July 9, 2014

Mr. John W. Stetkar, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: REVIEW AND EVALUATION OF THE NUCLEAR REGULATORY COMMISSION SAFETY RESEARCH PROGRAM

Dear Mr. Stetkar:

On behalf of the U.S. Nuclear Regulatory Commission (NRC), I am responding to your letter dated April 15, 2014, enclosing Volume 11 of NUREG-1635, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program."

I want to express my appreciation for the Advisory Committee on Reactor Safeguards (ACRS) continued valuable review and evaluation of NRC's safety research program. Throughout the years, the ACRS' evaluations and suggestions have helped to improve the program.

The agency agrees with your observation that the strategy for prioritizing research subjects through the User Need process has worked reasonably well, and the process has generated useful products to the line organizations in a timely manner. NRC will continue to strive to ensure that this approach does not hinder the development of in-depth understanding that is essential to future regulatory issues. I also appreciate the ACRS's reminder of the importance of collaborative research activities to leverage agency resources and to gain access to external facilities and expertise. The agency will continue to strive to utilize collaborative efforts to the extent practical in allocating its research resources.

The ACRS recommended that the Commission develop an integrated plan for providing the necessary technical basis for implementation of the lessons learned from events that occurred at the Fukushima Dai-ichi nuclear plant site. The NRC staff considers the relationships between various Fukushima-related activities and routinely provides the Commission with an integrated plan on the various recommendations. However, at this time, the NRC staff has no plans to create an integrated research plan for developing the technical bases associated with the implementation of the lessons learned. Many of the recommendations requiring support from the Office of Nuclear Regulatory Research (RES) are classified as Tier 3 recommendations. This designation indicates that these recommendations will be informed by the implementation of Tier 1 recommendations. It is possible that some Tier 3 recommendations will be subsumed by Tier 1 actions. Considering the uncertainty of any required actions for Tier 3 recommendations, the staff does not consider it necessary or prudent to develop an integrated plan for support from RES on the technical bases for the recommendations. Rather, as the need arises, the Office of Nuclear Reactor Regulation (NRR) will consult with RES on the support required to develop any technical bases associated with the post-Fukushima recommendations. NRR will then be responsible for integrating the technical support provided by RES with its own activities to determine the most appropriate regulatory action for each recommendation.

The enclosure contains the NRC staff responses to the ACRS' specific observations and recommendations. We will consider your recommendations as we execute the research program and develop research plans and budgets for upcoming fiscal years.

I look forward to continued interactions with the ACRS and thank the Committee for its efforts.

Sincerely,

/RA/

Mark A. Satorius
Executive Director
for Operations

Enclosure:
As stated

cc: Chairman Macfarlane
Commissioner Svinicki
Commissioner Magwood
Commissioner Ostendorff
SECY

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DATE	6/18/2014	06/19/2014	06/20/2014	07/ 9/2014

NRC STAFF RESPONSES TO RECOMMENDATIONS
ON SPECIFIC RESEARCH ACTIVITIES CONTAINED IN
NUREG-1635, VOL. 11, "REVIEW AND EVALUATION OF THE NUCLEAR
REGULATORY COMMISSION SAFETY RESEARCH PROGRAM"

On April 15, 2014, the Advisory Committee on Reactor Safeguards (ACRS) submitted a letter to Chairman Macfarlane enclosing Volume 11 of NUREG-1635, "Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program." That report contains a number of observations and recommendations about the U.S. Nuclear Regulatory Commission's (NRC's) safety research program. The NRC staff responses follow below.

CHAPTER 3 – FUKUSHIMA FORENSICS: UNDERSTANDING SEVERE ACCIDENT PROGRESSION

With regard to the post-Fukushima forensic analyses, the staff anticipates that the U.S. Government efforts will be coordinated by the U.S. Department of Energy (DOE). In addition, the staff anticipates the creation of an international program to assist the Government of Japan in the Fukushima reactors cleanup efforts. The international effort will include identifying information (improving severe accident modeling, informing severe accident management guidelines, etc.) to be gathered during the cleanup effort. The staff also recognizes that time and resource limitations require the information to be gathered and prioritized. NRC is engaged with DOE, the Organisation for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA) Committee on the Safety of Nuclear Installations, the International Atomic Energy Agency, and the Japanese regulatory and research organizations in discussing technical issues related to Fukushima cleanup efforts.

CHAPTER 4 – ADVANCED REACTOR DESIGNS

The NRC staff agrees with the ACRS' comments on issues and data needs for high-temperature gas-cooled reactors (HTGRs). These issues were recently identified through a well-structured Phenomena Identification and Ranking Table process when the DOE program in HTGR was very active. Currently, the NRC resources for HTGR research are very limited, pending, as ACRS noted, the identification of a specific very-high-temperature reactor design and an applicant. The Office of New Reactors (NRO) performs the NRC activities, although somewhat limited, on the resolution of licensing issues pertaining to the HTGR design.

CHAPTER 5 – DIGITAL INSTRUMENTATION AND CONTROL SYSTEMS

The ACRS noted the significant progress in analysis of digital system failures modes but was concerned that the use of these results is not coordinated with digital system probabilistic risk assessment (PRA) research. The staff continues to believe that ongoing research programs on digital instrumentation and control (I&C) reliability modeling for PRA and the analytical assessment of digital I&C systems in support of the current deterministic regulatory decisionmaking approaches are well coordinated. However, it should be recognized that while these programs are complementary in many ways, they are intended to support different applications. The staff intends to brief the ACRS Digital I&C Systems and Reliability and PRA subcommittees later in calendar year 2014 to describe the coordination and complementary nature of the programs.

The ACRS provided a comment that it continues to see a lack of integration of control of access, safety, and cyber security in the design stage and licensing as an impediment to ensuring secure digital I&C safety systems. The staff's response to this comment was included in a letter to the ACRS on April 3, 2014.

As the ACRS requested, the staff will inform the ACRS once the new 4-year digital I&C Research Plan for 2015 – 2018 has been developed. The staff also will continue cooperative nuclear safety research on digital safety systems under the terms of the memorandum of understanding (MOU) between the NRC Office of Nuclear Regulatory Research (RES) and the Electric Power Research Institute (EPRI).

CHAPTER 6 – FIRE SAFETY

The staff generally agrees with the ACRS comment that some fire research should anticipate additional emerging applications and potential intermediate- to long-term requirements that are not fully dictated by current User Need Requests. Although it is true that the majority of research projects are driven by other NRC Office User Need requests, RES does use other means to identify research projects. A recent example is NUREG/CR-7123, "A Literature Review of the Effects of Smoke from a Fire on Electrical Equipment," published July 2012. This project was performed under the NRC's long-term research program and was driven in part by previous ACRS reviews and evaluations of the NRC safety research programs. Another example is the Joint Analysis of Arc Fault (JOAN of ARC) testing program. This program was created by working with other countries as part of the OECD program. Testing will be taking place this year with a report to follow in 2015.

The staff agrees with the ACRS that the NRC should continue to encourage and support additional testing and fire experiments and fire prediction model development through collaborative efforts with U.S. industry and international organizations. The staff is currently pursuing other testing and fire experiments both with EPRI under the MOU and independently. NRC recently contracted with the National Institute of Standards and Technology (NIST) to perform a comprehensive series of experiments on the heat release rate from electrical enclosures. A report is expected to be published this year, and advances in the state of the art for fire modeling the electrical enclosure fires will be performed. In addition, the OECD JOAN of ARC program to explore the high energy arching fault phenomena is underway.

Regarding the further research recommended in NUREG/CR 6978, "A Phenomena Identification and Ranking Table Exercise for Nuclear Power Plant Fire Modeling Applications," NIST has completed a series of tests to evaluate the performance of incipient fire detection systems for the NRC. A draft report for public comment is expected to be published in 2014. Future testing programs identified in NUREG/CR-6978 currently are in the planning stages and will be scheduled in the future.

The staff recognizes the need for future research to be performed on digital I&C reactor systems. However, before that work can be started, the experts on the JACQUE-FIRE Phenomena Identification and Ranking Table panel identified near-term testing needed for the analog instrument systems currently in use at the majority of plants today. These experiments are now in the planning stages with NIST. After completion of these analog instrument experiments, the staff will juxtapose into planning for the digital I&C work. The performance of internal panel wiring currently is under consideration to be performed after the analog instrument testing.

CHAPTER 7 – REACTOR FUEL

The staff appreciates the ACRS' comments regarding maintaining an adequate fuels research program and continuing participation in collaborative domestic and international research programs. The staff is making preparations to continue participation in the Studsvik Cladding Integrity Program (SCIP-III) and Halden research programs, which will both provide information needed to evaluate and license future fuel designs. In addition, the staff has developed and plans to maintain strong working relationships with counterparts at the Institut de Radioprotection et de Surete Nucleaire in France and the Japan Atomic Energy Agency and Japan Nuclear Regulatory Authority in Japan. The technical information exchanges with these organizations also ensure that the staff maintains awareness of relevant research programs. In addition, the research staff participates in the Office of Nuclear Reactor Regulation (NRR) led annual fuel performance review meetings with each fuel vendor. Periodic participation in these meetings provides the staff with up-to-date information on the fuel development trends and highlights the areas where research is needed to ensure our capability to evaluate and license future fuel designs. Finally, the staff works collaboratively with DOE and EPRI on a few topics of shared effort, such as accident tolerant fuels, and these collaborations and working relationships often provide valuable insight related to fuel development trends and current industry research efforts.

The staff agrees with the ACRS assessment of the ongoing consequence assessment of fuel failure on the safety of spent nuclear fuel dry storage and transportation packages. In addition, NRC is waiting for the completion of static and vibration tests and the post investigation examination results from Oak Ridge National Laboratory to shed some light on the likelihood of the scenarios considered in the High Burnup Consequence Analysis investigation work.

As the ACRS noted, staff is cognizant of DOE and industry research in accident tolerant fuel and cladding development and meets regularly with DOE counterparts to receive the latest information. NRC staff also has been briefed by a number of industry entities who are developing accident tolerant fuels. NRC staff will continue to monitor DOE and industry activities in this area until the down selection process is complete. At this point, a more focused effort will be made to develop a licensing strategy and engage in regulatory research as needed.

The staff appreciates the assessment that the research addressing fuel fragmentation, axial relocation, and dispersal is well structured and well on its way to supporting regulatory decisions. Regarding the pellet-clad interaction (PCI) experimental database, the staff believes that the database assembled under the SCIP program is large and provides valuable insight. However, the staff also believes the database does not provide sufficient information to articulate a model or support new PCI-related guidance in the Standard Review Plan. This is evidenced by the fact that even collaboration between many worldwide experts on PCI within the SCIP program has still not produced a PCI failure criterion. Moreover, the staff believes that greater effort to develop a PCI model or to develop guidance for the standard review plan is not warranted because of the low probability and low safety significance of PCI failures, as well as the fact that the U.S. nuclear industry has very effectively managed PCI.

As described in the previous comment, the low probability and low safety significance of PCI do not warrant immediate action. If the nuclear power plant fleet actually transitions to load following, the staff will reconsider this position. In the meantime, the staff believes that focusing efforts on more safety significant issues such as fuel dispersal, loss-of-coolant accidents, and reactivity initiated accident is the right course of action.

The staff fully agrees with ACRS about the fact that it is important to be able to evaluate the safety of proposed changes in the nature and burnup limits of reactor fuels. To date, the FRAPCON and FRAPTRAN codes are capable of doing so and have capabilities that generally exceed those of licensed vendor codes. The staff believes that in their current state the FRAPCON and FRAPTRAN codes are adequate evaluation tools for burnups up to 70 GWd/MTU. Incremental fuel design changes also can be easily handled by FRAPCON and FRAPTRAN (e.g., new cladding alloys and a limited range of additives in the fuel pellet). The considerable resources required to develop codes such as those being developed by DOE Consortium for Advanced Simulation of Light Water Reactors and Nuclear Energy Advanced Modeling and Simulation are not currently justified considering the state of technology and codes being used or being proposed by industry for the foreseeable future.

CHAPTER 8 – HUMAN FACTORS AND HUMAN RELIABILITY

The staff agrees with the ACRS' recommendations regarding testing the Integrated Human Events Analysis System and generic human reliability analysis (HRA) methodologies on a large-scale application to ensure that it is practical. The staff plans to make these improvements as well as improving the guidance as part of its integrated research program from fiscal year (FY)14 to FY16.

In addition, as the ACRS noted, RES has expanded its work in degraded instrumentation and control, plans to publish a new medical HRA report in FY15, and plans to engage the User Offices to identify areas for further research in human factors and HRA in medical, security, and instrumentation and control. With respect to the Fukushima-related recommendations, the RES staff is also supporting rulemaking efforts managed by NRR that look into improvements to the command and control structure of severe accidents.

CHAPTER 9 – MATERIALS AND METALLURGY

The staff agrees with the ACRS comment that materials and metallurgy research programs be forward looking to stay abreast of major issues that may be important for long-term operation. The recently completed effort on Expanded Materials Degradation Assessment (EMDA) was specifically conducted to provide a proactive evaluation of potentially significant degradation mechanisms occurring after 60 years of plant operation. The results from this exercise will be used to ensure that future research activities performed by both the NRC and the industry are adequately addressing principal gaps. In addition, current research activities related to reactor internals, reactor pressure vessel (RPV) embrittlement, and piping degradation are already addressing issues related to material degradation beyond 60 years of operation.

The staff agrees with the ACRS that the exposure of stainless steels and nickel-base alloys to high-temperature water in the absence of radiation damage can result in a significant decrease in tearing resistance and fracture toughness. This issue is being addressed by ongoing research at Argonne National Laboratory as a result of NRR User Need Request NRR-2012-008.

The staff agrees that our RPV integrity effort has to be well aligned with the EMDA findings and recommendations. None of the EMDA findings is surprising, and most issues have been identified for several years. The staff has already started addressing many of these issues and continues to focus the RPV integrity research on topics that may affect long-term operation such as embrittlement and pressurized thermal shock. RES is also pursuing a joint MOU with EPRI to ensure coordination on research important for long-term operation.

The staff agrees with the ACRS that the use of high-density polyethylene (HDPE) will become more prevalent as certain piping systems are replaced. To ensure that we have the proper understanding of the mechanical behavior of HDPE, research programs will continue to investigate the slow crack growth behavior of these materials and the capabilities of nondestructive evaluation to locate defects in the base and joint materials. The staff will continue to take an active role in the American Society of Mechanical Engineers Boiler and Pressure Vessel code activities to ensure the NRC is involved in the continuing development of HDPE.

The staff acknowledges the importance of active degradation mechanisms such as primary water stress corrosion cracking (PWSCC) to the long-term operation of the operating fleet. We have ongoing and planned research on PWSCC so the NRC can retain the expertise in this area. Our current work on PWSCC growth rates will continue as we further understand the effects of chromium level on the resistance of nickel-based alloys to stress corrosion cracking in pressurized-water reactor (PWR) environments. As part of this effort, the NRC will continue to participate in the Alloy 690 crack growth rate expert panel, which has provided a cooperative environment for the development of crack growth testing standards and data. In addition, we are embarking on a cooperative research program with EPRI on the crack initiation behavior of these alloys in PWR conditions. Finally, the NRC staff will continue to be active in the international community by supporting and participating in conferences and technical exchange meetings such as the International Cooperative Group on Environmentally Assisted Cracking.

The staff agrees with the ACRS on the importance of maintaining an awareness of industry research programs related to dry fuel storage canisters. The NRC staff monitors the Extended Storage Collaboration Program research. Several staff members regularly attend the industry meetings in this area. If any significant gaps are identified, staff will address them appropriately.

The staff acknowledges the ACRS recommendation that staff not pursue additional research related to Alloy 690 with the exception for special cases that may arise in the future. While Alloy 690 steam generator tubes have not experienced stress corrosion cracking during service up to now, there is laboratory evidence that Alloy 690 is susceptible to PWSCC under certain conditions. The staff has ongoing research that is being conducted in coordination with industry to understand both crack initiation and growth in Alloy 690 and its weld materials to more fully identify the materials conditions that may lead to PWSCC in service. This research also aims to determine the relative susceptibility to PWSCC of Alloy 690 and Alloy 600 materials to confirm the acceptability of current inspection intervals as well as other existing and planned industry mitigation methods. While this research is primarily focused on PWSCC in primary pressure boundary piping, there will undoubtedly be knowledge gained that will be applicable to steam generator tube performance. Additionally, to provide staff with reasonable assurance in the overall integrity of steam generator tubes, and inspection methodology to ensure that integrity, there are many areas of industry practice which need to be both monitored and independently verified through NRC research programs. The staff is currently engaged in research activity in those areas as requested by NRR in User Need Request NRR-2012-010.

CHAPTER 10 – NEUTRONICS AND CRITICALITY SAFETY

The staff appreciates the ACRS' feedback on the neutronics and criticality safety research program and agrees on the importance of both near-term and long-term research to the improvement of our analysis tools.

The staff appreciates the ACRS endorsement of our continued international benchmarking cooperation for our codes validation and our ongoing plans to upgrade our codes to accommodate new reactor designs, core configurations, and spent fuel systems. In addition to the cited OECD/NEA program, NRC is pursuing other experimental data and validation sources including bilateral efforts with its Japanese and Spanish counterparts. These bilateral efforts will yield previously unavailable high burnup critical and radio-assay measurements to validate our criticality safety models and methods.

CHAPTER 11 – OPERATIONAL EXPERIENCE

The ACRS recommended that staff examine the impact of uncertainty in success data in their applications and analyses. ACRS' comments address two potential contributors to the increased uncertainty in component failure rates:

1. Inaccuracy in plant-specific reporting of successes at a given site, and
2. Variability in the reporting both successes and failures from plant-to-plant.

Regarding inaccuracy in plant-specific reporting of successes, the Equipment Performance and Information Exchange (EPIX) was developed as an industry response to the proposed Data Rulemaking and to help facilitate Maintenance Rule compliance in the mid-'90s. Estimates of equipment performance have always been necessary when using EPIX as licensees have latitude in their reporting. Idaho National Laboratory (INL) staff reviews the failure events reported for each Mitigating Systems Performance Index (MSPI) component and for many non-MSPI components. If necessary, they apply engineering judgment to adjust or eliminate EPIX data entries that appear inconsistent with the associated failure event narrative. Examples of adjustments required include: licensees reporting run hours exceeding 24 hours/day, run hours for standby components that were a factor of ten too high, and emergency diesel generator load demands greater than the start demands. In other cases, faulty demand counts result in the component being eliminated from further analysis.

Uncertainty distributions are developed and included as part of the reporting of component reliabilities, but a quantitative estimate of the potential incremental uncertainty associated with inaccuracy in the demand count or per-hour run times has not been performed. Assessing the extent of this inaccuracy would be relatively resource intensive (e.g., time consuming and expensive), and may not result in a significantly different characterization of the overall uncertainty associated with component reliability. In essence, the staff believes that as long as the under-reporting of demands is not too extreme, and it is not perceived to be so at this time, slight overestimation of failure frequencies is not considered to be an impediment to successful risk-informed regulation. Furthermore, no significant change in the overall uncertainty characterization is anticipated.

Regarding plant-to-plant variability in reporting, the staff believes that the statistical process employed in the development of reliability estimates, outlined in Table 5-2, "Process for Collecting and Analyzing EPIX Data," of NUREG-6928, is adequate to handle potential data disparities. For instance, to accommodate plant-to-plant reporting variability, INL employs an Empirical Bayes approach, if possible, that attempts to capture the variability and display it via appropriate uncertainty distributions. (A standard Bayes update is employed in the event that there is not enough plant-specific information available. In that case, a non-informative prior distribution is used instead of the plant-specific one.)

Regarding expanding the previous agency work on failure rate estimates as part of NUREG/CR-5500, the current approach employed by the staff and INL to handle data variability is discussed in NUREG/CR-6928 and its annual Web-based updates, which supersede the 11-volume NUREG/CR-5500 series. NUREG/CR-5500 was limited to “high worth” safety systems, such as the reactor protection system, in which all demands (successes and failures) were recorded, studied, and well understood. EPIX by comparison, covers thousands of components in dozens of plant systems, both safety and non-safety. The Institute of Nuclear Power Operations, in encouraging industry-wide voluntary EPIX participation, has reporting standards that are considered to be more flexible (potentially not as rigorous) as those employed in the old Nuclear Plant Reliability Data System, the data source for NUREG/CR-5500. Consequently, expert data analyst judgment is occasionally required.

CHAPTER 12 – PROBABILISTIC RISK ASSESSMENT

The staff agrees that RES should continue to explore ways to increase the engagement of regional and headquarters staff in Standardized Plant Analysis Risk (SPAR) model maintenance and related reactor oversight support activities. The staff also appreciates ACRS support of research projects that would advance the state of the art in risk assessment methods and practices. However, the regions and headquarters offices currently lack the necessary PRA staff and resources to fully share these technical support activities. Moreover, developing this capability would likely divert resources from higher priority reactor oversight and licensing tasks. In addition, sharing of this work among a number of regional and headquarters offices would result in increased inefficiencies and would significantly increase overall program costs. Advantages of the current centralized approach for SPAR model maintenance and development include minimizing the number of highly skilled staff needed to support the program, increased consistency across the SPAR models, better version and configuration control, and the ability to more fully integrate the SPAR models with other PRA-related activities such as data collection, system studies, and new modeling improvements (e.g., support system initiators and enhanced loss of offsite power modeling).

RES appreciates ACRS support of collaborative research programs with external organizations such as EPRI and The National Aeronautics and Space Administration as well as university programs. These programs, particularly university research, provides a unique opportunity to gain access to state-of-the-art research programs and also provides beneficial research results for the agency's PRA programs. RES will continue to support these programs in a manner consistent with budget priorities.

RES appreciates that the ACRS recognizes the shift in PRA research priorities toward a more balanced approach consistent with our goal-oriented framework for PRA research activities. The staff intends to periodically re-evaluate this framework and resource allocations to ensure they appropriately reflect evolving agency priorities.

With regard to advanced reactor PRA research, the staff agrees that a pilot study of a technology neutral framework could provide valuable insights for future risk-informed licensing applications. However, as discussed in the January 30, 2014, NRO memorandum to the Commissioners (Agencywide Documents Access and Management System Accession No. ML13212A222), the staff cannot justify further efforts to develop a technology-neutral regulatory structure as an agency priority. Therefore, pending a change in agency priorities in this regard, further work to test the technology neutral framework has been suspended.

The staff agrees with the ACRS that a potential role exists for risk-informed approaches in the security of licensed facilities. At this time, RES has not received a user need request from the Office of Nuclear Security and Incident Response (NSIR), but discussions are ongoing.

The staff also agrees with the ACRS that a potential role exists for risk-informed approaches in the decommissioning of licensed facilities, and it continues to discuss needs in this area with the program offices. The staff also agrees that consideration should be given to identifying additional goals, including support for the materials and security areas, in the underlying strategic framework used to assess future research needs.

In response to previous ACRS comments, the staff has released for comment Revision 1 of NUREG-1855. Section 2.3, "Risk-Informed Decisionmaking Process," and Section 2.4, "Assessing the Impact of the Uncertainties," were rewritten to emphasize and clarify the roles of staff and industry. Guidance for clarifying uncertainties in the risk-informed decisionmaking process is organized into three parts: (1) determining the approach to use in the treatment of the uncertainties (both the licensee and NRC involved); (2) identifying and assessing the uncertainties (initially licensee then both licensee and staff); and (3) clarifying the staff review component of the risk-informed decisionmaking process.

CHAPTER 13 – RADIATION PROTECTION

The staff appreciates the ACRS endorsement of the current NRC research activities in the area of radiation protection. These activities include development and maintenance of health effect/dose calculation tools, emerging health effects and dosimetry research, and participation in a number of national and international collaborative radiation protection activities.

In addition, the staff is embarking on a new initiative to maintain, distribute, and develop all NRC-sponsored radiation protection/dose assessment codes through the Radiation Protection Code Analysis & Maintenance Program (RAMP). The purpose of RAMP is to develop a framework to coordinate and share experiences and resources with our domestic and international partners to raise the quality and efficiency of such codes. Codes in RAMP include, but are not limited to, Radiological Assessment Systems for Consequence Analysis (RASCAL); Radionuclide Transport, Removal, and Dose; HABIT (a code used to model control room habitability); and VARSKIN (a code used to model and calculate skin dose).

CHAPTER 14 – NUCLEAR MATERIALS AND WASTE

The RES project on electrochemical reprocessing was a long-term research plan project that provided a review of the current state of knowledge in this area. As the ACRS suggested, no further actions in this area are currently planned until more definitive proposals from potential licensees are presented.

CHAPTER 15 – SEISMIC AND STRUCTURAL ENGINEERING

The staff appreciates the feedback on the seismic research plan and agrees with the assessment that this program will adequately support the NRC staff's capabilities to evaluate potential risks to U.S. nuclear power plants due to events such as earthquakes and tsunamis. RES is involved in efforts to improve assessment of the seismic hazard at new and existing nuclear power plants. Collaborative efforts have resulted in a new seismic source characterization (SSC) model for the Central and Eastern United States (CEUS) and ongoing development of a ground motion attenuation model for the CEUS (NGA-East). Selected ongoing activities include enhancing the capability of probabilistic seismic hazard assessment software to perform seismic hazard calculations using the CEUS-SSC model and the latest ground motion attenuation relationships; developing software to improve modeling of local site amplification effects; further developing guidance on the treatment of uncertainty and the use of experts in hazard assessment; understanding the phenomenon of paleo-liquefaction as an aid in predicting large, but rare, earthquakes in the CEUS region; and developing probabilistic methods to evaluate landslide-based tsunami sources to support a new regulatory guide on tsunami hazard assessment.

RES anticipates a renewed emphasis on the research related to seismic fragility assessment of structures, systems, and components of nuclear power plants to support review guidance for the seismic reevaluations of operating reactors as well as licensing of new reactors. To inform and support subsequent license renewal guidance, RES is updating and revising aspects of its structural engineering program to improve understanding of containment degradation. This includes corrosion of steel liners or containment steel shells and loss of pre-stress, irradiation effects on concrete structures, and alkali silica reaction degradation of concrete structures. In addition, RES plans to continue confirmatory assessments of analysis methods and software that inform the review of impact loads on spent nuclear fuel transportation casks, security-related regulatory activities, and structural aspects of severe accident studies.

RES will continue to periodically update and revise the Seismic and Structural Research Plan to reflect the completion of certain projects and the basis and framework for new projects.

CHAPTER 16 – SEVERE ACCIDENTS AND SOURCE TERM

RES concurs with the ACRS observation that the agency investments in severe accident research in recent years have been limited to the level that allows the RES staff to perform only the confirmatory analyses and risk-informed activities required by the program offices. In the aftermath of Fukushima, however, the staff recognizes the desirability to conduct additional research to address emerging issues and to reduce uncertainties. To this end, RES is working closely with our user offices (NRR, NRO, NSIR, etc.,) to best utilize available resources to enhance the code capabilities to provide technical basis for regulatory analysis.

RES appreciates ACRS' continued support to NRC's approach in leveraging resources in severe accident research through international collaboration. RES continues to evaluate existing and planned international projects for their values in providing experimental data to reduce uncertainties in our code predictions. RES also continues to participate in selected international projects under bilateral and/or cooperative agreements. Finally, RES welcomes the ACRS plan to review the process the staff uses to determine the efficacy of new data with regard to model development/improvement.

RES agrees with the ACRS comment that the current generation models in system codes had their beginning in experimental databases that were largely focused on PWR-specific accident progression phenomena as best understood from the Three Mile Island accident. Over the years, modeling improvements were made to simulate boiling water reactor (BWR)-specific phenomena based on expert opinion and engineering judgment. Based on the latest initial and boundary conditions known to date, the MELCOR code gave reasonable predictions of the accident progressions of the Fukushima accident. Since the Fukushima accident provides data specific to BWR accident progression, the staff intends to build on the Fukushima experience to develop improved models for BWR-specific in-vessel melt progression; ex-vessel melt progression, particularly the effect of metallic contents in the melt on melt coolability and concrete interaction; debris coolability; melt spreading; and fuel-coolant interactions. The staff also plans to understand the salt (and raw) water effects on in-vessel and ex-vessel fission products behavior under a proposed international program to obtain important data during the Fukushima clean-up efforts. Efforts are underway to benchmark RASCAL and MELCOR Accident Consequence Code System against Fukushima data.

The staff agrees with the ACRS recommendation to preserve valuable severe accident experimental data by leveraging domestic and international programs such as the ones (EPRI, DOE, and Severe Accident Research NETwork of Excellence (SARNET)) mentioned by ACRS. The staff further notes additional international programs such as the Nuclear Generation II & III Association (follow-on to SARNET) and Severe Accident Facilities for European Safety Target─both under the auspices of European Commission─and plans to maintain active interest in these programs.

CHAPTER 17 – THERMAL HYDRAULICS "The TRACE thermal-hydraulic analysis code"

The staff appreciates the ACRS recognition of the importance of the TRACE code to the regulatory process. RES has put forth a substantial effort and takes great satisfaction in the improvement and use of our analysis codes. The staff agrees that thermal- hydraulic models with greater than three primary fields should be further developed in TRACE, and work is ongoing to add and improve this capability.

Regarding the ACRS' assertion that interfacial area transport research should be phased out, the staff believes that value still exists in these efforts. Development of the TRACE thermal-hydraulics code has been focused on its applicability to conventional and advanced light-water reactors and its incorporation into the regulatory process. Advanced development, intending to replace ad hoc and empirical correlations with mechanistic models, has continued at lower priority. As part of this advanced development, the capability of modeling multiple fields has been added to TRACE. A four-field formulation was recommended by the TRACE peer review group and, as part of that formulation, improved closure models to account for interaction between the fields are necessary. Successful (accurate) implementation of the four-field approach depends on having a good estimate of the void fraction and structure of the flow. This is important because this void fraction and structure determine the interfacial area concentration necessary for calculation of interfacial heat and mass transfer.

In the current nuclear reactor system analysis codes such as TRACE, the interfacial area concentration is calculated using flow regime dependent correlations. The use of steady-state-based flow regime transition criteria imposes shortcomings in modeling the detailed nature of the flow. Several problems are associated with this flow-regime-based static approach:

- Because the flow regime transition criteria are algebraic relations for steady-state, fully developed flows, they reflect neither the true dynamic nature of changes in the interfacial structure nor gradual regime transitions.

- The existing flow regime dependent correlations and criteria are valid in limited parameter ranges for certain operational conditions. Often the geometrical scale effects are not taken into account correctly. Hence, these models may cause significant discrepancies and numerical instabilities.

Some of our current results indicate that the interfacial area based upon these conventional flow pattern maps can be significantly in error. Therefore, we feel it is vital to continue a limited amount of research into interfacial area transport. The research expenditure in this area has been and will continue to be quite modest. The staff would be pleased to discuss this issue with the ACRS further if requested.

The staff appreciates the ACRS' recognition of the staff's participation in international programs, and we will continue these efforts to share expertise and to leverage resources. While we acknowledge the ACRS recommendation that more domestic experimental facilities should be developed, it must be recognized that new experimental facilities are resource intensive and should only be considered if no other cost-beneficial alternative exists for the specific issue to be addressed. The staff plans to continue its current strategy of collaboration with international and domestic institutions to obtain the data needed for validation of both systems- level and high-resolution models. The staff has found this approach to be a very cost-effective strategy.

ACRS noted that NRC currently has a modest effort in the area of computational fluid dynamics (CFD) that is based on the use of commercial CFD codes. ACRS also noted that licensees will inevitably capitalize on the extraordinary advances in computer power and computational science to resolve ever more complex multidimensional and multiphase safety issues. The NRC thermal-hydraulic staff is prepared to address these types of developments in industry and actively considers options to leverage our limited resources through national and international cooperation. Our current reliance on commercial codes is considered to be the most advantageous at this time.