5 R

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 11, 2001

MEMORANDUM TO: Richard R. Rough, Director
Division of Planning, Budget and Analysis
Office of the Chief Financial Officer

FROM: Mabel F. Lee, Director
Program Management, Policy Development
and Analysis Staff
Office of Nuclear Regulatory Research

SUBJECT: RES INPUT TO THE NRC'S FY 2003 BUDGET ESTIMATES AND PERFORMANCE PLAN TO CONGRESS (GREEN BOOK)

The following provides the current status of the input requested in your memorandum of November 21, 2001:

1. CRDS FY 2001 actuals: RES input was provided to OCFO on November 14, 2001.

2. FY 2001 data for program output measures: RES provided input to OCFO via memoranda dated November 21, 2001, and December 10, 2001, and e-mail dated November 29, 2001. The November 21, 2001 memorandum also includes RES' FY 2001 Program Accomplishments in the Reactor, Materials, and Waste Arenas (copy attached).

3. Region direct FTE: RES has no regional FTE.

4. Cross-cutting functions: RES has no input.

5. Updates to CRDS, arena narratives, and reimbursable work agreements: As discussed with the OCFO, RES is not able to determine whether revisions will be required for these documents until receipt of information concerning NRC's appeal of the FY 2003 OMB passback, decisions have been made concerning NRC FY 2002 and 2003 activities and associated resources related to the September 11, 2001 events, and completion of office/agency prioritization and add/shed processes, if required.

If you have any questions, please contact me or Gina Thompson of my staff (415-6381).

Attachment: As stated

cc w/attachment:
T. Pulliam, OCFO
C. Abbott, OCFO
R. Nubgaard, OCFO
R. Baum, OCFO
R. Speigel, OCFO
J. Dorfman, OCFO

4/15

# RES' FY 2001 ACCOMPLISHMENTS FOR NRC's PERFORMANCE REPORT

## Reactors

1. RES provided significant support to NRR in the review and inspection of two Pressurized Water Reactor (PWR) operational events related to primary loop pipe cracking and control rod drive mechanism (CRDM) nozzle cracking. The pipe cracking occurred in a weld between a primary coolant nozzle and the attached piping. The CRDM nozzle cracking included cracking in the welds and base metal, resulting in reactor coolant pressure boundary degradation.

With respect to the pipe cracking event, RES has contributed both staff and contractor inspection expertise at the plant site. In addition, RES has assisted in review of the metallurgical and fabrication details of the weld joint. With respect to the CRDM nozzle cracking, RES organized an expert panel to evaluate the industry's and the agency's actions. The RES staff and this panel provided initial assessment on the cracking phenomenon and the adequacy and timing of inspection methods. RES also assisted in the development of NRC's bulletin requesting plant specific information related to this issue. Subsequent to the issuance of the bulletin, RES has provided assistance in reviewing the responses to the bulletin and has conducted analysis to estimate failure probabilities of the cracked nozzle. The outcome of this research supports the performance goal of maintaining safety, particularly as plants age and the effects of material degradation continue.

2. To improve NRC's understanding of steam generator tube integrity, RES conducted a multi-phase research program to study the potential of propagating steam generator tube failures due to erosion by steam jets or high temperature gas/particle streams emanating from an adjacent tube rupture. A detailed study of jet flow characteristics using computational fluid dynamics was performed covering several different tube sizes, crack sizes and geometries. In addition, steam jet erosion tests representative of design bases accident conditions and high temperature tests representative of severe accident conditions were performed. Results from the analytical and experimental studies provided additional insights on jet propagation and its effects on tube erosion. Based on the research result, the staff has concluded that potential damage by jet impingement does not appear to be a safety concern. The outcomes of this research support the performance goal of maintaining safety by providing independent assessment of steam generator tube integrity in support of NRC's licensing decisions.

3. RES provided support in the assessment of key technology, design, safety, licensing and policy issues related to advanced reactor designs. This will aid in the development of guidance on the application of the regulatory process to advanced reactors, and development of a regulatory framework for licensing an advanced reactor design. For the Pebble Bed Modular Reactor (PBMR), RES held a series of public meetings with the industry, DOE, and the Advisory Committee on Reactor Safeguards (ACRS) on regulatory framework requirements and technical issues related to safety analysis and risk assessment. For the PBMR and Gas Turbine Modular Helium Reactor (GT-MHR), RES accomplishments include the following: (a)

code development, validation, and safety analysis; (b) staff visits to South Africa, the United Kingdom, Germany, China, and Japan to exchange experiences and insights on gas-cooled reactor technology and identify opportunities for international collaboration and cooperation; and (c) attended, along with staff from NRR and NMSS, a DOE-sponsored one-week course on High Temperature Gas-Cooled Reactor technology and safety. Outcomes of these efforts will contribute to both maintaining safety and reduction of unnecessary regulatory burden through development of technically defensible positions for licensing of advanced reactors.

4. RES has conducted plant aging research and supported NRC's review of license renewal applications. The scope of research included aging mechanism on electrical cables, mechanical components, as well as systems and structures. The research result has contributed significantly to the review and approval of license renewal applications which extended the plant operation for an additional 20 years. Results from the aging research also provided technical bases for resolution of generic license renewal technical issues and the development of license renewal guidance documents.

The outcome of this research contributes to the performance goal of maintaining safety by ensuring that, during the extended period of plant operation, aging effects will be adequately managed and the plant licensing basis will be maintained.

5. RES completed the safety significance determination process notebooks for all sites, reaching an important milestone in support of the reactor oversight process. The reactor oversight process uses a risk-informed approach to initiate NRC's inspection and assessment efforts at reactor sites. These notebooks are used as an initial screening tool to estimate the risk significance of inspection findings. The risk information is provided in a standard format so that risk significance of inspection findings can be determined and characterized in an efficient, objective, and consistent manner. In addition, it will help to focus NRC and licensee attention and resources on the most risk significant areas. Objective and consistent characterization of the risk significance of inspection findings should enhance public and stakeholder confidence in the reactor oversight process.

6. During the last three years, improved computational models have become available for the evaluation of pressurized thermal shock. During FY 2001, these improved models were incorporated into a computer code which is used for performing risk-informed structural integrity evaluations of embrittled reactor pressure vessels (RPVs) subjected to potential pressurized thermal shock phenomenon. The updated risk-informed computational methodology in the code is currently being applied to selected domestic pressurized water reactors to evaluate the adequacy of current regulations and to determine if a technical basis can be established to support reduction of unnecessary conservatism in current regulations.

The outcomes of this research support the performance goal of maintaining safety through a better understanding of the pressurized thermal shock phenomenon which could lead to longer life for reactor pressure vessels.

7. RES issued NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program." The NRC had requested that each reactor licensee identify and report all plant specific vulnerabilities to severe accidents caused by external events. Examples of external events considered by licensees included seismic events, internal fires,

high winds and floods. This report documents the staff's findings and conclusions from the reviews of all licensees' IPEEE submittals. As a result of the IPEEE program, over 90% of the licensees identified and implemented or proposed plant improvements. These improvements included new procedures or procedural changes, hardware modifications, and enhanced training. In addition, perspectives from the IPEEE program can be used for prioritizing areas for plant inspections, providing insights on the risk importance of inspection findings, incorporating risk insights into PRA standards for external events, and prioritizing areas for additional research (e.g., fire risk research program and age-degraded structures and passive components).

8. RES provided recommendations on revisions to 10 CFR 50.46 (Emergency Core Cooling System (ECCS) Acceptance Criteria). These recommendations include: (a) replacing the current prescriptive acceptance criteria with a performance-based requirement; and (b) allowing voluntary use of more realistic models in performing ECCS analyses. This paper also indicates that additional changes to 50.46 may be warranted. These changes, which relate to the scope of pipe break sizes to be considered in the ECCS analyses, require further technical evaluation and feasibility study. This feasibility study could require significant staff and industry resources, but could result in considerable reduction of unnecessary regulatory burden.

Materials

1. RES issued NUREG-1717, "Systematic Radiological Assessment of Exemptions for Source and Byproduct Materials." This report provides the technical basis that will support Commission decisions regarding the need for changes in regulations exempting certain specific products or materials from regulatory control and will reduce regulatory burden for potential licensees.

Waste

1. RES completed the effort to provide the technical basis for reducing the unnecessary restrictions in the interim staff guidance for calculating the amount of spent fuel allowed in spent fuel transportation casks. More realistic analysis was used to account for spent fuel rather than fresh fuel in transportation casks. This analysis reduced unnecessary conservatism in earlier calculations and increased the number of spent fuel assemblies allowed to be loaded in each cask as well as loading spent fuel assemblies with higher enrichment. This revision provided a significant regulatory burden reduction by allowing transportation of more spent fuel assemblies in casks which results in increased safety by reducing worker exposure and requiring fewer casks to be transported.

2. RES completed the final report documenting the inspections and examinations performed on the Castor-V/21 cask that has been at the Idaho National Engineering and Environmental Laboratory (INEEL) since 1985. The purpose of this cooperative research was to provide the technical basis for renewal of licenses and Certificates of Compliance for dry storage systems for spent nuclear fuel and high-level radioactive waste. The fuel has been out of the reactor for approximately 20 years, and has been in continuous storage in this cask for approximately 15 years. The cask was reopened and the cask internals, fuel assemblies and several rods were

visually inspected. There is no evidence of significant degradation of the Castor-V/21 cask systems from the time of initial cask loading in 1985 up to the time of testing in 1999.

This RES project will support the performance goal of maintaining safety by providing data that can be used to confirm the predicted long-term integrity of dry cask storage components under dry storage conditions, and to augment the technical bases for evaluating the safety of spent fuel storage and transportation systems and for extending the dry cask storage licenses.