle positions, with baffle-to-heater distances being 6", 3", and 6"-to-3".

Table 2. Test conditions for runs in ULPU Configuration V - M sub-series.

ID #	Power Shape	$Q_{\max}$ Range °	Pre-CHF Flux (kW/m ²)	CHF (kW/m ²)	CHF Position °	Total Power (kW)	Upstream Input Power (kW)	Flow Rate (m ³ /min)	Baffle Position (Inch)	Sub-series
	<u>T08A</u>	10~16								
1.			1128	1156	15	345	19	0.689	3	M3
	<u>T24A</u>	38~43								
2.			1504	1598	43	271	71	0.625	3	M3
	<u>T40B</u>	66~71								
3.			1562	1672	71	219	121	0.690	6	M6
4.			1699	1782	71	233	129	0.644	3	M3
5.			1836	1919	71	251	139	0.689	6/3	M63
6.			1974	2028	67	266	147	0.713	6/3	M63
	<u>T40D</u>	66~71								
7.			1754	1782	67	170	129	0.589	3	M3X
	<u>T48A</u>	80~84								
8.			1672	1727	83	205	179	0.693	6	M6
9.			1645	1727	83	207	181	0.645	3	M3
10.			1782	1919	83	230	201	0.675	6/3	M63

At low power levels ~100 kW, this flashing was stable, confined within the inclined pipe at the top, and produced flow rates of ~0.5 m³/min. At high power levels (up to ~350 kW) the flashing occurred periodically, by successive penetrations and sweepouts (by single phase liquid) within the upper portion of the riser. The higher voids and associated gravity heads, with increasing power, were balanced by increased acceleration and frictional two-phase pressure losses at the exit (nozzle) section, thus effectively “saturating” the flow rate at only ~0.7 m³/min. Indeed, as seen in Tables 2 and 3, all burnouts occurred within a rather narrow range of flow rates, around 0.6 to 0.7 m³/min.

Table 3. Test conditions for runs in C and P series.

ID #	Power Shape	$Q_{\max}$ Range °	Pre-CHF Flux (kW/m ²)	CHF (kW/m ²)	CHF Position °	Total Power (kW)	Upstream Input Power (kW)	Flow Rate (m ³ /min)	Baffle Position (Inch)	Sub-Series
	<u>T08A</u>	10~16								
11.			916	987	10	309	17	0.654	3	C3
12.			902	930	83	265	265	0.696	6/3	C3
	<u>T24A</u>	38~43								
13.			940	1128	46	191	50	0.625	6/3	C63
	<u>T40B</u>	66~71								
14.			1645	1672	71	219	121	0.608	3	C3
15.			1233	1507	71	198	109	0.556	3	C3

16.			1370	1453	71	190	105	0.649	6/3	C63
17.			1370	1425	71	187	103	0.658	6/3	C63
	<u>T40D</u>	66~71								
18.			1425	1480	67	170	129	0.590	3	P3X
	<u>T40E</u>	66~71								
19.			1288	1343	71	196	117	0.681	3	P3
20.			1343	1398	71	204	122	0.684	3	P3
21.			1233	1370	71	200	120	0.671	3	P3
22.			1233	1370	67	198	120	0.633	6/3	P63
	<u>T40F</u>	66~71								
23.			1316	1370	71	243	162	0.690	3	P3
24.			1233	1316	71	233	156	0.693	6/3	P63
	<u>T48A</u>	80~84								
25.			1238	1507	80	179	156	0.656	3	C3
26.			1370	1480	83	178	155	0.679	3	C3
27.			1233	1370	83	165	144	0.572	3	C3
28.			1370	1507	80	181	158	0.601	6/3	C63
29.			1398	1453	83	175	152	0.633	6/3	C63
30.			1370	1507	83	181	158	0.611	6/3	C63
	<u>T48B</u>	80~84								
31.			1288	1343	83	209	188	0.677	3	P3
32.			1233	1288	80	200	180	0.650	3	P3
33.			1233	1370	83	213	192	0.657	3	P3
34.			1233	1370	83	213	192	0.666	6/3	P63
	<u>T48C</u>	80~84								
35.			1261	1288	80	238	228	0.692	3	P3
36.			1233	1370	83	253	242	0.685	6/3	P63

Auxiliary data on water chemistry were obtained in the basics-oriented (upwards-facing flat plate geometry) BETA experiment. The CHF data and understanding derived thereof robustly enhance the coolability limits found in the AP600 geometry.

The main conclusions found are

- A streamlined geometry such as employed in ULPU Configurations IV and V was shown to significantly increase the previously defined coolability margins in connection with the AP600 certification. Moreover, the deeper, more specific to AP1000 study in Configuration V revealed additional coolability margins relative to Configuration IV results (especially eliminating the there-found “exit phenomenon”).
- Under representative AP1000 exit geometry (at the RPV nozzle gallery), the natural circulation flow is dominantly subcooled, and is modulated by periodic flashing, and frictional phenomena at the exit. We recommend a closer examination of the effect of

exit flow resistance on these phenomena, the natural circulation flow rate, and on the CHF performance.

- Heated surface chemistry and nanomorphology, and coolant chemistry and impurities can have a profound influence on cooling performance, further enhancing this understanding from previous ULPU tests. We use the term “aging” to describe the aggregate of these not-yet-fully-understood, molecular scale phenomena. An aged copper surface such as that employed in ULPU exhibits a similar coolability performance as the bare external surface of the RPV steel. We recommend that peelable protective coating be used in shipment, and that it be removed on installation.
- At normal aging conditions as generally relevant in technical large scale systems, and waters containing microscopic impurities and dissolved substances, the AP1000-related coolability limits at the upper region ($66^\circ < \theta < 90^\circ$) of the lower head is in the 1.8 to 2.0 MW/m² range. The normally found in containment waters TSP can deliver outstanding further enhancement in performance, but the effect can be interfered by the presence of other substances and this remains unquantified at this time. We recommend a closer examination of this, in conjunction with reducing the exit flow resistance in ULPU as recommended above.
- The main energy content of the dynamic loads generated is due to cyclic boiling-condensing phenomena in the vicinity of the lower head, and flashing-sweeping phenomena at the upper end. Amplitudes and spectra provided in this report can be used in the structural design of the thermal insulation that is to form the flow path.

Figure 18. Critical Heat Fluxes (CHF) at the top ($66^\circ < \theta < 90^\circ$) in the three series of tests conducted in ULPU-V. The shaded region shows CHF range obtained in ULPU-III for AP600.

Figure 19. Critical Heat Fluxes measured in ULPU Configuration *V* experiments.

Sub-Task C3-B: The findings from the ULPU and BETA experiments at UCSB indicated the potential importance of vessel surface treatment, and containment water chemistry on the burnout performance. This has also influenced the AP1000 design. The objective of the extended C.3 task was to quantify the effect of water chemistry on burnout.

Of particular interest are Boric Acid (BA) and Tri-Sodium Phosphate (TSP), both chemicals naturally occurring in containment water during a postulated severe accident. Within the scope of the present paper is also to relate the performance of the copper material used in the ULPU facility, a large scale experiment intended to simulate burnout on a lower head, and actual steel material used in Reactor Pressure Vessel (RPV) manufacture. Specifically, the present work aims to address the following questions, two of which are of practical importance and a third one which is of fundamental character:

- (i) Whether and to what extent the copper block heaters in ULPU provide an adequate simulation of the reactor pressure vessel steel;
- (ii) Whether and to what extent containment water chemistry can alter the limits of coolability deduced from experiments using chemicals-free water; and
- (iii) Whether and to what extent the basic mechanism of contact line instability can be further elucidated through measurable effects of coolant chemistry.

Typical concentrations in containment waters are 2,500 to 3,000ppm and 3,500 to 6,000ppm for BA and TSP respectively. We examine the effect separately and in combination over concentration ranges that include the above stated ones. In addition, we experiment with pure De-Ionized Water (DIW), resistivity less than 1MΩ, and Tap Water (TW) with total

dissolved hardness (mainly Ca and Mg salts) of 333ppm (quality tests result reported by Goleta Water District). For heater material we use copper, stainless steel (304), and (carbon) steel material. The steel material is reactor (RPV) grade. All experiments were conducted in the BETA facilities made available for this work from a related NASA grant (Theofanous et al, 2002 a, b)¹⁻². It should be noted that in preparation of this work we performed also tests with various protective coatings normally used for reactor vessels. We found them to degrade performance, and on our recommendation it was decided that the In-Vessel Retention consideration for AP1000 will be based on bare steel material.

With respect to the BETA tests, a total of 27 CHF determinations in 8 series of runs were made on BETA facility for pool boiling. Each series starts with a fresh (clean) sand-blasted surface. Within each series the surface condition is as delivered from the previous run, after cooldown, and rinsing of the whole test section with deionized water (DIW).

(a) Effect of surface material and aging by exposure to air and to boiling. This pertains to all runs made with pure DIW (Figs 20-21). In several tests, we can see that with increasing aging the CHF increases from a low value of $\sim 750 \text{ kW/m}^2$ to over $\sim 1,300 \text{ kW/m}^2$. In one test we see that surface “pre-treatment” by boiling boric acid (BA) water to burnout yields a still further increase to over $\sim 1,500 \text{ kW/m}^2$. Finally we see that fresh steel is as good as well aged copper, and that further aging of steel produces rather modest improvement in CHF. All these results are very consistent with our recent basics-oriented BETA tests with nanofilm heaters and infrared thermography.

Figure 20. Aging by boiling in DIW

Figure 21 Cleansing of aged surface by boiling in DIW (replaced after each run)

¹ T.G. Theofanous, J.P. Tu, A.T. Dinh and T.N. Dinh, Aug. 2002, “The Boiling Crisis Phenomenon – Part 1: Nucleation and Nucleate Boiling Heat Transfer”, *J. Experimental Thermal Fluid Science*, pp.775-792, V.26 (6-7).

² T.G. Theofanous, T.N. Dinh, J.P. Tu, and A.T. Dinh, Aug.2002, “The Boiling Crisis Phenomenon – Part 2: Dryout Dynamics and Burnout”, *J. Experimental Thermal and Fluid Science*, pp.793-810, V.26 (6-7).

Figure 22. Effect of coolant chemistry on burnout on steel

Figure 23. Effect of coolant chemistry on burnout on copper

(b) Effect of alkaline tri-sodium-phosphate (TSP) water and acidic BA water (Figs.22-23). The TSP produced the highest CHF values yet, at $\sim 1,800$ to $1,900 \text{ kW/m}^2$, and this is consistent in both copper and steel surfaces. This is an important new result of basic as well practical significance. In BA water on the other hand the CHF exceeds that seen with DI water on aged heaters, reaching up to $\sim 1500 \text{ kW/m}^2$, again consistently with both copper and steel surfaces, fresh as well as aged. Diluting the TSP water seems to diminish the performance in steel but not in copper, but the data in this area are too few for definitive conclusion at this time.

(c) As secondary effect we note that boiling tap water in effect produces instant aging of copper surfaces, while it has no effect on the already aged steel surfaces. Also, from one run there is a hint that BA may interfere with the beneficial effect of TSP.

The overarching conclusion of these results is that TSP is very beneficial to CHF performance, while other chemicals found in tap water, and BA act mainly to age a copper surface to levels similar to those found naturally on a clean vessel steel surface. Also there is indication that these chemicals, acting principally through deposition processes, may interfere with the beneficial effect of TSP.

Furthermore, the BETA tests provide a principal connection between the aged copper heaters in ULPU and the prototypic steel material relevant to the AP1000 IVR. While established under pool boiling conditions, this connection is very tight in general, as both sets of data indicate the dominant influence of surface material aging, under both pool and flow boiling conditions, and convergence of CHF trends with degree of aging.

- (i) Reactor vessel steel provides a consistent burnout performance that is superior to copper and this performance is further enhanced in the presence of TSP and BA in the coolant.

- (ii) The presence of chemicals TSP and BA, as expected in containment waters, enhances the coolability of fresh surfaces well beyond even those that have been fully-aged. The so-enhanced fluxes, reaching up to $\sim 1.9 \text{ MW/m}^2$ motivate examination of such effects in large-scale, convective-boiling situations (ULPU, RPV).
- (iii) Recognizing that the effect of coolant chemistry can only be through molecular scale liquid-solid interactions at the contact line, this new evidence further supports the idea (Theofanous et al, 2002b) that the basic mechanism of burnout is microlayer rupture by contact line instability.

Task D1. Natural convection experiments in a hemisphere.

Lead organization: University of California, Santa Barbara

Task status. Task was removed from plan in exchange for extended C3 Task and E1 Task.

In a recent work on melt pool convection heat transfer, we were able to reconcile differences in high-Rayleigh-number natural convection heat transfer data. Specifically, we found that the differences reported in the literature are due to secondary currents found in large aspect ratio geometries (height/width > 1). The test section in the ACOPO facility is hemispherical and has the aspect ratio of $1/2$, as in the prototypic reactor geometry. Thus the issue of data discrepancy and applicability was resolved in ACOPO's favor, and we are quite satisfied that the ACOPO correlation would yield correct predictions for all reactor conditions of interest. In addition, recent work in an EU project that we participated shows that the lateral heat transport within the metal layer is negligible. This result confirms the validity of the heat transfer model we developed and applied in our assessment of in-vessel retention for AP600 (under DOE ARSAP program).

In our original proposal, we stated that the Raleigh numbers relevant to large reactors (AP1000, APR1400) are at the “bare edge” of the ACOPO range, and wished to extend the range by one order of magnitude. We also wanted to understand better the peaking in heat flux near the top of the oxide layer, and also in the metal layer. Relative to the former issue of some concern was the apparent scatter of upward heat flux data from various, smaller, experiments. For example, subsequent to the publication of the ACOPO data, a paper appeared in Nature reporting data for Raleigh-Bernard convection over a similar range of Raleigh numbers, which could be correlated by

$$\text{Nu} = 0.124 \text{ Ra}^{0.309} \quad (\text{D1.1})$$

as compared to

$$\text{Nu} = 0.206 \text{ Ra}^{0.303} \text{ Pr}^{0.084} \quad (\text{D1.2})$$

(for water $\text{Pr} = 7$, $\text{Nu} = 0.242 \text{ Ra}^{0.303}$) obtained from ACOPO. Also the COPO data in Finland showed a two-fold increase in upward Nusselt number.

In the year 2002 this situation has been modified as follows:

(a) In work that we carried out for the US NRC we were able to reconcile differences such as Eqs (D1.1) and (D1.2). These are due to secondary currents found in large aspect ratio geometries (height/width > 1). In the cylindrical test section that produced Eq.(D1.1) the aspect ratio was two, and in COPO (which is a thin slice geometry) the aspect ratio is 12. In ACOPO on the other hand the aspect ratio is $1/2$, as in the prototypic reactor geometry. Thus, the issue was resolved in ACOPO's favor, and we are quite satisfied that the ACOPO correlation would yield correct predictions for all reactor conditions of interest.

By the way in this recent work we also addressed theoretically the issue of peaking in the oxide layer (item i on p.10 of our proposal), and we are comfortable that the smaller, lower Raleigh number results are consequence of exactly these, non-prototypic features.

(b) Work carried out in France and in a EU project, in which we participated, shows that the lateral heat transport within the metal layer is negligible. This confirms the validity of the heat transfer model we developed and applied in our assessment of in-vessel retention for AP600 (under DOE ARSAP program).

(c) In our ULPU work for this project we found that the presence of a baffle promotes much improved resistance to burnout, but it also introduces some sensitivities that need better understanding. This is compared to the previous (AP600-related) work, where the baffle simulating the thermal insulation was found to have no significant effect. In addition, our current experiments in BETA, with chemical additives to match the chemical and pH environment of the containment water, show significant effects which deserve much closer examination.

(d) The stratification phenomenon found in small scale experiments carried out in Russia, while uncertain whether it is relevant to reactor scales, has attracted considerable interest. Accordingly, we believe some additional attention would be in order.

With respect to Burnout, in our ULPU work for this I-NERI project we found that the presence of a baffle promotes much improved resistance to burnout, but it also introduces some sensitivities that need better understanding. This is compared to the previous (AP600-related) work, where the baffle simulating the thermal insulation was found to have no significant effect. In addition, our current experiments in BETA, with chemical additives to match the chemical and pH environment of the containment water, show significant effects which deserve much closer examination.

In fact, we knew from the outset that the real "bear" to wrestle down is Burnout. As stated in our proposal there is little one can do about the thermal loads, and the refined definition of them was characterized (p.2) as an "auxiliary objective".

This means that our efforts in the second year of the I-NERI project could be better directed to addressing the stratified metallic pool, rather than refining the ACOPO correlations as

originally planned. So, we proposed to substitute the ACOPO experiments with direct numerical simulations of the metallic pool. This freed up some resources which we used for further experiments in the full-scale ULPU-2400 facility, to enlarge the CHF data base, and investigating more closely exit geometry effects, baffle positions, dynamic loads, and most importantly, effects of containment water chemistry on burnout performance.

Task D2. BETA Natural Convection Tests

Lead organization: University of California, Santa Barbara

Task status: Completed.

The objective of this task is to carry out visualization of the turbulent natural convection boundary layer, using infrared mapping of a heat generating nanofilm in the BETA experiment. Use these data to benchmark the DNS simulation to be conducted under Task E.1.

It is noted that while many experiments were conducted, and correlations are available for natural convection heat transfer, the physics of pattern formation is far from understood. In particular, Rayleigh-Benard convection has long been a paradigm for the study of complexity and remains a challenge for numerical simulations. The objective of the present study, as a task in the INERI project, is to take advantage of a novel experimental approach that the principal investigator and his team at UCSB have developed for NASA in a study of pool boiling, to investigate heat transfer at the onset of Rayleigh-Benard natural convection. The key idea is to characterize formations of thermal patterns in response to a power step. Namely, we use state-of-the-art, high-speed, high-resolution infrared camera to obtain instantaneous thermal patterns on a 140nm-thin Titanium film vapor-deposited on 130 μm glass substrate. The Titanium film is heated by alternative electric currents to provide a uniform heat flux. An extensive experimental program was conducted, and the thermal images obtained are documented and analyzed.

In this task we designed and constructed a natural convection cell for infrared-assisted experimentation. The test facility is named BETA-NC. The pulse heating approach was adopted. The test results showed that Titanium nano-film heaters on glass substrate are able to respond instantaneously to power surge. More importantly, we have conducted a series of experimental runs on BETA-NC to characterize thermal patterns on the heater surface upon the onset of natural convection in response to the application of constant heat flux on the bottom boundary of a fluid layer (Figs.24-26). Experimental data, including the thermal images of the bottom boundary surface are obtained and documented. The data, especially the pattern information, can be very useful for the development and verification of mathematical models and numerical schemes for Rayleigh-Benard natural convection as well natural convection in volumetrically heated liquid pools.

Figure 24. Temperature history at different locations in BETA-NC Ti46H14 experiment (33 kW/m^2 , 23 mm water height).

Figure 25. Temperature patterns at different time moments.

Figure 26. Temperature pattern and local temperature.

Task E1. Numerical simulation of natural convection flow

Lead organization: University of California, Santa Barbara

Task status: Completed.

In this task we examined the scale and Rayleigh/Prandtl number effects on natural convection heat transfer. Numerical results were compared to those of Task D.2, as well as other available data.

A critical analysis of past works as well as a review of recent advances in turbulence modeling and simulation was made, which revealed that previously intuitive, empirical selection of numerical schemes for use in “no model” LES model of turbulent natural

convection in internally-heated liquid pools has a root in Monotonically Integrated LES (or MILES) method. The MILES approach has recently received a substantial body of supporting results as well as theoretical basis. We believe that MILES offers a potential to capture turbulent mixing and heat transfer in high Rayleigh number regimes.

First, performance of different numerical schemes and solvers for Navier-Stokes equations was examined with respect to their ability to provide robust and accurate solutions for the Rayleigh-Benard natural convection in a fluid layer heated from below (Figure 27). The Fluent code is used, and the results confirm our recommendations from previous work about the QUICK scheme.

Figure 27. Temperature field in Rayleigh-Benard natural convection startup (calculation).

Figure 28. Temperature patterns on the BETA-NC heater surface (calculation).

Figure 29. Temperature history at different locations on the BETA-NC heater surface (calculation).

Second, we perform numerical simulations of the natural convection startup process in a fluid layer upon a sudden application of a thermal boundary condition. Numerical results revealed a regular pattern of instability that evolves into natural circulation loops. Most interestingly, we performed simulations of natural convection experiments, BETA-NC, performed at UCSB in the IVR project. In the BETA-NC experiments, natural convection forms in response to the

application of a constant heat-flux boundary condition on the fluid-layer's bottom surface. Temperature of the bottom surface is measured by a high-speed, high-resolution infrared camera (see Figure 24-25 above). Results of the 2D and 3D numerical simulation are compared with the thermal images from the infrared thermometry (Figs.28-29). The results confirm the capability of the CFD numerical scheme to capture pattern formation in natural convection.

Finally, we performed numerical analysis of heat transfer in a spherical-segment fluid layer with internal heat generation. The objective of this exercise was to quantify uncertainty in heat transfer model, used by Westinghouse Electric Company in their assessment of thermal margin of in-vessel retention for AP1000. We examined the Prandtl number effect and the effect of spherical-segment geometry to address uncertainties in the Westinghouse assessment. The analysis aimed to evaluate the importance of Prandtl number and spherical geometry in the assessment of energy splitting of a metal layer.

While natural convection heat transfer has been studied over the past decades by many authors, yet theoretical and computational works were largely limited to idealized and simple geometries and a majority of experimental data was obtained for easy-to-handle fluids, e.g. air and water. In the contrary, engineered systems often embody complex geometries and employ a broad range of fluids, including liquid metals and molten oxides. Usability of experimental heat transfer correlations obtained for air ($Pr \sim 1$) and water ($Pr \sim 3 \dots 7$) for liquid metal systems ($Pr \sim 0.01$) has been questioned, in particular, the Prandtl number effect on local heat flux distribution. Side-wall cooled descending flow in a liquid pool further complicates the pool's energy splitting and heat flux distribution.

The questions asked were (i) to what extent natural convection heat transfer correlations obtained for non-metal fluids are applicable in liquid metal scenarios, (ii) how the hemispherical segment geometry alters the pool's energy splitting (upward vs. downward) compared to a fluid layer, and (iii) what is a peaking factor of heat flux distribution along the curved boundary.

We used numerical experiments as a means to sort out the above uncertainties. The suitability of the computational tool used was examined, including our recent paper. Results of the numerical simulation confirm the Prandtl number effect on local heat flux distribution previously found in Nourgaliev et al (1997) over a limited range of Prandtl number (0.6 to 7). The descending flow along the curved wall was found to be responsible for the main difference of flow patterns in fluid layers and segment pools, namely the segment pool does not permit a large difference of temperatures in the pool's upper and lower boundaries.

Results of this work suggest the need for a full 3D conjugate transient analysis of a lower metal layer in the RPV bottom head in a stratified scenario, that accounts for upper oxidic pool behavior and multi-dimensional conduction in the vessel wall.

Task F1: Construct detailed MAAP4.04 model of AP1000 reactor pressure vessel internals, including core, core shroud or reflector and low head structures

Lead organization: Westinghouse

Task status: Completed.

MAAP4 detailed core and in-vessel model was assembled and test runs were performed. Documentation was prepared. The result is used as input for Tasks I.1-I.4.

Task F2: Model in-vessel core melting and relocation to the lower head with MAPP4.04

Lead organization: Westinghouse

Task status: Completed.

A series of MAAP4 analyses of the core melting and relocation was performed. Supporting calculations of the core heatup was completed. Modeling of the core geometry in more detail with a finite difference methodology using the FLUENCE code. The analysis was submitted for independent review. The result was then used as input for Tasks I.1-I.4.

Task F3: AP1000 Timeline and uncertainty

Lead organization: Westinghouse

Task status: Completed.

The objective of this task is to determine the core melt progression and outline uncertainties that may affect the assessment of IVR. Results of work in this task were then used as input for Tasks I.1-I.4.

Task F.4. SCDAP/MAAP calculation and analysis to support IVR for APR1400

Lead organization: Korea Hydro- and Nuclear Power

Task status: Completed.

KHNP performed SCDAP/MAAP calculations of core melt degradation, core melt relocation to the RPV lower head to support the analysis of thermal loadings in IVR scenarios. Results of this Task was reviewed and used in Tasks I.5-I.6.

Task G1: Develop boiling two-phase natural circulation flow model through a reactor vessel insulation configuration

Lead organization: Korea Hydro- and Nuclear Power

Task status: Completed.

The scaling model based on Reyes' two-phase natural circulation was developed. Independent review has been performed by an expert and by UCSB team. Several suggestions/comments made by the reviewers was implemented in a revised model.

Task G2: Estimate real plant thermal-hydraulic boundary condition at the outside of RPV

Lead organization: Korea Hydro- and Nuclear Power

Task status: Completed.

A sensitivity study using MAAP was performed to evaluate the effect of key parameters of melt progression on the behavior outside of the RPV. Based on the study, important parameters and their ranges were identified for the evaluation of IVR effectiveness (Task I.5). Also, a joint effort with INEEL to develop SCDAP/RELAP input deck for APR1400 was completed.

Task G3: Develop and optimize the final design of the RPV insulation configuration using CFD techniques

Lead organization: Korea Hydro- and Nuclear Power

Task status: Completed.

Initially, the limitation of proper modeling of two-phase flow with the existing CFD code and this task was put on hold. With the UCSB finding of the importance of the streamlined insulation configuration, this task was restarted with the following assumption: single-phase liquid is assumed and the focus is to identify the streamlines.

Task G4: Measurement of forced flow CHF, flow visualization, and comparison of flow and boiling phenomena with those of natural circulation

Lead organization: Korea Advanced Institute of Science and Technology (KAIST) through Korea Hydro- and Nuclear Power

Task status: Completed.

The objective of this task was to measure forced flow CHF, conduct flow visualization, and perform comparison of flow and boiling phenomena with those of natural circulation

A test loop was designed and construction at KAIST was completed. Visual information and 20 CHF data were obtained for the mass flow range of 0 ~200 kg/(m².sec). Two different inlet subcoolings, 2K and 14 K were examined. The result was compared with ULPU test data and SULTAN correlation.

The effect of mass flux and inlet subcooling on the critical heat flux has been investigated. The forced flow CHF correlation was modified based on the new data. Flow behavior was visualized under high void fraction and low inlet subcooling conditions. As shown in Figure 30, CHF increases linearly with the increase in the mass flux. Increase in inlet subcooling results in the increase in CHF. Based on the experimental data, a local condition basis CHF correlation is developed as following (Equation G.4-1);

$$q''_{CHF} = \frac{945 - 1200x}{(1.2 + 0.004G)^{-0.4} - 3.57G^{-1}} \quad (G.4-1)$$

where, $-0.05 < x < 0.50$, $0 < G < 300 \text{ kg/m}^2\text{s}$.

The CHF correlation correlates the data well with the RMS error 2.4% (Figure 31).

Figure 30. Effect of mass flux on forced flow CHF.

Figure 31. KAIST CHF correlation.

Task G5: Non-boiling and boiling natural circulation test.

Lead organization: Korea Hydro- and Nuclear Power

Task status: Completed.

The objective of this task is to carry out a half-scale non-boiling and boiling natural circulation performance tests with respect to flow stability, and analysis of the results. Using the scaling approach discussed in Task G1, the test loop was designed and constructed. Insulation configuration is based on the basic configuration. A streamlined insulation configuration was be incorporated into the design.

A non-boiling test by using air-water flow was performed. The sensitivity of major design parameters impacting the natural circulation performance was investigated; such as air injection rate, loop pressure drop, inlet flow area. At high air injection rate, carryover to downcomer section and the reduction in flow was observed. After the completion of shake-down tests using electrical heating, boiling tests were performed. Comparison with the air-water test was performed. Most notably, parameters examined are inlet flow area, heat flux (air injection rate for air-water tests), and inlet subcooling. Figures 32-33 show the effect of the heat flux and inlet flow area. For the test loop of interest, the effect of inlet subcooling is complex. It triggers flow oscillations with amplitude of 30 l/min in flow rate for a range of subcooling. The cause and the characteristics of the oscillations were investigated and the

flow stability regime map was constructed. An understanding of the mechanisms that caused the oscillations, analysis with transient code calculations was performed.

Figure 32. Effect of Heat Flux on the Mass Flowrate.

Figure 33. Effect of Inlet Flow Area on the Mass Flowrate.

Tasks H1, H2: Review latest tests and results of IVR phenomena, debris materials interaction and evaluate impact on modeling debris bed configuration in the RPV bottom head

Lead organization: Korea Hydro- and Nuclear Power

Task status: Completed.

Data and phenomenology of melt stratification observed during RASPLAV/MASCA prototypic corium experiments are being reviewed. Based on the review, two-layer configuration was proposed to be used as the baseline and three-layer configuration based on MASCA result to be used as a sensitivity study (Task I.5)

Task H3: Develop a new integral separate-layer debris model that can consider multidimensional effects inside the metallic layer and time dependent heat transfer at the water on top

Lead organization: Korea Hydro- and Nuclear Power

Task status: Completed.

CFD calculation was performed to simulate Rayleigh-Bernard convection in the metallic layer for various conditions such as heat flux from the oxide, radiative heat flux at top, and depth of the metallic layer.

Task H.4. Development and validation of an Integral Model of In-Vessel Debris for APR1400

Lead organization: Korea Hydro- and Nuclear Power

Task status: Completed.

The objective of this task is to verify the integral separate-layer debris model with available experimental data. The development and validation of an integrated model of IVR for APR1400 was made, which takes advantages of results from other tasks of the present project, such as analysis of experimental data on natural convection heat transfer in volumetrically-heated liquid pools, numerical simulation of heat transfer in metal layers (using AEA-T CFX code), analysis of CHF data on external vessel cooling. The verification exercise under task H4 was based on available information. Review of available literature survey was performed.

Tasks I.1-I4. ROAAM study of IVR for AP1000. Development of failure criteria of AP1000 IVR.

Lead organization: Westinghouse Electric Company

Task status: Completed.

The objective of this task is to perform analysis of the in-vessel retention phenomena using the Risk-Oriented Accident Analysis Methodology (ROAAM) and provide the basis for the application of the IVR accident management strategy to the AP1000 passive plant and quantification of vessel failure in the AP1000 PRA.

The present work is built upon a previous ROAAM analysis of IVR for the AP600. The ROAAMs included an analysis of the in-vessel melt progression and evaluation of the structural and thermal challenges to the vessel during the relocation to the lower head, including in-vessel steam explosion. Testing and evaluation of the uncertainties associated with the thermal loading produced by the molten debris pool and heat removal limitations due to boiling crisis on the exterior vessel surface were performed in the ACOPO and ULPU programs. The conclusions from the ROAAMs showed that the limiting challenge to the vessel integrity is the thermal loading produced during the steady-state heat transfer to the lower head wall after complete debris relocation to the lower plenum. The IVR ROAAM analyses and testing showed that the water in the AP600 cavity removes the heat produced by the molten debris bed in the lower head with significant margin while the structural integrity of the lower head was maintained.

Based on the ROAAM results, reactor vessel failure in the AP600 was considered to be physically unreasonable, and a probability of zero was applied to vessel failure in the AP600 PRA if the following conditions of the ROAAM analysis were met:

- The reactor coolant system was depressurized.
- The reactor vessel was submerged adequately to wet the heated surface.
- Reactor vessel reflective insulation and containment water recirculation flow paths allowed sufficient ingress of water and venting of steam from the cavity.
- The treatment of the lower head outside surface (painting, coatings, etc.) did not interfere with water cooling of the vessel.

As to AP1000, much of the passive plant design has not changed in the power uprating from the AP600 to the AP1000. The overall containment design below the operating deck remains essentially the same. The volume of the containment that floods to submerge the reactor vessel and the valves and piping used to drain the water from the IRWST are essentially the same. A check valve has been added to the refueling canal drain, which permits the containment to flood more rapidly by reducing the volume that fills with water. The reactor vessel diameter and lower head geometry has been retained from the AP600 design. Increasing the power output of the passive plant has resulted in several changes to the reactor

core, and reactor pressure vessel and internals, and all must be carefully considered in assessing the plant's capability for successful IVR:

- The reactor power is increased from 1933 MWt to 3400 MWt.
- The reactor core size is increased from 145 12-ft fuel assemblies to 157 14-ft fuel assemblies.
- The length of the reactor vessel was extended 19.5 inches through the core region to accommodate the longer fuel.
- The core reflector is modified to be a thick core shroud to accommodate the additional fuel assemblies within the core barrel.
- The lower core support plate is 1" thicker to accommodate the weight of the larger core.

AP1000 Base Case In-Vessel Retention of Molten Core Debris

Figure 34. AP1000 In-Vessel Retention with AP600 CHF Success Criterion.

These changes have an impact on the direct application of the methodology as performed in IVR ROAAM to the AP1000. Specifically, the large margin to vessel failure with respect to the critical heat flux (CHF) limits as demonstrated in the AP600 ROAAM has been reduced by the increase in reactor power in the AP1000. Figure 34 presents a conservative heat flux calculation for the AP1000 compared to the critical heat flux success criterion used in the

AP600 IVR ROAAM. The increased heat load in the AP1000 is offset somewhat by the increased mass of the core, but the margin to the AP600 critical heat flux limit is small.

To establish a strong basis for crediting in-vessel retention in the AP1000, the heat removal capacity of the water cooling the vessel wall needs to be increased to reclaim margin. To extend the IVR accident management strategy and the conclusion from the AP600 IVR ROAAM that vessel failure is physically unreasonable as applied to the AP1000, the following steps are taken:

- Increase the capability of the water in the reactor cavity to remove heat from the external surface of the reactor vessel (e.g. increase CHF).
- Demonstrate that for the increased critical heat flux, thermal failure remains the limiting failure over structural failure for the AP1000.
- Demonstrate that the AP1000 in-vessel melt progression does not change significantly from the AP600 melt progression in such a way as to challenge the vessel integrity during relocation.
- Demonstrate that the correlations used to calculate heat loads scale appropriately to the AP1000.

Each of these subjects is addressed in the technical report prepared as a separate deliverable for this project and distributed to project partners for comments. The report includes an analysis of core melt degradation, melt relocation, assessment of thermal loading on the RPV lower head, and thermal margins of IVR. The report was submitted to NRC and ACRS as part of licensing document for AP1000.

In particular, important conclusions from the analysis of the lower plenum debris pool formation analysis are:

- The lower plenum debris bed is cooled with water during the entire relocation process before contact with the support plate. Transient debris configurations are not predicted to threaten vessel integrity.
- The lower plenum oxide debris subsumes the lower core support plate before dry out in the lower plenum occurs. If the relocated debris is assumed to be instantaneously quenched in the lower plenum water, the oxide debris contacts the lower support plate before the debris can return to a superheated condition. Therefore, the lower core support plate, the core shroud, and a sizeable fraction of the core barrel are subsumed in the debris bed. The focusing effect is mitigated.
- The lower plenum debris bed is predicted to form a metal layer over oxide pool configuration.

- The potential for debris interaction, creating a bottom metal pool of uranium dissolved in zirconium, is expected to be small.
- The earliest time to achieve the fully molten, circulating debris bed in the lower plenum is 2.7 hours after shutdown. The timeline of events is presented in Table below.

DEBRIS RELOCATION TIME LINE			
Event	From Beginning of Accident (Second)	From Onset Melt	
		(Second)	(Hour)
Time of core uncover	2300	–	–
Onset of melting	3000	0	0.0
In-pool crust failure	6000	3000	0.8
Initial relocation to lower plenum starts	6000	3000	0.8
Oxide Blockage begins to melt	6300	3300	0.9
Lower plenum debris contacts lower support plate	6880	3880	1.1
Lower plenum dry out	7225	4225	1.2
Lower support plate melting ends	8000	5000	1.4

Most notably, the analysis presents bounding cases for challenges to the vessel integrity from two potential failure modes postulated from phenomena associated with lower plenum material interactions (Fig.35).

Figure 35. Bottom Metal Layer Lower Plenum Debris Bed Configuration to account for the formation of a dense bottom metal layer below the oxide layer.

Figure 36. The probability distributions of the heat fluxes normalized by the local critical heat fluxes are presented in Figure 39-14. The peak heat flux occurs at the top of the oxide layer at 70 percent of the critical heat flux. The results of the analysis show that, with the AP1000 insulation designed to increase the cooling limitation at the lower head surface and the cavity adequately flooded, the AP1000 provides significant margin-to-failure for IVR via external reactor vessel cooling

The first potential failure mode is a challenge from high heat fluxes generated by a bottom metal layer with a significant fraction of the fission products partitioned into the metal (Fig.36). The lower bound critical heat flux at the bottom point of the reactor vessel lower head for the AP1000 geometry is greater than 800 kW/m².

The upper bound of the heat flux from the bottom metal layer to the vessel wall is predicted to be less than 500 kW/m². Therefore, the heat flux from a bottom metal pool is not expected to exceed the critical heat flux at the bottom of the reactor vessel lower head. The ratio of the maximum heat flux to the minimum critical heat flux, q/q_{CHF} is 0.625. The second potential failure mode is from a thinned metal layer producing high heat fluxes via the focusing effect. For a bounding case of top metal layer thinning due to material interactions, the peak heat flux in the metal layer is 1720 kW/m². The lower bound critical heat flux in the metal layer at the azimuthal angle is 1890 kW/m². The q/q_{CHF} is 0.91, demonstrating that margin-to-failure is maintained for this bounding case (Fig.37). It is noted that actual heat fluxes expected from the debris bed to the lower vessel wall, even in the presence of material interactions and a bottom metal layer are significantly less than predicted in this analysis. The assumptions in this analysis are conservative with respect to increasing the heat loading to the vessel. Mechanistic assumptions are expected to result in milder thermal loading and larger margins-to-failure.

AP1000 Bounding Top Metal Layer Thinning from Material Interactions
Margin to Failure, q/q_{CHF}

Figure 37. Margin-to-failure in AP1000.

Therefore, the AP1000 reactor vessel is predicted to remain intact during IVR considering bounding conditions generated from material interactions postulated to occur in the lower plenum debris bed.

Tasks I.5-I6. In-vessel Retention Performance Evaluation for APR1400

Lead organization: Korea Hydro- and Nuclear Power

Task status: Completed.

From APR1400 PSA result, four scenarios (LOFW, SBLOCA, Medium break and large break LOCA) are chosen for the consideration. MAAP code calculations for these scenarios are completed. Similar calculations using SCDAP are in progress. The result of MAAP and SCDAP output was reviewed in determining the boundary conditions for the final IVR performance evaluation for APR1400.

Literature survey has been completed including works on miscibility gap, AP1000 assessment of IVR and on melt pool natural convection. A special attention is paid to the phase separation observed in MASCA tests. A sensitivity study was added to the IVR performance evaluation based on these surveys.

Overall, the result of the steam-water tests and the in-vessel retention performance evaluation provided additional insights in the finalization of APR1400 insulation design and its severe accident mitigation strategy development.

Summary

This project was carried out through collaboration between UCSB, Westinghouse and KHNP. Extensive exchanges between UCSB and Westinghouse occurred during the first year of the project when the ULPU-2400 Configuration facility was being upgraded to best simulate AP1000 plant design features. Also, UCSB and Westinghouse interacted intensively on Westinghouse's assessment and implementation of IVR. Westinghouse used the ROAAM methodology developed at UCSB for AP600.

During the second year of the project collaboration between US and Korean partners was intensified and included a series of information exchanges and technical meetings, which started with the PI's visit to KHNP facility in Taejon, Korea on May 12, 2003, and continued at the July 23-24 meeting at UCSB, and meeting of UCSB researchers with KHNP on October 6-9, 2003 during the NURETH-10 meeting in Seoul, Korea.

All the project tasks were carried out successfully and according to the project schedule; major project milestones and technical deliverables were completed in time to support In-Vessel Retention (IVR) Assessment for the license applications of AP1000 and APR1400.

MILESTONES/DELIVERABLE SUMMARY

W denotes Tasks by Westinghouse Electric Company.

K denotes Tasks by Korean Hydro- and Nuclear Power Company (KHNP).

Related Tasks ID	Milestone/Deliverable Description	Planned completion date	Actual completion date
A.1	<p>ULPU-2000 coolability limi tests</p> <p>Completed a series of burnout experiments on the full-scale ULPU-2000 Configuration IV facility.</p> <p><i>Measured critical heat flux (CHF) data showed a significant enhancement of the IVR coolability limit.</i></p>	May 30, 2002	June 30, 2002
A.2	<p>ULPU two-phase flow analysis</p> <p>Natural circulation flow data were obtained and analyzed.</p>	May 30, 2002	July 30, 2002
A.1-A.2 Deliverable/ Utility	<p>Issued a Technical Report and sent to Project partners and DOE as part of Annual Report (Jan 15, 2003)</p> <p>The ULPU data were used by Westinghouse for AP1000 licensing application submitted to NRC/ACRS</p> <p>ICAPP paper on two-phase flow in ULPU.</p>		Aug 30, 2002
A.3	<p>ULPU numerical simulation</p> <p>A method was developed and tested for interface capturing with a large density ratio.</p>	Aug 30, 2002	Nov 30, 2002
A.4-A.5, B.1	<p>Complete a series of flow path optimization experiments on the full-scale, power-upgraded ULPU-2400 Configuration IV facility (the facility was remodeled to simulate AP1000 design).</p> <p><i>The data demonstrates the potential enhancement of CHF in AP1000 design using an appropriate design of thermal insulation</i></p> <p><i>Found that the limit to coolability sensitive to water chemistry.</i></p>	Feb 30, 2003	Jan 15, 2003

A.4-A.5, B.1	Compilation of critical heat flux data and sent to partners and DOE as part of the Annual Project Report.		Jan 15, 2003
Deliverable/ Utility	The data and findings from the CHF experiments on the ULPU-2400 Configuration V facility were used by Westinghouse to update AP1000 licensing application.		
C.1-C.3	Complete a series of experiments on BETA/mini-ULPU facility on the effect of prototypic vessel surface coatings on CHF.	Feb 30, 2003	Jan 15, 2003
C.1-C.3	Data analysis and finding summary sent to Project partners and DOE as part of the Annual Project report.		Jan 15, 2003
Deliverable/ Utility	The findings suggested a need to modify the technology used to provide the vessel surface coating for AP1000. Westinghouse made a decision to use a strippable vessel painting. A need for new confirmatory tests is indicated.		
C.3 (extended scope)	Thermal insulation design optimization experiments on the ULPU-2400 facility and examination of containment water chemistry effects		
	Separate-effect experiments on the effect of boric acid, pH, trisodium phosphate (present in the reactor containment for iodine retention) on burnout performance	Apr 30, 2003	April 10, 2003
	Testing burnout performance on prototypic vessel steel for APR1400 and AP1000, using manufacturer's vessel coating procedures	May 30, 2003	May 30, 2003
	Testing the burnout performance with new baffle configurations	Jun 30, 2003	August 1, 2003
	Testing the burnout performance with prototypic water chemistries	Aug 30, 2003	Sep 15, 2003
	Testing the effect of exit restriction on natural circulation flow rate	Oct 15, 2003	Nov 30, 2003

C.3 Deliverable	Report on CHF data, on the effect of baffle design and water chemistry, with recommendation on the technology for vessel surface treatment, surface coating and thermal insulation design for AP1000. NURETH-10 paper on ULPU results.	Nov 30, 2003	ULPU-V Report issued on Jun 30, 2003.
D.1.	Task was removed upon an agreement with DOE and INERI project management. Replaced by tasks (C.3-extended scope) that were more critical to IVR performance in AP1000 than the originally planned task D.1		
D.2	BETA-NC natural convection tests Design, construction and testing of BETA facility for infrared-assisted natural convection experiments	April 30, 2003	April 30, 2003
D.2 Deliverable	Obtain natural convection heat transfer imagery for use in CFD validation exercises. ICAPP-2004 paper reporting BETA-NC data and results of analysis.		Oct 30, 2003 Jan 15, 2004
E.1 (extended scope)	Direct numerical simulation of natural convection flow under unstably stratified conditions characteristic of IVR Perform analysis of natural convection heat transfer and flow conditions of stratified melt pool configuration of interest to IVR Perform DNS and LES for natural convection flow in internally-heated liquid pools in a broad range of Raleigh number of relevance to IVR, with focus on stratified melt pool configurations	Jun 15, 2003 Oct 30, 2003	Nov 15, 2003 Dec 15, 2003
E.1 Deliverable	ICAPP-2004 paper provides an analysis of BETA-NC experiments, delineated the role of numerical schemes and discretization in predicting natural convection flow patterns.		Jan 15, 2004
E.2.	Task was removed upon an agreement with DOE and INERI project management. Replaced by tasks (C.3-extended scope and E-1) that were more critical to IVR performance in AP1000 than the originally planned task E.2.		

E.3	<p>Examine needs for 2D treatment</p> <p>Analysis of data from BETA-NC experiments (Task D.2) and from DNS/LES simulation (Task E.1) to reveal the significance of 2D effect, including stratified pools.</p> <p>Performed numerical simulation of natural convection flow and heat transfer in spherical segment for different Prandtl numbers.</p>	Oct 15, 2003	Nov 30, 2003
F.1 (W) F.2 (W)	MAAP setting and modeling for AP1000	Jun 30, 2002	Dec 30, 2002
F.3 (W)	AP1000 Time line and uncertainty	May 15, 2003	May 30, 2003
F.1-F.3 (W) Deliverable/ Utility	Westinghouse report on analysis of in-core melting and relocation using MAAP4.04 model of AP-1000.		June 30, 2003
F.4 (K)	SCDAP/MAAP Calculation for APR1400	Aug 15, 2003	Dec 15, 2003
G.1-G.3 (K)	IVR 2-phase flow modeling and analysis	Jan 30, 2003	Oct30, 2003
G.4(K)	APR1400 Insulation design	Nov 30, 2002	Dec 30, 2002
G.5(K)	<p>Design and construction of facility to test the insulation design.</p> <p>A half-scale test facility was built at KHNP and is ready for two-phase natural circulation experiments</p>	Jan 30, 2003	Feb 30, 2003
G.6-G.7 (K)	<p>Flow stability test performance and data analysis</p> <p>KHNP presentation on results of two-phase natural circulation experiments</p>	Nov 30, 2003	Jan 15, 2004
H.1-H.2 (K)	KHNP report. Review of IVR phenomena (in-vessel).	Apr 30, 2002	Apr 30, 2002

H.3-H.4 (K)	In-vessel phenomena: models and models validation	Nov 15, 2003	Dec 15, 2003
I.1-I.4 (W)	Westinghouse report. ROAAM study of IVR for AP1000	Oct 30, 2003	June 30, 2003
I.5-I.6 (K)	KHNP report on assessment of AP1400 IVR margin.	Jan 15, 2004	November 6, 2003