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UNITED STATES
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

In re:	Docket Nos. 50-247-LR; 50-286-LR
License Renewal Application Submitted by	ASLBP No. 07-858-03-LR-BD01
Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Operations, Inc.	DPR-26, DPR-64
	September 15, 2010

STATE OF NEW YORK'S MOTION FOR LEAVE TO FILE ADDITIONAL BASES FOR PREVIOUSLY-ADMITTED CONTENTION NYS-25 IN RESPONSE TO ENTERGY'S JULY 14, 2010 PROPOSED AGING MANAGEMENT PROGRAM FOR REACTOR PRESSURE VESSELS AND INTERNAL COMPONENTS

A. Introduction

Pursuant to 10 C.F.R. § 2.309(f)(2) the State of New York seeks leave to file the attached Additional Bases for Previously-Admitted Contention NYS-25. The Additional Bases are based on Entergy's filing with the Board of a proposed aging management program concerning the embrittlement of reactor pressure vessels (or RPV) and internal components, which was filed in an attempt to meet its obligations under 10 C.F.R. § 54.51(c)(1)(iii) with regard to RPV and internal components.[1] Entergy's initial License Renewal Application did not contain an aging management program concerning the embrittlement of reactor pressure vessels and internal components. On July 14, 2010 Entergy provided NRC Staff with the Ninth Amendment to the Indian Point License Renewal Application. That amendment, denominated NL-10-063, contained a description of an entirely new aging management program that did not exist before.

[1] Entergy's submission to the Board and the parties in this proceeding was dated July 15, 2010.

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The State of New York's Additional Bases (as well as the accompanying Additional Supporting Evidence and the September 15, 2010 declaration of Dr. Richard Lahey) are based on, and address deficiencies in, Entergy's July 14, 2010 NL-10-063 filing. Pursuant to the Scheduling Order issued by the Board on August 12, 2010, the filing of the proposed Additional Bases is timely, as they are being filed on September 15, 2010 -- the date by which the Board permitted the State to file new or supplemental contentions arising from Entergy's Ninth Amendment to the License Renewal Application concerning embrittlement of reactor pressure vessel components. *See* Scheduling Order dated August 12, 2010. Thus, the remainder of this pleading addresses the other factors in 10 C.F.R. § 2.309(f)(2) as well as the requirements of 10 C.F.R. § 2.309(f)(1) as required by the Board's July 1, 2010 Order.

B. The Contention Meets All The Requirements of 10 C.F.R. § 2.309(f)(2)

The contention fully meets 10 C.F.R. § 2.309(f)(2) which requires for admissibility, in pertinent part, a showing that:

> (i) The information upon which the amended or new contention is based was not previously available;
>
> (ii) The information upon which the amended or new contention is based is materially different than information previously available; and
>
> (iii) The amended or new contention has been submitted in a timely fashion based on the availability of the subsequent information.

Id.

1. Information Not Previously Available

Since the Additional Bases are based upon a document first filed with the parties to this proceeding in July 2010 and on the new information contained in that document regarding a proposed aging management program concerning the embrittlement of RPV internals, the State's submission relies on information not previously available and thus meets the first prong of the test set forth in 10 C.F.R. § 2.309(f)(2)(i).

2. The New Information Is Materially Different Than Previously Available Information

Before July 14, 2010 Entergy had not submitted an aging management program for reactor pressure vessels and internal components and embrittlement. As Entergy itself acknowledges in NL-10-063, "The Reactor Vessel Internals Program is a new plant-specific program." NL-10-063 at 84. Entergy's new July 2010 proposal and its contents differ materially from Entergy's previous proposal in its 2007 LRA. There, Entergy proposed to "manage loss of fracture toughness, cracking, change in dimensions (void swelling), and loss of preload in vessel internals" by participating in industry programs investigating aging effects of reactor internals and subsequently preparing an inspection plan for NRC review. Entergy LRA §§ A.2.1.41 (Reactor Vessel Internals Aging Management Activities (for IP2)); A.3.1.41 Reactor Vessel Internals Aging Management Activities (for IP3)).

C. The New Bases Meet All the Requirements of 10 C.F.R. § 2.309(f)(1)[2]

1. The Bases Are Within the Scope of License Renewal

New York State Contention NYS-25 claims that:

> Entergy's License Renewal Application Does Not Include An Adequate Plan To Monitor And Manage The Effects Of Aging Due To Embrittlement Of The Reactor Pressure Vessels ("RPVs") And The Associated Internals

[2] It is not clear that a showing needs to be made that new bases are allowable under 10 C.F.R. § 2.309(f)(1). Out of an abundance of caution, the State of New York provides the following demonstration that the new bases meet any requirements of § 2.309(f)(1) that might arguably be relevant to bases.

This contention and its bases have already been admitted by the Board. *Entergy Nuclear Operations, Inc.*, (Indian Point Nuclear Generating Units 2 and 3), Memorandum and Order (Ruling on Petitions to Intervene and Requests for Hearing) LBP-08-13 at 103-104, 68 NRC 43 (July 31, 2008). Now that Entergy has proposed a program, which it states will address aging management of the reactor pressure vessels and their internals due to embrittlement, the State of New York is expanding its original bases to address specific shortcomings in the newly offered plan. The RPV and internal components that are the subject of the recently-proposed AMP are plainly within the scope of Part 54. Thus the State's additional bases, which continue the challenge to Entergy's now-modified attempt to provide an adequate AMP for RPV and internals related to embrittlement, remain within the scope of this license renewal proceeding.

2. The Issues Raised Are Material to the Findings that the NRC Must Make to Support the Action that is Involved in this Proceeding

The issue of embrittlement of the Indian Point RPVs and their internals is material to this relicensing proceeding because, if the State is correct in its contention, the NRC must make certain findings to protect the public health and safety and the environment, and either deny the license renewal, or impose significant modifications on the applicant's operations. *See* 10 C.F.R. §§ 54.4(a)(1)and (3), 54.21(c)(1)(iii), and 54.29(a). The State has demonstrated in the Additional Bases, which are supported by the September 15, 2010 Declaration of Dr. Richard Lahey, that embrittlement is a significant safety and public health issue. Sept 15, 2010 Lahey Decl. at ¶¶ 8-12. Inadequate management of the effects of embrittlement on RPV and internals could lead to failures of those components to perform their intended safety functions and/or cracks in these components, which could result in the breaking away of parts which would interfere with other components and systems performing their safety functions. Sept 15, 2010 Lahey Decl. at ¶¶ 13-14, 16-18.

3. Adequate Bases Have Been Provided For the Contention

The State of New York today seeks leave to present Additional Bases in further support of a previously-admitted contention. These Additional Bases are detailed and exceed the regulatory requirement in 10 C.F.R. § 2.309(f)(1)(ii) for a "brief explanation" of the bases. The Additional Bases describe a number of deficiencies in Entergy's submission regarding its AMP for embrittlement of RPV and internal components. These bases are in addition to the bases previously accepted when Contention NYS-25 was admitted. The bases for this new contention allege not that Entergy has omitted something, but that what Entergy has presented does not meet its burden to prove that the AMP for RPV and internals embrittlement is adequate to meet the requirements of 10 C.F.R. § 54.51(c)(1)(iii).

4. A Concise Statement of Facts and Expert Opinion Support the Contention

Dr. Richard Lahey has offered his expert opinion that Entergy's July 14, 2010 Amendment to its LRA is flawed and does not provide an adequate AMP. For example, Dr. Lahey has provided his opinion that the proposed AMP does not adequately address the combined synergistic effects of embrittlement and fatigue on RPV and internal components. *See, e.g.,* Sept. 15, 2010 Lahey Decl. at ¶¶ 8-13. Dr. Lahey also states that proposed aging management plan as set forth in NL-10-063 lacks sufficient details to know precisely when the baseline inspections of the RPV and its internals will begin or when they will be completed. *See* Sept. 15, 2010 Lahey Decl. at ¶ 19. He also states that such baseline inspections and measurements should be completed before each reactor enters period of extended operation. *Id.* Dr. Lahey has based his opinion upon Entergy's own submissions, his review of NRC regulations, NRC and industry guidance, technical studies, and his extensive professional experience.

5. A Genuine Dispute Exists with the Applicant on a Material Issue of Law or Fact

The State of New York has provided sufficient information that a genuine dispute exists with Entergy regarding several material issues of the fact including whether the recently-proposed AMP is deficient for: (1) failing to consider the synergistic effects of embrittlement and metal fatigue on RPV and internals; (2) failing to provide sufficient objective details about when it will conduct and complete baseline inspections and measurements; (3) failing to provide sufficient objective details about how and when it will implement corrective actions to address problems identified with embrittlement; (4) failing to include adequate inspection techniques to identify embrittlement issues for certain RPV internals, including bolts; and (5) relying on vague future commitments to undertake corrective action when Entergy has encountered difficulties in tracking and completing commitments and corrective actions in a timely manner. Conversely, NL-10-063 reflects Entergy's view not to address such issues.

F. Conclusion

For the reasons stated, the State of New York respectfully requests that the Atomic Safety and Licensing Board grant leave to file the accompanying Additional Bases in support of the already-admitted Contention NYS-25.

Respectfully submitted,

s/

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dated: September 15, 2010

10 C.F.R. § 2.323 Certification

Pursuant to 10 C.F.R. § 2.323(b) and the Board's July 1, 2010 scheduling order, I certify that I have made a sincere effort to contact the other parties in this proceeding, to explain to them the factual and legal issues raised in this motion for leave, and to resolve those issues, and I certify that my efforts have been unsuccessful.

s/

John J. Sipos
Assistant Attorney General
State of New York

ADDITIONAL BASES FOR
PREVIOUSLY-ADMITTED CONTENTION NYS-25

(Embrittlement of Reactor Pressure Vessels and Associated Internals)

For addition after NYS-25, ¶ 3:

3.1 On July 15, 2010, Entergy provided the Board and the parties in this proceeding with Entergy's July 14, 2010 NL 10-063 communication to NRC Staff conveying Amendment 9 to the License Renewal Application which, in turn, proposed, for the first time, a plant-specific aging management program for reactor vessel internals.

3.2 Entergy's recently-proposed aging management program contained in the July 14, 2010 NL-10-063 submission is inadequate and violates 50 C.F.R. § 54.21(c)(1)(iii) because it does not address or manage the combined, synergistic aging effects of embrittlement and fatigue on reactor pressure vessel internal components including the core baffle, intermediate shells, former plates and bolts (particularly the re-entrant corners), and including the baffle-to-baffle bolt locations, the core barrel-to-former bolt locations, and baffle-to-former bolt locations, core barrel (and its welds), lower core plate and support structures, clevis bolts, fuel alignment pins, thermal shield, the lower support column and mixer, and the control rods and their associated guide tubes, plates, and welds. The failure of the recently-proposed aging management program to address or manage the combined, synergistic aging effects of embrittlement and fatigue on reactor pressure vessel internal components could have profound safety consequences for the State and its citizens.

3.3 While Entergy's recently-proposed aging management program proposes "baseline" inspections and measurements (NL-10-063 at 87) to analyze the embrittlement of various reactor vessel internal components, the proposed program is inadequate because it does not specify with any meaningful precision when such baseline inspections will be initiated or completed. Such baseline inspections and measurements should be completed before each Indian Point reactors enters 20-year extended operation period. The failure of the recently-proposed aging management program to require the completion of such baseline inspections and measurements violates 50 C.F.R. § 54.21(c)(1)(iii). The failure of the recently-proposed aging management program to specify when the "baseline" inspections and measurements must be completed could have profound safety consequences for the State and its citizens.

3.4 Entergy's recently-proposed aging management program is also inadequate because it:

> (a) does not specify with any meaningful precision when the replacement or repair of embrittled reactor vessel internal components will take place (NL-10-063 at 88);
>
> (b) disavows taking any preventative action to manage the effects of embrittlement aging of reactor vessel internal components (NL-10-063 at 86);
>
> (c) relies on less reliable remote-control VT-3 examinations to examine baffle-former assembly plates and edge bolts instead of the more reliable volumetric ultrasonic testing (UT) (which Entergy states it will use to examine the nearby baffle-to-former bolting) (NL-10-063 at 87; EPRI MRP-227 at 4-4 to 4-5, 4-14 to 4-16).

These deficiencies in the proposed aging management program violate 50 C.F.R. § 54.21(c)(1)(iii) and could have profound safety consequences for the State and its citizens.

3.5 Much of what Entergy has proposed in its new aging management program for reactor pressure vessel internals involves commitments to take certain corrective actions in the future (NL-10-063 at 88); however, there is growing evidence that Entergy is unable to meet its commitments, thus raising material questions about the adequacy of that portion of Entergy's recently-proposed aging management program that depends on future corrective actions.

ADDITIONAL SUPPORTING EVIDENCE FOR PREVIOUSLY-ADMITTED CONTENTION NYS-25

For addition after NYS-25, ¶ 7:

7.1 In response to Entergy's recently-proposed aging management program (NL-10-063), the State of New York also relies on the September 15, 2010 declaration of Richard T. Lahey, Ph.D.

7.2 Scope. Entergy describes the program as dividing internal components into four categories: (1) "primary" – components that are highly susceptible to effects of at least one aging mechanism; (2) "expansion" – components that are highly or moderately susceptible to the effects of at least one aging mechanism, but which show some degree of tolerance to those aging effects; (3) "existing" – components that are susceptible to an aging mechanism but are covered by an existing aging management program; and (4) "no measures" – components which will not be including within an aging management program. NL-10-063 at 85.

7.3 Preventative Actions. Entergy makes clear that the new aging management program "does not include preventative actions." NL-10-063 at 86.

7.4 Monitoring Frequency & Methods. Entergy states that the new program will use "periodic and conditional examinations" through visual examinations and volumetric ultrasonic (UT) examinations. NL-10-063 at pg. 86. Entergy also stated that in some instances it would make "baseline" measurements. NL-10-063 at 87. Entergy did not specify when it would make the "baseline" measurements or conduct the "periodic" examinations.

7.5 Entergy did not disclose that certain visual examinations (class VT-3 examinations) would be done by remote control. EPRI MRP-227 at 4-4. Moreover, Entergy did not disclose that other visual examination methodologies (class VT-1 and class EVT-1) have a greater degree of detection than class VT-3 examinations. *Compare* NL-10-063 at 87 *with* EPRI MRP-227 at 4-4.

7.6 Apparently because of radiation exposure concerns, Entergy plans to use remote VT-3 examinations to examine "baffle former assembly plates" and "edge bolts." NL-10-063 at 87. Nevertheless, Entergy plans to use volumetric ultrasonic tests (UT) to examine "baffle former bolting" which are also locate within the RPV and are in close proximity to the edge bolts and the baffle former assembly plates. *Id.* Entergy has not explained why it is using different examination methods for similar components.

7.7 Entergy states that it "may" use the results of inspections of the components in the "primary" group to conduct inspections of the components in the "expansion" group. NL-10-063 at 87.

7.8 Entergy also states that "In the case of significant conditions adverse to quality, measures are implemented to ensure that the cause of the nonconformance is determined and that corrective action is taken to preclude recurrence." NL-10-063 at 88. This is a vague commitment to some undefined action in the indefinite future that depends upon Entergy to be diligent in meeting this commitment. However, on September 14, 2010, NRC Staff released an audit of Entergy's Commitment Management System at the Vermont Yankee facility. NRC Staff stated:

> The NRC staff audit of the VY's Commitment Management System (CMS) determined that entries were not created in a timely manner for several commitments in the Table to adequately track the status of commitments. The condition reports for these commitments were generated and CMS entries have been created.
>
> A general observation was made of a process weakness in meeting implementation dates and entering CMS entries for each docketed site for

> commitments made at the fleet level. It appears that VY relied on the issuance of a fleet procedure to satisfy the commitment, but issuance date of the procedure exceeded the scheduled commitment date.

NRC Staff, Audit Report (Sept. 14, 2010) ML102420206. On July 31, 2008, Entergy released the "Independent Safety Evaluation Report" that observed (at 14, 43-45) a backlog of preventative and corrective maintenance work at Indian Point.

Respectfully submitted,

s/

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

In re:	Docket Nos. 50-247LR and 50-286LR
License Renewal Application Submitted By	ASLB No. 07-858-03-LR-BD01
Entergy Indian Point 2, LLC, Entergy Indian Point 3, LLC, and Entergy Nuclear Operations, Inc.	DPR-26, DPR-64 September 15, 2010

DECLARATION OF DR. RICHARD T. LAHEY, JR.

I, Richard T. Lahey, Jr., declare under penalty of perjury that the following is true and correct:

1. I am the *Edward E. Hood Professor Emeritus of Engineering* at Rensselaer Polytechnic Institute (RPI) in Troy, New York, a member of the National Academy of Engineering (NAE), a Fellow of the American Nuclear Society (ANS) and the American Society of Mechanical Engineers (ASME), and an expert in matters relating to the operations, safety, and the aging of nuclear power plants. I have previously submitted a declaration in support of the Notice of Intention to Participate and Petition to Intervene filed by the State of New York in this proceeding on November 30, 2007, which sets forth my qualifications in detail. By way of summary, I have held various positions in the nuclear industry and academia, and served on numerous panels and committees for the U.S. Nuclear Regulatory Commission (USNRC), Idaho National Engineering Laboratory (INEL),

Oak Ridge National Laboratory (ORNL), Electric Power Research Institute (EPRI), National Aeronautics & Space Administration (NASA), and the National Research Council (NRC). I have also held various positions in the nuclear industry and academia, including Dean of Engineering and Chair of the Department of Nuclear Engineering & Science at RPI. I have also been the lead engineer and manager of various departments responsible for safety analyses, Heat Transfer Mechanisms and Core & Safety Development for the General Electric Company (GE), including both military (*i.e.*, Naval) and commercial nuclear reactors. Over the last 40 years, I have also published numerous books, monographs, chapters, articles, studies, reports, and journal papers on nuclear engineering and nuclear reactor safety technology, and most of these publications have been peer reviewed. My *curricula vitae*, which more fully describes my educational and professional background and qualifications, is attached to this declaration and is available at: http://www.rpi.edu/~laheyr/laheyvita.html.

2. The factual statements and the expression of opinion in this declaration are based on, among other things, my best professional knowledge, my extensive professional experience in nuclear reactor technology, and my review of Entergy's April 2007 License Renewal Application, Entergy's July 15, 2010 submission to the Atomic Safety and Licensing Board (conveying Entergy's July 14, 2010 License Renewal Application Amendment No. 9 (communication NL 10-063 to NRC Staff), and other documents referenced in this declaration.

3. This declaration documents my recommendations and concerns about Entergy's new July 14, 2010 aging management program discussed in NL-10-063 and the detection and management of the embrittlement and/or corrosion-induced cracking of important structures and fittings within the reactor pressure vessel (RPV), and the age-related safety issues for the Indian Point reactors. My recommendations and concerns are based on my extensive experience and expertise in the field of nuclear reactor thermal-hydraulics and safety. Moreover, this input is based on, and expands upon, many of the concerns that I raised in my prior ASLB declaration concerning the re-licensing of IP-2 & 3 .

4. As I stated in my initial November 2007 declaration on these issues in support of the State of New York's Contention 25, in my professional judgment the applicant failed to demonstrate that it had adequately accounted for the aging phenomena of embrittlement for components inside the reactor pressure vessels at Indian Point Unit 2 and Unit 3. My professional judgment has not fundamentally changed based upon Entergy's July 14, 2010 submission of License Renewal Application, No. 9 [NL 10-063].

The Indian Point Reactors

5. Entergy's Indian Point Units 2 & 3 are currently under consideration for 20-year life extensions beyond their original 40-year design life. If approved, these plants will be licensed for operational levels of about 48 effective full power

years (EFPY). These Westinghouse designed plants are 4-loop PWRs and they are currently[1] rated at power levels of 3,216.4 MW_t.

6. They are sited on the east bank of the Hudson River in Buchanan, NY, which is about 24 miles north of the New York City (NYC) border.[2] Because of their close proximity to a very highly populated area (*i.e.*, the NYC metropolitan area), which is also the world's leading financial center, it is vital that IP Units 2 & 3 fully and unambiguously meet all reasonable and applicable criteria for safe operation. This is particularly true when considering life extension, since, like metal fatigue failures, failures due to embrittlement are much more likely as the plants age.

7. The USNRC Staff have prepared a guidance document entitled the "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, Rev. 1 (2005), in which Staff seeks to describe various Aging Management Programs (AMP) for the extended operations of nuclear power plants. That USNRC document does not specifically describe aging management programs for the embrittlement of internal components within the reactor pressure vessel (RPV), including, but not limited to, the: control rods and their associated guide tubes, assemblies, and seal welds, and many important in-core structures and fittings which will be discussed

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[1] The USNRC approved a stretch power increase of 3.26% for IP-2 in 2004 and a 4.85% increase for IP-3 in 2005; IP-2 and IP-3 also received 1.4% power uprates in 2003 and 2002, respectively.

[2] By way of additional reference, the Indian Point reactors are approximately 37 miles north of Wall Street in lower Manhattan, 3 miles southwest of Peekskill, 5 miles northeast of Haverstraw, 16 miles southeast of Newburgh, 17 miles northwest of White Plains, 23 miles northwest of Greenwich, Connecticut, 37 miles west of Bridgeport, Connecticut, and 37-39 miles north-northeast of Jersey City and Newark, New Jersey.

subsequently. *See* GALL, Chapter XI (Aging Management Programs); *see also* Entergy NL-10-063, at pg. 84 ("Revision 1 of NUREG-1801 includes no aging management program description for PWR reactor vessel internals."). Although the USNRC Staff did not include an aging management program to address reactor vessel internal components in GALL, I believe that all important safety concerns must be addressed to assure the health and safety of the American public during extended plant operations, and that the safety review for the requested licenses for extended operations of the two Indian Point reactors should include an analysis of the embrittlement of components inside the reactor pressure vessels and the implementation of a meaningful program to manage the embrittlement of such components during periods of extended reactor operation.

Embrittlement Phenomena

8. As previously discussed in my initial November 2007 declaration (¶¶ 6-18), one of the key age-related phenomena that must be considered in Entergy's License Renewal Application (LRA) is the embrittlement of the reactor pressure vessel's (RPV's) internal metal structures and fittings, which occurs due to the extended irradiation (*i.e.*, the neutron fluence, which is the neutron flux times the duration of the irradiation process) that will be experienced by these metal components, particularly those located within the so-called "belt line" region of the RPV (*i.e.*, the region of the RPV that is closest to the core) where the neutron flux is the highest. In addition, the reactor vessel internals may experience flow/thermal-transient-induced fatigue degradation, as well as embrittlement-induced

degradation due to various radiation damage mechanisms [Was, 2007], including damage due to void swelling which may occur because of transmutation and other effects [Was, 2007; NUREG/CR-6897; Barnes, 1964]. Also, some in-core components may experience irradiation assisted stress corrosion cracking (IASCC) [WCAP-14577, Rev.1-A, pgs,3-6 & 3-8, 2001], and/or primary water enhanced stress corrosion cracking (PWSCC) [WCAP-14577, Rev. 1-A, pgs. 3-4 & 3-5, 2001] due to prolonged exposure to the high temperature (*i.e.*, T > 400°F) borated primary coolant. In addition, cast austenitic stainless steel reactor components (*e.g.*, some reactor piping/fittings, pump casings, pressurizer spray heads, etc.) and various reactor pressure vessel (RPV) internals (*e.g.*, the upper mixing vanes, and the upper/lower core assemblies and support columns) are composed of a duplex stainless steel which contains both austenitic and ferretic phases, and are thus subject to embrittlement due to thermal aging [WCAP-14577, Rev.1-A, pgs. 3-12 & 3-13, 2001; EPRI Report TR-106092, 1997; NUREG/CR-4513, Rev.1]. Moreover, the heat affected zone (HAZ) of stainless steel and nickel alloy weldments may be more sensitive to embrittlement mechanisms than the base metals being joined [Hawthorne et al., 1986; NUREG/CR-6960; Carey, 2006].

9. In any event, embrittlement causes metals to lose ductility and become more susceptible to failures due to cracking or fracture. Also, reactor operations may be restricted by embrittlement since the temperature at which the embrittled metal structures and fittings change from non-ductile to ductile behavior (*i.e.*, the so-called nil ductility temperature, NDT) will increase as the reactor operates over

time and ages. Conversely, during reactor operations, the temperature at which the embrittled metal structures and fittings change from ductile to non-ductile behavior will increase. Significantly, this phenomenon implies that embrittled RPV internals will become progressively more vulnerable to failure due to thermal shocks as reactor operations continue. Obviously, irradiation damage is a serious age-related phenomena, and one that will not be annealed-out (*i.e.*, be healed) during reactor operations since PWR operating temperatures are too low for this to occur.

10. How the rather complex metal degradation mechanisms associated with fatigue and irradiation interact is still an area of active research, but it is known that the radiation-induced damage on reactor vessel internals can be extensive, since they may experience a neutron fluence of up to 10^{23} n/cm^2 at neutron energy (E) levels of E $\geq$ 1 MeV (i.e., $\geq$100 dpa)[3] [Was, 2007; Robinson, 2008] by the end of life (EOL) for extended operations. Indeed, the EOL Charpy impact Upper Shelf Energy (USE) for some thermally-aged cast stainless steel in-core components could be as low as 28 ft-lb$_f$ [WCAP-14577, pg. 3-13, 2001], which is well below the acceptable ASME code-specified minimum of 50 ft-lb$_f$, and even the 43 ft-lb$_f$ variance proposed by Westinghouse [WCAP-13587, Rev. 1, 1993], and endorsed by the ACRS [ACRS Letter, 9/23/09], as being acceptable for an Indian Point reactor pressure vessel (RPV) at the EOL.

[3] Displacements per atom (dpa) is a measure of radiation damage to a material.

11. Given that a variance from the applicable ASME Code is needed for the RPV itself, it is noteworthy that the RPV's inner wall experiences much less fluence than many of the in-core metal components that are inside the RPV and in closer proximity to the core and fuel rods. That is, the Indian Point RPVs are expected to experience $\sim 1.9\ 10^{19}$ n/cm^2 by the end of extended operations [Entergy Letter, 3/8/10], which, as noted previously, is much less than the $\sim 10^{23}$ n/cm^2 fluences that may be experienced by some RPV internals. The obvious conclusion is that RPV internal components will be significantly embrittled during the period of extended operations of IP-2 and IP-3; and much more so than the RPV inner wall. In-core components which are particularly vulnerable include the: core baffle, intermediate shells, former plates and bolts (particularly the re-entrant corners), and including the baffle-to-baffle bolt locations, the core barrel-to-former bolt locations and baffle-to-former bolt locations, core barrel (and its welds), lower core plate and support structures, clevis bolts, fuel alignment pins, thermal shield, and the lower support column and mixer. As discussed below, such in-vessel components also include the control rods and their associated guide tubes, plates, and welds.

12. Entergy acknowledges that, "PWR internals aging degradation has been observed in European PWRs, specifically with regard to cracking of baffle-former bolting." NL-10-063, at pg. 89. Indeed, EPRI has stated that, "considerable amount of PWR internals aging degradation has been observed in European PWRs." EPRI MRP-227, at A-4. Entergy also states: "As with other U.S. commercial PWR

plants, cracking of baffle former bolts is recognized as a potential issue for the [Indian Point] units." NL-10-063, at pg. 89. Moreover, material degradation has also been observed in control rod guide tube alignment (split) pins. EPRI MRP-227, at A-4.

13. As noted previously, any degradation in ductility will adversely effect the possible pressure-temperature (p-T) operating conditions (*i.e.*, there will be an increase in the nil ductility temperature, NDT). Also, it will adversely affect the ability of embrittled in-core components to withstand thermal shock transients and the decompression shock loads associated with a postulated design basis accident (DBA) loss of coolant accident (LOCA). Moreover, the metal structures and fittings within the RPV are subjected to many of the same transients (*e.g.*, a SCRAM - a rapid insertion of the control rods causing a rapid decrease in the core's power level), which are known to cause fatigue-induced degradation of the primary side piping, nozzles and structures [*see, e.g.*, my Sept. 8, 2010 ASLB Declaration on metal fatigue]. Hence, fatigue will also degrade the strength and ductility of many of the metal structures and fittings within the RPV, but virtually no fatigue analyses of this type have been presented by Entergy in their application for the extended operations of IP-2 and IP-3 [*see, e.g.*, my Sept. 8, 2010 ASLB Declaration on metal fatigue]. Entergy's recently-submitted aging management program amendment [NL-10-063], does not call for an analysis of the synergistic impacts of these different aging effects; I believe that this is a very serious omission and that

this deficiency should be corrected during the ASLB's re-licensing hearings for Indian Point Units 2 & 3.

14. In addition, severe pressurized thermal shocks can occur during postulated accidents which may rapidly depressurize the secondary side of the reactor system and cause a SCRAM. While pressured thermal shock of the reactor pressure vessel (RPV) itself was discussed in Entergy's re-licensing applications, there was no indication of what new accident analysis was done (if any) in which both embrittlement and fatigue were explicitly taken into account when assessing the effect of the accident-induced transient loads on RPV internals. This is quite important since thermal shock may cause highly embrittled and fatigued in-core components to fail, perhaps leading to an uncoolable core geometry and core melt. In my opinion, one of the most serious omissions from the USNRC's GALL Report and Standard Review Plan is that there was no mention at all of how highly embrittled and fatigued internal RPV structures and fittings will respond to the severe transient decompression shock loads associated with a DBA LOCA and the subsequent thermal shock loads associated with the discharge into the primary side of the reactor of relatively cold emergency core coolant (ECC) from the accumulators. It is well known [*e.g.*, Tong & Weisman, pgs. 147-149, 1970] that a strong decompression shock wave, created during the subcooled blowdown phase of a DBA LOCA, can cause significant transient pressure differentials across various internal RPV structures. Detailed experiments (*e.g.*, LOFT) and analyses have

shown that, when ductile, these in-vessel metal structures are not likely to fail or deform to the point where a coolable geometry can not be maintained for the core. In contrast, no such experiments and analyses have been presented by Entergy to justify that highly embrittled and fatigued in-vessel components will not fail and that a coolable core geometry will be maintained subsequent to a DBA LOCA. This is a very serious and, in my opinion, a totally unacceptable omission since brittle and fatigue-weakened structures are known not to tolerate shock loads well (*e.g.*, they may break loose or fracture) and, if a coolable geometry of the core is not maintained, it can melt, releasing a significant amount of radiation and possibly causing a breach of the lower head of the RPV. It is incumbent on Entergy to prove this will not happen, since Federal regulation, 10 C.F.R. § 54.4(a), clearly states that reactor operators must: "provide the capability to shut down the reactor and maintain it in a safe shut-down condition." It is also important to stress that while the USNRC Staff noted in their review of the Safety Evaluation Reports (SERs) for IP-2 & 3 that, "…if certain reactor vessel internals failed, they could potentially inhibit core coolability during an accident." [SER, Dockets 50-247 and 50-286, pg. 2-40, (Nov. 2009)], their primary focus was on the reactor's sample tubing systems, and did not encompass the more critical RPV internal components such as those listed previously in paragraph 11. Also, the industry programs, which Entergy has proposed to follow under the AMP for RPV internals, are mute on the serious age-related safety concern of the coolability of PWR cores subsequent to an accident-induced failure of highly embrittled and fatigued PRV internals. Unfortunately, the new aging management plan submitted by Entergy [NL-10-063] does not address or manage the synergistic aging effects of embrittlement and fatigue on RPV internals and the impact of accident-induced shock loads on these components..

15. In summary, I believe the USNRC has made a major error in not highlighting the above age-related safety issues in the GALL Report [NUREG-1801. Rev. 1] and the Standard Review Plan [NUREG-1800, Rev. 1]. Perhaps this is because of "stove piping" of the safety evaluations and the various AMP issues, in which each is discussed and analyzed separately, and thus the integrated and synergistic effect of accident-induced shock loads on highly embrittled and fatigued RPV internals was not considered. In fact, the aging phenomena of embrittlement <u>and</u> fatigue acting together and in concert with one another has apparently not been considered. In addition, there has apparently been some confusion associated with the USNRC's leak- before-break (LBB) ruling [NUREG/CR-4572, NUREG-1061, Vol. 3, 10 CFR 50, Appendix-A], in which the USNRC's the rules were changed for some of the dynamic ex-vessel LOCA loads (*i.e.*, for the pipe whip and jet loads) associated with the design basis accident (DBA). In particular, the USNRC now allows reactor operators to <u>not use</u> the ex-vessel loads associated with a double-ended pipe break if they can show that the primary side piping would be expected to leak well before it breaks. It is significant to note that the LBB ruling <u>does not apply to the in-vessel DBA LOCA decompression and thermal shock loads</u>. Unfortunately, the implications of this ruling have apparently been misunderstood by many in the nuclear industry, and it appears to have led USNRC staff to not be

overly concerned about the effect of DBA LOCA decompression loads, and emergency core cooling system (ECCS) or secondary side LOCA induced thermal shock loads, on highly embrittled and fatigued metal components within the RPV. As a consequence, it appears that this significant safety issue has been totally overlooked in the Standard Review Plan for Licensing Renewal Applications (LRAs).

Integrity of Control Rods, Guide Tubes, and Plates

16. Any aging management program concerning the embrittlement of reactor pressure vessel internals should include control rods and their associated guide tubes, plates, and welds within the scope of such program. The control rods and their associated guide tubes, plates, and welds are also very important RPV internals and their integrity is an extremely important safety concern. They are located in the core region of the RPV, and are inserted into the RPV through the upper head via so-called stub tubes. Their function is to absorb excess fission neutrons (*i.e.*, those not need to achieve a chain reaction) so that the power level of a reactor can be controlled.

17. With respect to control rods and their associated guide tubes and plates, of particular concern is the significant and reoccurring stress corrosion cracking that has been observed in the J-groove seal welds on the control rod drive (CRD) stub tube penetrations of the upper head of PWR RPV's. By way of example, according to USNRC documents, earlier this year the operator of the Davis-Besse reactor, "found evidence boric acid deposits and indications of primary water stress corrosion cracking in their nozzles and welds." NRC Staff, Division of Component

Integrity, NRC Perspectives on PWR Materials Issues, at 7 (June 2010) ML101520577. According to the USNRC, "the timing and extent of cracking was unexpected." *Id.*; *see also* NRC News, III-10-123 (May 26, 2010). It is significant to note that this type of leakage had been found earlier (*i.e.*, in 2002) at Davis Besse, and it nearly resulted in a major LOCA due to a massive corrosion-induced failure of the upper RPV head. In fact, the stress corrosion cracking of these type welds is widely considered to be one of the biggest challenges currently facing operating PWRs [NEI 03-08 [Addenda], at D-5 (June 2009)].

18. In addition, because of geometric considerations, many PWRs (including IP-2 and IP-3) can not meet the USNRC's required minimum coverage for the non-destructive testing (NDT) of these important J-groove welds [Walpole, 2009], and thus the integrity of these stub tube welds can not be confirmed.. It appears that to help address this chronic problem Entergy has ordered two new RPV heads [Telecon-USNRC/Entergy Report, March 18, 2008], but they have not yet been scheduled for installation at Indian Point [Telecon-USNRC/Entergy, March 18, 2008]]. In any event, unlike the rather superficial treatment given this important safety concern by Entergy [NL-10-063], I believe that a tangible, enforceable, and viable aging management program must developed and implemented before re-licensing the Indian Point reactor plants for extended operations since the integrity of these welds must be assured. If not, due to the leakage of borated primary coolant through cracked welds, there can be aggressive corrosion and wasting of the unclad outer surface of the upper head of the RPV

(such as the serious event that occurred at Davis-Besse and was identified in 2002). Worse yet, there might be an inadvertent control rod ejection (*e.g.*, due to a massive failure of the welds in the upper RPV head), which could cause a relatively major reactivity excursion, leading to core melting and significant radiation releases.

Baseline Inspections

19. With respect to Entergy's proposal to conduct baseline examinations of the RPV internals, I note that I previously called on Entergy to conduct such examinations and for NRC Staff to require the conduct of such examinations before entering the period of extended operations. In particular, in my initial November 2007 declaration in support of the State of New York's motion to intervene in this proceeding, I stated:

> As part of the relicensing review, and prior to the commencement of any extended operations, the NRC should require Entergy to conduct a thorough baseline inspection of both IP2 and IP3. These inspections should involve both visual and physical characterization and the non-destructive testing (NDT) of structures and components, including but not limited to the RPV, the RPV heads/fittings, the control rod drive mechanisms and associated RPV penetrations, most RPV internal hardware, and all key welds and fittings in the primary and secondary systems of the reactors.
>
> ....Thorough baseline inspections should examine the changes that the plants' systems, structures, and components have experienced during the first three and a half decades of operation. Without proper inspections, the NRC, the applicant, and the public will not have the necessary information to assess whether these plants are in any condition to continue to operate for an additional 20 years.... Routine, sound engineering practice requires a thorough baseline inspection for the license extension of a nuclear power plant to establish the state of the reactor facility, systems, structures and components at the end of their design life and disclose degradation which may have occurred. The failure to conduct thorough baseline inspections prior to life

extension is reckless and runs counter to rudimentary engineering practice.

See November 2007 Declaration of Richard Lahey, at ¶¶ 24, 25; *see also* State of New York Notice of Intention to Participate and Petition to Intervene, at pgs. 217-220, State of New York Contention 23 (Baseline Inspections).[4] While Entergy now seems to have embraced the concept of baseline inspections for RPV internals, the text of the proposed aging management plan as set forth in NL-10-063 lacks sufficient details to know when the baseline inspections of the RPV and its internals will begin or when they will be completed. In my opinion, this should occur before the onset of extended operations.

[4] Both Entergy and the USNRC Staff opposed this proposal and the State's Contention 23.

20. For the all reasons given above, I do not believe the Entergy's July 15, 2010 communication to the Board [NL-10-063] concerning a new AMP for RPV internals is adequate to address the new safety concerns and technical issues that I have raised herein.

Pursuant to 28 U.S.C. § 1746, I declare under penalty of perjury that the foregoing is true and correct.

September 15, 2010
Troy, New York

R.T. Lahey

Dr. Richard T. Lahey, Jr.

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Attachment

curricula vitae
Richard T. Lahey, Jr., Ph.D.

September 15, 2010
Lahey Declaration

VITA

Dr. Richard T. Lahey, Jr.

The Edward E. Hood Professor Emeritus of Engineering

Rensselaer Polytechnic Institute

Troy, New York

Education

B.S. Marine Engineering-1961, U.S. Merchant Marine Academy
M.S. Mechanical Engineering-1964, Rensselaer (RPI)
M.E. Engineering Mechanics-1966, Columbia University
Ph.D. Mechanical Engineering-1971, Stanford University

Professional Experience

7/61 - 9/61 Cities Service Company, New York, New York

Third Assistant Engineer - Operating Engineer on "jumbo" tanker, S/S Fort Hoskins also had responsibility for maintenance of all electrical equipment.

9/61 - 8/64 Knolls Atomic Power Laboratory, Schenectady, NY

Engineer - Various assignments on advanced naval nuclear submarine (S5G) design.
Thermal Development Group - Experimental work on DNB and hydrodynamic instability.
Fluid Systems Group - Systems design and documentation.
Safety Analysis Group - Analytical investigation of hypothetical accident conditions development of analog and digital computer models.

9/64 - 6/66 Columbia University, New York, New York

Research Assistant - University research in the area of biomechanics (blood flow, pulmonary mechanics, etc.)

8/61 - 8/67 U.S. Navy

Naval Officer - USNR

7/66 - 6/71 General Electric, San Jose, California

Principal Development Engineer - Responsible for experimental and analytical investigations in two-phase flow and boiling heat transfer phenomena, including: hydrodynamic stability, subchannel analysis, CHF and Pressure drop.

6/71 - 6/72 General Electric Company, San Jose, California

Manager, Heat Transfer Mechanisms - Responsible for applied research in the are of subchannel analysis, transient analysis and detailed Boiling Water Nuclear Reactor (BWR) heat transfer mechanisms.

6/72 - 11/73 General Electric Company, San Jose, California

Manager, Core Development - Responsible for all non-safety related thermal-hydraulic development work in support of the boiling water nuclear reactor.

11/73 - 10/75 General Electric Company, San Jose, California

Manager, Core and Safety Development - Responsible for all heat transfer and fluid flow and reactor physics development work in support of the boiling water nuclear reactor. Responsible for all foreign and domestic safety R&D programs.

10/75 - 6/87 Rensselaer Polytechnic Institute, Troy, New York

Chairman, Department of Nuclear Engineering & Science - Teaching, research and management of academic department concerned with nuclear technology.

5/87 - 4/89 Rensselaer Polytechnic Institute, Troy, New York

Professor, Department of Nuclear Engineering and Engineering Physics, and, Professor, Department of Chemical Engineering - University teaching and research.

4/89 - Present, Rensselaer Polytechnic Institute, Troy, New York

The Edward E. Hood, Jr. Professor of Engineering (4/89-9/08 ;**Emeritus** 9/08-Present), Department of Mechanical, Aerospace & Nuclear Engineering and, Chemical Engineering - University teaching and research.

5/91 - 6/94 Rensselaer Polytechnic Institute, Troy, New York

Director, Center for Multiphase Research - University teaching, research and administration.

7/94 - 3/98 Rensselaer Polytechnic Institute, Troy, New York

Dean of Engineering - Academic administration and research.

Consulting

Argonne National Laboratory

Long Island Lighting Company

Battelle Northwest Laboratories
Brookhaven National Laboratory
Babcock & Wilcox Company
Combustion Engineering (ABB)
Corning, Inc
Creare, Inc.
EG&G Idaho, Inc. (INEL)
Electric Power Research Institute
Exxon Nuclear Company, Inc.
General Electric
General Public Utilities
International Atomic Energy Agency
Air Products

NYS-DEC ; NYS- OAG

Norhtrop Grumman

Nuclear Associates International
Oak Ridge National Laboratory
PJM Interconnection(***Board Member***)
Sandia Laboratories
Savannah River Laboratory
Singer Link-Miles
Stauffer Chemical Company
Stone & Webster
U.S. Department of Energy
U.S. Nuclear Regulatory Commission
Westinghouse (NED)
Yankee Atomic Electric Company

Jason Associates

NYC (Couch White , LLP)

Professional Memberships and Technical Review Groups

American Nuclear Society (ANS)

President, Northeastern New York Section (78-79)

Member, Board of Directors (79-82)

Member, ANS Executive Committee (80-82)

Member, Executive Committee - Power Division (79-82)

Chairman, Technical Group for Thermal-Hydraulics (79-80)

Member, Executive Committee - Thermal-Hydraulics (80-81)

Member, E.E.&A. Accreditation Committee (84-87)

Member, NHTC Coordinating Committee (86-89)

Member, ANS Nominating Committee (86)

American Society of Mechanical Engineers (ASME)

Nucleonics Heat Transfer Committee (ASME K-13)

Chairman (78-81)

American Society of Engineering Education (ASEE)

Chairman, Program Committee (86-87)

Chairman, Nuclear Engineering Division (87-88)

American Institute of Chemical Engineers (AIChE)

Chairman, Energy Transport Field Research Committee (87-91)

Association of Engineering Colleges in New York State (AECNYS)

Secretary/Treasurer (96)

ECPD Council

ASME Representative, (76-79)

Engineering Manpower Commission (EMC)

Commissioner (81-84)

Council on Competitiveness

Member (94-98)

Nuclear Engineering Department Heads Organization (NEDHO)

Chairman (82-83)

Liaison with USNRC and USDOE (82-87)

International Center for Multiphase Flow - Japan

Corresponding Member (USA)

The New York Academy of Sciences (NYAS)

Member (90-09)

Society of the Sigma Xi

Member (70-Present)

Editorial Boards

Journal of Nuclear Engineering & Design (Formerly Editor -NE&D , 83-94)

International Journal of Heat & Mass Transfer

International Communications in Heat & Mass Transfer

Journal of Multiphase Science & Technology

Nuclear Safety Review Board (RPI)

Chairman (76-87)

EG&G Scientific Advisory Committee

Member (76-83)

Review Group Membership

USNRC Advanced Code Review Group (76-84)

USNRC Two-Phase Instrumentation Review Group (76-84)

USNRC Containment Code Review Group (77-84)

USNRC Two-Phase Flow Calibration Review Group (78-84)

USNRC LOFT Review Group (77-83)

USNRC EBTF Research Review Group (79-82)

USNRC 2D/3D Review Group (79-84)

USNRC BWR BDHT Review Group (79-84)

EPRI Design Review Committee Member for MAAP Code (88-93)

EPRI Design Review Committee Member for BWRSAR Code (88-90)

LILCO Peer Review Committee Member (88-92)

USDOE Savannah River Laboratory (SRL) Review Group Member (88-92)

USDOE Advanced Neutron Source (ANS) Review Panel (88-92)

ORNL Engineering Technology Division Advisory Committee - Chairman (89-92)

ORNL Advanced Neutron Source (ANS) Reactor Advisory Committee - Chairman (92-93)

ORNL CASL Science Council - Member (2010 - Present)

National Association of Corporate Directors (NACD) - Member (97-Present)

National Academy Activities

Member, National Research Council (NRC) -Space Science Boards Committee on Microgravity Research (1997-2008)

Member, National Research Council (NRC) Study on: "Microgravity Research in Support of Technologies for the Human Exploration and Development of Space and Planetary Bodies" (1998-2000).

Member, National Research Council (NRC)I Study on: "The Safety and Security of Commercial Spent Fuel Storage" (2004 - 2006).

Member, National Research Council (NRC) Decadal Study on: "Biological and Physical Sciences in Space" (2009 - 2010).

Honors

- Elected Fellow of ANS (1980)
- Elected Life Fellow of ASME (1980)
- Elected ***National Academy of Engineering*** (1994)
- Elected ***Russian National Academy of Sciences-Baskortostan*** (1995)
- Graduated (with Honors) - USMMA (1961)
- Nominated: G.E.s Steinmetz Award (1975)
- Whos Who in Engineering
- Whos Who in the East
- International Whos Who in Energy & Nuclear Sciences
- Whos Who in Technology Today
- American Men & Women of Science (17th Edition)
- The International Whos Who of Intellectuals (Vol. VI)
- *Fulbright-Hays* Fellowship (1983-1984)
- Elected Senior Fellow-Magdalen College of Oxford University (1983-1984)
- Keynote Lecture, 5th Indian Heat & Mass Transfer Conference, Hyderabad, India (1980)
- Editor, ***Journal of Nuclear Engineering and Design***, (1983-1994)
- Appointed IAEA Expert to Assist Argentina in Nuclear Power Research (1985-Present)
- Keynote Lecturer, International Workshop on Two-Phase Flow Fundamentals, NBS, Gaithersburg, MD (1985)
- People-to-People Delegation Leader to the PRC on Nuclear Reactor Safety (11/4-25/85)
- Keynote Lecture, 4th International Symposium on Multi-Phase Transport & Particulate Phenomena - Miami, Florida (1986)
- Keynote Lecturer, International Centre for Heat and Mass Transfer, Dubrovnik, Yugoslavia (1987)
- Chairman, DOE/EPRI Second International Workshop on Two-Phase Flow Fundamentals (3/87)
- Visiting Professor, University of Pisa, Pisa, Italy (1987)
- Visiting Professor , Universite Claude Bernarde, Lyon, France (1987)
- Appointed External Dissertation Reviewer - Univeriti Malaya (1987)
- Visiting Professor of Engineering , Centro Atomico, Bariloche, Argentina (1988)
- Keynote Lecture, Japan Society of Multiphase Flow - Tokyo, Japan (1988)

- Appointed Honorary Senior Fellow - Magdalen College of Oxford University (1989-Present)
- Elected Chairman of RPI Faculty Council (1989-1991)
- Japan Society for the Promotion of Science (JSPS) Fellowship (1990)
- Keynote Lecture, American Society of Mechanical Engineers, Dallas, TX (1990)
- Alpha Nu Sigma Honor Society (1992)
- Plenary Lecturer, International Symposium on Instabilities in Multiphase Flows - Rouen, France (5/92)
- Member, Editorial Advisory Board - ***International Journal of Heat and Mass Transfer***
- Member, Editorial Advisory Board - ***International Communications in Heat and Mass Transfer***
- U.S. Representative, International Information Center for Multiphase Flow (ICeM)
- Mark S. Mills Award of the ANS [Advisee: Susana Kalkach-Navarro] (1993)
- Elected Member of RPI Engineering Research Council (1993-1994)
- General Chairman, International Topical Meeting on Nuclear Reactor Thermal-Hydraulics (NURETH-7), Sept. 10-15, 1995
- Member, Advisory Editorial Board - ***Heat Transfer Research***
- Keynote Lecture, MFTP-2000, Antalya, Turkey (2000)
- Member, Presidium, ICMS-2000, UfA, Russia (2000)
- Listed as an Expert Knowledge Provider , Intota web site, *www.intota.com* (2001)
- Keynote Lecture , HEAT-2002 , Kielce,Poland (2002)
- Member - Engineering Advisory Board , U.S. Merchant Marine Academy (2003-2005)
- Elected to Palmer C. Ricketts Society of Patroons - RPI (2004-present)
- Co-Chair , Japan/US Seminar on Two-Phase Flow Dynamics , Nagahama, Japan (2004)
- Plenary Lecture , PISA' 04 , Pisa , Italy (2004)
- Keynote Lecture , Yadigaroglu Retirement Seminar , ETH-Zurich , Zurich, Switzerland (2004)
- Keynote Lecture, ISMF' 05, Xi'an , China (2005)
- Keynote Lecture, NURETH-11 , Avignon, France (2005)
- ***Alexander von Humbolt*** Senior Scientist Fellowship-FZK (2005-2006)
- Keynote Lecture , NURETH-12 , Pittsburgh , PA. (2007)

Awards

- Meritorious Service Award of the ANS (1983)
- **Glenn Murphy Award** of the ASEE (1985)
- Technical Achievement Award of the ANS (1985)
- United States Merchant Marine Academy Alumni Association, Outstanding Professional Achievement Award (1986)
- ***E.O. Lawrence Memorial Award*** of the USDOE (1988)
- **Arthur Holly Compton Award** of the ANS (1989)
- ***Donald Q. Kern*** Award of the AIChE (1989)
- ***Glenn T. Seaborg*** Medal of the ANS (1992)
- ASME/ANS NHTC Best Paper Award (1993)
- **William H. Wiley Distinguished Faculty Award**-RPI (2004)

Books and Monographs

"Out-of-Pile Subchannel Measurements in a Nine-Rod Bundle for Water at 1000

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"An Analysis of Density-Wave Oscillations in Ventilated BWR Fuel Rod Bundles," Multi-Phase Flow and Heat Transfer III, Part A: Fundamentals, eds. Veziroglu & Bergles, Elsevier, 1984 (with R. Taleyarkhan and M. Podowski).

"Three Dimensional Conical Probe Measurements in Turbulent Air/Water Two-Phase Pipe Flow," TSI Quarterly (Flow Lines), 1987 (with S.K. Wang, S.J. Lee and O.C. Jones, Jr.)

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"A Comprehensive Subgrid Air Entrainment Model for Reynolds-averaged Simulations of Free-Surface Bubbly Flows", Proc. 62nd Meeting of the APS-Division of Fluid Dynamics, Minneapolis, MN , Nov. 22-24, 2009 (with Jingsen Ma , A. Oberai, D. Drew and M. Hyman).

"The Numerical Simulation of Two-Phase Annular Flow", Proc. ICMF 2010, Tampa, Florida, May 30-June 4 , 2010 (with Joseph Rodriguez and Ken Jansen).

"Direct Numerical Simulation of Turbulent Two-Phase Bubbly Channel Flow" , Proc. ICMF 2010, Tampa, Florida, May 30-June 4 , 2010 (with Igor Bolotnov, Ken Jansen , Donald Drew , Assad Oberai and Michael Podowski).

"DES and RaNS Modeling of a 3-D Hydraulic Jump with Air Entrainment using a Two-Fluid Model", Proc. ICMF 2010, Tampa, Florida, May 30-June 4 , 2010 (with Jingsen Ma, Assad Oberai and Donald Drew).

"A Generalized Sub-Grid Air Entrainment Model for RaNS Modeling of Bubbly Flows", Proc. ICMF 2010, Tampa, Florida, May 30-June 4 , 2010 (with Jingsen Ma , Assad Oberai and Donald Drew).

"Sub-grid Air Entrainment Model for RANS and LES Simulations of Free Surface Turbulence Bubbly Flows", Proc. 28th ONR Symposium on Naval Hydrodynamics, Cal Tech , Pasadena , CA , Sept. 12-17, 2010 (with Assad Oberai, Jingsen Ma, and Donald Drew).

" A Detached Direct Numerical Simulation of Two-Phase Turbulent Bubbly Channel Flow" , Proc. 7th Int. Conference on Multiphase Flow (ICMF 2010), Tampa, FL. May30 - June 4, 2010 (with I. A. Bolotnov, K.E. Jansen, D.A. Drew, A.A. Oberai, M.Z. Podowski).

" The Simulation of Air Entrainment in a Hydraulic Jump using Two-Fluid DES and RaNS Models", Proc. 7th Int. Conference on Multiphase Flow (ICMF 2010), Tampa, FL. May30 - June 4, 2010 (with J. Ma, A.A. Oberai, D.A. Drew).

"A Generalized Subgrid Air Entrainment Model for RaNS Modeling of Bubbly Flows around Ship Hulls", Proc. 7th Int. Conference on Multiphase Flow (ICMF 2010), Tampa, FL. May30 - June 4, 2010 (with J. Ma, A.A. Oberai, M.C. Hyman, D.A. Drew).

" ATwo-Way Coupled Polydispersed Simulation of Bubbly Flow Beneath a Plunging Liquid Jet", Proc. ASME Fluids Engineering Division (FED) Summer Meeting , Montreal, Quebec-Canada, August 1-5, 2010 (with J. Ma, A.A. Oberai, D.A. Drew).

"Acoustic Chambers for Sonofusion Experiments : FE - Analysis Highlighting Performance Limiting Factors" , Proc.17th International Congress on Sound and Vibration (ICSV 17), Cario, Egypt , July 18-22, 2010 (with Markus J. Stokmaier, Andreas G. Class, Thomas Schulenberg).

" Influence of Bubbles on Liquid Turbulence Based on the Direct Numerical Simulation of Channel Flows", Proc. 63rd Annual APS Meeting - Division of Fluid Dynamics , Long Beach, CA, Nov. 21-23, 2010 (with Igor Bolotnov, Donald D. Drew and Michael Z. Podowski).

Unrefereed Publications

"Control Rod Oscillator Tests: Garigliano Nuclear Reactor," GEAP-5534, August 1967.

"BWR Stability Considerations Resulting from Garigliano Research and Development Program," International Symposium on Dynamics of Two-Phase Flow, presented at University of Eindohoven, The Netherlands, 1967 (with J. Hodde).

"Representation of Space-Time Velocity and Pressure Fluctuation Correlations by a Tentative Phenomenological Model," Stanford University Report MD-22, August

1968.

"Subchannel and Pressure Drop Measurements in a Nine-Rod Bundle for Diabatic and Adiabatic Conditions," GEAP-13049, March 1970 (with B. Shiralkar, et al)

"A Stochastic Wave Model Interpretation of Correlation Functions for Turbulent Shear Flows," Stanford University Report MD-26, May 1971.

"The Analysis of Transient Critical Heat Flux," GEAP-13249, 1972 (with J. Gonzalez).

"General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application,"NEDO-10958, November 1973.

"A Turbine-Meter Evaluation Model for Two-Phase Transients (TEMPI)," EG&G Idaho, Inc. Topical Report, 1977 (with P. Kamath).

"Transient Analysis of a Drag-Disk in Two-Phase Flow," EG&G Topical, NES-483, 1978 (with P. Kamath and D.R. Harris).

"The Measurement of Phase Separation in Wyes and Tees," USNRC Topical Report, NUREG/CR-0557, 1978 (with T.J. Honan).

"The Development of a Side-Scatter Gamma Ray System for the Measurement of Local Void Fraction," USNRC Topical Report, NUREG/CR-0677, 1978 (with S. Schell).

"A Review of Selected Void Fraction and Phase Velocity Measurement Techniques," Proceedings of the FDI Two-Phase Instrumentation Course, Dartmouth College, 1978.

"The Analysis of Proposed BWR Inlet Flow Blockage Experiments at PBF," EG&G Idaho, Inc., Topical Report, 1978 (with K. Ohkawa).

"Virtual Mass Effects in Two-Phase Flows," USNRC Topical Report, NUREG/CR-0020, 1979 (with L. Cheng and D.A. Drew).

"Flow Patterns & Phase Distribution Phenomena," Invited paper given at Two-Phase Flow Summer Course, Munich, Germany, 1979.

"Two-Phase Flow Instability," Invited paper given at Two-Phase Flow Summer Course, Munich, Germany, 1979.

"The Measurement of Void Fraction and Phase Velocity using Electrical Impedance Probes," Invited paper given at Two-Phase Instrumentation course, Grenoble, France, 1979.

"Radioactive Tagging Techniques in Two-Phase Flow," Invited paper given at Two-Phase Instrumentation Course, Grenoble, France, 1979.

"Photon Attenuation and Scattering Techniques in Two-Phase Flow," Invited paper given at Two-Phase Flow Instrumentation Course, Grenoble, France, 1979.

"Two-Phase Flow Phenomena in Nuclear Regulatory Technology," USNRC Topical Report, NUREG/CR-0677, 1979 (with S. Schell and R.R. Gay).

"Force & Torque Flow Measurement Methods," Proceedings of Stanford Summer Course on Two-Phase Flow Instrumentation, 1980.

"Transit Time Techniques," Proceedings of Stanford Summer Course on Two-Phase Flow Instrumentation, 1980.

"The Design of Photon Attenuation and Scattering Systems," Proceedings of Stanford Course on Two-Phase Flow Instrumentation, 1980.

"Local Void Probes," Proceedings of Stanford Summer Course of Two-Phase Flow Instrumentation, 1980.

"The Analysis of Linear and Nonlinear Instability Phenomena in Heated Channels," USNRC Topical Report, NUREG/CR-1718, 1980 (with J.L. Achard and D.A. Drew).

"Flow Regime Identification and Void Fraction Measurement Techniques in Two-Phase Flow," USNRC Topical Report, NUREG/CR-1692, 1980 (with M.A. Vince).

"An Assessment of the Literature Related to LWR Instability Models," NUREG/CR-1414, 1980 (with D.A. Drew).

"Transient Analysis of DTT Rakes," USNRC Topical Report, NUREG/CR-2151, 1981 (with P.S. Kamath).

"The Analysis of Countercurrent Two-Phase Flow Pressure Drop and CCFL Breakdown in Diabatic and Adiabatic Conduits," NUREG/CR-2386, 1981 (with A. Ostrogorsky and R.R. Gay).

"Parallel Channel Effects During the Emergency Core Cooling of a BWR," Proceedings of the 9th Water Reactor Safety Information Meeting, Washington, DC 1981.

"Transient and Sustained Instabilities in Multiphase Flows," Proceedings of the 2nd Multiphase Flow and Heat Transfer Symposium Workshop, 1981 (with J.L. Achard).

"The Measurement of Two-Dimensional Phase Separation Phenomena," USNRC Topical Report, NUREG/CR-1936, 1981 (with M. Barasch).

"Two-Fluid or Not Two-Fluid," Guest Column, ***Heat Transfer and Fluid Flow Service,*** UKAEA, UK,1981.

"The Analysis of Proposed BWR Inlet Flow Blockage Experiments Using

MAYU4b," USNRC Topical Report, NUREG/CR-2260 and EG&G Topical Report, EGG-2181, 1982 (with M.E. Nissley and R.R. Gay).

"The Analysis of Pulsed Neutron Activation Technique," USNRC topical Report, NUREG/CR-2471, 1981 (with M.L. Griffo and R.C. Block).

"An Experimental Investigation of Boiling Water Nuclear Reactor Parallel Channel Effects During a Postulated Loss-of-Coolant Accident," USNRC Topical Report, NUREG/CR-2971, 1982 (with W.M. Conlon).

"An Analysis of Density-Wave Oscillations in Ventilated Channels," USNRC Topical Report, NUREG/CR-2972, 1982 (with R. Taleyarkhan and M. Podowski).

"Phase Separation Phenomena in Branching Conduits," USNRC Topical Report, NUREG/CR-2590, 1982 (with N. Saba).

"The Development of NUFREQ-N, An Analytical Model for the Stability Analysis of Nuclear Coupled Density-Wave Oscillations in Boiling Water Nuclear Reactors," USNRC Topical Report, NUREG/CR-3375, 1983 (with G.C. Park, M. Podowski and M. Becker).

"An Analysis of Wave Dispersion, Sonic Velocity and Critical Flow in Two-Phase Mixtures," USNRC Topical Report, NUREG/CR-3372, 1983 (with L. Cheng and D.A. Drew).

"Air/Water Subchannel Measurements of the Equilibrium Quality and Mass Flux Distribution in a Rod Bundle," USNRC Topical Report, NUREG/CR-3373, 1983 (with R. Sterner).

"Parallel Channel Effects and Long-Term Cooling During Emergency Core Cooling in a BWR/4," USNRC Topical Report, NUREG/CR-3376, 1983 (with M. Fakory).

"The Development of Gamma Ray Scattering Densitometer and Its Application to the Measurement of Two-Phase Density Distribution in an Annular Test Section," USNRC Topical Report, NUREG/CR-3374, 1983 (with K. Ohkawa).

"An Analysis of Boiling Water Nuclear Reactor Stability Margin," USNRC Topical Report, NUREG/CR-3291, 1983 (J. Balaram, C.N. Shen and M. Becker).

"The Measurement of Phase Distribution Phenomena in a Triangular Conduit," USNRC Topical Report, NUREG/CR-3576, 1983 (with S. Sim).

"Mechanistic Core-Wide Meltdown and Relocation Modeling for BWR Applications," NUREG/CR-3525, 1983 (with M.Z. Podowski and R. Taleyarkhan).

"Mathematical Modeling of U-Tube Steam Generator Dynamics for Slow Transients and Small Break Loss-of-Coolant Accidents," EPRI report RP11, 63-5, 1983.

"The Measurement of Countercurrent Phase Separation and Distribution in a Two-Dimensional Test Section," USNRC Topical Report, NUREG/CR-3577, 1984 (with K.M. Bukhari).

"Current Understanding of Phase Separation Mechanics in Branching Conduits," Proceedings of the U.S.-Japan Seminar on Two-Phase Flow Dynamics, Lake Placid, NY 1984.

"Advances in Analytical Modeling of Linear and Nonlinear Density-Wave Instability Modes," Proceedings of the U.S.-Japan Seminar on Two-Phase Flow Dynamics, Lake Placid, NY 1984.

"Modeling Two-Phase Flow Division at T Junctions," Proceedings of the H.T.F.S. Symposium, Coventry, England, 1984 (with B. Azzopardi and M. Cox).

"NUFREQ-NP: A Digital Computer Code for the Linear Stability Analysis of Boiling Water Nuclear Reactors," NUREG/CR-4116 USNRC Topical Report, 1984 (with S.J. Peng and M.Z. Podowski).

"Analytical Methods for Multicomponent Systems," Proceedings of Workshop on Industrial Applications of Multiphase Flow, UCSB, 1985.

"Light Water Nuclear Reactor LOCA Technology," Proceedings of Workshop on Industrial Applications of Multiphase Flow, UCSB, 1985.

"Condensation Heat Transfer," Proceedings of the RPI Summer Course on Two-Phase Heat and Mass Transfer in Single and Multicomponent Systems, 1985.

"Multicomponent Condensation," Proceedings of the RPI Summer Course on Two-Phase Heat and Mass Transfer in Single and Multicomponent Systems, 1985.

"Multicomponent Boiling," Proceedings of the RPI Summer Course on Two-Phase Heat and Mass Transfer in Single and Multicomponent Systems, 1985.

"The Modeling of BWR Core Meltdown Accidents - For Application in the MELRPI.MOD2 Computer Code," NUREG/CR-3889, 1985 (B.R. Koh, S.H. Kim, R. Taleyarkhan and M.Z. Podowski).

"Basic Conservation Equations," Proceedings of the RPI Summer Course on Computer Simulation of Multiphase Flows, 1986.

"Interfacial Transfer Laws," Proceedings of the RPI Summer Course on Computer Simulation of Multiphase Flows, 1986.

"Closure Conditions for Two-Fluid Models of Two-Phase Flow," Proceedings of the Sixth Symposium on Energy Engineering Sciences, ANL, 1988 (with G. Arnold and D.A. Drew).

"The Relationship Between Microstructure and the Averaged Equations of Two-

Phase Flow," EUROMECH 234, Toulouse, France, May, 1988 (with G. Arnold and D.A. Drew).

"The Analysis of Phase Separation Phenomena in Branching Conduits," Proceedings of the JAPAN/US Seminar on Two-Phase Flow Dynamics, Kyoto, Japan, July 1988.

"An Analysis of Wave Propagation Phenomena in Two-Phase Flow," Proceedings of the JAPAN/US Seminar on Two-Fluid Flow Dynamics, Kyoto, Japan, July, 1988

"Phase Distribution and Phase Separation Phenomena in Two-Phase Flows," Proceedings of the Japan Society of Multiphase Flow, 1988.

"An Analysis of Wave Propagation Phenomena in Two-Phase Flow," Proceedings of RPI Summer Course of Modern Developments in Boiling Heat Transfer and Two-Phase Flow, 1988.

"An Analysis of Phase Distribution Phenomena in Two-Phase Flow," Proceedings of RPI Summer Course of Modern Developments in Boiling Heat Transfer and Two-Phase Flow, 1988.

"An Analysis of Phase Separation in Branching Conduits," Proceedings of RPI Summer Course of Modern Developments in Boiling Heat Transfer and Two-Phase Flow, 1988.

"The Development of APRIL.MOD2 - A Computer Code for Core Meltdown Accident Analysis of Boiling Water Nuclear Reactors," NUREG/CR-5157, July, 1988 (with S. Kim, et al).

"An Analysis of Wave Propagation Phenomena in Two-Phase Flow," Proceedings of RPI Summer Course of Modern Developments in Boiling Heat Transfer and Two-Phase Flow, 1989.

"An Analysis of Phase Distribution Phenomena in Two-Phase Flow," Proceedings of RPI Summer Course of Modern Developments in Boiling Heat Transfer and Two-Phase Flow, 1989.

"An Analysis of Phase Separation in Branching Conduits," Proceedings of RPI Summer Course of Modern Developments in Boiling Heat Transfer and Two-Phase Flow, 1989.

"Degraded BWR Core Modeling - Physical Simulations of Selected Components," ESEERCO EP84-4 Final Report, September 1989 (with M.Z. Podowski).

"The Analysis of Void Wave Phenomena," Proceedings of the Eighth Symposium on Energy Engineering Sciences, pp. 27-34, ANL Report CONF-9005183, 1990 (woth J-W. Park and D.A. Drew).

"Degraded BWR Core Modeling - APRIL.MOD3 Severe Accident Code," ESEERCO EP84-4 Final Report, July 1990 (with M.Z. Podowski).

"Multiphase Thermal Science," Proceedings of the NSF Workshop on Thermal Sciences, Chicago, April 19-21, 1991.

"A Four Field Model for Two-Phase Flow," 12th Symposium on Energy Engineering Sciences, 4/27-29/94, Argonne National Laboratory (with D.A. Drew).

"Synchronic Nonlinear Forcing of a Sonoluminescent Microbubble using Fast Ultrasonic Pulses," Proceedings of the APS, March 1996 (with F.J. Bonetto and G.A. Delgadino).

"A CFD Analysis of Multidimensional Two-Phase Flow and Heat Transfer Using a Four Field Two-Fluid Model," Proceedings of the Thirteenth U.S. National Congress on Applied Mechanics, U of Florida, June 21-26, 1998.

"A CFD Analysis of Multidimensional Two-Phase Flow & Heat Transfer with a Four Field Two-Fluid Model," Proceedings of IMUST Meeting, Santa Barbara, CA, March 18-20, 1999.

"A Center-Averaged Two-Fluid Model for Wall-Bounded Flows," ONR Free Surface and Bubbly Flows Workshop, La Jolla, CA, Feb. 24-26, 1999 (with A.E. Larreteguy and D.A. Drew).

"Multidimensional Two-Fluid Modeling of Two-Phase Flow and Heat Transfer In a Boiling Channel with Applications to CHF Modeling in Forced-Convection Sucooled Boiling," National Science Agency of Tiawan Report, August 1999 (with C. Pan and D. A. Drew)

"An Analysis of Two-Phase Flow and Heat Transfer using a Multidimensional, Multi-Field, Two-Fluid Computational Fluid Dynamics (CFD) Model", Proceedings of the Japan/US Seminar on Two-Phase Flow Dynanmics, Santa Barbara, California, June 5-8, 2000 (with D.A. Drew).

"An Analysis of Rectified Diffusion in a Sonoluminescing Gas Bubble",, Proceedings of the Japan/US Seminar on Two-Phase Flow Dynamics, Santa Barbara, California, June 5-8, 2000 (with S. Bae and R. Nigmatulin).

"On the Multidimensional Analysis of Two-Phase Flows" , Proceedings of the USDOE Workshop on Scientific Issues in Multiphase Flow , U. Illinois-CU ,May 7-9 , 2002 (with D. Drew).

"Sonoluminescence and the Search for Sonoluminescence" , ANS Panel on Advances in Fusion Technology , ANS Annual Meeting , Hollywood , Florida , June 9-13 , 2002

"Response - Tabletop Fusion Revisited (by: A. Galonsky)" , *Science* on-line , 2002 (with R. Taleyarkhan , R. Block and C. West).

"Response - Questions Regarding Nuclear Emissions in Cavitation Experiments (by: M. Saltmarsh and D. Shapira)" , *Science* on-line , 2002 (with R. Taleyarkhan , R. Block and C. West).

"Energetics of Nano-to-Macro Scale Triggered Tensioned Metastable Fluids", ORNL/TM-2022/233 , 2002 (with R. Taleyarkhan , C. West , J. Cho and I. Akhatov).

"The Modeling of Bubbly Flows Around Ship Hulls" , Maui High Performance Computing Center, Application Brief , 2002 (with F. Moraga and D. A. Drew).

"Full-Scale Simulations of the Research Ship Roger Revelle", Maui High Performance Computing Center, Application Brief , 2003 (with F. Moraga and D. A. Drew).

"The Development of Interfacial Drag and Non-Drag Laws for Stratified Flow using PHASTA-2I" , Proceedings of the American Physical Society, East Rutherford , NJ ,Nov.23-25 , 2003

"Computational Multiphase Fluid Dynamics (CMFD) Analysis of a Single ESBWR Riser Channel," ISL Final Topical Report, 2004 (with S. Antal, M. Popowski).

"Research in Support of the Use of Rankine Cycle Energy Conversion Systems for Space Power and Propulsion," NASA/CR-2004-213142, 2004 (with V. Dhir)

"Safety and Security of Commercial Spent Nuclear Fuel Storage," Classified National Research Council (NRC) Topical Report, 2004.

"Nuclear Engineering External Review Committee Report," Purdue University Report, 2004.

"The CMFD Analysis of Three-Field Chemical Reactors," CREL Topical Report, 2004 (with S. Antal).

"The Sonofusion Research Project at KIT and RPI" , Proceedings of the 62nd Meeting of the American Physical Society - Division of Fluid Dynamics, Minneapolis, Minnesota, November 22-24, 2009 (with Markus Stokmaier, Bernard Malouin, Andreas Class , Thomas Schulenberg).

Special Courses Taught

- RPI Summer Program on Nuclear Reactor Design & Basic Nuclear Technology (RPI sponsored), Troy, NY 1997-1983
- Short course in Introduction to Nuclear Power, Continuing Education Center, (CEC sponsored) Sheraton Motor Inn, East Brunswick, NJ, 1978
- Two-Phase Flow and Heat Transfer (B&W sponsored), Alliance, OH, 1978
- Two-Phase Flow and Heat Transfer (EG&G sponsored), Idaho Falls, ID, 1979-1983
- Two-Phase Flow Instrumentation course (FDI sponsored), Dartmouth University, 1978

- Multiphase Flow Instrumentation course (CEA sponsored), Grenoble, France, 1979
- Workshop on Transient Analysis of Reactors (FRG sponsored), Munich, Germany, 1979
- Reactor Thermal-Hydraulics, AIChE short course, 1976 -1983
- Stanford summer course on Two-Phase Flow Instrumentation, 1980
- Course on Two-Phase Flow and Boiling, Yankee Atomic Electric Company, 1980
- Summer school on Reactor Thermal-Hydraulics (ICHMT sponsored), Dubrovnik, Yugoslavia, 1980
- Stanford summer course on Two-Phase Flow & Heat Transfer (Stanford sponsored), Stanford University, 1982
- Simposio Internacional Sobre Flujos Bifasicos en Tuberias (Mexican sponsored), Cuernavaca, Mexico, 1983
- Lecture Series No. 8, Construction Aspects of Two-Phase Flow Equipment (Norwegian sponsored), Trondheim, Norway, 1984
- Workshop on Industrial Applications of Multiphase Flow (UCS sponsored), Santa Barbara, CA 1985
- Workshop on Two-Phase Heat and Mass Transfer in Single and Multicomponent Systems (RPI sponsored), Troy, NY 1986
- Modern Developments in Boiling Heat Transfer and Two-Phase Flow (CMR sponsored), Troy, NY 1988-present
- An Introduction to Applied Nonlinear Dynamics - Bifurcations, Fractals and Chaos in Heat Transfer and Fluid Flow (ETH sponsored), Zurich, Switzerland, 1994-1996
- Short Course on Multiphase Flow and Heat Transfer (ETH sponsored), Zurich, Switzerland, 1994-1996
- 2001 Frederic Joliot/Otto Hahn Summer School , Karlsruhe , Germany , August 20-29 , 2001
- Short Course on "Transient Multiphase Flow and Heat Transfer at Microgravity" , NASA , Glenn Resrarch Center , Cleveland , Oh. Sept. 17-19 , 2002 (with M.Z. Podowski)

Research Funding

USNRC

Two-Phase Flow Phenomena in Nuclear Reactor Technology

$1,006,240 6/1/76-5/31/80.

Technical Assistance Program for the Thermal-Hydraulic Stability Analysis Relating to Light Water Nuclear Reactors
$676,425 3/15/76-94/83.

Multidimensional Effects in LWR Thermal-Hydraulics
$176,000 6/1/80-1/31/81.

The Development of Thermal-Hydraulic Stability Methods for BWR's
$100,000 9/5/81-9/4/83.

ONR

An Experimental Study of Plunging Liquid Jet Induced Air Carryunder and Dispersion
$122,883 11/1/90-10/30/91.

A Study of Spreading Two-Phase Jets
$120,000 1/1/94-12/31/95.

A Study of Spreading Two-Phase Jets
$138,892 3/1/95-12/31/95.

Bubbly Flow Dynamics and Numerical Implementation in Complex Flows
$707,701 2/1/96-6/30/2000.

The Modeling of Two-Phase Flow Around Ship Hulls
$900,841 July 1, 2001 - June 30 , 2006

The Modeling of Two-Phase Flow Around Ship Hulls
$1,1280,000. July 1, 2006 - June 30 , 2010.

EG&G

An Investigation of Turbine-Meter Drag Disc Devices in Transient Two-Phase Flow
$11,000 10/1/76-9/30/77.

Analysis of BWR Inlet Flow
$20,000 10/1/77-9/30/78.

An Investigation of Turbine-Meter Drag Disc Devices in Transient Two-Phase Flow
$121,764 10/15/76-9/30/79.

Analysis of BWR Inlet Flow
$78,549 2/1/80-1/31/81.

The Analysis of PNA Techniques
$29,323 2/1/80-2/28/81.

The Development of a Global Transient Model for DTT Rakes
$55,764 10/1/78-9/30/79.

The Analysis of BWR Inlet Flow Blockage using MAYU-4B
$81,258 1/1/80-12/31/80.

NSF

Three-Dimensional Turbulence Structure Measurement in Two-Phase Flow

$218,000 3/1/81-11/30/83.

Three-Dimensional Turbulence Structure Measurements in Two-Phase Flow
$78,800 5/1/82-4/30/83.

Three-Dimensional Turbulence Structure in Two-Phase Flow Measurements
$77,700 5/1/83-4/20/84.

Phase Separation Mechanisms in Branching Conduits
$175,000 12/1/84-1/31/87.

Phase Distribution Phenomena in Complex Geometry Conduits
$85,655 3/1/88-2/28-89.

A Study of Phase Separation in Branching Conduits
$95,000 2/1/89-1/31/90.

Phase Distribution Phenomena in Multiphase Systems
$88,446 9/1/89-8/31/90.

The Modeling of Two-Phase Turbulence
$252,351 11/1/01 -10/31/04

ORNL

The Development of Mechanistic Models for the MARCH-based Analysis of BWR Cores
$30,950 9/1/81-8/31/82.

The Development of Mechanistic Models for the MARCH-based Analysis of BWR Cores
$117,990 9/1/81-8/31/83.

The Development of Mechanistic BWR Hydraulics and Structural Component Failure Models for the MARCH Code
$46,000 9/1/82-8/31/83.

Development of Improved Models for BWR Thermal-Hydraulics and Core Degradation Phenomena
$170,748 9/1/83-8/31/85.

Perform Bubble Fusion Analysis and Experiments at RPI and ORNL
$95,000 4/3/98-9/30/99.

Analysis of Sonoluminescence/Sonofusion Phenomena to Support ORNL Experiments
$164,784 , 7/28/99-7/27/2003.

Westinghouse

The Analysis of Thermal-Hydraulic Instabilities in Quad-Plus Fuel
$97,445 9/1/81-8/31/82.

EPRI

The Development of Analytical Modules for Nuclear Reactor Simulators
$49,085 6/1/81-5/31/82.

Workshop on Two-Phase Flow Fundamentals
$15,000 9/1/86-6/30/87.

ESEERCO

An Analysis of BWR/4 and BWR/5 Pressure Boundary Failure Modes During Core Meltdown and Its Impact on Mark-II Containment
$304,667 2/1/84-1/31/86.

The Propagation and Failure Modes in Severely Degraded BWR Cores
$210, 568 9/1/85-8/31/86.

Degraded BWR Code Modeling: Radionuclide Transport and Additional Thermal Hydraulics Models for the APRIL Code
$399,608 6/1/87-5/31/88.

Modeling and Analysis of Severe Accidents in BWRs Using the APRIL

Computer Code
$356,712 1/1/90-12/31/90.

Degraded BWR Code Modeling: The Upgrading and Validation of APRIL as an Interactive Computer Code for BWR Severe Accident Analysis
$428,862 1/1/92-12/31/92.

USDOE

An Analysis of the Closure Conditions for Two-Fluid Models of Two-Phase Flow
$115,000 4/1/86-3/31/87.

Workshop on Two-Phase Flow Fundamentals
$98.000 9/1/86-6/30/87.

An Investigation of the Closure Conditions for Two-Fluid Models of Two-Phase Flow
$120,000 4/1/87-3/31/88.

An Analysis of the Closure Conditions for Two-Fluid Models of Two-Phase Flow
$120,000 4/1/88-3/31/89.

The Continuum Modeling of Two-Phase Systems
$532,088 4/1/89-3/31/93.

A Nonintrusive Measurement System for Multiphase Flows
$134,549 6/30/89-6/29/90.

Analysis of Nuclear Reactor Instability Phenomena
$83,278 6/1/91-5/31/92.

The Continuum Modeling of Two-Phase Systems
$128,000 4/1/92-3/31/93.

Analysis of Nuclear Reactor Instability Phenomena
$88,384 4/15/93-4/14/94.

The Development of Multidimensional Two-Fluid Modeling Capabilities
$129,551 4/1/94-3/31/97.

Multidimensional Analysis of Bubble Dynamics Associated with Bubble Fusion Phenoma
$404,203 7/1/99-6/30/2002.

KAPL

The Development of Improved Models for BWR Thermal-Hydraulics and Core Degradation Phenomena

$50,000 9/1/85-8/31/85.

Turbulent Phenomena in Two-Phase Flows
$32,000 4/15/86-4/14/87.

The Development of Improved Models for BWR Thermal-Hydraulics and Core Degradation Phenomena
$14,997 9/1/86-8/31/87.

A Review of KAPL Methodology in the Area of Multiphase Flow and Heat Transfer
$30,163 7/8/91-7/7/92.

The Mechanistic Analysis of Critical Heat Flux Using Two-Fluid Models
$73,796 4/1/93-9/30/93.

The Mechanistic Analysis of Critical Heat Flux Using Two-Fluid Models
$184,529 10/1/93-9/30/94.

The Mechanistic Analysis of Critical Heat Flux Using Two-Fluid Models
$149,080 10/1/94-9/30/95.

The Mechanistic Analysis of Critical Heat Flux Using Two-Fluid Models
$80,300 12/1/95-4/30/96.

The Analysis of Annular Two-Phase Flows
$227,049 7/1/03-6/30/06

CAAPS

Enabling Technology for Multiphase Flow Food Processing
$48,927 9/1/92-8/31/93.

Enabling Technology for Multiphase Flow Food Processing
$51,305 9/1/93-8/31/94.

Enabling Technology for Multiphase Flow Food Processing
$54,333 9/1/94-8/31/95.

NASA

The Analysis of Phase Distribution Phenomena in Microgravity Environments
$119,438 5/1/92-4/30/93.

The Analysis of Phase Distribution Phenomena in Microgravity Environments

$120,000 12/1/93-11/30/95.

The Analysis of the Effect of Gravity on Multiphase Flows
$396,544 7/26/04 - 9/30/07

GE

Two-Phase Flow Analysis
$10,152 7/15/92-11/1/92.

Two-Phase Flow Analysis
$24,180 12/14/92-3/1/93.

The Development of Multidimensional Modeling Capabilities for Annular Flows in Heated Assemblies
$203,756 8/1/96-9/1/98.

Master's Degrees

Completed

- Vea, Henry W., "An Exact Analytical Solution of Pool Swell Dynamics During Depressurization by the Method of Characteristics" (1977)
- Kamath, Pradeep, "A Turbine Meter Evaluation Model for Two-Phase Transients" (1977)
- Sim, Suk Ku, "An Analysis of Phase Distribution Mechanisms in Turbulent Two-Phase Pipe Flow" (1977)
- Cheng, Lap-Yan, "Virtual Mass Effects in Two-Phase Flow" (1977)
- Saba, Nematollah, "An Experimental Technique for the Determination of Steam/Air Fraction" (1977)
- Ohkawa, Katsu, "The Analysis of BWR Inlet Flow Blockage" (1978)
- Schell, Susan, "The Development of A Sid-Scatter Gamma Ray Technique for the Measurement of Void Fraction" (1978)
- Lombardo, Nicholas, "The Instrumentation and Data Processing of a Large Freon Loop (1978)
- Honan, Timothy J., "Phase Separation in Wyes and Tees" (1978)
- Shum, Feibiu, "The Development of a 4-Equation Drift-Flux Computer Code (DRIFT-4)" (1978)
- Drozd, Andrew , "NUFREQ-S: A Frequency Domain Drift-Flux Technique for the Evaluation of the Thermal-Hydraulic Stability of Boiling Systems" (1980)
- Ostrogorsky, Alexander ,"The Analysis of Countercurrent Two-Phase Flow Pressure Drop and CCFL Breakdown in Diabatic and Adiabatic Conduits" (1980)
- Nissley, Mitchell ,"The Analysis of Proposed BWR Inlet Blockage Experiments Using MAYU-4b" (1981)
- Antal, Steven, "An Evaluation of Boiling Water Nuclear Reactor Stability Using Systems Pressure Perturbation (1982)
- Balaram, J., "Stability Margins for Boiling Water Nuclear Reactors: A Multi-Input Multi-Output Analysis" (1982)

- Shatas, Jr., Joseph F., "A Streak Photography Method for Use in Two-Phase Flow" (1982)
- Lee, Sung Jin, "The Development of a Digital Data Processing System for Two-Phase Turbulence Data" (1982)
- Sterner, R.W., "Air/Water Subchannel Measurements of the Equilibrium Quality and Mass Flux Distribution in a Rod Bundle" (1983)
- Petzold, J.F., "Mathematical Modeling of U-Tube Steam Generator Dynamics for Slow Transients and Small Break for Loss-of-Coolant Accidents (1983)
- Koh, Byung R., "The Model of Emergency Core Cooling Systems for Severe Accident Analysis of Boiling Water Reactors (BWRs)" (1984)
- Bryam, Morris, "Modification of Brookhaven National Lab BWR Plant Analyzer to Accommodate BWR-2 Design of Nine Mile Point-1 Nuclear Power Station" (1987)
- Campanella, Rick, "Wave Propagation Phenomena in Bubbly Two-Component Two-Phase Flows" (1988)
- Kalkach-Navarro, Susana, "A Mechanistic Analysis of Phase Separation in Branching Conduits Using Three-Dimensional Two-Fluid Models" (1988)
- Kimpland, Robert, "A Phenomenological Model for the Prediction of Phase Separation in Branching Conduits" (1990)
- Haley, Thomas C., "A Characteristic Analysis of Mathematical Models of Two-Phase Flow in One Space Dimension" (1990)
- Mayo, Stephen, "Void-Fraction and Velocity Measurements in Three-Phase Flow Using a Hot-Film Anemometer" (1994)
- Potterf, Samuel, "Modeling of Heat Transfer Through Stratified Multiphase Flow in a Horizontal Pipe Using Drift-Flux Techniques, with Special Applications to Aseptic Food Processing" (1995)
- Bolotnov, I., "A Non-Linear Spectral Model for Bubble-Induced Turbolent Viscosity in Bubbly Two-Phase Flow" (2003)
- Saglime, Frank, "Experimental Results for the RPI Bubble Fusion Project" (2004)
- Trofimova , Alisa, "The Direct Numerical Simulation of Turbulent Channel Flows" (2007)

In Progress

- None

Doctoral

Completed

- Vince, Mark A., "Flow Regime Identification and Void Fraction Measurement Techniques in Two-Phase Flow" (1980)
- Kamath, Pradeep, "The Transient Analysis of DDT Rakes" (1981)
- Saba, Nematollah, "Phase Separation Phenomena in Branching Conduits (1981)
- Perez-Griffo, Maria, "Analysis of the Pulsed Neutron Activation Technique (1981)
- Conlon, William M., "An Experimental Investigation of Boiling Water Nuclear Reactor parallel Channel Effects during a Postulate Loss-of-Coolant Accident (1982)
- Taleyarkhan, Rusi P., "An Analysis of Density-Wave Oscillations in Ventilated Channels (1982)
- Ohkawa, Katsuhiro, "The Development of a Gamma-Ray Scattering Densitometer and its Application to the Measurement of Two-Phase Density Distribution in an Annular Test Section" (1983)
- Cheng, Lap-Yan, "An Analysis of Wave Dispersion, Sonic Velocity and Critical Flow in Two-Phase Mixtures" (1983)

- Park, Goon C., "The Development of NUFREQ-N, an Analytical Model for the Stability of Nuclear Coupled Density-Wave Oscillations in Boiling Water Nuclear Reactors" (1983)
- Fakory, Mohammad, "Parallel Channel Effects and Long-Term Cooling During Emergency Core Cooling in a BWR/4" (1984)
- Peng, Shie-Jeng, "NUFREQ-NP - A Digital Computer Code for the Linear Stability Analysis of Boiling Water Nuclear Reactors" (1984)
- Lin, Thomas Fu Yuan, "An Application of Pulsed Photon Activation Technique in Single- and Two-Phase Flow Measurements (1984)
- Bukhari, Khalid M., "Phase Separation and Distribution Phenomena in a Two-Dimensional Test Section" (1985)
- Wang, Shou-Kong, "Three-Dimensional Turbulence Structure Measurements in Air/Water Two-Phase Flow" (1985)
- Koh, Byung R., "The Modeling of BWR Thermal-Hydraulics for Severe Accident Analysis (1986)
- Sim, Suk K., "Phase Distribution Phenomena in Triangular Conduits (1986)
- Hwang, Sun-Tack, "A Study on Phase Separation in Branching Conduits" (1986)
- Lee, Sung-Jin,"Turbulence Modeling in Bubbly Two-Phase Flows" (1986)
- Ruggles, Arthur E., "The Propagation of Pressure Perturbations in Bubbly Air/Water Flows" (1987)
- Kim, Seok H. ,"The Modeling and Analysis of BWR Severe Accidents (1987)
- Arnold, Gary S., "Entropy and Objectivity as Constraints Upon Constitutive Equations for Two-Fluid Modeling of Multiphase Flows" (1988)
- SiAhmed, El-Khider, "An Experimental and Theoretical Investigation of Melt Propagation in a Rubble Bed with Applications to Degraded Nuclear Reactor Cores" (1988)
- Kim, Dong-Ha, "An Analysis of Corium/Structure Interaction during Severe Accidents in Boiling Water Nuclear Reactors" (1990)
- Lopez de Bertodano, Martin, "Turbulent Bubbly Two-Phase Flow in a Triangular Duct" (1992)
- Park, Jee-Won, "Void Wave Propagation in Two-Phase Flow" (1992)
- Cho, Chang-Sok, "Modeling and Analysis of Radionuclide Transport during Severe Accidents in Boiling Water Nuclear Reactors" (1992)
- Valenti (Kalkach-Navarro), Susana, "The Mathematical Modeling of Flow Regime Transition in Bubbly Two-Phase Flows" (1992)
- Jia, Hong, "Multidimensional Effects During Corium Spreading and Corium-Concrete Interaction in BWRs" (1992)
- Yoon, Juh-Yeon,"A Streamline Upwinding Finite Element Method for Multiphase Flow and Heat Transfer Analysis" (1993)
- Antal, Stephen P., "Phase Distribution in Bubbly Two-Phase Flows" (1994)
- Alajbegovic, Ales, "Phase Distribution and Turbulence Structure for Solid/Fluid Upflow in a Pipe (1994)
- Chang, Chin-Jang, "The Analysis of Chaotic Instabilities in Boiling Systems" (1994)
- Kim., Shin-Whan, "Computer Simulation of Severe Accidents in Boiling Water Nuclear Reactors" (1994)
- Assad, Amir, "An Experimental Study of Phase Distribution and Turbulence Structure for Solid/Liquid Flow in Horizontal and Vertical Pipes" (1995)
- Garea, Veronica B., "Nodal Analysis of Two-Phase Flow Instabilities" (1998)
- Moraga, Francisco, "Lateral Forces on Rigid Spheres in a Turbulent Uniform Shear Flow" (1998)
- Nigmatulin, Tagir, "An Experimental and Theoretical Study of Interfacial Area Density for different Two-Phase Flow Regimes" (1999)
- Bae, Song H., "A Theoretical Investigation of the Dynamics of Single Bubble

Sonoluminescence" (1999)
- Delgadino, Gerardo, "The Dynamics of Single Bubble Sonolominescence" (1999)
- Singhal, Maneesh, "The Effect of Dispersed Phase Buoyancy on Phase Distribution" (2004)
- Yin, Juan, "Modeling and Analysis of Multiphase Instabilities" (2004)
- Cancelos , Silvina, "Effect of Acoustically-Induced Pressures on the Permeability of a Bullfrog Urinary Bladder" (2007)
- Galimov , Azat , "An Analysis of Interfacial Waves and Air Ingestion Mechanisms" (2007)
- Bolotnov, Igor , " Cascade Modeling of Single and Two-Phase Turbulence" (2008)
- Ortega-Gomez , Tino, "Stability Analysis for High Performance Light Water Reactor" - U. Karlsruhe , Germany (2008)
- Rodriguez , Joseph M., "Numerical Simulation of Two-Phase Annular Flow" (2009)
- Schlichting, William R. , " An Analysis of the Effect of Gravity on Interacting DWO/PDO Instability Modes" (2009)

In Progress

- Markus Stokmaier (FZK)

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

ATOMIC SAFETY AND LICENSING BOARD

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In re:	Docket Nos. 50-247-LR and 50-286-LR
License Renewal Application Submitted by	ASLBP No. 07-858-03-LR-BD01
Entergy Nuclear Indian Point 2, LLC, Entergy Nuclear Indian Point 3, LLC, and Entergy Nuclear Operations, Inc.	DPR-26, DPR-64 September 15, 2010

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CERTIFICATE OF SERVICE

I hereby certify that on September ~~9~~ 15, 2010, copies of (1) the State of New York's Motion for Leave to File Additional Bases For Previously-Admitted Contention NYS-25, (2) the State of New York's Additional Bases For Previously-Admitted Contention NYS-25, and (3) Declaration of Dr. Richard Lahey were served upon the following persons via U.S. Mail and e-mail at the following addresses:

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John J. Sipos

Dated at Albany, New York
this 15th day of September 2010