October 24, 2003

MEMORANDUM TO: William D. Travers
Executive Director for Operations

FROM: Ashok C. Thadani, Director ***/RA/***
Office of Nuclear Regulatory Research

Janice Dunn Lee, Director
Office of International Programs

SUBJECT: SUMMARY OF 2nd NRC-AERB NUCLEAR SAFETY PROJECTS MEETING

The 2nd NRC-AERB Nuclear Safety Projects Meeting was held between the U.S. Nuclear Regulatory Commission (NRC) and the Atomic Energy Regulatory Board of India (AERB) at the NRC Headquarters from September 8-17, 2003. Presentations were made by the RES and NRR staff and by the Indian delegates in the areas of license renewal and aging, design modifications, fire safety, PRA technology, and Emergency Operating Procedures. All discussions and NRC staff presentations were based on publicly available information. We also arranged tours of the NIST fire safety facility, the University of Maryland, and the NRC Operations Center. A tour of the Surry Nuclear Power Plant was arranged by NRR. A summary of the meeting and the individual presentations is attached.

Meeting discussions were viewed as mutually beneficial and further focused dialogue was accepted. A proposal is being prepared to seek funding from the Indo-U.S. Science and Technology Forum.

We will keep you informed of progress in this interaction.

Attachment: As stated

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DATE	10 /20/03		10/24/03		10/24/03	10/24/03	10/24/03	

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2nd NRC-AERB Nuclear Safety Projects Meeting
Held in
Rockville, Maryland
September 8 – 17, 2003

Record of Discussions

The 2nd NRC-AERB Technical Discussion Meeting on Nuclear Safety Topics between the U.S. Nuclear Regulatory Commission and the Atomic Energy Regulatory Board of India was held at the Nuclear Regulatory Commission offices in Rockville, Maryland, from September 8 to September 17, 2003. The key participants from each side were:

USNRC

1. Ashok C. Thadani, Director, Office of Nuclear Regulatory Research
2. Michael Mayfield, Director, Division of Engineering Technology, Office of Nuclear Regulatory Research
3. Karen Henderson, Office of International Programs
4. P. T. Kuo, Director, License Renewal and Environmental Impacts, Office of Nuclear Reactor Regulation
5. Nilesh Chokshi, Chief, Materials Engineering Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research
6. Jitendra Vora, Materials Engineering Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research
7. William Cullen, Materials Engineering Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research
8. Carol Moyer, Materials Engineering Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research
9. Mark Kirk, Materials Engineering Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research
10. Keith Wichman, Materials and Chemical Engineering Branch, Division of Engineering, Office of Nuclear Reactor Regulation
11. Sunil Weerakkody, Chief, Fire Protection and Special Studies Section, Plant Systems Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation
12. Anthony Hsia, Assistant Chief, Engineering Research and Applications Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research
13. Stephen Bajorek, Safety Margins and Systems Analysis Branch, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research
14. Robert Tregoning, Materials Engineering Branch, Division of Engineering Technology, Office of Nuclear Regulatory Research
15. John Hannon, Chief, Plant Systems Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation
16. Mark Cunningham, Chief, Probabilistic Risk Assessment Branch, Division of Risk Analysis and Assessment, Office of Nuclear Regulatory Research
17. Jack Rosenthal, Chief, Safety Margins and Systems Analysis Branch, Division of Systems Analysis and Regulatory Effectiveness, Office of Nuclear Regulatory Research

18. William Beckner, Chief, Reactor Operations Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation
19. James Bongarra, Reactor Operations Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation
20. Jennifer Uhle, Chief, PWR Systems Section, Reactor Operations Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation
21. Robert Palla, Reactor Operations Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation
22. Jeffrey Miller, Office of the Senior Coordinator for Nuclear Safety, U.S. Department of State
23. Elizabeth Rood, Foreign Affairs Officer, Bureau of Nonproliferation, Office of Regional Affairs

India

1. S. K. Sharma, Vice Chairman, Atomic Energy Regulatory Board
2. S. K. Chande, Director, Operating Plants Safety Division, Atomic Energy Regulatory Board
3. R.K. Sinha, Associate Director, Reactor Design and Development Group, Bhabha Atomic Research Centre
4. H. K. Kushwaha, Associate Director, Safety and Environment Group, Bhabha Atomic Research Centre
5. S. A. Bhardwaj, Executive Director (Engineering), Nuclear Power Corporation of India, Ltd.
6. S.S. Bajaj, Executive Director, Reactor Safety and Analysis, Nuclear Power Corporation of India, Ltd.
7. K.K. Dwivedi, Counselor, Science and Technology, Embassy of India, Washington, D.C.

Summary

A. Mr. Thadani welcomed Mr. Sharma and his colleagues from the Atomic Energy Regulatory Board (AERB), the Nuclear Power Corporation of India, Ltd. (NPCIL), and the Bhabha Atomic Research Center (BARC). He noted that the 2nd meeting to discuss the nuclear safety projects was a major milestone in fulfilling the goal of enhanced nuclear safety cooperation between the United States and India.

Mr. Thadani briefly described the history of USNRC and AERB interactions, beginning in 1995, with subsequent visits by USNRC delegations in 1998 and in February 2003. He noted that the USNRC participants in those meetings have consistently been impressed with the technical content of the nuclear safety program in India. Mr. Thadani noted that over time the USNRC participants have gained a reasonably good understanding of the Indian regulatory program, and that, hopefully, the AERB participants also had gained good understanding of the USNRC's program as well.

Mr. Thadani noted that during the February 2003 meeting Professor Sukhatme, Chairman, AERB, offered a number of proposals for the expansion of the nuclear safety dialogue. Clearly, there are many areas of mutual interest between USNRC and AERB and the challenge is to pursue these interests within the constraints of relevant domestic law, regulations, policies and international obligations. Mr. Thadani recognized AERB interest in joint standard problem analyses using safety codes to assess accident progression and any radiological releases. He further noted that the USNRC is very interested in the AERB analyses and operational experience with PHWRs that could be of value in the USNRC review of the ACR-700 design.

Mr. Thadani stated that the meeting agenda built on the discussions and briefings from the February 2003 meeting and the previous discussions. He noted that the meeting agenda included visits to two research facilities in the Washington, D.C., area and a visit to the Surry Nuclear Power Plant. He further noted that the technical discussions address the five project areas agreed to: License Renewal, design modifications, fire safety; emergency operating procedures; and PRA technology.

Mr. Thadani concluded his remarks by noting that finding a mechanism to provide funding to support the active cooperation is a critical issue for USNRC. He noted that he was pleased to receive the AERB draft proposal to the Indo-U.S. Science and Technology Forum, and that he is looking forward to agreeing on the joint proposal to the U.S. and India sides of the Forum to provide the needed funding.

B. Mr. Sharma opened by noting that he appreciates the fact that the 2nd meeting is being held. He noted that the February 2003 briefing gave a glimpse of the work being done at AERB and also the R&D work at BARC to support the AERB regulatory program. He also emphasized the intent to prepare a joint proposal to the Science and Technology Forum and that USNRC and AERB should strive to finalize this proposal by the end of the meeting.

Mr. Sharma noted that the AERB is a small organization, with approximately 100 scientific staff, and that they draw heavy input from BARC, the safety groups of NPCIL, and various academic institutes in the country. AERB is also making use of retired staff for regulatory support functions. These inputs are necessary for efficient conduct of regulatory activities in India where the nuclear power program is growing with 14 operating NPPs and 9 units of various designs under construction presently.

Mr. Sharma recalled a plan to have a review near the end of 2004 and then to expand the cooperation program as warranted. He noted that AERB wants to expand beyond the five topics previously accepted, and specifically to include inter-comparison analysis exercises for comparing computer codes for safety analysis and discussions on activity transport thru fuel, clad, coolant and primary coolant pressure boundary to the containment atmosphere. He also expressed that AERB would be interested in obtaining certain safety analysis computer codes from USNRC, particularly those related to radiological safety aspects under accident conditions and advanced codes for probabilistic safety analysis.

Mr. Sharma closed by reaffirming the need to seek funding to support the cooperative activities. He noted that it will require 6-8 months to get approval from the Forum, and that it is essential we submit proposals for funding in the very near future.

C. The agenda for the meeting is included as Attachment 1. The technical discussions were held on September 8-12, and were limited to the five technical topics previously accepted, but provided more detail of both the USNRC and AERB programs in each of the topics than previously presented. Summaries of each technical presentation are provided in Attachment 2.

Visits to the National Institute for Standards and Technology and the University of Maryland were made on September 15 to discuss fire safety and PRA technology, respectively. At NIST the delegation received a briefing summarizing the NRC-NIST benchmark experiments for validation of fire models. The briefing included a description of the test objectives, equipment configuration, and preliminary results. A tour of the NIST large fire test facility was conducted after the brief. Scott Newberry, Director of the Division of Risk Assessment and Application, Office of Nuclear Regulatory Research then briefed the delegation on the major elements of the fire research plan which includes tasks to improve the methods to assess fire risk.

In the afternoon, the delegation visited the University of Maryland and was greeted and briefed by Dr. Mohammed Moddares. He provided an overview of the University history and their engineering program. Dr. Carol Smidts described her research on digital software reliability and tours were provided of the University research reactor, Gamma irradiation facility, and linear accelerator. Three Indian university students were invited to attend and participate in part of the discussion.

D. A visit to the Surry Power Station was made on September 16 to provide the perspective of a nuclear power plant licensee on license renewal and emergency operating procedures. A tour of the facility was conducted by the licensee. The visit and the presentations made by Surry staff were highly appreciated by the Indian delegation. The opportunity to see the plant areas, the new RPV head that is to be shortly installed on Unit 2, including the preparatory work for this major activity was also appreciated.

E. The meeting was concluded on September 17 with summaries provided by Mr. Thadani and Mr. Sharma. Both noted that the discussions had been very constructive, providing a strong basis for further detailed dialogue.

Discussions of the joint proposal to the Indo-US Science and Technology Forum led to agreement on a draft proposal, Attachment 3. The draft will be reviewed by both USNRC and AERB to develop the final version. A target date of November 1 for submitting the proposal to the US and India sides was agreed upon.

A time frame of January/February 2004 for the 3rd NRC-AERB Technical Discussion Meeting on Nuclear Safety Topics was accepted. Mr. Thadani and Mr. Sharma agreed to finalize the meeting dates and to develop a proposed technical agenda by November 30, 2003. Mr. Thadani agreed to explore the possibility of expanding the discussion to include code comparisons through performance of standard problem exercises and discussions on activity transport from fuel to containment atmosphere to facilitate the discussions of nuclear safety issues and the capability to conduct realistic analyses of the potential for radiological releases to the public. Mr. Thadani also agreed to explore the possibility of making certain safety analysis codes available to AERB, and any opportunities for Indian experts to work at specified institutions in the U.S. for durations of about 6 months as per Mr. Sharma's suggestion, within the constraints described in Section A.

Mr. Thadani suggested that discussions in future meetings could also include safety insights gained by AERB from analysis and experience with PHWRs. This was agreed to by Mr. Sharma.

F. The meeting concluded with mutual agreement that the discussions and agreement on further cooperation had been a positive step toward the goal of enhanced nuclear safety cooperation between the United States and India.

<u>Original signed by A. Thadani</u>	<u>Original signed by S.K. Sharma</u>
Ashok C. Thadani, Director	S. K. Sharma, Vice Chairman
Office of Nuclear Regulatory Research	Atomic Energy Regulatory Board, India
U.S. Nuclear Regulatory Commission	

ATTACHMENT 1

2ND NRC - AERB NUCLEAR SAFETY PROJECTS MEETING

SEPTEMBER 8 - 17, 2003

Monday, September 8
Room T-2-B1

1:00 - 2:00	Welcome and Introductions	A. Thadani, RES S. K. Sharma, AERB
2:00 - 4:00	License Renewal Process and Experience in the U.S. - Regulatory Process - Status of License Renewal - Description of Example Renewal Application and Process	P.T. Kuo, NRR
4:00 - 5:00	Summary of NRC's Aging Research Activities	N. Chokshi, RES

Tuesday, September 9
Room T-2-B1

8:30 - 11:30	Aging Effects and Aging Research - Nuclear Plant Aging Program - Cable Aging - Materials Aging Research Environmentally Assisted Cracking Non-Destructive Examination Fracture Mechanics and Reactor Pressure Vessel Integrity	N. Chokshi, RES J. Vora, RES W. Cullen, RES C. Moyer, RES M. Kirk, RES
11:30 - 1:30	Lunch	
1:30 - 2:30	Overview of AERB Activities	S. K. Sharma, AERB
2:30 - 3:30	License Renewal Process & PSR	S. K. Chande, AERB
3:30 - 5:00	License Renewal of TAPS	S. S. Bajaj, NPCIL

Wednesday, September 10
Room O-9-B2

9:00 - 10:30	Leak-Before-Break Research and Regulatory Applications	K. Wichman, NRR N. Chokshi, RES
10:30 - 11:00	Fire Safety	S. Weerakkody, NRR
11:00 - 12:00	PWR and BWR Sump Blockage Issues	A. Hsia, RES
12:00 - 1:30	Lunch	
1:30 - 3:00	Redefinition of LOCA Break Size and Frequency	S. Bajorek, RES R. Tregoning, RES
3:00 - 5:00	Plant Retrofits After TMI Accident	J. Hannon, NRR M. Cunningham, RES

Thursday, September 11
Room O-9-B2

9:00 - 10:00	Emergency Operating Procedures	S. S. Bajaj, NPCIL
10:00 - 12:00	Design Modifications and Retrofits In Indian NPPs	S. A. Bhardwaj, NPCIL
12:00 - 1:30	Lunch	
1:30 - 5:00	Symptom Based EOP's and Severe Accident	J. Uhle, NRR J. Rosenthal, RES M. Cunningham, RES

Friday, September 12
Room O-9-B2

9:00 - 10:30	Coolant Channel Life Management	R. K. Sinha, BARC
10:30 - 12:00	Leak-Before-Break Research in India	H. S. Kushwaha, BARC
12:00 - 1:30	Lunch	
1:30 - 3:00	Summary of Discussions and Identification of Future Interaction on These Subjects	A. Thadani, RES S. K. Sharma, AERB
3:00 - 5:00	Preparation of Meeting Summary	M. Mayfield, RES

Monday, September 15
AERB Rep.

8:30	Depart NRC	
9:00 - 12:00	NIST Tour	S. Newberry, RES M. Cunningham, RES NIST Staff
12:00 - 1:00	Lunch - NIST	
1:00	Depart NIST	
2:00	University of Maryland Tour	S. Newberry, RES M. Cunningham, RES University of MD
Staff		
5:00	Depart University of Maryland	
6:00	Arrive NRC	

Tuesday, September 16

6:00	Depart NRC	
9:00 - 3:00	Tour - Surry NPP	C. Gratton, NRR M. Mayfield, RES
3:00	Depart Surry NPP	
6:30	Arrive NRC	

Wednesday, September 17
T10-F3

9:00 - 11:00	Meeting Summary and Discussion	A. Thadani, RES S. Sharma, AERB
11:00 - 12:00	Finalize Memorandum of Meeting	A. Thadani, RES S. Sharma, AERB
12:00 - 1:30	Lunch	
1:30 - 2:00	Tour NRC Operations Center	R. Wessman, NSIR K. Henderson, OIP
2:00: - 4:00	Agree on Proposal to Indo-US Science And Technology Forum	A. Thadani, RES S. Sharma, AERB

ATTACHHMENT 2

SUMMARY OF PRESENTATIONS

Summary of NRC Presentations

License Renewal Process and Experience in the U.S. **P.T. Kuo, NRR**

The presentation focused on the license renewal process and the U.S. experience to date. The presentation described the Atomic Energy Act of 1954 and the NRC regulations, 10CFR Parts 54 and 51, that specify requirements for safety review and environmental review, respectively. The presentation provided an overview of the background and the key provisions of the license renewal rule, 10 CFR Part 54. This rule was developed based on the significant Commission determinations that the existing regulatory process is adequate for ensuring safety of operating plants; that current licensing basis is adequate and carries forward into the period of extended operation; and that issues relevant to the current operation of plants will be addressed by the regulatory process. The presentation also addressed the rational for the evolution of the rule from the 1991 version to the 1995 amended version. In addition, the presentation provided an update of the NRC's license renewal guidance documents which include the Standard Review Plan for License Renewal, the Generic Aging Lessons Learned (GALL), and Regulatory Guide 1.188. Specifically, the GALL report was highlighted as the technical basis for the NRC staff's acceptance criteria for aging management programs. An interim staff guidance process was also discussed that has been implemented to quickly disseminate the lessons learned from previous reviews to future license renewal applicants. Finally, the status of past, current and future license renewal application was reviewed and the presentation was concluded with the observation that the US license renewal process is stable and predictable and allows public scrutiny and participation.

Summary of NRC's Aging Research Activities **N. Chokshi, RES**

This presentation consisted of an overview of the aging research being conducted in the Division of Engineering Technology, RES. The presentation consisted of two parts. The first part dealt with the generic aspects of the aging research such as motivations, confirmatory and anticipatory aspects, how it supports regulatory functions, what are the broad issues associated with the aging degradations, and general NRC approach in dealing with degradations. The second part focused on specific areas of research, again in an overview fashion. The specific areas included: reactor pressure vessel integrity, pipe fracture, environmentally assisted cracking, barrier integrity action plan, steam generator tube integrity, and containment and other structural aging research. The format was to describe regulatory issues and needs in each of the areas followed by brief descriptions of research programs to address these issues. For example, the issue of pressurized thermal shock was discussed, followed by discussion of the research programs in fracture mechanics, embrittlement, and non-destructive examination to illustrate how the research is being used to develop a technical basis for potential revision of the regulation in this area to reduce unnecessary conservatism. The presentation was summarized with a brief discussion of a recent initiative on pro-active management of the degradations. More detailed presentations were subsequently made on the research related to reactor pressure vessel, environmentally assisted cracking, non-destructive examination, and cable aging.

Nuclear Plant Aging Program and Cable Aging Research **J. Vora, RES**

Nuclear Plant Aging Research (NPAR): The presentation gave an overview of RES/NRC sponsored NPAR program that was implemented in 1984 and completed in 1994. The presentation covered NPAR approach to understanding and mitigating aging in systems, structures and components (SSCs). The NPAR approach is applicable to any SSC of interest. It was emphasized that aging is an issue for license renewal and plant life extension considerations but also for the current license term of forty years. The SSCs that were studied as part of the NPAR program were reviewed and the presentation provided a summary listing (NUREG/CR-1377) and illustrative examples. As a part of the presentation, the elements of the RES technical staff and research support to the overall license renewal process, specifically in the development of the guidance documents for license renewal were also discussed.

Cable Aging: The following items were reviewed in this presentation: (a) The Federal Programs for Wire System Safety; (b) Technical and Reliability Issues Related to Wire System Safety; and (c) Completed and Ongoing Research Topics in Electrical Area. The presentation provided some background information on wire system safety interagency working group (WSSIWG), discussed the purpose and functions of WSSIWG, scope of interagency focus and conclusions and recommendations. A copy of the WSSIWG report was provided to the AERB staff. Subsequently, four specific technical and reliability issues related to wire system safety were discussed. This included: (I) Reliability Physics Modeling; (ii) Fire Risk Assessment; (3) Risk Significance of Wire System Aging; and (iv) Cable Prognostics and Diagnostics. During this discussion a copy of the Proceedings of the International Conference on Wire System Aging, NUREG/CP – 0179, was provided to the AERB staff.

Environmentally Assisted Cracking **W. Cullen, RES**

The purpose of the presentation was to present a summary of the research programs addressing the various aspects of environmental degradation. The description of the larger of two Argonne programs highlighted the major tasks: (a) Environmental Effects on Fatigue Life, (b) Irradiation-Assisted Cracking of Stainless Steel in a BWR Environment, (c) Irradiation-Assisted Cracking of Stainless Steel in a PWR Environment, and (d) Stress-Corrosion Cracking of Nickel-Base Alloys. Several data sets were presented to illustrate the results of IASCC testing in BWR environments. The plans for IASCC testing in PWR environments which will involve 304 and 316 stainless steels irradiated to 5, 10, 20 and 40 dpa were discussed. The second program at Argonne that will measure the corrosion effects of concentrated boric acid solutions on reactor pressure boundary materials was described. The goals of that program are (a) to conduct crack growth rate testing of Alloys 600 and 182 removed from the discarded Davis-Besse head, (b) to develop a probabilistically-based model for the improved calculation of inspection intervals, and measurement of (c) electrochemical potentials and (d) wastage of pressure boundary materials in concentrated boric acid solutions, both aerated and deaerated. Lastly, the presentation covered a brief review of the plans for examination of cracks and flaw indications from both the discarded Davis-Besse and North Anna 2 vessel heads, and a schematic of the proposed approach to develop a proactive materials degradation assessment.

Non-Destructive Examination (NDE) Issues and Programs C. Moyer, RES

The presentation focused on a summary of two non-destructive examination (NDE) programs being conducted by the NRC. Both of the programs presented are directed toward understanding the reliability of NDE techniques and improving the effectiveness of in-service inspections (ISI). The presentation highlighted the following topics and issues which are the focus of the research programs: (1) quantification of the effects of such complications as surface roughness, cast structures and human factors on NDE quality; (2) investigation of advanced NDE techniques to determine their accuracy and reliability for potential ISI use; (3) determination of density and distribution of flaws in reactor pressure vessels and piping, both from fabrication and from repairs, that are needed for structural integrity evaluations and to calculated failure probabilities; and (4) the particular NDE challenges presented by coarse grained materials and primary water stress corrosion cracking (PWSCC) that are being pursued. In addition, the presentation also discussed the NRC staff and contractors interactions with the ASME Code committees, in order to translate research results into suggested improvements in the Code, particularly with regard to ISI requirements needed to ensure that structural integrity is maintained.

Fracture Mechanics and Reactor Pressure Vessel Integrity M. Kirk, RES

This presentation covered topics of Advanced Fracture Mechanics, Pressure Vessel Embrittlement, and RPV Integrity. The presentation focused on the current work at ORNL, the U.S. Navy, the University of Illinois, the University of California, and Modeling and Computing Services Incorporated that contribute to addressing some of the regulatory issues, such as the development of technical basis for revision of the regulations 10CFR50.61, 10CFR50 Appendix G and the Regulatory Guide 1.99. Specific topics included the technical basis of the PTS rule, the project aimed at developing Revision 3 to Reg. Guide 1.99, and on projects in which the Master Curve (as proposed by Wallin of VTT in Finland) has been approved and/or used by NRR and RES. Key points discussed included the use of a risk informed approach to revise 10CFR50.61, the use of physical insights in the development of a revised embrittlement trend curve for Regulatory Guide 1.99, and the development of strategies for incorporating Master Curve techniques into existing regulations. Members of the delegation expressed particular interest in RES's development of a characterization of the upper shelf fracture toughness for ferritic steels, and on demonstration that the temperature dependency of upper shelf fracture toughness is common across a broad range of ferritic alloys.

LBB Research and Regulatory Applications K. Wichman, NRR

This presentation covered the background of LBB development in the U.S., the definition of LBB as used in the U. S. plus some technical details which included methodology and applicable margins. It also listed the benefits that accrue to LBB and enumerated the LBB applications that have been approved by the NRC. It mentioned some of the research programs instrumental in LBB development and outlined the process for applying LBB to advanced reactors. Finally, the presentation discussed the generic implications and the resulting generic activities (ongoing and planned) of the NRC and the industry to assess the effects of PWSCC in Alloy 82/182 pipe butt welds as it may affect LBB.

Fire Safety **S. Weerakkody, NRR**

The presentation on Fire Safety began with discussion of the current regulations dealing with this issue. 10 CFR 50.48, Fire Protection, General Design Criterion 3, Fire Protection, and Appendix R to Part 50, Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979, were discussed. The presentation then focused on risk insights and current initiatives on risk-informing requirements in this area. The specific initiatives discussed included: rulemaking to adopt Standard NFPA 805; rulemaking to allow feasible manual actions; use of circuit analysis inspection and resolution plans; and improvement to Significance Determination Process. The presentation concluded with brief discussion of status of each of the initiatives. During the discussions, a question was raised regarding rationale for moving to risk-informed approach in light of sparsity of quantitative data on large fires.

PWR and BWR Sump Blockage Issues **A. Hsia, RES**

This presentation covered issues concerning debris generation in PWR and BWR containments following a LOCA and the effect of those debris on containment sump performance. The presentation included discussion of: NRC regulation in 10 CFR 50.46 requiring all light water reactors to ensure long term core cooling following a LOCA; operational events at PWRs and BWRs as a result of debris generation and consequent containment sump blockage; NRC's research programs to gain knowledge of the sump blockage issue; and NRC's regulatory actions to ensure reactor safety and provide resolution of the issue. The presentation also included examples of technical approaches used to evaluate potential debris generation, debris transport in the containment, and its impact on sump or strainer screen head loss which may reduce the available net positive suction head of downstream pumps. Discussions were held with the AERB representatives regarding potential vulnerability of their containment designs and periodic testings for recirculation pumps in U.S. nuclear plants.

Redefinition of LOCA Break Size and Frequency **S. Bajorek, RES**
R. Tregoning, RES

In this joint presentation, the first part focused on the NRC perspective and motivation for redefining the LB LOCA and discussed the plant response implications of the current requirements and how the LB LOCA dominates the analysis methods and ECCS research at the expense of more risk significant smaller break sizes. The presentation also discussed risk insights and described potential safety and other benefits. The presentation outlined some anticipated operational changes that may be feasible after redefinition.

The second part of the presentation focused on issues associated with establishing the spectrum of break sizes and frequencies to redefine the LB LOCA, and described details of NRC's approach to address these issues. The presentation concentrated on the motivation behind this rigorous assessment of passive system LOCAs and the need to consider both piping and non-piping contributions in order to obtain a comprehensive assessment as continued material degradation occurs. Technical challenges were addressed and the approach developed to address these challenges was presented. The approach uses expert elicitation to combine operational experience data and the probabilistic fracture mechanics. The elicitation structure was discussed in detail and upcoming portions of the process were noted. The presentation also contained information to address research issues associated with the risk analysis which will utilize the insights gained by increased technical understanding of LOCA frequencies and plant system response.

The discussion surrounding the presentation centered on the validity of elicitation approaches. Past NRC elicitations were discussed and steps taken to validate this methodology were discussed. The discussion also focused on probabilistic fracture mechanics code development as well as the establishment and assessment of relevant operating experience in order to provide confirmatory validation of the elicitation findings.

Plant Retrofits After TMI Accident **J. Hannon, NRR**
M. Cunningham, RES

The presentation included the following:

1. TMI Post Accident Studies: following the TMI accident, several investigations occurred and studies were performed. The most important studies include; (a) Kimeny Commission Report to the President of United States, and (2) NRC's Rogovin Report.
2. Process of Implementation: U.S. Nuclear Regulatory Commission (NRC) staff developed the Action Plan, NUREG-0660, by collecting a comprehensive list of TMI related items to improve the safety at power reactors. Later, specific items were identified in NUREG-0737, for industry to implement. NRC issued generic letters and confirmatory letters to the licensees of both Boiling Water Reactors and Pressurized Water Reactors. Guidelines for Technical Specification were provided by NRC to the licensees for implementation.
3. Highlights of the Presentation included: Human performance (e.g., operator training); control room habitability; NRC, industry, and public interaction; and systems upgrade.
4. Implementation Examples: three specific examples were presented depicting the process of TMI Action Plan implementation.

Symptom Based EOP's and Severe Accident **J. Uhle, NRR**
J. Rosenthal, RES
M. Cunningham, RES

This presentation was divided into three specific topic areas as follows.

EOP Process Summary: This presentation focused on the following topics related to the NRC's Emergency Operating Procedures (EOP) Upgrade Program: 1) Post TMI requirements for the US nuclear industry to upgrade EOPs, 2) the Procedures Generation Package (PGP) concept and application, and 3) the EOP Audit and Inspection Program. In response to a question from the Indian delegation, it was clarified that, although the NRC conducted an EOP audit and inspection program as part of its overall program to ensure that licensees upgraded their pre-TMI, event based procedures to symptom-based procedures, it remains the licensees' responsibility to develop, implement, and maintain high quality EOPs. The NRC assures compliance with commitments made by the licensees in their PGPs.

Generic Technical Guidelines Development: This presentation covered an overview of the Westinghouse Emergency Response Guidelines (ERGs) as an example of vendors' development of generic technical guidelines. The presentation highlighted the following points: (1) each vendor has developed a set of technical guidelines that the licensee develops into plant specific EOPs; (2) the ERGs provide a well-defined framework for emergency operations, and the operator's role is addressed through a network of predefined symptom-based strategies for responding to any emergency; and (3) event-related recovery strategies provide guidance to obtain the optimal end state while maintaining the Critical Safety Functions (CSFs) within acceptable limits independent of event sequences.

In response to a question from the Indian delegation, it was clarified that probabilistic risk assessment-based techniques were used to assess the coverage of events provided by the Optimal Recovery Guidelines (ORGs) of the ERGs. Westinghouse used a functional failure probability value of 10^{-8} as the cut-off limit for identifying functional failure sequences for the events considered in the ORGs. It was further stressed that regardless of the probability basis used in defining the required extent of the ORGs, assurance that the residual risk sequences are covered is provided by the CSF Restoration Guidelines in the ERG program.

The Indian delegation questioned if the ERGs have been demonstrated to work under complex event scenarios. In response, it was indicated that the steam generator tube rupture is the event that has been exercised the most and the ERGs were shown to be effective. The ERGs have also been validated on plant simulators.

Severe Accident Management Summary: This presentation consisted of an overview of the NRC's involvement with the US nuclear industry in developing and implementing the US Severe Accident Management Program. The discussion addressed topics such as, defining the concept of accident management, explaining the process by which accident management is implemented by licensees, describing key industry commitments related to accident management implementation, and reviewing the relationship of accident management with the NRC's Integration Plan for Closure of Severe Accidents/Individual Plant examinations. There was substantial dialog between the Indian delegation and the NRC staff on the topic of accident management implementation.

Summary of AERB Presentations

Overview of AERB Activities — S.K. Sharma, AERB

The presentation provided an overview of AERB activities. AERB was established in 1983 under the Atomic Energy Act, 1962 and is charged with the responsibility of regulation of nuclear and radiation facilities in the country. The safety reviews for this purpose are done through multi-tier review process and cover various stages from siting through decommissioning. AERB issues authorizations for various facilities after review, conducts regulatory inspections and ensures emergency preparedness at the nuclear facilities. It also develops safety documents, informs the public about safety matters and supports safety research. The safety review process for nuclear power projects and for operating nuclear power plants was described in detail including the process of renewal of authorization every three years after brief review and exhaustive periodic safety review every nine years. Some details of developing safety documents, especially for the PHWR based NPPs were presented. The presentation included examples of some recent and important regulatory reviews conducted for PHWRs. These included safety and life management of zirconium alloy coolant channels, incidences of steam generator tube leaks, issue of thinning observed in coolant outlet feeder pipes, fire incident in the turbine building of Narora Atomic Power Plant in 1993 and flooding incident in Kakrapar NPP in 1994. Safety upgradations implemented in RAPS and MAPS and those identified in TAPS for licence renewal were also touched upon.

Licence Renewal Process and Periodic Safety Review of NPPS in India **S.K. Chande, AERB**

In India, the operating licence is issued after a detailed review of design, commissioning activities and initial operations for a period of 30 years. However, within the licence period, a periodic reauthorization is necessary. Presently at 3 years' interval, a brief review of operational safety, OEF and major modifications carried out for issue of reauthorization. In addition, every nine years, a comprehensive review covering safety status of plant ageing management and safety analysis among other factors is carried out. The PSR requires assessment of these factors in comparison to current standards and practices to identify deviations. Based on safety significance of these deviations a plan of upgrades is drawn up, as necessary.

Such reviews carried out for two plants have indicated that safety performance of these plants is satisfactory and the original safety analyses are conservative.

The license renewal exercise has been carried out for Tarapur 1& 2 units which are in operation since 1969. During this period, there have been significant changes in safety requirements and hence a more fundamental approach has been used for this review. The details of this review, assessments made and schedule of implementation of corrective measures are described in a subsequent presentation.

Licence Renewal of TAPS – 1&2 S.S. Bajaj, NPCIL

Tarapur Atomic Power Station Unit 1&2 (TAPS 1&2) were recently subjected to a detailed regulatory review for its licensing renewal, and this presentation served as an example of the Indian practice in this regard. The two TAPS units, based on pre-mark-I design, have been in operation since 1969; in the year 2000. The Indian Atomic Energy Regulatory Board called for a comprehensive review as a pre-requisite for continued operation, covering review of operational performance, ageing management and design basis and safety analysis, against current safety practices. The review brought out requirements for upgrades, especially in station electric power supplies, including rerouting of cables, and augmentation of inspection. Based on the review, AERB has permitted operation of TAPS 1&2 units beyond 2005 only after implementation of the identified upgrades.

The presentation and the ensuing discussion served to bring out the Indian approach to licence renewal.

Emergency Operating Procedures for Indian PHWRs **S.S. Bajaj, NPCIL**

Emergency Operating Procedures (EOPs) for Indian PHWRs are covered by about 60 event based procedures, grouped under functional classification of initiating events. The procedures include tabulation of anticipated event progression and required operator actions, at various times, as well as operator action flow charts, which guide the operator to identify the event and take action based on symptoms. The EOP for station black out and its actual use in the NAPS fire incident during 1993 was discussed.

Adaptation of the procedures to actual incidents is facilitated by training and the fact that senior control room operators are graduate engineers.

The EOPs are being implemented on power plant simulator.

Design Modifications / Retrofits in Indian NPPs **S.A. Bhardwaj, NPCIL**

The presentation detailed some of the important design modifications and backfits made in Indian PHWRs. It was explained that such backfits are usually in response to precursors / actual incidents experienced in India or abroad. As also the routine and special periodic safety reviews conducted by AERB bring out need for cerain design modifications. Broad results of reviews conducted in India following TMI, Chernobyl accidents ; Pickering coolant tube failure, NAPS fire incident and the design changes proposed as a consequence were enumerated. Backfits and design modifications in RAPS and MAPS reactors like separation of electrical power supplies and associated cable routes, incorporation of high pressure emergency core cooling system, supplementary control room, augmentation of fire protection provisions, protection against external floods, seismic qualification, etc. were explained. The replacement programme of Zircaloy 2 coolant channels was also elaborated.

Consequent to certain degradations to important equipment, operation of some units was restricted. Examples of backfits conducted to restore the normal operation of these plants were also discussed. In this context Calandria Inlet Manifold failure in MAPS and restoration of normal moderator flow by use of a sparger concept was described. Another example covered was arresting of leak experienced in calandria over pressure relief device of RAPS – 1.

Life Management and Safety of Pressure Tubes in Indian PHWRs **R.K. Sinha, BARC**

The design features of coolant channels of Indian PHWRs range from coolant channels with zircaloy-2 pressure tubes, two loose fit garter spring spacers and open annulus in the older reactors, to coolant channels with Zr-2.5% Nb pressure tubes, four tight fit garter springs and closed annulus in the new reactors.

On account of their simple geometry, the pressure tubes are amenable to extensive manufacturing, pre-service and in-service inspection. They can be replaced at the end of their design life which, in current generator designs, is limited by leak-before-break considerations, rather than considerations arising out of dimensional changes on account of irradiation enhanced creep and growth.

Even though zirconium alloys are not covered by ASME code, the pressure tube material has been well characterized to the extent required for enabling the design of pressure tubes to meet the intent of the ASME B&Nr code section III. The design is based on rigorous analysis to keep the wall thickness as low as permissible anywhere along the length of the tube under any design condition.

Delayed hydride cracking (DHC) is the most important mechanism for growth of defects in pressure tubes. For a postulated flaw growing from the inside of the pressure tube, leak before break can be predicted with confidence. In case of pressure tube – calandria tube contact when, under certain conditions, brittle hydride blisters may from on the outer surface of pressure tubes, leak before break can be defeated. Such a potential existed in early generation Indian PHWRs with loose fit garter spring spacers, vulnerable to shifting from design axial locations. The problem is being successfully handled through development and implementation of several technologies of in-service-inspection and post-irradiation examination, computer codes for modelling the creep, hydrogen pick up and blister growth phenomena, and the regulatory acceptance criteria. The Indian strategy for life management of pressure tubes has been rather unique in the sense that starting with very conservative acceptance criteria, the margins of conservatism on these criteria were progressively trimmed as the volume of inspection and PIE data, and validity of computer codes used for life management, grew with time, inspection, hydrogen pick up monitoring examination, and garter spring repository 8.5 full power years and more, for the zircaloy-2 pressure tubes of Indian PHWRs. In two of the reactors, RAPS-2 and MAPS-2 pressure tubes of Indian PHWRs. In two of the reactors, RAPS-2 and MAPS-2, large scale coolant channel replacement has been completed and this work for MAPS-1 is currently in hand.

Leak-Before-Break Research in India H.S. Kushwaha, BARC

The presentation focused on the research work done in India related to Leak-Before-Break (LBB concept) for nuclear reactor piping. This research work involved experimental and analytical work on fatigue and fracture behaviour of piping components. As a part of this work large number of fatigue and fracture tests were carried out on Standard Compact Tension (CT) specimen and piping components of actual size of SA 333 Gr.6 carbon steel and SS 304 LN Stainless Steel. The tests on piping components were carried out by machining through-wall and part-through notch in base and weld regions. The main aim of these tests was to understand several issues related with material, geometry and loading on piping. It was also concluded that Dynamic strain ageing (DSA) in SA 333 Gr.6 piping material is not very significant. The part-through crack under cyclic load grows significantly in thickness direction of pipe for higher aspect ratio (crack length to depth ratio). This will produce leak in pipe under service condition. It was also concluded that mode of failure of carbon and stainless steel pipes having through wall crack is net section collapse, if yield stress is considered as limit stress of the material. This will always produce conservative resistance irrespective of size of piping. The effect of seismic load and compliance were also considered. It was found that seismic load reduces the crack resistance whereas compliance will provide more margin. The net result is that they compensate each other. The LBB methodology was applied to 540 MW(e) PHWR reactor coolant piping. It was demonstrated that the main reactor coolant piping of 540 MW(e) meet required margin on load and crack size.

Probabilistic fracture assessment using R-6 and Net Section collapse was also presented. It was emphasized that choice of limit state function may produce different results. Therefore, it is, necessary to consider the limit state function which represent correct failure mode. Future work will include tests at operating temperatures and pressures.

Attachment 3

PROPOSAL TO INDO-US SCIENCE AND TECHNOLOGY FORUM
FOR
ATOMIC ENERGY REGULATORY BOARD–NUCLEAR REGULATORY COMMISSION
TECHNICAL CO-OPERATION WORK

A. Title of Activity

The India Atomic Energy Regulatory Board (AERB) – U.S. Nuclear Regulatory Commission (NRC) Nuclear Safety Workshops

B. Executive Summary

Since 1994 -1995, the NRC and AERB have attempted to develop joint nuclear safety projects. Despite joint government approval in principle, a combination of lack of funding and changing government policies have hindered the projects from being implemented. In November 2001, President Bush and Prime Minister Vajpayee committed to renewing the nuclear safety dialogue, focusing on the NRC-AERB projects. The projects are anticipated to be the forerunners of a longer-term nuclear safety relationship between the U.S. and Indian Governments.

During February 2003, an USNRC delegation visited the AERB, in Mumbai, India. During this visit brief discussions were held pertaining to the safety of nuclear power plants covering the topics of fire safety, design issues, emergency operating procedures, license renewal and risk informed, performance-based regulation. In September 2003, an AERB delegation visited the USNRC, in Washington D.C. and had further discussions on these topics. From these discussions, it was realized that considerable benefit can be obtained by AERB and USNRC if structured workshops and other co-operative activities on these nuclear safety topics are conducted. Jointly, NRC and AERB are seeking Indo-U.S. Science and Technology Forum funding to begin the process of implementing these projects. Accordingly, it is proposed to conduct two workshops in 2004 for detailed discussions on nuclear safety issues: the first workshop to be held in Mumbai and the second in Washington D.C.

Both the U.S. (A) and India have considerable experience in design, construction and operation of nuclear power plants (NPPs). While the U.S. nuclear power program is based on Boiling Water Reactors (BWR) and Pressurized Water Reactors (PWR), the Indian program is primarily based on Pressurized Heavy Water Reactors (PHWR). Further, both countries have experience in extension of life of NPPs and their safety upgrades. Thus, exchange of information on nuclear safety issues through organized workshops is considered useful for enhancing nuclear safety knowledge amongst experts of AERB and USNRC.

C. Co-PIs of the Organizations

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D. Concept and Purpose

Based on earlier discussions between AERB and USNRC in February and September 2003, it is proposed to hold two workshops in 2004, one in Mumbai and the other in Washington, D.C. The workshops are aimed at exchanges of information and expertise at advanced levels in typical areas of current interest. The first workshop is proposed for January/February 2004 in Mumbai. That workshop will focus on two areas:

i) License Renewal for NPPs

The discussions will address the technical bases supporting license renewal evaluations, the aging effects that must be addressed to support extended operation of nuclear power plants, and supporting technologies such as pipe fracture analysis methods, degradation of structures, and nondestructive inspection technologies.

ii) Fire Safety

The discussions will address methods for analyzing fires, fire risk analyses, and applications to nuclear power plant safety. The discussions also will address fire test results and insights from nuclear power plant fires in the U.S. and India.

The second workshop is proposed for August/September 2004 in Washington, D.C. That workshop will focus on two areas:

i) Risk Informed and Performance Based Regulation

The discussions will address the relationship of Safety Goals to risk informed regulation, risk analysis methods, and the technical basis for specific changes to regulations that have been and are being pursued based on risk informed and performance based concepts.

ii) Design Retrofits and Safety Upgrades

The discussions will address design retrofits and safety upgrades that have been made in plants based on operating experience and information and insights obtained since the plants were designed and built. The discussions will include both regulatory changes that led to the retrofits and upgrades, and the technical information developed to either motivate or support the changes.

E. Need for the Bilateral Workshops and Expected Benefits

Vast experience exists in the USNRC in overseeing safety and regulation of NPPs based on BWR and PWR designs. In India, similar experience exists for BWR and PHWR based NPPs. Considerable improvements in NPP safety have been achieved in both organizations through systematic reviews and application of modern scientific tools like Probabilistic Safety Assessments. In this background, a need is felt for organizing structured workshops to facilitate information exchange on the topics listed above. The benefits that will accrue from these activities include better understanding of safety matters and obtaining directions for future advanced work for NPP safety.

F. Number and Names of Participants

Numbers and names of delegates will be communicated later.

G. Proposed Venue and Dates

(a)	Workshops in Mumbai	- Reputed hotel in Mumbai, January/February 2004
(b)	Workshops in Washington D.C.	- Reputed hotel in Washington, D.C., August/September 2004

H. Expected Outcome

The workshops will provide a platform for exchange of information at advanced levels between AERB and USNRC experts in the area of safety of Nuclear Power Plants. The acquired knowledge at these workshops is expected to contribute to enhancement of safety and improved regulation of NPPs.

I. Total Estimated Cost

The total costs for the two proposed workshops are under development.

J. CVs of Co-Pis

These are given in Attachment II & III. (To be provided)