S. Hrg. 114–159

OVERSIGHT OF THE NUCLEAR REGULATORY COMMISSION

HEARING

BEFORE THE

COMMITTEE ON ENVIRONMENT AND PUBLIC WORKS UNITED STATES SENATE

ONE HUNDRED FOURTEENTH CONGRESS

FIRST SESSION

OCTOBER 7, 2015

Printed for the use of the Committee on Environment and Public Works

Available via the World Wide Web: http://www.gpo.gov/fdsys

U.S. GOVERNMENT PUBLISHING OFFICE

98–268 PDF WASHINGTON : 2016

For sale by the Superintendent of Documents, U.S. Government Publishing Office
Internet: bookstore.gpo.gov Phone: toll free (866) 512–1800; DC area (202) 512–1800
Fax: (202) 512–2104 Mail: Stop IDCC, Washington, DC 20402–0001

COMMITTEE ON ENVIRONMENT AND PUBLIC WORKS

ONE HUNDRED FOURTEENTH CONGRESS
FIRST SESSION

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WITNESSES

OVERSIGHT OF THE NUCLEAR REGULATORY COMMISSION

WEDNESDAY, OCTOBER 7, 2015

U.S. SENATE,
COMMITTEE ON ENVIRONMENT AND PUBLIC WORKS,
Washington, DC.

The full committee met, pursuant to notice, at 9:34 a.m. in room 406, Dirksen Senate Office Building, Hon. James M. Inhofe (chairman of the full committee) presiding.

Present: Senators Inhofe, Boxer, Barrasso, Capito, Boozman, Fischer, Rounds, Carper, Cardin, Whitehouse, Gillibrand, Booker, and Markey.

OPENING STATEMENT OF HON. JAMES M. INHOFE, U.S. SENATOR FROM THE STATE OF OKLAHOMA

Senator INHOFE. The meeting will come to order.

If you remember the last time we met, I made the comment that there are nine people who are on both the Armed Services Committee and this committee, so we set up something where we are not going to coincide. Historically, we have always had the meeting at 9:30 on Armed Services on both Tuesday and Thursday. Well, they decided to have one today. So that shows how much influence I have over there.

This hearing is part of an ongoing oversight on NRC's decision-making on fiscal and policy matters.

I would like to begin by welcoming our four commissioners. We appreciate very much your being here. We have received the President's nomination of Mrs. Jessie Robertson for the open seat, and I expect to proceed with a hearing on her nomination once my colleagues have had a chance to visit with her in person. So you might share that with her so we can make that happen.

We will continue with the committee's practice of a 5-minute opening statement for the chairman and then 2 minutes for each commissioner, and then we will be asking questions.

The NRC's mission is a vital one and must be adequately funded. I want our nuclear plants to be safe, and they are safe. Following Fukushima, I urged the Commission to perform a gap analysis to assess the difference between the basic regulations that they had in Japan, as opposed to what we had in this country, because a lot of people were laboring under the misconception that it was the same, and it wasn't. So we were far ahead of them to start with.

Four and a half years later, the industry has spent more than $4 billion and the NRC staff has repeatedly sent proposals to the Commission, which they admit are not safe, significant, or cost-justified. I believe this shows the NRC's bureaucracy has grown beyond the size needed to accomplish the mission.

Now, this is a chart that we are using here, and those who have been on this committee for a while know that we beefed up because we are anticipating something that never did happen, and then you don't beef up after that. So that is kind of the thrust, at least my thrust, in this committee hearing today.

Ten years ago, the NRC accomplished a lot more work with fewer resources. Despite the shrinking industry, the NRC continued to grow, and you can see that in this chart. Over the last few years we have increased our oversight of the NRC's budget and raised concerns about: one, the NRC's extreme level of corporate overhead costs; two, the reactor oversight, spending increasing, despite the decline in operating reactors; three, over-budgeting for the new reactors, work that no longer exists; and, four, persistent carryover funds.

In response to this scrutiny, the Commission initiated Project Aim 2020 to right-size the agency, and I would like to take the NRC for its word. However, I am struggling to reconcile this with the NRC's recent response to the Senate appropriators.

Lamar Alexander spent a lot of time looking at this, saying what we should do from an appropriation perspective. Then I have the response. I do want to make this response, without objection, a part of the record; and I think several of my colleagues here are going to be asking some questions about that. So it is now part of the record.

[The referenced information follows:]

NUCLEAR REGULATORY COMMISSION 2005-2015

workload

UNITED STATES NUCLEAR REGULATORY COMMISSION

resources

Operating Reactors - **5%** Decrease
Licensing Actions - **40%** Decrease
Materials Licensees - **36%** Decrease
License Renewal - **43%** Decrease

NRC Budget - **52%** Increase
Full-Time Employees - **22%** Increase
Licensee Fees - **68%** Increase
Corporate Overhead - **41%** Increase

Senator INHOFE. Rather than seize this as an opportunity to be proactive in the spirit of Project Aim, the NRC took the posture of a bureaucracy, fighting to maintain every nickel of spending. I consider this irresponsible. The situation is strikingly similar to the state of the agency when I took over.

I took over as chairman of this subcommittee in 1997. At that time, there had not been an oversight hearing in 4 years. Four years. And that can't happen. So we did, we put targets out there as to how often we were going to be having them. I think we need to go back to that and pay a little bit more oversight attention.

Now, given the NRC's response to appropriators, I don't have confidence that the agency will diligently address the need to reform on its own. I believe it is time for oversight to take place.

I intend to draft legislation to reform the NRC's budget structure and fee collection in an effort to instill fiscal discipline in the agency and ensure that resources are properly focused on safety, significant matters, timely decisions are made on matters.

Senator Boxer.

[The prepared statement of Senator Inhofe follows:]

STATEMENT OF HON. JAMES M. INHOFE,
U.S. SENATOR FROM THE STATE OF OKLAHOMA

This hearing is part of our ongoing oversight into the NRC's decisionmaking on fiscal and policy matters. I'd like to begin by welcoming the four commissioners.

We have received the President's nomination of Mrs. Jessie Roberson for the open seat, and I expect to proceed with a hearing on her nomination once my colleagues and I have visited with her.

We will continue with the committee's practice of a 5-minute opening statement from Chairman Burns and 2 minutes for each of the commissioners.

The NRC's mission is a vital one and must be adequately funded. I want our nuclear plants to be safe, and they are safe.

Following Fukushima, I urged the Commission to perform a "gap analysis" to assess the differences between our regulations and those of the Japanese, in order to guide what regulatory changes might be needed. Instead of taking that approach, the Commission empowered the NRC staff to develop a wish list of more than 40 items including restructuring the regulatory framework.

Four and a half years later, the industry has spent more than $4 billion and the NRC staff has repeatedly sent proposals to the Commission, which they admit are not safety significant or cost justified.

I believe this shows the NRC's bureaucracy has grown beyond the size needed to accomplish its mission.

Ten years ago, the NRC accomplished a lot more work with fewer resources. Despite a shrinking industry, the NRC has continued to grow.

Over the last few years we have increased our oversight of the NRC's budget and raised concerns about:

- The NRC's extreme level of corporate overhead costs;
- Reactor oversight spending increasing despite the decline in operating reactors;
- Over-budgeting for New Reactors work that no longer exists; and
- Persistent carry-over funds.

In response to this scrutiny, the Commission initiated "Project Aim 2020" to "right-size" the agency. I would like to take the NRC at its word.

However, I am struggling to reconcile this with the NRC's recent response to Senate appropriators when asked about the impact of a possible $30 million decrease for fiscal year 2016—a mere 3 percent of their budget.

Rather than seize this as an opportunity to be proactive in the spirit of Project Aim, the NRC took the posture of a bureaucracy fighting to maintain every nickel of spending. I consider this irresponsible.

This situation is strikingly similar to the state of the agency when I took over as subcommittee chair in 1997.

Given the NRC's response to appropriators, I don't have confidence the agency will diligently address the need for reform on its own. I believe it's time for Congress to step in.

I intend to draft legislation to reform the NRC's budget structure and fee collection in an effort to instill fiscal discipline in the agency and ensure that resources are properly focused on safety-significant matters and timely decisionmaking.

OPENING STATEMENT OF HON. BARBARA BOXER, U.S. SENATOR FROM THE STATE OF CALIFORNIA

Senator BOXER. Thank you so much, Mr. Chairman. I wanted to thank you and the staff because you moved this up to 9:30 because we asked you to because we thought we had something at 10, and it turns out we didn't.

Senator INHOFE. But in Armed Services we do, so that is the problem.

Senator BOXER. It is hard to do all this.

I respect your looking at the fiscal issues surrounding the Commission. As you know, my focus has been really the slow pace at which the NRC is implementing measures to protect American nuclear plants in the wake of the earthquake, tsunami, and nuclear meltdowns that occurred in Japan in March 2011. So we have different focuses, which is fine.

Only one of Japan's 43 nuclear reactors has been turned back on since the Fukushima disaster. A recent Reuters analysis found that of the other 42 operable nuclear reactors in Japan, only 7, only 7 out of 42 are likely to be turned on in the next few years.

For the last 4 years I have been saying that in order to earn the confidence of the public, we must learn from Fukushima and do everything we can to avoid similar disasters here in America. Following the last NRC oversight hearing in April, I met with Chairman Burns to discuss the Commission's progress on implementation of the Fukushima Near-Term Task Force recommendations. I do appreciate the letter that you sent to me after our meeting outlining the status of the Commission's work and timelines for completing each of the recommendations.

While I recognize progress has been made in some of the areas, I am frustrated and disappointed with the overall slow pace. Not one of the 12 task force recommendations has been fully implemented, and I think we have a chart that shows this. Many of the recommendations still have no timeline for action.

I am also concerned with some of the decisions NRC is making on whether to implement important safety enhancements. For example, the Commission overruled staff safety recommendations. They overruled their staff and voted not to move forward with multiple safety improvements. By a 3 to 1 vote, the Commission decided to remove a requirement that nuclear plants have procedures in place for dealing with severe accidents.

What is wrong? How can we vote that way? How does this make any sense?

This requirement was identified in the aftermath of Fukushima, but, after years of work, the Commission chose not to move forward. This is unacceptable.

The Commission, in my view, is not living up to its own mission, which I always read to you to instill in you this burning desire for safety. This is your mission: "To ensure the safe use of radioactive materials for beneficial civilian purposes, while protecting people and the environment." That is your goal. Not to build new nuclear plants as fast as you can, or walk away from your own ideas on how to make plants safer.

We need to look no further than the two nuclear power plants in my State. At California's Diablo Canyon Power Plant, NRC has repeated declared the plant safe, even after learning of a strong earthquake fault near the plant, which wasn't known about when the plant was approved. If you asked the average person on the street, I don't care if they are Republican, a Democrat, a liberal, a conservative, or anything in between, do you think you ought to build a nuclear power plant near a really big earthquake fault, I think they would say no. And I don't think they would need a degree in nuclear science to get the fact that that is not safe. So when you hear of a new fault, and for you not to take any action is very shocking to me.

At the San Onofre Nuclear Generating Station in San Diego, which has closed permanently, the NRC recently issued exemptions to emergency planning requirements. We still have a lot of nuclear waste there. There are so many millions of people who live around that plant. The plant's operator, because of your decision, will no longer be required to maintain detailed plans for evacuation, sheltering, and medical treatment of people residing in the 10-mile zone around the plant should something go wrong.

I am aware that NRC is planning a rulemaking on decommissioning issues, but rubber-stamping exemptions the way the Commission is the wrong approach. I believe it is wrong to relax emergency planning requirements with thousands of tons of extremely radioactive spent fuel remaining at the site. The millions of people, my constituents, they write to me. They are scared. They are really glad that place closed, but they are scared because they don't see the kind of attention being paid to their safety.

The NRC owes it to the citizens of California and to the Nation to make safety the highest priority, and I urge all the commissioners to rethink this, refocus. Think about why you are there.

And I do look forward to discussing these issues with you today. I know you don't look forward to it, but I look forward to it.

Thank you.

Senator INHOFE. Thank you, Senator Boxer.

We will take a moment to congratulate Victor McCree, hold your hand up so everyone knows who you are, on his promotion as Executive Director. It is kind of coincidental; last night I was at an event and three different people came up to me and were singing your praises. So we are looking for great things, and I am hoping that after this meeting concludes you won't change your mind.

[Laughter.]

Senator INHOFE. He is a graduate of the Naval Academy. That gives you and Commissioner Ostendorff something to talk to him about, so I think you will be a welcome addition there.

Senator Rounds.

Oh, I am sorry, we will start with the chairman for your 5 minutes, and then we will go down and hear from the rest of the commissioners. You are recognized.

STATEMENT OF STEPHEN BURNS, CHAIRMAN, U.S. NUCLEAR REGULATORY COMMISSION

Mr. BURNS. Thank you, Chairman Inhofe, Ranking Member Boxer, Chairman Capito, Ranking Member Carper, and distinguished members of the committee. We are pleased to provide an update this morning on the Nuclear Regulatory Commission's activities.

As you know, in response to earlier industry plans to construct a new fleet of reactors, the NRC recruited staff and enhanced our licensing capability. Today, only 6 applications remain active, out of 18 combined applications originally submitted. Two early site permit requests are under review, not the expected four, and two standardized plant design certifications, instead of the anticipated four, remain on the docket.

The focus of the NRC's work has also shifted in other areas over the last decade. Interest in new reactors is growing. There has been a focus on security, of course, after the events of 9/11. We are also working on license renewal, looking at power uprates, overseeing decommissioning, and, importantly, implementing safety enhancements spurred by the Fukushima Daiichi Nuclear Power Plant accident.

To meet the workload challenges, we are instituting organizational and budget realignments under Project Aim 2020. We are identifying the work most important to our mission, as well as the activities that can be shed, deprioritized, or performed with a reduced commitment of resources.

Rebaselining is a central element of the Project Aim initiative. The NRC has about 3,628 full-time equivalent staff, down from about 3,960 in fiscal year 2010. Our target is 3,600 by the end of this fiscal year. This excludes the Office of the Inspector General in those numbers.

But, importantly, Project Aim will improve our ability to respond to change, to plan and to execute our important safety and security mission. But we must monitor attrition and recruit with care to retain appropriate expertise in the agency. Our success as an agency is due to our highly trained and knowledgeable staff and their commitment to our mission has established worldwide our reputation as a strong, independent, and competent regulator.

Overseeing the most safety-significant enhancements stemming from the Fukushima accident remains a priority. Most licensees will complete the highest priority work by the end of 2016. This will substantially improve the already significant capabilities of U.S. nuclear plants and provide further assurance that they can cope with extreme natural hazards or events.

The NRC technical staff is reevaluating plans for the remaining longer-term or lower priority recommendations and will present the Commission with a paper later this month or next month, and we will be meeting on that in the near future.

The Commission has also directed its staff to submit a proposal for increasing the Commission's involvement in the rulemaking process. The goal is for the Commission to be more involved early in the process, before significant resources are expended.

Being prepared to evaluate applications for light water-based small modular reactors, as well as non-light water technologies, presents challenges, but we are prepared to review any applications under our existing framework. Within budget constraints, the agency is working on advanced reactor activities with the Department of Energy, industry standard setting organizations, and the Generation IV International Forum. We expect to receive a small modular reactor design application in late 2016.

Finally, I would like to touch on this topic of spent nuclear fuel. The NRC has received two letters from potential applicants indicating intent to apply for a consolidated interim storage facility license. The NRC does not have resources budgeted for either review this fiscal year but could reprioritize work if need be. The NRC has previously issued a license to authorize an independent spent fuel storage facility—private fuel storage in Utah, but construction of that facility did not go forward.

In conclusion, as I have noted many times since becoming chairman, I am very proud to be part of this organization. The NRC has a prestigious history and is viewed worldwide as a premier regulator. I am repeatedly reminded of the NRC's importance and the excellence with which we pursue our work. We are in a sustainable path toward reshaping the agency, while retaining the skill sets necessary to fulfill our safety and security mission.

Thank you, and I would be pleased to answer your questions.

[The prepared statement of Mr. Burns follows:]

STATEMENT OF STEPHEN G. BURNS, CHAIRMAN

U.S. NUCLEAR REGULATORY COMMISSION

BEFORE THE

SENATE ENVIRONMENT AND PUBLIC WORKS COMMITTEE

Oct. 7, 2015

Chairman Inhofe, Ranking Member Boxer, Chairwoman Capito, Ranking Member Carper, and distinguished Members of the Committee, my colleagues and I appreciate the opportunity to testify this morning to provide an update on the U.S. Nuclear Regulatory Commission's (NRC) licensing and regulatory activities. I will be providing a brief update on the status of our "rebaselining efforts," as well as the progress in achieving post-Fukushima safety enhancements and improvements in our rulemaking process. I will also provide updates on advanced reactor, decommissioning and spent nuclear fuel storage activities.

As you know, the nuclear industry has been in a period of change since the early 2000s. At that time, in response to the industry's plans to construct a new fleet of reactors, the NRC aggressively recruited staff and restructured the agency's licensing organization for reactors. The intent was to ensure the continued safety and security of the operating units even as the agency reviewed new plant designs and reactor license applications that we expected would exceed 20 for more than 30 new reactors.

It is a different picture today. Only six applications remain active out of the 18 combined license applications that were filed. Five units have been issued combined licenses authorizing their construction and operation, with but a handful still under review. The agency has two early site permit requests under review – not the expected four – and two standardized plant design certifications – not the anticipated four. Two design certification renewal applications remain under review.

The focus of the NRC's work also shifted in other areas. Interest in advanced reactors is growing, and by late 2016, the agency expects to receive a small modular reactor design certification application and an application for an early site permit for a small modular reactor. The agency also is now reviewing two construction permit applications for facilities that would produce medical isotopes and expects to make a licensing decision on one of the applications by early next year; the nation currently has no such facility and is dependent on imports.

The agency also put a greater focus on security, safeguards, and emergency preparedness since the terror attacks of September 11, 2001. Work on license renewals, power uprates, and the Yucca Mountain high-level waste repository application required resources, as did implementing the safety enhancements precipitated by the March 2011, accident at the Fukushima Dai-ichi nuclear power station in Japan. Work also came from the unexpected decommissioning of several reactors before the end of their licensing term, and the shift in nuclear materials work, propelled by an increase in licensing activities related to uranium recovery facilities.

Despite these workload challenges, the agency has remained a competent and respected regulator. The NRC has responded decisively to the new challenges before us and adjusted the trajectory of the agency to institute organizational and budget realignments to better position ourselves for our present and future workload.

Our Project Aim 2020 initiative leveraged the talents of a team of senior officials who worked with staff throughout the agency, and consulted with external experts, to draft recommendations to streamline processes, reduce the size of the workforce, and improve the effectiveness and timeliness of regulatory decision-making. The Commission directed the staff to reassess the agency's workload and to prioritize activities that could be reduced or eliminated. The staff submitted several papers to the Commission in late August regarding its efforts, and a public Commission meeting was held last month to discuss these efforts. We've received more than 400 comments; fewer than 100 came from external stakeholders with the remaining ones from NRC employees on how to achieve Project Aim objectives.

A central element of the Project Aim effort is the rebaselining process. In our direction to staff, my colleagues and I made clear that the focus should be on identifying what work is most important to the safety and security mission of the agency, and what activities can be shed, de-prioritized, or performed with a less intense resource commitment.

While Project Aim will build an organizational structure that improves the NRC's ability to respond to change, plan, and execute our mission, we must be careful to maintain the expertise needed to do our job. The NRC currently has approximately 3,628 full-time equivalents (FTE).

This is down from a peak of about 3,960 FTE in fiscal year 2010. Under Project Aim, our staffing target is 3,600 FTE by the end of fiscal 2016. These numbers do not include the NRC Office of the Inspector General, which has a separate staffing allocation.

While reaching that goal, we must retain key personnel. The NRC has acquired expertise in mission-critical areas such as nuclear, chemical, structural, and fire protection engineering; health physics and physical science; earth sciences including hydrology, meteorology, seismology, and geology; economics; information technology systems; and computer and physical security, among others. As we monitor our attrition and recruit with care, we must remain vigilant to retain appropriate expertise.

While this effort is ongoing, the Commission must continue to emphasize both the importance of our mission and the excellence with which we achieve it. Our success is largely due to the dedicated, highly trained, and knowledgeable NRC staff. It is the staff's professionalism and commitment to maintaining the safe and secure use of nuclear materials and facilities that has established NRC's worldwide reputation as a strong, independent, and competent regulator.

Fukushima-Related Safety Activities

Even as we rebaseline, we remain committed to ensuring the most safety significant of the enhancements that stemmed from the Japanese nuclear accident at Fukushima Dai-ichi remain a priority. Most licensees will complete the majority of the highest priority enhancements by the end of 2016. This will be a significant achievement.

You may recall just two weeks after the accident at Fukushima Dai-ichi, the Commission directed a task force of senior NRC staff members to make recommendations for strengthening safety at U.S. nuclear power plants. This Near-Term Task Force provided a preliminary, first-cut set of 12 recommendations after a 90-day review. Those recommendations became the starting point for a more in-depth assessment that considered input from the public, stakeholders, additional NRC staff members, and the Commission. The result of the more detailed assessment was prioritization of the most significant work, which was implemented through a series of NRC orders, requests for information, and rulemaking.

The highest-priority work focused on: strategies for mitigating impacts of events that are beyond those the plant was originally designed to withstand; improved instruments for measuring the water level in spent fuel pools; seismic and flooding walk downs (visual inspections); updated reevaluations of flooding and earthquake hazards at each site; severe-accident capable vents for BWR reactors with Mark I and II containments (similar types of containments to those at the Fukushima station); and enhancements to emergency preparedness communications and staffing.

These safety enhancements will substantially improve the already robust prevention, mitigation, and emergency response capabilities of U.S. nuclear power plants and provide further assurance that these plants can effectively cope with extreme natural hazards or other events.

Some Task Force recommendations were merged into ongoing or completed work, and other recommendations, upon reevaluation, were assessed as not providing sufficient, substantial safety enhancements that would merit further regulatory actions. The NRC technical staff is currently reevaluating the plans for the remaining longer-term or lower-priority recommendations and will provide a paper to the Commission later this year.

Rulemaking Process and Other Regulatory Improvements

As we streamline the organization as a whole, the Commission is also working to improve the effectiveness and efficiency of our regulatory processes. The Commission recently directed the staff to submit a proposal for increasing the Commission's involvement in the rulemaking process. The goal is for the Commission to be more involved during early stages of the rulemaking process -- before significant agency resources are expended. We are mindful of Congress's interest in this goal as well. The staff's proposal, due in just a few weeks, will include a recommendation on whether to reintroduce Commission approval of the "Rulemaking Activity Plan," as was the practice in the late 1990s and early 2000s.

Separately, the agency has been examining ways over the past several years to mitigate the cumulative effects of regulations and to improve the assessment of benefits, costs, and timing associated with implementing new regulations. The NRC staff has increased public input through all phases of the rulemaking process. There would also be an opportunity for the regulated community to provide feedback about potential adverse impacts from the implementation of proposed new requirements. In addition, the agency has engaged with the industry to develop more accurate cost estimates of new requirements, since these estimates inform the agency's decision about whether and how to pursue new requirements.

The agency's use of quantitative and qualitative factors in its regulatory decision-making has been of high interest to stakeholders in recent years. The Commission recently approved the staff's plans for updating guidance regarding the use of qualitative factors to improve the clarity, transparency, and consistency of the agency's regulatory and backfit analyses.

Specifically, the updated guidance should support regulatory analyses that clearly present the analyst's consideration of qualitative factors in a transparent way that decision-makers, stakeholders, and the public can understand. This approval does not authorize an expansion of the consideration of qualitative factors in regulatory analyses and backfit analyses. The Commission specifically directed that the revised guidance encourage quantifying costs to the extent possible and use of qualitative factors to inform decision-making, in limited cases, when quantitative analyses are not possible or practical (i.e., due to lack of methodologies or data). As stated in the Commission's direction to the staff, the appropriate weighting of qualitative factors in regulatory decision-making ultimately lies with the Commission. As this work is ongoing, the Commission will continue to pay close attention to this element of our work.

It is important to note the agency has a statutory mandate to provide reasonable assurance of adequate protection of public health and safety, and when establishing that level of adequacy, the Commission does not consider costs, although the Commission may consider costs in selecting between alternative methods of achieving adequate protection. Most of the NRC's regulatory framework today has been established on the basis of adequate protection. That said, the Commission has recognized that it must be deliberate, judicious, and predictable when it comes to establishing new regulatory requirements on the basis of adequate protection.

Another initiative instituted last year focused on decreasing the agency's backlog of power reactor licensing activities, with the ultimate goal to eliminate it. Already, in less than a year, we've seen improvement in this area, as we have reallocated resources from lower priority work and expanded the use of contractor support.

Advanced Reactors

Being prepared to evaluate potential applications for light water-based small modular reactors and non-light water reactor technologies presents some challenges for the NRC, but the NRC is prepared to receive and review any such applications under its existing framework. To this end, the NRC has been proactive within the framework of its largely fee-based approach to regulatory reviews. Within the constraints of our budget, the agency is working on advanced reactor activities with the Department of Energy, industry standard-setting organizations, and with the Generation IV International Forum. The NRC expects to begin reviewing one small modular reactor design application in late 2016. The NRC is also preparing for potential advanced, non-light-water reactor power applications in the future.

However, because the NRC's current reactor licensing regulations and guidance documents were developed based primarily on light-water reactor technologies, the agency recognizes the potential knowledge gaps for both the staff and prospective applicants. In addition, if the NRC were to receive an advanced reactor application within the next five years, there may be challenges related to research and modeling work in both the technical issues and code development for non-light-water reactor designs, as well as some critical skill gaps.

Decommissioning

Over the past few years, five reactors permanently ceased operation earlier than anticipated and began the process of decommissioning. These reactors joined 14 other units in some stage of decommissioning under NRC oversight. In addition, Oyster Creek announced it plans to close in 2019, and there are indications other plants may shut down before the expiration of their operating licenses due to economic conditions. The NRC has traditionally used operating reactor regulations for plants undergoing decommissioning, thereby requiring the plants to seek exemptions when the regulations for operating reactors are no longer relevant or appropriate. While this approach is sound from a safety standpoint, the Commission has directed the NRC staff to initiate a process for developing a reactor decommissioning rulemaking, with a final rule to be issued by early 2019. We expect this rulemaking will improve the effectiveness and transparency of the decommissioning process.

High-Level Waste, Spent Nuclear Fuel

Finally, I'd like to touch on the important topic of high-level waste. The NRC has been responsive to judicial direction to review the construction authorization application for the spent fuel repository at Yucca Mountain with the carryover resources the NRC has available. On August 13, 2015, the NRC issued a draft to the Department of Energy's supplemental Environmental Impact Statement on potential groundwater impacts. To date, the agency has held three public meetings to solicit input. The comment period will end on November 20, 2015, and a final supplement is anticipated to be issued in early 2016.

Regarding interim storage facilities, in the past several months, the NRC has received two letters from potential applicants indicating their intent to submit a filing for a consolidated interim storage facility. One facility would potentially be located in Andrews County, Texas, and the other in southeastern New Mexico. The NRC does not have resources budgeted for either review in fiscal year 2016, but could reprioritize work if applications are submitted. The NRC has previously issued a license that would authorize an independent spent fuel storage facility – Private Fuel Storage in Skull Valley, Utah – using its current regulatory structure, although construction of that facility did not ultimately go forward.

Conclusion

As I have noted numerous times since becoming Chairman, I am extremely proud to be a part of this organization. The NRC has a long, prestigious history and is viewed world-wide as a premier regulator. I dedicated the majority of my career to this agency and its mission, and am repeatedly reminded of the NRC's importance and the excellence with which it pursues its work. We must not lose sight of our critical mission, and the high esteem with which we are held by our counterparts in the United States and around the world.

We are on a sustainable path toward reshaping the agency to meet our changing environment while retaining the right skill sets to fulfill our safety and security mission.

Thank you, and I would be pleased to answer your questions.

The Honorable James Inhofe

QUESTION 1. **Of the research projects listed in the NRC's October 6, 2015 letter response to Senator Inhofe's requests for research information, please rank them according to their relative risk reduction or safety significance.**

ANSWER.

The agency does not calculate risk reduction or safety significance related to its research projects and therefore cannot rank these projects in such terms. The NRC primary research focus is on independent confirmatory work. The research is intended to ensure that current regulatory requirements remain valid and continue to adequately protect public health and safety.

QUESTION 2. **For each of the research projects listed in the NRC's October 6, 2015, letter response to Senator Inhofe's requests for research information, please identify whether the project resulted from: program office user need requests; recommendations from technical advisory groups; recommendations by research contractors; lessons from operating experience and inspection findings; research results; or peer review.**

ANSWER.

The attachment to this question provides details on the driver for the projects listed in the NRC's October 6, 2015 response.

QUESTION 3. **Of the research projects listed in the NRC's October 6, 2015, letter response to Senator Inhofe's requests for research information, 99 of them are listed to have durations of "greater than a year." Considering the NRC's draft budget for FY 2017 has already been submitted to the Office of Management and Budget, more detail should be available regarding the duration of these projects. Please provide additional detail regarding:**

a. The date these projects began; and

b. Whether the research projects will continue beyond FY 2017 and, if so, for how long?

ANSWER.

a. The attachment to this question provides estimated start dates for the 99 projects noted to have durations "greater than a year."

b. The President has not yet proposed a budget for FY 2017. Consequently, the NRC cannot definitively state which research projects will continue beyond FY 2017 or the lifespans of those projects. Once the NRC receives its fiscal year appropriation, it will prioritize its resources at the project level, based on the project status and the NRC's current priorities.

QUESTION 4. **The total cost of the research projects listed in the NRC's October 6, 2015 letter response to Senator Inhofe's requests for research information is $48.7 million and was described as "the contract support funds executed in the cost center as reflected in the budget table provided" in the NRC response dated September 17, 2015. However, that amount is $55 million. Please clarify the discrepancy.**

ANSWER.

The October 6, 2015, response reflected contract support funds managed by the Office of Nuclear Regulatory Research and did not include the infrastructure items related to International Research (~$3 million), and Mission Information Technology (~$4 million), which were part of the $55 million reported in the September 17, 2015, response.

QUESTION 5. **Please provide the ratio of the number of Office of Research's FTEs to the number of contractor FTEs utilized by the Office of Research.**

ANSWER.

Information responsive to this question will take additional time to prepare. We will provide that additional information not later than January 12, 2016.

QUESTION 6. **In 2005, the NRC IG reported a finding that the NRC needed to update its budget formulation procedure. Please provide a date by which the Commission will implement an up-to-date budget formulation procedure and the fiscal year for which budget development will utilize the new procedure.**

ANSWER.

Annually, in the February timeframe the formal budget instructions are issued by the Office of the Chief Financial Officer, which provide an up-to-date process for budget formulation procedures and are issued to guide the development of budget input. These instructions provide specific program direction, including fiscal guidance for each business line, and workload expectations for projected workload and outputs. This ensures the budget developed by business and product line lead offices are aligned with the agency's strategic direction. The budget instructions also include Commission direction that impacts budget formulation for the current year.

The agency is in the process of revising Management Directive (MD) 4.7, "Budget Formulation," which provides an overview of the roles and responsibilities for the Commission and agency offices in the budget formulation process. The Management Directive also provides a description of the various steps and work products related to the NRC's internal and external budget formulation process. On November 3, 2015, the Commission began its review of a revised Management Directive 4.7, "Budget Formulation." The agency intends to use the MD for the FY 2018 budget formulation process if approved by the Commission.

QUESTION 7. **The 2015-2016 Rulemaking Activity Plan lists 93 rulemakings with only nine ranked as low priority.**

a. **Please describe actions the Commission is taking to address the cumulative impact of regulation inherent in such a robust list where the vast majority of items are listed as medium and high priority.**

b. **Please also describe the process for assigning the priority ranking.**

ANSWER.

a. While the 2015-2016 Rulemaking Activity Plan lists 93 rulemakings, a number of those rulemakings update existing regulations by adopting revised industry codes and standards or renewing or amending approved designs. In the Staff Requirements Memorandum to SECY-11-0032, "Consideration of the Cumulative Effects of Regulation in the Rulemaking Process," dated 22October 11, 2011, the Commission approved the staff's process changes to address the cumulative effects of regulation in the rulemaking process. The process changes included interacting with stakeholders throughout the rulemaking process, beginning with the development of the regulatory basis. This "early and often" stakeholder interaction allows the staff to understand the cumulative effects of regulatory actions under development, and to appropriately structure requirements and set compliance dates. This process is designed to mitigate, when possible, the impact of compliance with new regulatory requirements, not to reduce or eliminate the need for new regulatory requirements.

At the request of the Commission, the staff recently provided recommendations on how to improve the agency controls on the rulemaking process. The Commission is currently considering these recommendations.

The Executive Director for Operations (EDO), oversees the rulemaking prioritization process, which is conducted by an agency-wide Rulemaking Coordinating Committee (RCC). In general, only high priority and medium-priority rulemakings are funded by the Commission in their annual budget request. Consequently, there are relatively few low priority rulemakings at the NRC. The RCC includes staff from each lead rulemaking program and support offices; the development of the list of prioritized rulemakings is a collaborative effort. Each year the RCC updates the list to add new rulemakings, reprioritize current rulemakings as appropriate, and remove rulemakings that have been completed or will no longer be pursued. Rulemaking activities are prioritized using a prescribed methodology and set of criteria that considers four factors: (1) Factor A - NRC Strategic Plan strategic goals (safety and security), (2) Factor B - NRC Strategic Plan cross-cutting strategies (regulatory effectiveness and openness), (3) Factor C - Governmental priority, and (4) Factor D - Public priority. Each rulemaking is scored based on its support for each of the four factors and within a separate range of values for each factor: (1) Factor A - 0 to 20, (2) Factor B - 0 to 10, (3) Factor C - 0 to 10, and (4) Factor D - 0 to 5. The prioritization score for a rulemaking activity is determined by summing the scores for Factors A, B, C, and D. Each rulemaking is then assigned a priority category based on its prioritization score: High - 31-45, Medium - 16-30, and Low - 0-15. The priorities recommended by staff are reviewed and approved by Office Directors and the EDO. The Commission reviews the prioritized list of rulemaking activities and associated proposed resources annually in the budget provided by the Chairman.

QUESTION 8. **Please provide a copy of the 2005 Rulemaking Activity Plan.**

ANSWER.

A copy of the 2005 Rulemaking Activity Plan and related documents are attached. Please note these documents are marked "Official Use Only" and should not be released to the public.

QUESTION 9. **The Office of Personnel Management approved early retirement incentives for NRC senior managers. When will NRC know how many employees elected to utilize it? As soon as the information is available, please provide the number of staff separating, the date on which they will be required to separate, and the NRC's staffing level on that date. Please indicate the amount of savings in salaries and benefits expected to result from the buyout authority.**

ANSWER.

The NRC offered early retirement authority and buyout incentives for supervisors and managers at the grade 15 level, and some senior project managers at the grade 15 level. Forty-nine employees were approved for the buyout and/or early out. Of those, 10 employees were supervisors or managers, and 10 employees were senior project managers. The remainder of the employees performed corporate functions.

All employees taking the buyout and/or early out have selected separation dates of January 9, 2016, or earlier. After factoring in the reductions from the early out/buy out, and considering other projected attrition, as well as gains to support critical mission and skill needs, it is anticipated that the NRC will use about 3,585 FTE by the end of FY 2016 (not including the Office of Inspector General). This number may fluctuate slightly throughout the year, depending on other attrition as well as gains to support critical mission and skill needs.

Based on the average per FTE salary and benefits rate budgeted in FY 2016, this represents an annual cost savings of approximately $8.1 million. The immediate savings in FY 2016 will be lower because employees who are separating will have been on board for part of the year, and the agency will need to pay separation costs from current year salary and benefits funds.

QUESTION 10. **Please provide the total number of hours the NRC spent inspecting operating reactors for FY 2005 to the present. Please provide a breakdown of those hours by the following categories: resident inspectors; routine ROP inspections; inspections triggered by increased oversight; and other.**

ANSWER.

The completion and performance of the Reactor Oversight Process (ROP) inspections are tracked by Calendar Year (CY) instead of by Fiscal Year (FY). Therefore, the response is presented by CY. Resident inspectors communicate any issues that are identified as well as the daily status of the plant to the regions and headquarters on a regular basis that is documented below as "Communications."

The following are the hours NRC spent inspecting operating reactors from CY 2005 to the present (as of November 19, 2015):

Resident Inspectors baseline inspections:	1,649,382
Routine ROP inspections:	2,962,099
Inspections triggered by increased oversight:	36,832
Other Resident Inspector hours:	
Plant Status:	523,787
Communications:	369,216
Total Hours as of Nov 19	5,541,316

QUESTION 11. **Has the Department of Energy notified the NRC of its decision NOT to support the NRC's review of the Yucca Mountain license application? If so, please provide all related documentation including but not limited to correspondence, email, and meeting notes.**

ANSWER.

The Department of Energy has not provided the NRC with such a notification. Nonetheless, the Department of Energy recently informed the agency that it would not prepare a supplement to the Yucca Mountain Impact Statement, as requested by the Commission. Instead, the Department of Energy agreed to provide an updated technical report to support the NRC's work in preparing the updated Environmental Impact Statement. A copy of that correspondence is attached to this question.

QUESTION 12. **Together, corporate support costs and research costs represent roughly half of the NRC's total budget. In the document "NRC FY 2016 High Level Impacts of Further Reductions," the NRC threatens to *"Defer licensing reviews of new applications for medical radio-isotope production facilities."* Why didn't the NRC consider pursuing corporate support cost reductions rather than targeting an important domestic medical priority?**

ANSWER.

The "NRC FY 2016 High Level Impacts of Further Reductions" was prepared for use in discussions with Congressional staff. It presented a preliminary list of potential reductions, and it was not an agency budgeting document. The Commission will make final budget decisions on the basis of fact-of-life information at the time those decisions are made. This document reflects the NRC is scaling back corporate support costs. This document assumes a baseline level of $990 million and 3,600 FTE. This is a reduction of $30 million and 78 FTE from the FY 2016 President's Budget. The baseline level includes a reduction of $10.3 million to corporate support, or 34% of the baseline reduction. Additional reductions in impact levels 1 through 3 would decrease the corporate support budget by another $10.1 million.

QUESTION 13. **Please provide a list of the power uprate applications that are pending and those anticipated to be received in 2016 with the projected amount of the power increases.**

ANSWER.

The NRC is currently reviewing two license amendment requests (or applications) from licensees requesting power uprates that would affect a total of four (4) units:

1. Duke Energy Corporation; Catawba, Unit 1 – Projected power uprate is 58 Megawatts-Thermal (MWt) or 1.7% of rated core power.

2. Tennessee Valley Authority (TVA); Browns Ferry Nuclear, Units 1, 2, and 3 – Projected power uprate for each unit is 494 MWt, or 14.3% of rated core power.

The NRC currently has no power uprate applications scheduled for review in 2016. In 2017, the NRC anticipates receipt of licensing applications that would impact 7 units, with a total increase in power output of 383 MWt.

QUESTION 14. **In the document "*Request for Information Regarding Recommendations 2.1, 2.3, and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Event*" dated July 21, 2015, the NRC estimates the time required to respond to the NRC's information request to be 617,705 hours. Using the NRC's hourly rate of $268 for approximation produces an industry cost estimate of $166 million. Please provide an estimate of the NRC resources necessary to review the information that will be provided. Please describe in detail the safety improvement the NRC expects to gain from this expenditure.**

ANSWER.

For FY 2016, NRC has budgeted 20 FTE and $800K in contract support for seismic reevaluations and 25 FTE and $2,200K in contract support for flooding hazard reevaluations. This budget supports the NRC's review of seismic probabilistic risk assessments for 20 sites and flooding integrated assessments for 10 sites to determine if additional plant safety enhancements are necessary. Budget information for FY 2017 will be available in early February with the FY 2017 Congressional Budget Justification. Resource needs for subsequent years will be evaluated through the NRC's normal budget process.

The NRC expects that the information requests and reviews will allow the NRC to focus on plants with the greatest potential for safety enhancements in light of the reevaluated external hazards. Not all licensees are required to complete these assessments; rather, they will be completed by a subset of licensees depending on how the reevaluated hazard compares to a facility's design-basis hazard. As part of completing these assessments, licensees will assess plant capacities to cope with higher seismic or flood hazards, and identify safety enhancements, if necessary. Examples of potential seismic enhancements include upgrades to equipment anchorages, removal of seismic interferences from plant equipment, and replacement of, or compensatory actions for low capacity electrical relays. Examples of potential flooding enhancements could include revised procedures to respond to predicted flooding events, compensatory actions to protect buildings and equipment (e.g., constructing flood barriers or regrading the site's topography to improve drainage), and improving water removal capabilities. Because the assessments have not yet been completed, specific safety enhancements have not yet been identified. Any potential nuclear plant enhancements beyond current NRC requirements would be evaluated using safety and risk insights, consistent with Title 10 of the *Code of Federal Regulations* Section 50.109, "Backfitting."

<u>QUESTION 15.</u> **Please describe actions the NRC is taking to bring the costs of its office space in line with its staffing level.**

<u>ANSWER.</u>

Since 2012, the NRC has been reducing its office space and corresponding costs at its headquarters location in Rockville, MD. To date, the NRC has released approximately 365,000 square feet of usable office space, reducing the headquarters office space from approximately 1,074,000 to 709,000 usable square feet. Going forward, the NRC has identified additional reductions of approximately 68,000 square feet of usable office space. These reductions begin in FY 2016 and go through FY 2020.

The NRC will continue to work with the Office of Management and Budget and the General Services Administration (GSA) to review requirements and identify opportunities for space and cost savings based on staffing levels as appropriate.

QUESTION 16. **The document, "Streamlining the FY 2016 Congressional Budget Justification," indicates that product line specific discussions will be eliminated and some important information may be excluded.**

a. Was the NRC's effort to "streamline" future Congressional Budget Justification documents, made in response to a congressional request?

b. Considering the increased Congressional scrutiny of the NRC's budget, why is the NRC decreasing the level of detail and transparency in its budget?

ANSWER.

a. No, the effort to streamline the Congressional Budget Justification (CBJ) document was a proactive initiative to improve budget process efficiency and produce a budget document that better focuses on the justifications for resources rather than past accomplishments that are described elsewhere.

b. It was not the NRC's intent to reduce transparency in its budget. The streamlining effort will reduce duplication of information and enhance the CBJ format to provide meaningful information to decision makers. For example, the product line introductory and resource paragraphs were considered to be duplicative and eliminated because this information is discussed in the business line introductory paragraphs and displayed in the business line resource tables. The product line-specific narrative was replaced with a "Major Activities" section. This section focuses on resource-significant activities and activities that the NRC believes to be of interest to external stakeholders.

The Honorable John Barrasso

QUESTION 1. **Has the NRC documented any cases where recovery solutions from a uranium in-situ recovery facility has migrated beyond the permit area and contaminated drinking water?**

ANSWER.

Information responsive to this question will take additional time to prepare. We will provide that additional information not later than January 12, 2016.

QUESTION 2. **What is the policy basis for selecting 10 years as the license duration for uranium recovery facilities?**

ANSWER.

Prior to 1996, it was Commission policy to issue five-year licenses to uranium recovery facilities. In the mid-1990s, the staff investigated ways to reduce the regulatory burden on uranium recovery licensees. In SECY-96-112, "Ten-Year License Terms for Uranium Recovery Licensees,"[1] the staff proposed lengthening the license term for uranium recovery facilities to 10 years to help reduce the regulatory burden on licensees. The Commission approved SECY-96-112 on July 2, 1996. The Commission subsequently published 62 Federal Register 5656, "10-Year License Terms for Material Licenses," on February 6, 1997, which increased the terms for most other materials licensees from a 5-year period to a 10-year period.

[1] SECY-96-112 can be found on the NRC's website at http://www.nrc.gov/reading-rm/doc-collections/commission/secys/1996/secy1996-112/1996-112scy.pdf.

<u>QUESTION 3.</u> **Please describe how the characteristics of a uranium recovery facility age or otherwise change within ten years such that a full review is required to extend the license.**

ANSWER.

Information responsive to this question will take additional time to prepare. We will provide that additional information not later than January 12, 2016.

<u>QUESTION 4.</u> **Is it feasible to implement a longer license duration [for uranium recovery facilities]?**

ANSWER.

NRC staff has not evaluated extending license terms greater than 10 years. NRC staff would need to analyze changing the licensing cycle, using an approach similar to the one provided in SECY-96-112. This might include such determinations as whether operational, environmental, and health and safety concerns will remain the same over the additional time.

<u>QUESTION 5.</u> **How might a longer license duration help the NRC manage its workload better in this area uranium recovery facilities?**

ANSWER.

Information responsive to this question will take additional time to prepare. We will provide that additional information not later than January 12, 2016.

QUESTION 6. **Considering the NRC's lengthy reviews both for initial licenses and license extensions, what steps will the NRC take to improve efficiency in these reviews [uranium recovery]?**

ANSWER.

The NRC staff continues to offer pre-application interactions to potential applicants to inform them of NRC regulations and the level of detail needed in an application. These pre-application meetings offer NRC staff an initial high-level look at the potential application for completeness and an opportunity to ask questions if information appears to be missing or incomplete. The NRC staff has seen improvements in application quality since starting these pre-application interactions, which can improve the efficiency of the NRC's review. NRC staff also encourages applicants to review previous NRC uranium recovery applications and NRC review comments, such as requests for additional information, to see the level of detail that is required in an application.

In addition, the NRC staff has implemented process improvements to address varied and complex issues in its reviews under the National Environmental Policy Act (NEPA) and Section 106 of the National Historic Preservation Act. The NRC staff uses its generic environmental impact statement (GEIS) for in-situ uranium recovery facilities to inform its environmental analyses. Site-specific supplemental environmental impact statements and environmental assessments incorporate applicable GEIS discussions by reference and by adopting relevant GEIS conclusions, allowing the staff to focus on site-specific issues in its analyses. In addition, the NRC has enhanced its coordination with other agencies and stakeholders, including BLM and Native American Tribes. For example, the Memorandum of Understanding with the U.S. Bureau of Land Management allows both agencies to coordinate their NEPA and Section 106 reviews, as appropriate, and reduce duplicative efforts.

The NRC has also partnered with the Advisory Council on Historic Preservation (ACHP) to establish an NRC/ACHP liaison to support the NRC's Section 106 activities.

QUESTION 7. **For the last 15 years, please provide a list of all uranium recovery applications for new facilities and license extensions with the following information for each:**

- **The duration of any National Historic Preservation Act Section 106 reviews;**
- **The duration of any hearings;**
- **The number of sequential rounds of Requests for Additional Information issued; and**
- **The total number of RAI's issued.**

ANSWER.

The requested information is provided in the attached table.

QUESTION 8. **Please provide a copy of the NRC's process for carrying out National Historic Preservation Act Section 106 reviews. Please explain any differences between the NRC's process and the regulations and guidance produced by the Advisory Council on Historic Preservation.**

ANSWER.

The NRC uses the process in the Advisory Council on Historic Preservation (ACHP) implementing regulations at 36 CFR Part 800, "Protection of Historic Properties" for compliance with the National Historic Preservation Act Section 106 Consultation requirements. In June

2014, the NRC staff published draft interim staff guidance, "Guidance for Conducting the Section 106 Process of the National Historic Preservation Act for Uranium Recovery Licensing Actions" for public review and comment (copy attached). This interim guidance follows the process established in the ACHP's regulations, while describing and clarifying each step in the process. The NRC staff is currently reviewing the comments it received, and will finalize the guidance in early 2016.

QUESTION 9. **Please describe the NRC's process for providing uranium recovery applicants with transparency regarding the status of application reviews for both new facilities and license extensions.**

ANSWER.

Information responsive to this question will take additional time to prepare. We will provide that additional information not later than January 12, 2016.

QUESTION 10. **The NRC's "Generic Environmental Impact Statement (GEIS) for In-Situ Leach Uranium Milling Facilities" was expected to improve the efficiency of environmental reviews for these facilities leading to completion of most licensing reviews within two years. Please indicate the cost of producing the GEIS and describe why those expected benefits have not materialized.**

ANSWER.

Information responsive to this question will take additional time to prepare. We will provide that additional information not later than January 12, 2016.

QUESTION 11. **In SECY 11-0159, "Status of the Decommissioning Program – 2011 Annual Report", the NRC staff stated that "...24 months is usually insufficient..." for remediation of groundwater during decommissioning. Does the NRC plan to revise or eliminate the 24-month deadline for decommissioning of in-situ uranium recovery facilities?**

ANSWER.

No. The 24-month deadline does not need to be eliminated since the pertinent regulation, Title 10 of the Code of Federal Regulations (CFR) § 40.42, allows for alternate schedules, as appropriate. Furthermore, although NRC-licensed uranium recovery facilities normally request decommissioning schedules for groundwater remediation that are greater than 24 months, these schedules may be shortened or lengthened as remediation progresses.

QUESTION 12. **On July 2, 2015, the NRC staff provided Committee staff with a table "Uranium Recovery Resource Summary" which listed enacted budgetary resources for the last six fiscal years. These amounts do not match the amounts listed in the annual fee recovery rules with regard to the "total budgeted resources" line in Table IX. Please explain any discrepancies.**

ANSWER.

The differences between the Uranium Recovery Resource Summary (Resource Summary) and the Fee Rule Table IX (Fee Rule) are primarily due to the following:

1. The Resource Summary was utilizing an FTE rate that was not full costed, while the Fee Rule follows the guidance of OMB Circular A-25: *User Charges*, which recommends a full costed FTE rate.

2. During the development of the Fee Rule, certain Uranium Recovery resources were allocated to the fee relief category called Generic Decommissioning. The NRC does not charge annual fees to materials program licensees that are undergoing decommissioning or site reclamation. Fee relief allocations of budgeted resources are not reported within the Resource Summary.

QUESTION 13. **On July 2, 2015 the NRC staff also provided Committee staff with a table "Part 171 Billing Costs for Uranium Recovery Licensees." The amounts listed in this table do not seem to match the numbers listed in the annual fee recovery rules with regard to the "Net 10 CFR Part 171 resources" line in Table IX. Please explain any discrepancies.**

ANSWER.

The differences between the Part 171 Billing Costs for Uranium Recovery Licensees and the Fee Rule Table IX for Net 10 CFR Part 171 resources for uranium recovery facilities are primarily due to the timing of the NRC's billing of the licensees. The information for the Part 171 Billing Costs for Uranium Recovery Licensees reported what was actually billed for each year. The information reported in Table IX for Net 10 CFR Part 171 is the total budgeted resources to be recovered through annual fees assessed to the Uranium Recovery Fee Class. The new fee rule becomes effective in the fourth quarter of each fiscal year. For those licensees subject to a fee of $100,000 or more, the NRC bills the actual current year fee after the new fee rule is in effective in the current fiscal year. However, for those licensees subject to fees of less than $100,000, the NRC bills them annually in the month in which their license was issued, not necessarily by the end of the current fiscal year. Because uranium recovery licensees have annual fees less than $100,000, they are billed their annual fees in the month in which their license was issued.

QUESTION 14. **The EPA's January 26, 2015, proposed rule "Health and Environmental Standards for Uranium and Thorium Mill Tailings" sets forth new standards for uranium in-situ recovery facilities. Did the NRC comment on this proposed rule during the inter-agency process?**

ANSWER.

Yes. EPA informed the NRC that it was undertaking its rulemaking in 2010 and provided the NRC with periodic updates regarding the status of the rulemaking during its development. During the preparation of EPA's draft proposed rule, the NRC responded to EPA's requests for technical information. After EPA completed its draft proposed rule, the NRC staff reviewed and provided written comments on the proposed rule during the interagency review process led by the Office of Management and Budget. In addition, the NRC communicated its views in a July 28, 2015, letter sent by the NRC's General Counsel to the EPA's General Counsel.

QUESTION 15. **Did the NRC request that the EPA initiate the proposed rule "Health and Environmental Standards for Uranium and Thorium Mill Tailings"?**

ANSWER.

No. The NRC initiated a rulemaking for groundwater protection at uranium recovery *in-situ* leach (ISL) facilities in 2006. Although the NRC essentially completed its draft proposed rule, it was not able to obtain EPA concurrence as required by section 84a (3) of the Atomic Energy Act of 1954 (AEA), as amended. At that time, the two agencies could not reach consensus regarding long term monitoring and the establishment of site background water quality.

In 2010, EPA decided that it would be more practical to undertake its own rulemaking to develop generally applicable standards for byproduct material as allowed by the AEA. As a result, the NRC decided to defer its own proposed rulemaking until EPA promulgated its rule because the NRC is required to apply the EPA standards after they are promulgated.

QUESTION 16. **Is the NRC already preparing to implement the EPA's proposed rule "Health and Environmental Standards for Uranium and Thorium Mill Tailings" in anticipation of it becoming final?**

ANSWER.

No. The NRC staff is closely monitoring EPA's proposed rulemaking. The NRC has raised jurisdictional concerns with EPA resulting from the NRC's review of EPA's published proposed rule and is working with EPA to resolve those concerns. Assuming the NRC's concerns are resolved, and once EPA has promulgated its rule, the NRC will begin work on conforming regulations pursuant to Section 84a (3) of the Atomic Energy Act (AEA) of 1954, as amended.

The Honorable Shelley Moore Capito

QUESTION 1. **The NRC gave a document to appropriators entitled: "NRC FY 2016 High Level Impacts of Further Reductions." These potential reductions would be in addition to the 1.3% reduction the Senate Appropriations Committee has already approved. The document contained a list of actions the NRC would take in response to a $30 million dollar decrease. The first item listed is:**

> ***"Delay operating reactor license amendment reviews with less safety or security significance, resulting in a decrease in licensing actions of 5%. This would impact licensees' operational needs such as outage planning and reliability improvements."***

Between 2012 and 2016, corporate support costs are projected to have increased $26 million in operating reactors alone to reach a level of $211 million. Agency-wide corporate support spending has reached $422 million, or 42% of the NRC's total budget authority. Why would the Commission choose to inflict economic strain on the industry rather than scaling back escalating corporate support costs?

ANSWER.

The "NRC FY 2016 High Level Impacts of Further Reductions" was prepared for use in discussions with Congressional staff. It presented a preliminary list of potential reductions, and it was not an agency budgeting document. The Commission will make final budget decisions on the basis of fact-of-life information at the time those decisions are made. This document reflects the NRC is scaling back corporate support costs. This document assumes a baseline level of $990 million and 3,600 FTE. This is a reduction of $30 million and 78 FTE from the FY 2016 President's Budget. The baseline level includes a reduction of $10.3 million to corporate support, or 34% of the baseline reduction. Additional reductions in impact levels 1 through 3 would decrease the corporate support budget by another $10.1 million.

QUESTION 2. **That same document, *"NRC FY 2016 High Level Impacts of Further Reductions,"* indicates the NRC would delay review of domestic licensing actions prior to suspending the review of a foreign reactor design for construction in a foreign country. How does the NRC justify giving foreign work a higher priority than domestic licensees' operational needs?**

ANSWER.

The NRC does not prioritize design certification application review over the safety and security of the operating fleet. Although the NRC is currently reviewing the APR1400 reactor design, which was submitted by a Korean company for application in the United States, the NRC has not prioritized this review over the needs of domestic licensees. To the contrary, the NRC has assigned all of the following new reactor activities higher priority: (1) overseeing the construction of new reactors at Vogtle and Summer, (2) inspecting vendors to ensure the safety of the supply chain for operating and new reactors in the United States, (3) reviewing active applications for combined licenses or early site permits for domestic plants, and (4) reviewing any design certification application for which there is a domestic applicant referencing the design.

QUESTION 3. **Under law, the federal government funds only 10% of the NRC's budget while licensees are required to fund the other 90%. The NRC**

collects these fees in two ways. The first is 10 CFR Part 170 under which the NRC assesses fees based on work for specific applicants or licensees such as new plant applicants or license renewals. The other category of fees, Part 171, recovers generic regulatory costs that are not otherwise recovered under Part 170. Here is a quote from the NRC's 2014 Fee Recovery Rule which illustrates this situation:

> ***"The annual fees for power reactors increase primarily as a result of: (1) Decreased 10 CFR Part 170 billings due to the decline in current year licensing actions and delays in major design certification applications and combined license applications (this decline in 10 CFR Part 170 billings means that 10 CFR Part 171 fees need to increase to make up the difference and ensure that the NRC collects approximately 90% of its budgetary authority);..."***

For FY 2016, the NRC over-budgeted by one-third the number of applications it expects to review in the Office of New Reactors. This could repeat the 2014 problem requiring operating reactors, and consequently their ratepayers, to pay for the NRC's failure to budget accurately.

The NRC could have taken the opportunity to address this discrepancy when it provided *"NRC FY 2016 High Level Impacts of Further Reductions"* to appropriators but chose not to. Instead, the NRC chose:

- ***"Terminate the program to supply potassium iodide (KI) tablets to States in support of nuclear power plant emergency preparedness programs"***
- ***"Reduce confirmatory research in support of the operation of power reactors beyond 60 years"***

Why did the NRC target national safety and energy security matters rather than simply suggesting a correction to address the New Reactors over-budgeting?

ANSWER.

The document titled "NRC FY 2016 High Level Impacts of Further Reductions" was prepared for use in discussions with Congressional staff. It presented a preliminary list of potential reductions, and it was not an agency budgeting document. Please be assured the Commission remains committed to ensuring that appropriate levels of resources are provided to carry out its mission, including all the activities cited in this question. The Commission will make final budget decisions on the basis of fact-of-life information at the time those decisions are made. This document does address the over budgeting of the Office of New Reactors before taking other reductions. The document describes the impacts of potential *further* reductions beyond the baseline level, but does not describe the impacts of the reductions in the baseline level which includes a reduction of $7.2 million to reflect the reduced workload in the Office of New Reactors as a result of the decrease in the number of applications expected for review. The reductions in the Agency baseline level total $30 million and 78 FTE from the FY 2016 President's budget, and these reductions would be taken before any of the additional reductions described in the document.

<u>QUESTION 4.</u> **Since 2012, five reactors have permanently closed and more closures are possible. In both the 2014 and 2015 Fee Recovery Rules, the NRC has accounted for the reactor closures and the resulting loss of those fees by billing the remaining operating reactors more to make up the difference. For example, the NRC stated in their 2015 Fee Recovery Rule:**

"The permanent shutdown of the Vermont Yankee reactor decreases the fleet of operating CFO reactors, which subsequently increases the annual fees for the rest of the fleet."

What actions is the NRC taking to ensure that, in the future, operating reactors will not be penalized for plant closures?

<u>ANSWER.</u>

The Omnibus Budget Reconciliation Act of 1990 requires the NRC to recover 90 percent of its budget from fees. If there are fewer reactors in the United States, the NRC must allocate the program infrastructure costs of the NRC's operating reactor activities across a smaller number of reactors, resulting in increased fees for individual licensees. The closure of a single reactor will not necessarily result in a commensurate reduction in resources within the Operating Reactor Business Line (BL). For example, when a plant first enters the decommissioning process, the NRC continues to have a significant regulatory role. In other words, if 10% of reactors cease operations, the Operating Reactor business line would not automatically be reduced by 10%.

At the same time, the NRC continually assesses its needs, and adjusts resources accordingly, to minimize any fee increases resulting from plant closures. The NRC is focused on rebaselining its activities and resources as part of the Project Aim 2020 activities.

QUESTION 5. **In the document, *"NRC FY2016 High Level Impacts of Further Reductions,"* the NRC warned that a potential $30 million budget reduction and 140 FTEs would require "reductions in force." In FY 2015 the NRC was authorized to have 3,778 FTEs. The NRC's written testimony states that the NRC has 3,628 FTEs, a decrease of 150 FTEs. Given the current staff level and the NRC's 5% attrition rate, please describe why it would be necessary for the NRC to instigate reductions in force.**

ANSWER.

The NRC has taken a proactive approach to position management and strategic workforce planning. The agency has specifically targeted critical skill and position gaps as well as surpluses, reassigning staff internally whenever possible to fill gaps and reduce surpluses, appropriately utilizing contractor resources, and thus limiting external hires to those necessary to fill critical needs that cannot be met internally. This approach has enabled the NRC to reduce on-board and projected FTE utilization while not impacting the NRC's mission. Additionally, OMB approved an early out and/or buyout to specific groups. A total of 49 employees were approved for the buyout and/or early out and will be leaving the agency by January 9, 2016.

Based on these strategies, the NRC expects to utilize approximately 3,585 FTE in FY 2016, excluding the Office of the Inspector General, and therefore does not anticipate a need to conduct a reduction-in-force to meet the Commission's target of 3,600 FTE assumed in the baseline level of the "NRC FY 2016 High Level Impacts of Further Reductions" document. However, FTE reductions assumed in the first impact level in the document would result in an FY 2016 FTE budget significantly below projected agency utilization. Excluding resources for the Office of the Inspector General, the NRC requested 3,678 FTE in the FY 2016 President's Budget. The first impact level in the document assumes a budget reduction of 140 FTE, resulting in an FTE ceiling of 3,538. At 3,585 FTE, total projected FY 2016 utilization is 47 FTE above the 3,538 FTE ceiling, with a cost of $7.8 million in salaries and benefits in the first impact level.

QUESTION 6. **According to its 5% attrition rate, the NRC loses approximately 180 FTEs per year. In response to project Aim 2020, the Commission set a staffing goal of 3,600 by the end of 2016. Considering the attrition rate and current staff of 3,628, please describe why the Commission did not set a more proactive target than a reduction of 28 FTEs.**

ANSWER.

The 3,600 FTE target is an interim step towards transitioning to an eventual target for 2020. The Commission deferred setting a 2020 FTE target until after the Project Aim 2020 effort to rebaseline agency work is completed. The Commission expects further changes in needed resources, including staffing levels, will result from the rebaselining effort that is in progress as part of Project Aim 2020. The agency has taken steps to closely manage human capital through early outs/buy outs and limited hiring for only critical skills or positions, and therefore, the NRC expects to utilize 3,585 FTE in FY 2016, excluding the Office of the Inspector General, and does not anticipate a need to conduct a reduction-in-force to meet the staffing goal of 3,600 FTE.

QUESTION 7. **Please provide a monetized estimate of the potential savings resulting (overall and by fiscal year) from the EY Recommendations, as discussed in SECY-15-0109, "Assessment of the Recommendations in the April 30, 2015 Ernst and Young Overhead Assessment."**

ANSWER.

Although the NRC found merit in all of the cost reduction recommendations, in the EY report, the agency is not prepared to quantify those savings at this point. Evaluations of the recommendations are underway as part of Project Aim 2020.

QUESTION 8. **In response to SECY 13-0132, the Commission decided against revising the regulatory framework to more fully incorporate defense-in-depth considerations. Please provide a list of all staff recommendations to the Commission since that decision that have cited defense-in-depth as a justification for regulatory changes.**

ANSWER.

Staff rulemaking recommendations to the Commission since May 19, 2014, that cited defense-in-depth as a justification for regulatory change are set forth in the following table:

Document	Staff Recommendation	Location of Regulatory Analysis	Commission Response
SECY-15-0065, Proposed Rule: Mitigation of Beyond-Design-Basis Events	Staff recommends that the Commission consider severe accident management guidelines (SAMGs) to be a substantial safety enhancement by providing additional protection for defense-in-depth to satisfy the requirements under 10 CFR 50.109(a)(3) and that the direct and indirect costs of implementation are justified in view of this increased protection.	ML15049A212	The Commission approved the staff's recommendation and directed the removal of the proposed requirements for Severe Accident Management Guidelines (ADAMS Accession No. ML15239A767).

Document	Staff Recommendation	Location of Regulatory Analysis	Commission Response
SECY-15-0085, Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities	Staff recommends proceeding with developing the proposed rulemaking for improved protection for BWR Mark I and Mark II containments to make generically applicable the containment protection measures imposed by Order EA-13-109, including the planned implementation for Phase 2 of that order that uses external water addition (i.e., SAWA/SAWM). [The staff recommended this path based primarily on defense-in-depth reasons.]	ML15022A214	The Commission disapproved the staff's plan to issue a *Federal Register* notice requesting public comments on the draft regulatory basis for the Containment Protection and Release Reduction rulemaking. The Commission approved Alternative 1, Order EA-13-109 implementation. (ADAMS Accession No. ML15231A471).

QUESTION 9. **On June 11, 2014, the NRC Inspection Manual Chapter [308 Attachment 3][2], Appendix M, *"Technical Basis for the Significance Determination Process SDP Using Qualitative Criteria,"* was revised to: "...provide a technical basis for using qualitative criteria in determining the safety significance of an inspection finding."**

a. Was this revision initiated by the Commission or by NRC staff?

b. Was the Commission involved in the development of this revision?

c. Were stakeholders given an opportunity to participate in the development of the revision?

d. Does the revision incorporate additional subjectivity into the SDP and the Reactor Oversight Process?

e. Should the Commission's direction in response to SECY 14-0087 apply to modifications of the Reactor Oversight Process

[2] The title of the referenced document is corrected by the brackets.

ANSWER.

a. The June 11, 2014, issuance of Inspection Manual Chapter (IMC) 0308, Attachment 3, Appendix M was the initial issuance of the document so it has not been revised. The staff developed it to officially document its technical basis for IMC 0609, Appendix M, "Significance Determination Process Using Qualitative Criteria." The staff uses IMC 0609, Appendix M to provide deterministic guidance for assessing the significance of inspection findings when quantitative methods cannot either adequately address the finding's complexity or provide a reasonable estimate of the significance due to modeling uncertainties within the timeliness goals. This document was initiated by the staff without consultation with the Commission.

b. The Commission was not involved in the development of the document.

c. External stakeholders (industry and the public) were given opportunities to participate in the development of the document through a series of public meetings.

d. IMC 0609, Appendix M simply provided a more qualitative approach to assess the safety significance of findings when more quantitative approaches would not support the objective of the SDP to produce a timely regulatory decision. IMC 0308, Attachment 3, Appendix M provided the technical basis for IMC 0609, Appendix M.

e. The Commission's direction in the SRM for SECY 14-0087 is specific to regulatory analyses and backfit analyses for new or changed requirements or NRC interpretation of requirements imposed on entities such as nuclear power plant licensees who are protected by backfitting and/or issue finality provisions of NRC's regulations. Therefore, the Commission direction in response to SECY 14-0087 does not apply to the SDP guidance and to the ROP. The appropriate weight given to qualitative factors in regulatory decision making ultimately lies with the Commission.

The Honorable Michael Crapo

QUESTION 1. **Advanced Reactor technologies hold tremendous promise for providing even safer, more reliable, more affordable forms of energy for the future. Industry is taking a close look at these reactor concepts as well. Ultimately, advanced designs would need to be certified by the NRC before they could be constructed and operated in the United States. In recent testimony before the House Science Committee, one expert explained that the industry needs a "more efficient regulatory framework for licensing of advanced reactors." It is absolutely critical that we have a regulatory framework that protects public safety while not stifling energy innovation.**

a. **On September 1, 2015, there was a joint NRC-DOE workshop on advanced non- light water designs. What steps have been taken to ensure an efficient licensing process?**
b. **As Gen IV reactors are researched and designed, how would the NRC proceed with the design of the regulatory framework? Do you think the NRC would consider a "technology-neutral regulatory framework" to be best? Do you have enough flexibility under the current NRC regulations for developing a new framework for advanced reactors?**

ANSWER.

a. The NRC could license a non-light water reactor today, provided the design meets our safety and security requirements, using General Design Criteria available in Appendix A to 10 CFR Part 50 and other specific regulatory requirements, as appropriate. Because the

NRC's reactor licensing regulations and guidance documents were developed based primarily on light-water technology, the review would not be as efficient as the review of a light-water design. To improve the efficiency and effectiveness of the staff's review of advanced reactor designs, and to provide guidance to potential vendors, we are developing regulatory guidance documents that are specific to advanced reactors. This work is being done in coordination with the Department of Energy and has been informed by the agency's collaboration with our international counterparts. During the White House Nuclear Energy Summit on November 6, 2015, Chairman Burns announced that the agencies are in the early stages of planning a second joint NRC-DOE workshop for the spring of 2016. In addition, the NRC plans to provide DOE, through the Gateway for Accelerated Innovation in Nuclear initiative, guidance that can used to assist prospective applicants in navigating the NRC's regulatory processes efficiently.

b. The NRC has worked in the past on a technology-neutral framework and some of the current NRC regulations are already applicable to a variety of technologies, including non-light water reactors. Depending on the proposed design, the NRC would determine the extent to which current regulations are applicable and where new regulations may be needed for Gen IV reactors.

<u>**QUESTION 2.**</u> **In October 2014, the NRC certified a reactor design after nearly ten years of review and a cost $68 million. Another reactor vendor asked the NRC to suspend its design certification reviews after spending over seven years and $82 million. In contrast, the NRC has estimated it can review a modular reactor design in 39 months. This seems optimistic compared to the results of the experienced vendors.**

a. Why has the NRC concluded that certifying a modular design will be more efficient?

b. What steps will the NRC take to ensure the certification process for modular reactor designs is more predictable than previous efforts?

c. There are several generic issues related to differences between SMRs and conventional reactors. What issues remain and what progress has been made in resolving those issues? Will they be resolved by late 2016 when NuScale intends to file its design certification application?

ANSWER.

a. The NRC has performed a lessons-learned assessment of previous review activities for new large light-water reactor applications and has implemented several improvements to its internal processes that are designed to promote efficiency in the review of the anticipated NuScale SMR design certification application. For example, the agency has engaged in more extensive pre-application interactions with prospective applicants to identify potential issues and mechanisms for resolving them prior to the submission of an application. The agency has used the information obtained in pre-application engagements with potential applicants to prepare design-specific review standards tailored toward an individual modular design to promote predictability and efficiency in the safety review of the modular reactor design certification application. To ensure that the applicant has submitted all the information required to support a timely review, the agency has also revised the guidance for determining when an application is complete enough to be accepted for an in-depth technical review. This process improvement is expected to result in fewer requests for additional information during the staff's review of an application and a more predictable review schedule. The agency has also modified other staff guidance documents to help perform more efficient technical reviews. The NRC staff expects to gain efficiencies in the review of future reactor applications as a result of these process improvements. That being said, these projected timeframes remain estimates.

b. In August 2014, the NRC staff completed an assessment of its readiness to review small modular reactor applications. This assessment discusses the activities the staff has completed to prepare for new reviews and better ensure a predictable review schedule, including preparing design-specific review standards tailored toward an individual modular design. The NRC has also emphasized to prospective applicants that maintaining an efficient, predictable review schedule requires that the applicant: 1) support productive pre-application interactions, 2) submit high quality and complete information, and 3) reply to NRC requests for additional information and other questions about the design in a timely manner.

c. The readiness assessment completed in August 2014, contains a listing and status of the generic issues related to the differences between SMRs and conventional large reactors. The agency is making progress in resolving these technical and policy issues. One example of the agency's progress is the Commission's direction for the staff to initiate a rulemaking to revise regulations and guidance for emergency preparedness for SMRs. Specific to the anticipated NuScale design, the NRC has had extensive pre-application interaction with NuScale to develop a design-specific review standard and the staff is working to develop regulatory positions on key technical and policy issues that are critical to an efficient review of the design. The staff is preparing responses to public comments on the design specific review standard, which will be finalized and issued publicly in a timely manner to support NuScale's application schedule.

<u>QUESTION 3.</u> **The Idaho National Lab does significant work related to advanced reactors. I understand there is considerable interest in developing a "step-wise" approach for NRC's review: a process that would allow companies to pursue NRC approval for development of a design in stages. Such a process would also provide opportunities for developers to seek investment as they proceed.**

a) Is NRC evaluating how to provide such a framework?

b) If so, please provide a description of the agency's progress and any conclusions reached to date.

c) How can the NRC utilize the expertise of the Idaho National Lab to prepare staff to evaluate advanced reactor technologies?

<u>ANSWER.</u>

a&b. Yes, the NRC recognizes that new technologies may require innovative approaches to licensing. The NRC's current regulatory framework provides many options for the licensing and certification of new designs. To this end, the NRC is exploring specific alternatives that would accommodate the needs of applicants with varying degrees of design maturity.

c. The NRC has used, and will continue to use, the extensive experience of DOE and the national labs when preparing to evaluate advanced reactor technologies. The NRC and DOE are working to adapt the general design criteria, which form a foundational piece of the NRC's regulatory framework, to make them more relevant for advanced non-light water designs. DOE, along with the Idaho (INL), Oak Ridge (ORNL), and Argonne (ANL) National Laboratories, are providing technical support for this initiative. This work is expected to clarify the NRC's safety and security requirements for new advanced reactor designs. The NRC also hosted an advanced and accident-tolerant fuels seminar for NRC and DOE staff in July 2015. Representatives from DOE, INL, and ORNL provided NRC and DOE staff with information on advanced and accident-tolerant fuels, and the NRC provided information on fuels licensing and fuel cycle.

<u>QUESTION 4.</u> **Advanced reactors encompass a broad range of technologies, but the NRC has limited resources for developing needed expertise. Resources are constrained because the NRC's funds must be recovered from existing licensees under NRC's mandate to recover 90 percent of their budget through fees.**

a. Do you have any recommendations on how to overcome this dilemma?

<u>ANSWER.</u>

a. Being prepared to evaluate potential applications for advanced reactor technologies presents some challenges for the NRC, but the NRC is prepared to receive and review any such applications under its existing framework. To this end, the NRC has been proactive within the framework of the largely fee-based approach to regulatory reviews. Within the constraints of our Congressional appropriation and fee collection requirements, the agency is working on advanced reactor activities with the Department of Energy, industry standard-setting organizations, and the Generation IV International Forum. The NRC will continue to leverage our existing knowledge base as we develop expertise while regulating emerging technologies. As noted, such infrastructure development work would need to be recovered through fees unless the requirements under Omnibus Budget Reconciliation act of 1990 were changed, or some other mechanism were identified to fund this work. One approach could be to establish a separate line-item of the fee basis for regulatory development for advanced reactor technology.

The Honorable Deb Fischer

QUESTION 1. **On August 19, 2015, the Commission voted to not require the installation of external containment vent filters for U.S. boiling water reactors with Mark I and II containments. The Cooper Nuclear Station 1 in Brownsville, Nebraska, has a boiling water reactor with Mark I containment. According to an industry report, "installation of external engineered vent filters would cost each of 29 affected boiling water reactors between $37 million and $55 million." Those are costs that, if unwarranted, would add unjustified costs to electricity customers. What level of scrutiny do costly proposed requirements like these undergo at the NRC?**

ANSWER.

In preparing its rulemakings, the NRC staff performs preliminary cost assessments and backfitting assessments using established methods and the best available data to estimate the future costs of a proposed regulatory action. A backfit analysis is a systematic and documented analysis that shows a substantial increase in the overall protection of the public health and safety or the common defense and security where the direct and indirect costs of implementation are justified in view of the increased protection. These estimates are available for public review at the time these draft regulatory bases are published for public comment. Information received on the preliminary cost assessments is used to prepare a draft regulatory analysis, which accompanies the proposed rule. If the Commission decides that rulemaking is the appropriate regulatory decision, then the staff updates the cost estimate using reliable data as it becomes available, explicitly addressing uncertainties and recognizing excluded costs, as well as conducting an independent review of selected cost estimates to ensure realism, completeness, and consistency. Additional cost data received during a proposed rule's public comment period are used to refine the cost estimates for the final regulatory bases. The comments from the public are reviewed and the appropriate changes are made to the cost-benefit analysis. All public comments are addressed by the NRC in the final regulatory documentation, and the final cost-benefit analysis is reviewed and approved by the Commission.

<u>**QUESTION 2.**</u> **In recent correspondence to the NRC, the Nuclear Energy Institute noted that:**

"The 2015-2016 Rulemaking Activity Plan shows prioritization results for 93 rulemakings. Of these, only nine rulemakings received a LOW priority. The remaining 84 rulemakings were ranked MEDIUM or HIGH with 56 being funded in FY2015. "

a. Did the Commission approve this rulemaking plan?

b. Please indicate the number of rulemakings that were initiated by the industry, the NRC staff, or other stakeholders.

<u>ANSWER.</u>

a. No. Since 2001, the staff has submitted the annual Rulemaking Activity Plan to the Commission for information only. Ultimately, the Commission approves the agency's prioritization of rulemakings through its role in approving the agency's budget.

b. In the 2015-2016 Rulemaking Activity Plan, 27 rulemakings were requested by the industry (including 5 petitions for rulemaking), 52 were initiated by the NRC (including 7 petitions for rulemaking), and 2 were requested by other stakeholders. In addition, there were 9 rulemakings where required to comply with Congressional or Executive mandate. Finally, 3 administrative rulemakings were reported that would make corrections or administrative updates to the NRC's regulations.

<u>QUESTION 3.</u> **Has the Commission reviewed its engagement in rulemaking initiation as a tool for mitigating the cumulative effects of regulation?**

<u>ANSWER.</u>

Although the Commission did not specifically review its engagement in rulemaking initiation as a tool for mitigating the cumulative effects of regulation, early Commission involvement would enhance the Commission's ability to evaluate the cumulative effect of its regulations and their impact on licensees. In response to Commission direction the NRC staff provided on October 19, 2015, recommendations for enhancing the Commission's role in initiating and approving the development of rules. The Commission is currently considering these recommendations.

<u>QUESTION 4.</u> **I understand the NRC staff presented a plan to the Commission with recommendations regarding Commission engagement in rulemaking.**

- **a. Is it true that the Commission essentially asked the NRC staff to evaluate whether the Commission should more closely scrutinize the initial stages of the staff's rulemaking efforts?**
- **b. How can the Commission be confident that it will get unbiased recommendations?**
- **c. Is the Commission seeking any stakeholder input in developing its recommendations?**
- **d. Please describe any other efforts made to ensure transparency**

in the development of the NRC staff's recommendations.

ANSWER.

a. In the Staff Requirements Memorandum (SRM) for COMSGB-15-0003, "Commission Involvement in Early Stages of Rulemaking", the Commission directed the staff to provide a proposed options for increasing the Commission's involvement early in the rulemaking process, with the objective of ensuring early Commission engagement before significant resources are expended. The staff provided its proposed plan in SECY-15-0129, which is currently under Commission review.

b. In the SRM the Commission sought to limit the potential for staff bias unintentionally affecting the staff's recommendation by directing the staff to consider certain specific options. Thus the Commission has confidence that the proposed plan and options are based on an analysis of a range of options, from a variety of perspectives, and that the resulting recommendations provided for Commission approval are balanced. In all events, the Commission will ultimately determine whether the proposed actions satisfies the Commission's objectives.

c. In the SRM for COMSGB-15-0003, the Commission did not direct staff to seek external stakeholder input when developing its recommendations. However, as stated in (a) above, the recommendations were developed by a diverse group of rulemaking staff. This group worked closely with the Executive Director for Operations, the Deputy Directors for Operations, and the Office Directors to develop its proposed plan for increasing the Commission's involvement in the rulemaking process, with the objective of ensuring early Commission engagement before significant resources are expended.

d. To ensure transparency in this process, COMSGB-15-0003 and SECY-15-0129 are publicly available.

<u>QUESTION 5.</u> **Does the Commission's direction in the March 4, 2015, Staff Requirements Memorandum[4] regarding the use of qualitative factors reinforce current practice?**

([4]http://pdaupws.nrc.gov/docs/ML1506/ML15063A568.pdf)

<u>ANSWER.</u>

Yes.

<u>QUESTION 6.</u> **The NRC's Committee to Review Generic Requirements (CRGR) has a stated mission to:**

"...recommend either approval or disapproval of the staff's proposed backfits and to guide and assist the NRC's program offices in implementing the Commission's backfit policy."

Please describe how the CRGR is working to ensure consistent implementation of the Commission's direction regarding the use of qualitative factors provided the March 4, 2015, Staff Requirements Memorandum for SECY-14-0087.

<u>ANSWER.</u>

As specified in the CRGR charter, the CRGR will ensure consistency with the Commission's backfit policy for those actions under its consideration, in accordance with the established CRGR review procedures. The Commission has recently approved the NRC staff's plans for updating guidance to improve its methods for regulatory analyses, including the treatment of uncertainties, to develop realistic estimates of the costs of implementing proposed requirements. This guidance will include the Commission's direction on qualitative factors contained in SRM-SECY-14-0087. It will address qualitative factors assessment methodology, cost estimating best practices, and the treatment of uncertainty in regulatory and backfit analyses. This guidance is scheduled to be released in July 2016 following Commission review. After the updated guidance is issued, it will be applied by the staff to any regulatory actions, as appropriate. A key CRGR function is to ensure that the agency documents reviewed by the committee comply with all applicable Commission guidance, which would include the Commission direction on qualitative factors in SRM-SECY-14-0087.

QUESTION 7. **Of the staff assigned to serve on the CRGR, please indicate how many have been actively involved in proposing new regulatory requirements in the last 5 years. Please describe what protections are taken to ensure that the CRGR's conclusions remain objective, independent and disciplined, rather than acquiescent to the views of those advocating for new requirements.**

ANSWER.

Four of the seven CRGR members are actively involved in proposing new regulatory requirements. The CRGR comprises the deputy office directors (SES managers) drawn from the Offices of Nuclear Regulatory Research (RES), Nuclear Reactor Regulation (NRR), New Reactors (NRO), Nuclear Material Safety and Safeguards (NMSS) and Nuclear Security and Incident Response (NSIR), the deputy regional administrator from one of the four NRC regional offices selected on a rotating basis (currently the Region I deputy regional administrator), and one member from the Office of the General Counsel (OGC). The CRGR reports to the Executive Director for Operations (EDO) and conducts its activities in accordance with Revision 8 of the CRGR charter, dated March 2011. By virtue of their normal line organization responsibilities, the CRGR members from NRR, NRO, NMSS, and NSIR are involved in proposing new regulatory requirements. Although the CRGR member from OGC would not be a decision maker for proposing new regulatory requirements, this member would review the legal aspects of such decisions. Similarly, the RES CRGR member, who chairs the CRGR, would not normally be directly involved in decisions to pursue new regulatory requirements. However, because RES prepares the technical bases to support many proposed rules, this member could be involved in the development, review, and approval of such technical bases. The CRGR member from the regional office would not normally be directly involved in making or approving proposals for new regulatory requirements as part of his or her normal job.

It is important to highlight the CRGR, by its current charter, reviews only selected staff guidance and regulatory requirements that could impose a generic backfit. More specifically, the CRGR ensures any generic backfits that are proposed for NRC-licensed power reactors, new reactors, and nuclear materials facilities, and that fall within its charter, are appropriately justified on the bases of the backfit provisions of the applicable NRC regulations, the NRC's regulatory analysis guidelines, and the Commission's backfit policy. The CRGR does not have a role in deciding whether or not the newly proposed requirement itself should be adopted. Those decisions are made through other agency processes.

The EDO and the Commission provide oversight of the CRGR and ensure that it remains objective and adheres to its Commission-approved charter, which specifies its scope, roles, and responsibilities. Since 1997, the CRGR has evaluated and reported its activities and accomplishments to the EDO and the Commission in an annual report that is publicly available. Each report summarizes the backfit reviews performed by the committee during the assessment period and provides the results of the committee's annual self-assessment, which includes input from the regulatory offices that use CRGR services. The most recent CRGR annual report is SECY-15-0107, "Annual Report of CRGR Review Activities," of August 20, 2015.

The CRGR is also subject to audit by the NRC Office of the Inspector General (OIG). The most recent OIG audit was completed in 2009. The OIG made two recommendations—one involving the backfit review process and one concerning whether the CRGR function is still needed—which the CRGR resolved. The OIG did not identify any issues associated with the objectivity of the CRGR's work or the results of its reviews.

QUESTION 8. **Please indicate whether the CRGR reviewed any of the following issues.**

- **Staff recommendations in SECY 12-0157 regarding filtered vents;**
- **Staff recommendation to require Severe Accident Management Guidelines as part of the Mitigating Strategies rulemaking;**
- **Staff recommendations regarding the revision of Part 61 for low-level waste disposal facilities;**
- **Staff recommendations regarding reevaluated flooding hazards per COMSECY 14-0037;**
- **New requirements for Part 70 licensees regarding dermal and ocular exposure; and**
- **Staff recommendations in SECY 14-0087 regarding the use of qualitative factors.**

For those issues reviewed by the CRGR, please provide the date of the review and indicate whether the CRGR conducted a formal or informal review.

ANSWER.

These rules and documents were not reviewed by the CRGR because the proposing offices did not request CRGR review of these rules and documents. This is consistent with the current CRGR charter, which was revised to eliminate CRGR mandatory review of proposed rules. The reasons for this change are set forth in SECY-07-0134, "Evaluation of the Overall Effectiveness of the Rulemaking Process Improvement Implementation Plan".

QUESTION 9. **Please describe the situations in which a CRGR review would not be beneficial to Commissioners in making decisions.**

ANSWER.

In the context of new regulatory requirements, of the regulatory products specified in the CRGR charter for potential CRGR review, those involving rulemaking are the only ones that the Commission would normally act on. As such, the Commission has the final say in determining whether new requirements are backfits and should be imposed. The other products specified in the charter, for example, NRC generic communications, are almost always handled at the staff level. The staff and the Commission most recently provided their positions on the value of CRGR reviews of rulemakings in SECY-07-0134, "Evaluation of the Overall Effectiveness of the Rulemaking Process Improvement Implementation Plan" and it's Staff Requirements Memorandum (SRM).

In SECY-07-0134, the staff stated: "For rulemaking tasks, CRGR review has limited value, because rulemaking packages are subject to an extensive concurrence process which typically allows all office to concur prior to CRGR review. The package includes a rigorous regulatory analysis that supports the rulemaking, and the legal perspective of backfit issues are adequately addressed by OGC with the support of the associated technical offices. This thorough vetting of the product significantly diminishes the opportunity for CRGR to add value. On the basis of this assessment, the staff recommended the Commission completely eliminate review by CRGR for all rulemaking tasks". In the SRM for SECY-07-0134, the Commission approved the staff's recommendation to remove the CRGR from the review of current and future rulemaking packages. In response to this Commission direction, the CRGR revised its charter to eliminate the requirement that CRGR review proposed rulemaking packages. However, the revised (current) charter allows an office director or the EDO to request CRGR review of a proposed rule.

Subsequently, in SECY-15-0129, "Commission Involvement in Early Stages of Rulemaking" (ADAMS Accession No. ML15267A759), the staff incorporated the Commission's direction that CRGR not review rulemakings. This SECY paper is currently under Commission review.

<u>QUESTION 10.</u> **Please describe the opportunities in the CRGR review process for stakeholder input.**

<u>ANSWER.</u>

Under existing NRC processes, the staff publishes most, if not all, of its proposals for public review and comment before submitting them to CRGR for its review. As part of their review, the CRGR members consider any public comments, paying particular attention to any backfit concerns raised by the commenters. Stakeholders are encouraged to provide comments and raise any backfit concerns during the public comment phase. If the CRGR finds that the staff did not fully address any public comments associated with a backfitting concern, it can direct the staff to take additional actions to resolve the concerns. Such directed actions could include an additional comment period with focused questions, a public meeting to discuss the backfitting concern with all interested stakeholders, or both. The CRGR has not elected to direct such actions in its recent activities.

The CRGR has also engaged directly with licensees and the nuclear industry. As an example, last year, after learning about a staff proposal concerning tornado missile protection that could apply to a number of operating reactors, the Nuclear Energy Institute (NEI) raised a backfit concern and asked the CRGR to review the proposed action. In response to NEI's requests, the CRGR Chairman, with the support of other CRGR members and staff, held a conference call with NEI and licensee representatives. At the end of the call, NEI stated that the CRGR had addressed its request for CRGR engagement. The CRGR documented this engagement in SECY-15-0107, "Annual Report of CRGR Review Activities," of August 20, 2015 (ADAMS Accession No. ML15167A513).

In their roles as senior executives and technical managers, the CRGR members will continue to take advantage of appropriate opportunities to engage with members of the public, licensees and industry representatives to discuss the role of the CRGR, the NRC's generic backfit management process, and industry questions and issues associated with backfitting. In addition, at a recent internal meeting, the committee decided to meet with all interested stakeholders to discuss the role of the CRGR and generic backfitting. The CRGR will hold this public meeting during calendar year 2016.

QUESTION 11. **The CRGR charter states: "...rulemakings will only be reviewed at the request of the proposing office or as directed by the EDO." Presumably, the "proposing office" would have little or no incentive to seek a CRGR review. If the EDO does not request a review, how can the CRGR fulfill its stated purpose to implement the Commission's backfit policy?**

ANSWER.

The CRGR recently identified a need to provide guidance to the regulatory offices about when CRGR review of a rulemaking should be requested. The NRC staff addressed this CRGR initiative in SECY-15-0129, Commission Involvement in Early Stages of Rulemaking, as follows:

> Since October 2007, subsequent to the Commission's approval of the removal of CRGR from the review of rulemaking packages, the NRC staff has not requested CRGR review of any proposed rule packages. This may have been caused in part by a lack of guidance or criteria available to assist the EDO or office directors in deciding when to request CRGR review or involvement in a particular proposed rulemaking.

The NRC staff is not aware of instances in which CRGR review would have resulted in different outcomes. However, given the recent focus on ensuring thorough backfitting and regulatory analysis reviews, and in light of the recent Commission direction on qualitative factors, it is appropriate to revisit guidance on CRGR review of rulemaking packages. Consequently, the CRGR has begun addressing this gap in its operating procedures and the NRC staff's implementing procedures by developing appropriate criteria and guidance. The criteria will provide clarity on when the NRC staff would request CRGR review of proposed rules.

In addition, in this same paper SECY-15-0129, the staff has proposed to include CRGR in the distribution of upcoming rulemakings to give CRGR the opportunity to request briefings early in the rulemaking process. This staff recommendation is under deliberation by the Commission. The development of appropriate criteria and guidance, and CRGR notification of rulemakings will enhance the CRGR's ability to implement the Commission's backfit policy.

QUESTION 12. **Please provide a list of actions taken by the NRC staff or recommendations made to the Commission that cite qualitative factors as a partial or crucial justification since March 4, 2015. Please indicate whether the items on that list were reviewed by the CRGR. If so, please indicate which office requested the review and whether a formal or informal review was conducted.**

ANSWER.

The actions listed below cite qualitative factors and were issued since March 4, 2015. As approved by the Commission, the CRGR charter does not require CRGR review of rulemakings. Additionally, the proposing offices did not request CRGR reviews of the actions and consequently, the CRGR did not review them.

In SRM-SECY-14-0087, the Commission directed the staff to adhere to four high-level principles in the development of its guidance:

- The staff should continue to strive to improve its methods for quantitative analyses, including the treatment of uncertainties.
- The staff should use the best information available to develop realistic estimates of the cost of implementing proposed requirements.
- To ensure that qualitative factors are used in a judicious and disciplined manner, the revised guidance should continue to encourage quantifying costs to the extent possible and use qualitative factors to inform decision making, in limited cases, when quantitative analyses are not possible or practical (i.e., due to lack of methodologies or data).

- To improve transparency and decision making, any revised guidance should outline how the staff will articulate its rationale for the selection of qualitative factors and describe with specificity how these factors were used in the analysis, including the use of sensitivity analyses.

This guidance is scheduled to be released in July 2016 following Commission review.

In preparing its rulemakings, the NRC staff performs preliminary high-level cost assessments and backfitting assessments using established methods and valid data to estimate the future costs of a proposed regulatory action. These estimates are available for public review at the time the draft regulatory bases are published for public comment. Additional cost data received during a proposed rule's public comment period are used to refine the cost estimates for the final regulatory bases. If the Commission decides that rulemaking is the appropriate regulatory decision, then the staff updates the cost estimate using reliable data as it becomes available, explicitly addressing uncertainties and recognizing excluded costs, and conducting an independent review of selected cost estimates to ensure that they are realistic, complete, and consistent.

Since March 4, 2015, the staff has cited qualitative factors in the following actions:

- Revisions to Transportation Safety Requirements and Harmonization with International Atomic Energy Agency Transportation Requirements; Final Rule; 10 CFR Part 71 (Regulation Identifier Number (RIN) 3150-AI11; NRC-2008-0198)
- Cyber Security Event Notifications: Final Rule; Part 73 of Title 10 of the *Code of Federal Regulations* (10 CFR) (RIN 3150-AJ27; NRC-2014-0036)

- Supplemental Proposed Rule: Enhanced Weapons, Firearms Background Checks, and Security Event Notifications (10 CFR Part 73; RIN-3150-AI49)
- Proposed Revisions to Policy Statement on Reporting Abnormal Occurrences Criteria
- Request for Approval of Staff Recommendation to Authorize Babcock and Wilcox Nuclear Operations Group-Lynchburg to Use Section 161A Preemption Authority
- Proposed Rule: Mitigation of Beyond-Design-Basis Events (RIN 3150-AJ49)
- Evaluation of the Containment Protection & Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities (10 CFR Part 50) (RIN-3150-AJ26)
- Rulemaking: Proposed Rule: Incorporation by Reference of Institute of Electrical and Electronics Engineers Standard 603-2009, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating"

Proposed Rule: Incorporation by Reference of American Society of Mechanical Engineers Codes and Code Cases (RIN 3150-AI97)

QUESTION 13. **Please provide a list of all post-Fukushima items either considered or acted upon, noting which ones received a CRGR review. For those items reviewed by the CRGR, please indicate which office requested the review, and whether it was a formal or informal review.**

ANSWER.

The CRGR conducted an informal review of a March 12, 2012, request for information issued pursuant to Title 10 of the *Code of Federal Regulations* Paragraph 50.54(f). This request for information addresses Near-Term Task Force Recommendations 2.1, 2.3, and 9.3, and includes activities associated with seismic and flooding walkdowns, seismic and flooding hazard reevaluations, and assessment of emergency preparedness staffing and communication capabilities. The review was requested by the Office of Nuclear Reactor Regulation. The CRGR also formally reviewed Generic Letter 2015-01, "Treatment of Natural Phenomena Hazards in Fuel Cycle Facilities," prior to issuance of the letter on June 22, 2015. This review was requested by the Office of Nuclear Material Safety and Safeguards.

The CRGR, which is primarily comprised of deputy office directors and deputy regional administrators, did not review any of the other post-Fukushima items. However, the NRC's Fukushima-related activities are being conducted under the oversight of the Japan Lesson-Learned Division (JLD) Steering Committee. The JLD Steering Committee is primarily comprised of office directors and regional administrators, many of whom have prior experience as members of the CRGR.

<u>QUESTION 14.</u> **In correspondence to the NRC dated September 15, 2015, Mr. John Butler of the Nuclear Energy Institute wrote: *"The original charter of the Committee to Review Generic Requirements (CRGR) stated its role as: "The CRGR will develop means for controlling the number and nature of the requirements placed by the NRC on reactor licensees. The objectives of these controls are to eliminate the unnecessary burdens placed on reactor licensees, reduce the exposure of workers to radiation in implementing some of these requirement, and conserve NRC resources while at the same time not reducing the levels of protection of public health and safety. Since its inception in the early 1980's the role of CRGR has been reduced from a critical "gatekeeper" to one that is focused solely on backfit implications of generic communications."* Are the CRGR's**

original objectives as cited by Mr. Butler still valid and beneficial? If not, please explain.

ANSWER.

The NRC agrees that such objectives are valid and beneficial, and it continues to strive to meet such objectives today. However, through changes to existing processes and the adoption of new processes since the original CRGR charter was issued 33 years ago, the NRC now achieves such objectives through the CRGR in combination with other regulatory processes.

QUESTION 15. **Please provide a copy of the CRGR's original charter.**

ANSWER.

The original 1982 CRGR Commission-approved charter is attached.

QUESTION 16. **What other resources or established processes does the NRC utilize to ensure adherence to the Commission's direction in its March 4, 2015, Staff Requirements Memorandum for SECY-14-0087? Does the NRC staff receive training regarding the backfit Rule requirements?**

ANSWER.

The NRC's concurrence process is the formal processes that ensure adherence to all Commission directions. The NRC staff has been informed of the Commission's direction in SECY-14-0087 through the wide distribution of the SRM within the NRC.

Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," requires the NRC to conduct backfit training. Specifically, Office Directors and Regional Administrators are directed to ensure that beginner, advanced, and refresher courses are available.

The NRC staff's qualification program includes training regarding backfitting and issue finality requirements. The NRC also has online backfitting and issue finality training available, and it periodically provides in-person backfit and issue finality training to the staff. In addition, the staff is currently updating the agency's cost-benefit guidance, and plans to issue the draft for comment in summer 2016. The NRC provides training on cost-benefit to staff during its semi-annual rulemaking training.

QUESTION 17. **Both the Government Accountability Office and the NRC's Office of the Inspector General have criticized the NRC's ability to develop realistic estimates of the costs of implementing proposed requirements. Please describe the actions being taken to implement the recommendations from both offices. Or, if the NRC is choosing to reject any of these recommendations, please describe the basis for having the confidence to do so.**

ANSWER.

The NRC staff generally agrees with the Government Accountability Office's (GAO's) recommendation that the NRC could improve its regulatory analyses including cost estimates, and has efforts underway to do so. These efforts have been discussed with GAO. However, the NRC does not believe that the standards used by GAO to assess our program, set forth in GAO-089-3SP, *GAO Cost Estimating and Assessment Guide: Best Practices for Developing and Managing Capital Program Costs* (March 2009), are the only appropriate standards. The standards in that GAO document are applicable to evaluating federal agency capital expenditures (acquisitions). By contrast, the regulatory analyses prepared by the NRC are used to evaluate proposed regulatory actions affecting commercial entities that are within NRC's regulatory jurisdiction. For these types of proposed regulatory actions, the NRC follows federal agency practice by using Office of Management and Budget (OMB) guidance on the attributes of a regulatory analysis (OMB Circular A-4, "Regulatory Analysis"). The NRC uses OMB Circular A-4 as an essential reference for the NRC guidance on preparing regulatory analyses for its rulemakings and other regulatory actions. Consistent with OMB Circular A-4 and in response to GAO report GAO-15-98 recommendations, the NRC staff is updating the content and structure of the NRC's cost-benefit guidance documents, primarily NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," and NUREG/BR-0184, "Regulatory Analysis Technical Handbook." As it pursues these efforts, the staff is considering the applicable best practices from the *GAO's Cost Estimating and Assessment Guide*, as discussed in the GAO report.

With respect to the OIG report OIG-15-A-15 recommendations, the NRC staff has provided its responses to the OIG and is currently working on completing the actions to (1) formalize its existing regulatory analysis training/qualification program, (2) implement established knowledge management techniques for the regulatory analysis program, (3) update and implement the cost-benefit guidance documents, and (4) develop and implement procedures to consistently document stakeholder input prior to the proposed rule stage. The NRC staff has fully responded to OIG's recommendations, as documented in OIG letter, "Status of Recommendations: Audit of NRC's Regulatory Analysis Process (OIG-15-A-15)," dated August 31, 2015.

The Honorable Edward Markey

QUESTION 1. Can you please clarify whether the NRC draft EIS Supplement took into account the nuclear industry's current approach to nuclear fuel management, which results in spent fuel with higher burn-up, than was assumed by DOE in the Final EIS submitted to the Commission as part of the Yucca Mountain license application in 2008? If so, what conclusions were reached? If not, why not?

QUESTION 2. Can you please clarify whether the NRC draft EIS Supplement took into account the nuclear industry's current approach to storing spent nuclear fuel at reactors, using large welded storage canisters that are not compatible with the transportation, aging and disposal (TAD) canisters assumed by DOE in the Final EIS submitted to the Commission as part of the Yucca Mountain license application in 2008? If so, what conclusions were reached? If not, why not?

QUESTION 3. Can you please clarify whether the NRC draft EIS Supplement took into account the transportation impacts of using storage canisters that are not compatible with the TAD canisters assumed by DOE in the Final EIS submitted to the Commission as part of the Yucca Mountain license application in 2008? If so, what conclusions were reached? If not, why not?

QUESTION 4. **Is the NRC draft EIS Supplement accurate if the DOE-proposed TAD canister is never made available, or will the NRC need to do another supplemental EIS? Please fully explain your response.**

QUESTION 5. **Does the NRC draft EIS Supplement assume, as the Yucca Mountain license application assumes, that titanium drip shields will be installed roughly 100 years in the future after the proposed repository is constructed and loaded, using robotic installation equipment that would not be designed, constructed or available before a license application reaches a final decision? If not, what alternative assumptions are used regarding the drip shield installation?**

ANSWER TO QUESTIONS 1-5.

The draft EIS supplement does not address nuclear fuel management (including the management of spent high-burnup fuel), transportation impacts (including the use of canisters not compatible with the TAD canisters referenced in DOE's 2008 application), or issues associated with the proposed titanium drip shields. Following the decision of the U.S. Court of Appeals for the District of Columbia Circuit in *Aiken County*, 725 F.3d 255 (D.C. Cir. 2013) directing the NRC to resume the licensing process, the Commission directed the NRC staff to develop and issue a limited scope EIS supplement with the monies remaining in the Nuclear Waste Fund. The staff had previously reported to the Commission that it had determined that one aspect of its environmental review—the post-closure impacts of a repository on groundwater—required supplemental analysis. The scope of the draft EIS supplement is therefore limited to this discrete issue.

Several parties have challenged the application, as well as the scope of the NRC staff's analysis, by raising issues for litigation, called contentions, in the adjudication before the NRC's Atomic Safety and Licensing Board. Contentions pertaining to the issues mentioned above, including storage and transportation risks, the use of TAD and non-TAD canisters, and the proposed drip shields, have been admitted for litigation. At this time, the adjudication is suspended. Parties to the adjudication would have the opportunity to continue to pursue these contentions before the Atomic Safety and Licensing Board, or raise new issues in the form of new or amended contentions, should the adjudication resume.

<u>QUESTION 6.</u> **NRC Certification of Transportation Packages for Shipping Spent Nuclear Fuel to a Consolidated Interim Storage Facility or to a Geologic Repository.**

The use of transportation packages (shipping casks) certified by the NRC would be required for shipments to a consolidated interim storage facility or to the proposed repository at Yucca Mountain. The National Academy of Sciences (NAS) 2006 report, <u>Going the Distance? The Safe Transport of Spent Nuclear Fuel and High-Level Radioactive Waste in the United States</u>, strongly endorsed the use of full-scale testing to determine how packages will perform under both regulatory and credible extra-regulatory conditions. The Blue Ribbon Commission on America's Nuclear Future 2012 Report to the Secretary of Energy in turn endorsed this NAS recommendation. However, full-scale testing of transportation packages is not currently required by NRC for certification under 10 CFR Part 71. Please provide the following information:

1. A list of all approved transportation packages for spent nuclear fuel and high-level nuclear waste, currently certified by the Commission under 10 CFR Part 71;

2. A physical description of each approved transportation package design, including the intended transportation mode, the design waste volume, and associated impact limiter;

3. A list of all physical tests performed to determine the safety of each approved transportation package, including the associated impact limiter;

4. The criteria used to determine the safety of each approved transportation package for certification;

5. A description of the methods or analyses used in tests to measure transportation package performance for each criteria; and

6. **The results of all tests on each approved transportation package and associated impact limiter.**

ANSWER.

1. The following packages have been approved, or are under review, by the NRC for transport of spent nuclear fuel or high-level waste:
   - NAC-LWT (Docket No. 71-9225)
   - GA-4 (Docket No. 71-9226)
   - 2000 (Docket No. 71-9228)
   - NAC-STC (Docket No. 71-9235)
   - TN-FSV (Docket No. 71-9253)

- NUHOMS® MP187 Multi-Purpose Cask (Docket No. 71-9255)
- HI-STAR 100 System (Docket No. 71-9261)
- UMS Universal Transport Cask Package (Docket No. 71-9270)
- FuelSolutions™ TS125 Transportation Package (Docket No. 71-9276)
- TN-68 Transport Package (Docket No. 71-9293)
- NUHOMS®-MP197, NUHOMS®-MP197HB (Docket No. 71-9302)
- TN-40 (Docket No. 71-9313)
- HI-STAR 180 (Docket No. 71-9325)
- HI-STAR 60 (Docket No. 71-9336)
- BEA Research Reactor (BRR) Package (Docket No. 71-9341)
- TN-LC (Docket No. 71-9358
- HI-STAR 180D (Docket No. 71-9367)
- M-140 (Docket No. 71-9793) (Naval Reactors)
- M-290 (Docket No. 71-9796) (Naval Reactors)

2. See Attachment 1 to this question. Note that the design waste volume for spent fuel contents is generally expressed as authorized contents (i.e., number of fuel assemblies) where possible. Also note that the information provided by Naval Reactors to the NRC for review of its two packages is classified as "Confidential – Restricted Data"; therefore, the NRC is not including information on these two packages in its response.

3. Information responsive to question 3 will take additional time to prepare. We will provide that additional information not later than January 12, 2016.

4. The regulatory requirements applicable to spent fuel and high level waste transportation packages, which are contained in Title 10 of the *Code of Federal Regulations*, Part 71, include three main safety criteria for package performance: maintaining package radiation dose rates; maintaining release of radioactive material below the maximum allowable limits; and ensuring the contents remain subcritical. NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," summarizes the requirements for package approval and describes the procedures used by NRC staff to determine if these requirements have been satisfied. Attachment 2 contains details regarding these three safety criteria.

5. Information responsive to question 5 will take additional time to prepare. We will provide that additional information not later than January 12, 2016.

6. Information responsive to question 6 will take additional time to prepare. We will provide that additional information not later than January 12, 2016.

The Honorable Bernie Sanders

QUESTION 1. **The decommissioning fund was set up for one purpose – to clean up the nuclear site. In Vermont, we have seen the Nuclear Regulatory Commission (NRC) approve the withdrawal of decommissioning funds for expenses unrelated to decommissioning Vermont Yankee. The NRC's own regulations make clear that any other expenses the plant might be facing, including the costs of managing spent fuel, cannot come out of this fund. However, the NRC has granted inappropriate exemptions, with no due process or recourse to appeal a decision by NRC staff or Commissioners. This is a flawed process we hope to remedy not only for Vermont, but for other states that have as well. Why has the NRC failed to follow its own regulations? What grounds does the NRC have to continue to waive its own regulations?**

ANSWER.

The NRC has a decommissioning funding oversight program in place to provide reasonable assurance that sufficient funds will be available for the radiological decommissioning of all U.S. commercial nuclear reactors. Decommissioning trust funds are designed to protect the funds from withdrawals for expenditures other than those specifically authorized by NRC regulations. The intent of the trust funds is to cover the costs associated with the radiological decommissioning of the reactor facility, resulting in the termination of the NRC-issued license. As permitted under NRC regulations, some licensees choose to place funds in their decommissioning trusts to pay for costs associated with spent fuel management and site restoration. In some cases, including the case of the Vermont Yankee Nuclear Power Station, licensees have sought regulatory exemptions to use decommissioning trust funds for spent fuel management expenditures. In the case of Vermont Yankee, the use of decommissioning trust funds was the subject of a recent decision by an NRC Atomic Safety and Licensing Board, which the state of Vermont has asked the Commission to review in its adjudicatory capacity, and also is the subject of a lawsuit in the U.S. Court of Appeals for the D.C. Circuit.

As a general matter, the NRC regulations at 10 CFR 50.12, "Specific exemptions," allow the NRC to grant exemptions from the requirements of its regulations in 10 CFR Part 50, which includes its nuclear reactor decommissioning regulations. The NRC may grant such exemptions if they are authorized by law, will not present an undue risk to the public health and safety, are consistent with the common defense and security, and involve special circumstances such as that the application of the regulation is not necessary to achieve the underlying purpose of the regulation or that compliance with the regulation will result in undue hardship or other costs that are significantly in excess of those contemplated when the regulation was adopted. Thus, when considering an exemption from the NRC's decommissioning regulations to allow the use of decommissioning trust funds for purposes other than radiological decommissioning (e.g., spent fuel management), the NRC must determine that sufficient funds are (or will be) available for the radiological decommissioning activities required by NRC regulations. If there is reasonable assurance that additional funds are available beyond what is necessary to support radiological decommissioning, then the Commission may grant an exemption.

The NRC is committed to ensuring that the radiological decommissioning of any site is completed within the time period and in a manner consistent with the NRC's regulations. Compliance with decommissioning and funding assurance regulations for reactors that have permanently ceased operations is verified through a broad monitoring program that includes an onsite inspection program and the review of annual licensee-prepared decommissioning funding status reports. If, through this monitoring, the NRC staff determines that there is no longer reasonable assurance that there is sufficient funding to complete radiological decommissioning, any previously granted exemption may be revoked.

<u>QUESTION 2.</u> **I am concerned about the lack of input Vermont communities have had in the Vermont Yankee decommissioning process, particularly compared to the level of input the Nuclear Energy Institute has enjoyed.**

a. How does the NRC plan to actively engage members of the communities that host decommissioning nuclear plants in the current decommissioning policymaking process? If no such plans exist, please provide a timeline and steps the agency intends to take to increase community engagement in the decommissioning process.

b. Once the NRC has fully engaged with all stakeholders in the decommissioning process, how will the NRC integrate the advice and opinions of the stakeholders? What recourse do stakeholders have if they feel their feedback has not been reflected in the NRC's decommissioning policy making process or in the active decommissioning of a nuclear plant in their community?

<u>ANSWER.</u>

a. The NRC recognizes that states and local communities have a strong interest in the decommissioning of nuclear power plants within their boundaries. NRC regulations provide interested parties the opportunity to comment on the licensee's Post-Shutdown

Decommissioning Activities Report (PSDAR), which includes planning, schedule, cost, and environmental impact information; and on the License Termination Plan (LTP). In addition, the NRC conducts public meetings in the vicinity of the facility following the licensee's submission of its PSDAR and also following the licensee's submission of its LTP. The NRC also provides interested parties an opportunity to request an adjudicatory hearing regarding the LTP.

Throughout the decommissioning process, the NRC staff continues to encourage licensees to involve members of the local community. While licensees are not required to create a community advisory board, the NRC recommends the creation of a site-specific community advisory board and has provided recommendations for methods of soliciting public advice. The NRC also provides comments and suggestions for effective public involvement in the decommissioning process. In addition, the NRC maintains an active State Liaison Program, which provides states with opportunities for open communication with the NRC to make comments, ask questions, and express concerns at any time. Further, the NRC staff conducts or participates in a number of meetings where members of the public are invited to participate and provide comments, such as the NRC Annual Assessment Meetings, as well as Government-to-Government Meetings.

Whenever a licensee requests an amendment to a decommissioning facility's license, there is an opportunity for public comment and the opportunity to request a hearing. Furthermore, before granting any license amendment request involving a no-significant-hazards consideration determination, the NRC project manager must make a good-faith attempt to consult with the State.

The NRC is also in the early stages of developing a power reactor decommissioning rulemaking. The NRC will be providing a number of opportunities for public input on this proposed rulemaking. For example, the Advance Notice of Proposed Rulemaking (ANPR) was issued in the *Federal Register* on November 19, 2015, for a 45-day public comment period. The NRC has received several requests for extension of the comment period. ,. The ANPR requested specific comment on a variety of topics, including: the current regulatory approach to decommissioning, decommissioning trust fund, emergency preparedness, certified fuel handler training, staffing, aging management, security and cybersecurity, fitness for duty, onsite and offsite insurance, backfit, and regulatory analysis. Further, the staff conducted a public meeting on December 9, 2015, to discuss and answer questions from the public regarding the content of the ANPR. Approximately 105 people attended the meeting, either in person, by teleconference or webinar. The ANPR's purpose is to engage stakeholders to inform the staff's development of options for rulemaking, which will be incorporated into the regulatory basis for the rule. Additional opportunities for public participation will be provided during the development of the regulatory basis for the power reactor decommissioning rulemaking. The NRC will also publish a proposed rule to obtain public comments, and it will resolve public comments received prior to publication of the final rule.

b. The NRC will consider the advice and opinions of stakeholders regarding reactors currently undergoing decommissioning as they relate to applicable regulations and guidance. The NRC will also address comments that may indicate that the common defense and security or the health and safety of the public may not be adequately protected. With regard to recourse that stakeholders have if they feel their feedback is not addressed in the NRC's current decommissioning process, members of the public can participate in the ongoing rulemaking activity by attending public meetings and providing formal written comment on the ANPR, draft regulatory basis, and proposed rule. Members of the public may submit petitions for rulemaking in accordance with section 2.802 of Title 10 of the *Code of Federal Regulations* (10 CFR). Members of the public can submit 10 CFR 2.206 petitions to request that the NRC modify, suspend, or revoke a license, or take any other action as may be proper. Finally, members of the public with specific safety-related concerns regarding a plant and its regulated activities associated with the decommissioning process can bring these concerns (allegations) directly to the NRC.

The Honorable Dan Sullivan

QUESTION 1. Rulemaking is usually a regulator's primary tool to impose requirements on its regulated community. However, regulators also use guidance documents to clarify interpretation of their rules. For the NRC:

"The Regulatory Guide series provides guidance to licensees and applicants on implementing specific parts of the NRC's regulations, techniques used by the NRC staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits or licenses."

According to the NRC website, there appear to be 221 such guides for power reactors, 75 for fuel facilities, and 84 for materials.

a. How does the Commission decide when to revise a rule to achieve a desired outcome or to revise the relevant regulatory guidance for a rule?
b. Is the Commission involved in the development or revision of regulatory guides?
c. Are stakeholders consistently involved in the development or revision of regulatory guides?

ANSWER.

a. Rulemaking is initiated either through Commission approval of the agency's budget or through a Staff Requirements Memorandum Issued by the Commission. Currently, the Commission is deliberating on SECY-15-0129, "Commission Involvement in Early Stages of Rulemaking," dated October 19, 2015.

   Once resources are budgeted to the rulemaking activity, the staff begins by developing a regulatory basis. The regulatory basis document provides the problem statement and also describes potential alternatives to regulations (including guidance development and/or modifications to existing guidance). The NRC interacts with the public in the development of the regulatory basis, either through public meetings, public comment opportunities, or a combination of both. The conclusion of the final regulatory basis is whether rulemaking is warranted.

   If rulemaking is warranted, then the staff proposes the rule and prepares a draft regulatory analysis. That draft regulatory analysis evaluates alternatives to the proposed action, including guidance updates, if applicable. The Commission reviews this draft regulatory analysis as part of its deliberation on the proposed regulation. At the final rule stage, the regulatory analysis is also finalized (including any necessary updates to the evaluation of alternatives). The Commission reviews the final regulatory analysis when it deliberates on the final regulation.

b. The Commission provides oversight of the agency's regulatory program, including the development, revision, and issuance of regulatory guides. The Commission directed, as part of controlling the cumulative effects of regulation, that the staff develop and issue regulatory guidance concurrent with proposed and final regulations.

For reviewing and updating regulatory guides, the Commission is not normally involved unless there is significant stakeholder interest or the revised regulatory guide involves a change in regulatory policy (*for example,* backfitting). The NRC's Advisory Committee on Reactor Safeguards (ACRS) reviews revisions to existing guides as well as regulatory guides associated with rulemakings and provides advice to the Commission. In some cases, the NRC staff or external stakeholders may independently bring issues associated with regulatory guides to the Commission's attention.

c. Yes, stakeholders are consistently involved in the development and revision of regulatory guides. Draft regulatory guides associated with proposed regulations are noticed in the *Federal Register* for public comment. The NRC reviews the comments received on draft regulatory guides and provides formal responses to them. Similarly, draft updates to existing guides are also noticed in the *Federal Register* for public comment, and the comments received are reviewed and dispositioned by the NRC staff as part of the regulatory guide development and revision process.

<u>QUESTION 2.</u> **The NRC's Standard Review Plan "*...provides guidance to US Nuclear Regulatory Commission staff in performing safety reviews...*"**

a. Is the Commission involved in the development or revision of the Standard Review Plan?

b. Are stakeholders consistently involved in the development or revision of the Standard Review Plan?

<u>ANSWER.</u>

a. While the Commission does not have a direct role in the development or revision of SRPs, it does sets policy regarding guidance documents such as SRPs. For example, in 2005, the Commission directed the staff to revise applicable sections of the NUREG-0800, other guidance documents and office procedures to ensure up-to-date guidance would be available for the next generations of staff that would be responsible for reviewing and licensing activities.

b. Yes, stakeholders are consistently involved in the development or and revision of the Standard Review Plan. A draft of each revised Standard Review Plan is made for public comment through the publication in the *Federal Register.* Additionally, the NRC staff has, on occasion, held public meetings for stakeholders to ask question and provide comments orally about the draft Standard Review Plan section.

The staff develops a comment response document that presents the public comments received for a draft Standard Review Plan section and provides the staff's responses to those comments. The *Federal Register* notice of the NRC's issuance of the final version of the Standard Review Plan includes information on the availability of the comment response document.

QUESTION 3. **In addition to Regulatory Guides, there is also Interim Staff Guidance (ISG):** ***"Documents issued to clarify or to address issues not discussed in a Standard Review Plan (SRP)."***

a. How many ISG documents does the NRC have?

b. How many were initiated by the NRC staff?

c. Some of the ISG documents listed on the NRC website date back to 2000, 2002, and 2005. Why have "interim" documents been allowed to linger for so long rather than being incorporated into one of the NRC's more established regulatory tools?

d. What is the NRC's process for eliminating ISG's that don't warrant formal inclusion in the NRC's more established regulatory tools?

e. Are stakeholders or the Commission involved in the development of these documents?

ANSWER.

a. The NRC has approximately 100 ISGs in use today within the NRC. They can be found on the NRC's public website and are organized by technical/subject matter: http://www.nrc.gov/reading-rm/doc-collections/isg/.

b. While most ISGs are initiated by NRC staff, both internal and external stakeholders can recommend their development of ISGs.

c. ISGs are incorporated into regulatory guides (RG) and standard review plans (SRP) as these documents are updated. The ISGs might may be applicable to more than one specific such document, and therefore cannot be withdrawn until the ISG guidance is incorporated into all of the applicable documents.

d. ISGs are used to clarify established regulatory tools (i.e., NUREGs, regulatory guides, other regulations) and thus will be eliminated once they are incorporated into the appropriate document(s). Changes that do not warrant inclusion in the more established regulatory tools are handled by other means.

e. The staff follows a public participation process in finalizing and issuing ISGs. The staff evaluates the issue, develops a proposed ISG, issues the ISG for public comment, evaluates any comments received and, as appropriate, issues a final ISG. The Commission does not have a direct role in the ISG process, in part because ISGs are not supposed to contain do not contain new or changed guidance constituting backfitting or a violation of issue finality requirements in 10 CFR Part 52. policy direction.

QUESTION 4. **In additional to the above-mentioned documents, the NRC has several additional regulatory tools including:**

- **Generic Letters;**
- **Information Notices;**
- **Regulatory Issue Summaries;**
- **Branch Technical Positions;**
- **Request for Information issued under 10 CFR 50.54(f); and**
- **Task Interface Agreements.**

For each tool, please describe the following:

- **The extent to which the Commission is involved in the issuance or revision of such documents;**
- **The extent to which development of such documents is initiated by the NRC staff;**
- **Whether stakeholders are afforded opportunities to comment during development of the documents of any subsequent revisions; and**
- **Whether the Committee to Review Generic Requirements reviews the documents to ensure compliance with the backfit rule.**

ANSWER.

The level of internal and external stakeholder review varies for each regulatory tool listed above. The documents referenced below establish policies and procedures for Commission involvement, who may initiate the action, when and how stakeholders should be engaged, and CRGR review, as applicable, for each regulatory tool.

- Management Directive (MD) 8.18, "Generic Communications Program," establishes the process to develop and issue NRC generic communications (i.e., Generic Letters, Regulatory Issue Summaries, and Information Notices). The Commission reviews information papers from the Executive Director for Operations (EDO) informing the Commission of the staff's intent to issue a generic communication and retains the option to take action on the proposed generic communication. Development of a generic communication can be initiated by staff. The NRC generally makes new draft generic communications, as well as updates and revisions to generic communications, available for public comment prior to issuance. The CRGR reviews new or revised generic communications in accordance with the responsibilities outlined in the CRGR Charter.
- Branch Technical Positions are provided as appendices to the Standard Review Plan, and are controlled within the scope of the Standard Review Plan process. [See question 2 response].
- MD 8.4, "NRC Program for Management of Plant-Specific Backfitting of Nuclear Power Plants," establishes the process for NRC staff implementation of 10 CFR 50.54(f). The Commission is not directly involved in the issuance of requests for information under 10 CFR 50.54(f); however, 10 CFR 50.54(f) provides that the NRC staff must prepare the reasons for the information request to ensure that the burden imposed on licensees is justified in view of the potential safety (or security) significance of the issue to be addressed unless the information is being sought to verify licensee compliance with the current licensing basis. The EDO must approve the staff's justification before the information request may be issued. The NRC staff may engage with stakeholders during the development of 50.54(f) requests for information. Requests for information under 50.54(f) are not reviewed by the CRGR.

- NRR Office Instruction COM-106 (ADAMS Accession No. ML15219A174) describes the process to ensure that NRR responses to questions or concerns raised within the NRC (Technical Assistant Agreement requests) regarding nuclear reactor safety and the related regulatory and oversight programs are promptly and appropriately communicated to both internal and external stakeholders. The Commission is not directly involved in the resolution of Technical Interface Agreements (TIA). TIAs are initiated by NRC staff. Because TIAs relate to a specific licensee or group of licensees, licensees whose facilities are the subject of a TIA are informed of the TIA. The licensee or licensees are afforded the opportunity to voluntarily provide any information they feel would assist in the TIA review. If it is determined that the staff's response to a TIA can be viewed as a new or changed staff position, then the TIA process is not appropriate. The regulatory function of backfitting and its associated activities are outside the scope of the TIA process. Consequently, TIA responses are not reviewed by the CRGR.

<u>**QUESTION 5.**</u> **The NRC's Reliability Principle of Good Regulation states:**
"Once established, regulation should be perceived to be reliable and not unjustifiably in a state of transition. Regulatory actions should always be fully consistent with written regulations and should be promptly, fairly, and decisively administered so as to lend stability to the nuclear operational and planning processes."

With many of the documents listed above, their development or revision creates the potential to create de facto regulatory requirements by redefining what constitutes compliance. How does the NRC ensure regulatory discipline in these processes, that any

changes are truly warranted and consistent with the NRC's Reliability Principle of Good Regulation?

ANSWER.

The NRC pursues regulatory reliability and the other principles of good regulation through a variety of actions. The NRC's approach includes developing regulatory guidance simultaneously with the development of proposed new or changed regulations. In addition, the NRC addresses backfitting and issue finality, when applicable, and prepares a regulatory analysis to support the new or changed regulation. The NRC often provides opportunities for public involvement during its development of the draft proposed regulation and guidance before they are published as the official NRC proposals for public comment. During the public comment period, and in some cases after the public comment period closes, the NRC provides additional opportunities for public involvement (meetings) with respect to the development of the draft final regulation and guidance before they are published as final rules and guidance. These public involvement opportunities allow the public to comment on the need for the new or changed regulation and the backfitting implications (if any), as well as the substance of the new or changed regulation.

For revisions to existing guidance not associated with a rulemaking, the NRC addresses backfitting and issue finality if the guidance is directed at entities who are subject to backfitting or issue finality protection. In addition, for revisions to "durable" guidance (as opposed to interim or temporary guidance), the NRC prepares a regulatory analysis to support the revised guidance. The NRC provides an opportunity for public comment on all substantive revisions to existing guidance.

All of these actions help to ensure both regulatory discipline and that changes are truly warranted and consistent with the NRC's Reliability Principle of Good Regulation.

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

January 12, 2016

The Honorable James M. Inhofe
Chairman, Committee on Environment and
Public Works
United States Senate
Washington, DC 20510

Dear Mr. Chairman:

The U.S. Nuclear Regulatory Commission appeared before the Committee on Environment and Public Works on October 7, 2015. From that hearing, you forwarded questions for the hearing record to the Commission. All but seven of the responses were provided to you on December 17, 2015. Enclosed with this cover letter are the remainder of the responses to Senator Inhofe question 5; Senator Barrasso questions 1, 3, 5, 9, and 10; Senator Markey's requests 3, 5, and 6 regarding transportation packages for spent nuclear fuel and high-level radioactive waste. If I can be of further assistance, please do not hesitate to contact me at (301) 415-1776.

Sincerely,

Eugene Dacus, Director
Office of Congressional Affairs

Enclosures:
(As stated)

cc: Senator Barbara Boxer

The Honorable James Inhofe

QUESTION 5. **Please provide the ratio of the number of Office of Research's FTEs to the number of contractor FTEs utilized by the Office of Research.**

ANSWER

The NRC cannot provide the information requested. Although the agency controls the number of Full Time Equivalents (FTEs) assigned to the Office of Nuclear Regulatory Research (RES), which had 219 FTE in 2015, contractors for the agency are not required to itemize their FTE under the contracts and interagency agreements. In addition, contractors may further subcontract portions of the initial request. In Attachment 1, RES provided FTE budget information in the NRC's October 6, 2015, response.

Senator Inhofe Attachment 5

U.S. Nuclear Regulatory Commission
Research Product Line Activities
As of June 30, 2015

U.S. Nuclear Regulatory Commission
Research Product Line Activities [1,2,3]
As of June 30, 2015

				FY 2015			FY 2015			FY2015				FY 2016		
				Enacted			CE - Enacted			Estimated Actual (3Q actual/1Q forecast)				President's Budget		
Business Line	**Product Line**	**Products**	**Office**	**CS $ K**	**FTE**	**Total Amt**	**CS $ K**	**FTE**	**Total Amt**	**Cost Center**	**CS $ K**	**FTE**	**Total Amt**	**CS $ K**	**FTE**	**Total Amt**
BL-11 Operating Reactors	PL-6 Research	142 International Research	RES	2,946	5.9	3,908	2,946	6.0	3,924	1015 Halden Reactor[5]	2,002	4.4	3,931	2,658	6.0	3,654
										1024 International Cooperation[6]	1,213					
BL-11 Operating Reactors	PL-6 Research	206 Generic Issues Program	RES	225	7.7	1,480	225	7.5	1,448	N/A - $0 executed	0	3.0	489	225	4.0	889
BL-11 Operating Reactors	PL-6 Research	156 Mission IT	RES/OIS	3,006	2.5	3,414	3,006	2.5	3,414	1126 Infrastructure Services and Support[8]	727	1.1	3,151	1,125	2.5	1,540
										6000 Scientific Software Licenses[9]	562					
										6001 HPC Laptops, Workstations and Servers[10]	651					
										6012 NUREG Composite Authoring Platform[11]	1,032					
BL-11 Operating Reactors	PL-6 Research	212 Fukushima NTTF	RES	1,651	6.9	2,776	1,651	4.5	2,385	1062 Tier 1[12]	257	0.9	404	1,342	4.5	2,089

Senator Inhofe Attachment 5

U.S. Nuclear Regulatory Commission
Research Product Line Activities
As of June 30, 2015

				FY 2015			FY 2015			FY2015				FY 2016		
				Enacted			CE - Enacted			Estimated Actual (3Q actual/1Q forecast)				President's Budget		
Business Line	Product Line	Products	Office	CS $ K	FTE	Total Amt	CS $ K	FTE	Total Amt	Cost Center	CS $ K	FTE	Total Amt	CS $ K	FTE	Total Amt
BL-11 Operating Reactors	PL-6 Research	213 Reactor Research	RES	14,689	128.6	35,651	14,689	124.5	34,983	1003 Consequence Analysis	1,079	122.5	61,915	38,987	128.0	60,235
										1006 Digital I&C	1,094					
										1007 Electrical Research	744					
										1013 Fire Safety Research	3,682					
										1014 Flooding (non-Fukushima)	393					
										1031 Long Term Research	227					
										1032 Materials Performance Research	13,006					
										1038 Operational Events Analysis	800					
										1044 Radiation Protection	2,257					
										1045 Reactor Safety Codes and Analysis	8,143					
										1052 Risk Analysis	9,345					
										1058 Seismic and Structural Research	1,177					

Senator Inhofe Attachment 5

U.S. Nuclear Regulatory Commission
Research Product Line Activities
As of June 30, 2015

				FY 2015			FY 2015			FY2015				FY 2016		
				Enacted			CE - Enacted			Estimated Actual (3Q actual/1Q forecast)				President's Budget		
Business Line	Product Line	Products	Office	CS $ K	FTE	Total Amt	CS $ K	FTE	Total Amt	Cost Center	CS $ K	FTE	Total Amt	CS $ K	FTE	Total Amt
BL 17 New Reactors	PL-6 Research	100 Advanced Reactor Research	RES	910	6.7	2,002	820	3.5	1,391	1061 Technical Assistance[13]	517	0.6	614	820	3.5	1,401
BL 17 New Reactors	PL-6 Research	161 New Reactor Research	RES	3,371	14.0	5,653	4,031	13.0	6,150	1031 Long Term Research	150	5.0	5,413	4,240	13.0	6,398
										1061 Technical Assistance[13]	4,448					
BL 38 Fuel Facilities	PL-6 Research	P-155 Materials Research	RES	0	0.6	98	0	0.5	82	N/A - $0 executed	0	0.0	0		0.5	83
BL-33 Spent Fuel Storage and Transportation	PL-6 Research	P-199 Waste Research	RES	1,553	3.5	2,124	1,553	3.5	2,124	1036 Non-Waste Confidence Research	1,378	5.5	2,439	1,453	4.0	2,117
										1068 Waste Confidence Research[7]	164					
BL-33 Spent Fuel Storage and Transportation	PL-6 Research	P-199 Waste Research	NMSS	1,435	8.0	2,739	1,435	4.5	2,169	N/A - $0 executed[4]	0	4.4	717	1,435	3.5	2,016
BL-34 Nuclear Materials	PL-6 Research	155 Materials Research	RES	350	2.0	676	350	2.0	676	1044 Radiation Protection	424	2.2	783	450	2.0	782
BL-35 Decommissioning and LLW	PL-6 Research	199 Waste Research	RES	0	2.0	326	0	2.0	326	1061 Technical Assistance[13]	840	1.6	1100.4	0	2.0	332

Senator Inhofe Attachment 5

U.S. Nuclear Regulatory Commission
Research Product Line Activities
As of June 30, 2015

				FY 2015			FY 2015			FY2015				FY 2016		
				Enacted			CE - Enacted			Estimated Actual (3Q actual/1Q forecast)				President's Budget		
Business Line	Product Line	Products	Office	CS $ K	FTE	Total Amt	CS $ K	FTE	Total Amt	Cost Center	CS $ K	FTE	Total Amt	CS $ K	FTE	Total Amt

Foot Notes:

1. The FY 2015 implemented budget data includes $22 million of prior-year funding authorized in the FY 2015 appropriation language and transfers from other product lines for higher priority user needs.

2. The budget data does not include mission direct supervisory staff, allocated corporate overhead, and travel resources

3. Table includes budget data for the Research Product Line. Therefore, resources for generic homeland security and support for Rulemaking are not included.

4. Resources in FY 2015 implementation budget for Nuclear Materials/Materials Research were reduced in the Appropriations Plan

5. The Halden Reactor Project (HRP) is a cooperatively funded international research and development project that operates under the Organisation for Economic Co-operation and Development's Nuclear Energy Agency (OECD/NEA). The HRP performs research in the areas of nuclear fuels, irradiation-assisted degradation of nuclear reactor materials, digital systems, and human factors.

6. International Cooperation captures resources typically used for participation in OECD/NEA multilateral cooperative research programs.

7. The work is for "Technical Assistance for Extended Storage of Spent Fuel Thermal Analysis for a Vertical Cask." This applies to waste confidence in the broad sense, e.g., to include activities associated with extended storage of spent fuel.

8. Infrastructure Services and Support includes relocation of the RES High Performance Computing system from Church Street to Three White Flint. Additional expenditures include costs for the relocation and establishment of the secure enclave room in Two White Flint.

9. Scientific Software Licenses refers to software licenses (e.g., FLUENT, ANSYS, STAR-CD) for use in modeling complex fluid engineering scenarios that affect the design, construction, and safe operation of nuclear facilities and safe transfer, storage, and transportation of nuclear materials and waste, consistent with the mission of the NRC.

10. HPC Laptops, workstations and servers cost center captures the investment in equipment associated with the high-performance computer systems that are used in support of the scientific software (e.g., Fluent, Ansys, STAR-CD) that models complex fluid engineering scenarios that affect the design, construction, and safe operation of nuclear facilities and safe transfer, storage, and transportation of nuclear materials and waste, consistent with the mission of the NRC. Systems administration and FISMA cybersecurity compliance are also included in this line item.

11. NUREG composite authoring platform includes costs for establishing infrastructure built on the Business Process Automation Stack (BPAS), to provide document management, composite and collaborative authoring, web publishing, and a revamped and more intuitive public website. This platform supports the Generic Issues program, knowledge management and the Agency Lessons Learned activities.

12. Tier 1 cost center includes work in support of the Fukushima Tier 1 activities, specifically the FY15 funds support the Scoping Study for PRA method for seismically induced fires and floods.

13. Technical Assistance is a high level cost center capturing work performed for regulatory offices in support of user needs.

The Honorable John Barrasso

QUESTION 1. **Has the NRC documented any cases where recovery solutions from a uranium in-situ recovery facility has migrated beyond the permit area and contaminated drinking water?**

ANSWER

No. The NRC staff has not documented any case where recovery solutions from an NRC-licensed uranium in-situ facility have migrated beyond the licensed (permit) area and contaminated drinking water.

QUESTION 3. **Please describe how the characteristics of a uranium recovery facility age or otherwise change within ten years such that a full review is required to extend the license.**

ANSWER

The analysis for license renewals is less complex than the analysis for the original licensing. A "full review" occurs only during initial facility licensing, which requires an Environmental Impact Statement (EIS) or Supplemental EIS. An application to renew (extend) a license usually requires only an Environmental Assessment that focuses on environmental impacts not previously evaluated. For example, uranium recovery facilities are often co-located with other types of energy projects or in areas where new technologies allow for energy sources to be developed or utilized that were previously absent. Facilities also are located in areas where plant or animal species may be added or removed from federal protection lists since the previous license renewal, the need for groundwater resources may change in an area over time, and historic resources near a site may be identified that were not previously identified.

Likewise, staff's safety evaluation for license renewal focuses on changes in the facility or changes to regulations or guidance that would warrant changes to the license to protect health and safety. For example, deep disposal wells associated with in-situ uranium recovery facilities can lose their disposal capacity or may fail over time; uranium recovery production, monitoring, and remedial technologies change over time; land application disposal methods may lose their effectiveness; and facilities may have accidents during operations and resulting lessons learned that can and should inform the license renewal review.

<u>QUESTION 5.</u> **How might a longer license duration help the NRC manage its workload better in this area [uranium recovery facilities]?**

<u>ANSWER.</u>

Extending the license term would reduce the administrative burden associated with the license renewal process for both NRC staff and the uranium recovery licensees. However, in order to extend the license term, the staff would first have to determine that operational, environmental, and health and safety issues can be appropriately accounted for over the extended term of a license. Such a change would likely reduce or eliminate the NRC staff's current practice of deferring or delaying review of renewal applications based on limited resources.

<u>QUESTION 9.</u> **Please describe the NRC's process for providing uranium recovery applicants with transparency regarding the status of application reviews for both new facilities and license extensions.**

<u>ANSWER.</u>

The NRC's uranium recovery licensing Project Manager (PM) communicates regularly with the applicant during application reviews and provides schedule updates as the reviews progress.

This includes informing the applicant if its application review has been deferred or delayed due to the NRC's staff workload. Review schedules are posted publicly on the NRC Uranium Recovery website and updated as schedule dates are determined or changed. The NRC staff PM and applicant PM discuss scheduling and review status weekly, on average, but they can be more or less frequent depending upon the stage of the review. The NRC PM also schedules publicly noticed meetings at the request of either NRC staff or the applicant to discuss and resolve issues. These meeting notices are posted on the NRC's public meeting webpage. The results from these meetings are documented in a meeting summary and made publicly available in the NRC's Agencywide Documents Access and Management System (ADAMS).

QUESTION 10. **The NRC's "Generic Environmental Impact Statement (GEIS) for In-Situ Leach Uranium Milling Facilities" was expected to improve the efficiency of environmental reviews for these facilities leading to completion of most licensing reviews within two years. Please indicate the cost of producing the GEIS and describe why those expected benefits have not materialized.**

ANSWER.

The GEIS for In-Situ Leach Uranium Milling Facilities, which was completed in May 2009, cost approximately $1.62 million to prepare. The GEIS has been helpful during the NRC's licensing reviews, and the NRC has issued five Supplemental EISs (SEISs) that tier off of the GEIS. The NRC staff has effectively incorporated applicable GEIS discussions and adopted relevant GEIS conclusions into the SEISs, allowing the staff to focus on site-specific issues. The first SEISs did, however, present challenges and the staff has improved its review process based on lessons learned from this experience. The GEIS has helped expedite the staff's SEIS development schedule and the NRC has and continues to identify lessons learned and improve its processes.

Although the GEIS has helped expedite the staff's SEIS development schedule and the NRC has and continues to identify lessons learned and improve its processes, there are other factors that have impacted review schedules and costs. These factors include, but are not limited to: (i) quality of license applications; (ii) timing and quality of responses from an applicant to the NRC staff's Requests for Additional Information (RAIs); (iii) number of cooperating agencies and interested stakeholders; (iv) number of public comments received on the draft SEIS; (v) size, complexity, and location of the proposed project; (vi) number and complexity of an applicant's changes to the project design or to the affected environment during SEIS development; and (vii) whether a hearing is associated with the licensing action and the number of issues to be addressed in the hearing. Each application processed since the GEIS was issued has involved at least one of these factors, contributing to a lengthened review process. To partially mitigate potential future delays, the NRC staff has conducted two workshops, in Denver and NRC Headquarters, to discuss lessons learned with applicants and licensees on the environmental reviews that can be incorporated into future actions.

The Honorable Edward Markey

QUESTION 6. NRC Certification of Transportation Packages for Shipping Spent Nuclear Fuel to a Consolidated Interim Storage Facility or to a Geologic Repository.

The use of transportation packages (shipping casks) certified by the NRC would be required for shipments to a consolidated interim storage facility or to the proposed repository at Yucca Mountain. The National Academy of Sciences (NAS) 2006 report, Going the Distance? The Safe Transport of Spent Nuclear Fuel and High-Level Radioactive Waste in the United States, strongly endorsed the use of full-scale testing to determine how packages will perform under both regulatory and credible extra-regulatory conditions. The Blue Ribbon Commission on America's Nuclear Future 2012 Report to the Secretary of Energy in turn endorsed this NAS recommendation. However, full-scale testing of transportation packages is not currently required by NRC for certification under 10 CFR Part 71. Please provide the following information:

1. A list of all approved transportation packages for spent nuclear fuel and high-level nuclear waste, currently certified by the Commission under 10 CFR Part 71;

2. A physical description of each approved transportation package design, including the intended transportation mode, the design waste volume, and associated impact limiter;

3. A list of all physical tests performed to determine the safety of each approved transportation package, including the associated impact limiter;

4. The criteria used to determine the safety of each approved transportation package for certification;

5. A description of the methods or analyses used in tests to measure transportation package performance for each criteria; and

6. The results of all tests on each approved transportation package and associated Impact limiter.

ANSWER.

Our responses to your requests 1, 2, and 4 were transmitted to the Committee on December 17, 2015. Enclosed are the responses to your requests 3, 5, and 6. For each transportation package, we include information regarding all physical tests performed, a description of methods and analyses used, and test results.

Response to Request for Information Items 3, 5, and 6:

Physical Tests Performed on Spent Fuel Packages

Enclosure

Table of Contents

Physical Tests Performed on Spent Fuel Packages

1. Model No. NAC-LWT (Docket No. 71-9225)

The information below was obtained from Section 2.10.8 in NAC International, Inc., consolidated application dated November 30, 2014 (see Agencywide Documents Access and Management System (ADAMS) Accession No. ML15020A548).

A. Tests performed

A series of free drop tests were performed on a quarter-scale model of the NAC-LWT package. Thirty-foot drop test orientations included a vertical top end drop, top corner drop at an angle of 15.7 degrees from vertical (center of gravity over corner), side drop, and a bottom oblique drop at an angle of 60 degrees from vertical. After these tests, the scale model was dropped 40 inches onto a puncture pin at the mid-point side of the package. After the tests, measurements were performed on the package internal pressure to determine whether the containment boundary leaked.

A series of quasi-static crush tests were performed on quarter-scale models of the bottom impact limiter for the NAC-LWT package. These crush tests were performed to document the force-deflection and energy absorption characteristics of the honeycomb material used in the impact limiter. The limiter orientations tested were end (axial), side, and center of gravity over corner (15.7 degrees from axial).

B. Description of the methods or analyses used in tests

The package was instrumented with nine strain gauges to determine maximum bending stresses and maximum plastic strains in the scale model. The strain gauges were attached at three axial locations, 120 degrees apart around the circumference of the outer surface of the scale model.

C. Test Results

Data obtained from the tests consist of both qualitative information with respect to observations about the package and the limiter and quantitative data obtained from the recorders. The data for each test include measured impact limiter deformation, strain gauge data and stress calculations (for the end drop and side drop only), and observations of the package and attachments. The strain gauge data were only presented for the end drop and the side drop since the loads developed in those tests are the most severe from an overall structural consideration. The end drop corresponds to the maximum axial loading condition, while the side drop developed the maximum lateral loading on the overall package body.

a. Top Drop

For the top drop the aluminum honeycomb impact limiter was crushed to an average depth of 1.2 inches out to a diameter of 7.2 inches. The

1.2-inch crush corresponded to an overall crush strain of 33 percent. It was also observed that the attachment lugs for the limiter failed; however, once the limiter is engaged in the crushing action, the lugs do not provide additional functionality in the end drop condition.

The maximum axial strain observed in the test was 560 microstrains. The axial stress is calculated by multiplying the axial strain by the modulus of elasticity. The maximum axial stress in the drop is 15.8 ksi, which is much less than the yield strength of the material.

As the limiter was driven onto the package, the test valve used to pressurize the interior was broken off, which prevented the pressure from being measured after the test. A new valve was installed and the cavity was pressurized to 30 psig and was maintained to determine if the package containment boundary was damaged. There was no measurable pressure loss. The loss of this valve has no implications for the full-scale package, since the full-scale version does not have this valve.

b. Top Corner Drop

For the top corner drop, the maximum amount of crush was 1.38 inches, which corresponds to a strain of 38 percent. Similar to the top drop, the attachment lugs for the limiter failed; however, once the limiter is engaged in the crushing action, the lugs do not provide additional functionality in the corner drop condition.

Based on the pre-test and post-test dimensional measurements, nearly all permanent deformation of the packaging model was limited to the impact limiters, as expected and desired. The only significant deformation of the package body model occurred in the side puncture test, where local deformation of the outer shell and the lead shielding did occur. As designed, the outer shell was not punctured. This local deformation was fully expected and produced a very slight, local dimple in the inner shell wall (0.05 in) of the model. These local puncture deformations are of no consequence since the containment vessel and its contents are protected. Because no deformation to the package was observed, except as previously discussed, and the containment vessel sustained essentially no deformation, leakage was not expected and did not occur.

The coupling connected to the valve used to pressurize the interior was loosened, which prevented the pressure from being measured after the test. The coupling was tightened and the cavity was pressurized to 29.2 psig and was maintained to determine if the package containment boundary was damaged. There was no measurable pressure loss.

c. Side Drop

Limiters in the side drop condition were crushed only in regions that are backed by the package body. The impact limiters were crushed on both the inside next to the scale model and outside where it made contact with the impact surface. The maximum crush on the bottom impact limiter was 2.64 inches near the package and the minimum was 0.2 inches at the bottom. The thickness of the as-built bottom impact limiter is 3.89 inches. The maximum crush on the top impact limiter was 1.52 inches and the minimum was 0.52 inches at the top. The thickness of the as-built top impact limiter is 4.52 inches.

The maximum axial strain observed in the test was 2500 microstrains, which also resulted in a permanent set of 750 microstrains. Due to the manner in which the package is loaded, the maximum strains occurred at the midpoint. The maximum calculated stress for the side drop is 52.2 ksi. The pressure in the package was checked prior to the test and afterwards. It was found to be 30.4 psig. This indicates that pressure containment was maintained during impact.

d. Bottom Oblique Drop

In the 60-degree oblique drop test, the maximum crush on the outside of the bottom impact limiter is 1.25 inches, and the inside which contacts the package is approximately evenly crushed at 1.39 inches. The maximum crush on the outside of the top impact limiter is 0.5 inches, and the minimum is 0.25 inches. The inside of the impact limiter is crushed approximately evenly at 1.5 inches. The pressure measurement indicated that the cavity pressure did not change as a result of the oblique drop test. Cavity pressure was measured to be 30.2 psig before and after the test.

e. One (1) Meter Puncture test

As a result of dropping the package body onto the steel pin, the containment of the package body was not violated. This was confirmed by the pressure measurements before and after the test. All permanent strain was local to the region of impact of the pin. The maximum deformation of the outer shell was 0.5 inch deep and a localized depression of 0.05 inch occurred in the inner shell. Note that the pin puncture impact was directed against the portion of the outer shell that was stressed most severely in the 30-foot drop tests to show that the outer shell retained its puncture capability after the dynamic stress of the 30-foot fall impact.

f. Force-Deflection Tests

Quasi-static force-deflection tests were performed on quarter-scale model impact limiters used in drop testing thee quarter-scale model. Limiter samples were selected for a particular test based on the limiter having no damage for test orientation. Three limiter orientations were tested – 0, 15, and 90 degrees. While each impact limiter tested was being compressed, it was instrumented with two calibrated linear variable differential transformers (LVDT) mechanically attached to test fixtures to obtain crush force as the impact limiter deformed. Deformation of the limiter proceeded well into honeycomb lock-up. As the force on the limiter decreased after the limiter locked up, force and deflection continued to be monitored, revealing the amount of elastically stored energy. The static force for each data point is multiplied by 1.196, a static to dynamic scaling factor, enabling comparison values computed with the RBCUBED computer program.

Forces calculated from the RBCUBED computer program are higher in all cases except the end drop. The end drop forces were higher due to a shearing previously unaccounted for, causing 10 percent higher forces than calculated by RBCUBED. The shearing and shear force generation occurs simultaneously with crushing, and is a small force compared with the crush force. The average maximum/peak forces and g-loads calculated using RBCUBED was compared with corresponding values from each of the quasi-static tests. Structural margins are all positive.

2. Model No. GA-4 (Docket No. 71-9226)

The information below was obtained from the application dated January 31, 1997 (see ADAMS Accession Nos. ML15351A261, ML030860360, ML15351A295, and ML030860386).

A. Tests performed

1. Honeycomb Impact Limiter Force Deflection Tests

Four quarter-scale versions of the impact limiter designs were tested at different crush angles to provide data on the load-versus-deflection curve of the impact limiter. Three impact limiters were tested twice, on opposite sides. The tests performed range from end to side crush. The table below shows the tests performed.

Impact Limiter No.	Test on Impact Limiter	Test Crush Orientation (degrees)	Energy Dissipated During Tests (lb-in)
1	1	60	626,178
	2	15	480,743
2	1	0 (side)	324,570
	2	45	464,708
3	1	75	689,051
	2	35	218,119
4	1	90 (end)	717,358

2. Half-Scale Model Drop Tests

The tests consisted of three regulatory test sequences:

1: Side drop
- 30-foot side drop
- 40-inch puncture test in horizontal orientation.

2: Slapdown
- 30-foot drop with longitudinal axis at 30° from horizontal
- 40-inch puncture test in horizontal orientation.

3: Center-of-gravity (CG) over closure
- 30-foot CG over closure corner drop with the longitudinal axis tilted 12° from the vertical position,
- 40-inch puncture drop with the longitudinal axis oriented 7° from the vertical and the punch striking the closure in the vicinity of the gas sample port and closure bolts, and
- 40-inch puncture drop with the package oriented horizontally with the punch striking a longitudinal edge of the model body near mid-length at the location of a joint between two depleted uranium rings.

3. Full-scale Closure Seal Tests

The primary O-ring seal of the package was tested for leakage using a full-scale mockup of the package closure and flange.

B. Description of the methods or analyses used in tests

1. Honeycomb Impact Limiter Force Deflection Tests

i. Test Methods

The tests were performed on a compression testing machine using a quarter-scale model of the impact limiters. The impact limiters were directly backed by a solid aluminum test fixture. The test set-up was instrumented during the entire event to provide a complete record of the load applied to the specimen as a function of deflection. Graphs of the load-versus-deflection data were produced.

2. Half-Scale Model Drop Tests

i. Tests Conditions

All tests were performed at ambient temperature. The initial pressure in the model's fuel cavity was 80 psig (0.55 mpa). The model was not disassembled or parts replaced during any test sequence. Impact limiters, impact limiter bolts, and closure seals were replaced after each sequence.

ii. Test Methods

GA constructed a concrete and steel drop pad that meets International Atomic Energy Agency's guidelines for an unyielding surface. For puncture events, a 3-inch diameter mild steel puncture pin was bolted to the pad.

Prior to any testing, the model was marked to locate the exact points at which the measurements were to be taken. For each drop, the model was rigged in the proper orientation, lifted to a height of 30 ft (or 40 inches for the puncture tests) by a crane, and released by simultaneously firing multiple explosive cable cutters. Triggering of the high speed cameras and the strain gage and accelerometer data acquisition system was synchronized with the cable cutter firing.

The closure O-ring seals and the gas sample port seals were leak tested before and after each test sequence. The impact limiters were inspected after each test to determine the damage caused by the test. Dimensional checks and helium leakage tests of the containment boundary and cavity liner were performed on the model package after all testing was completed.

The dimensional measurements included are overall length measurements and package body profile. Complete dimensional checks were performed before and after testing was completed. The model was disassembled and all the removable parts inspected. The fuel support structure was not removed from the cavity.

3. Full-scale Closure Seal Tests

i. Test Conditions

Four tests were performed at temperatures of ambient, -42 °F, 250 °F, and 380 °F. Shim plates between the fixture lid and flange, ranging from 0 to 0.038 in., simulated gaps resulting from thermal-induced distortion. The leakage testing was carried out by means of a helium mass spectrometer leak detector, following the guidelines in American National Standards Institute American National Standards Institute (ANSI) N14.5-1987, "Radioactive Materials - Leakage Tests on Packages for Shipment."

ii. Test Set-up

The test fixture consisted of a lid and flange and was a full-scale representation of the cross section of the package closure end. Two dovetail grooves in the lid held the primary and secondary O-ring seals. The grooves and O-ring seals precisely modeled the full-scale package. All fixture materials were fabricated from 304 stainless steel. The fixture lid weighed approximately 170 lb. and the flange 180 lb. The fixture lid was attached to the flange with twenty, 1-inch bolts that thread into nuts tack-welded to the bottom of the flange. The bolts were torqued to 100 ft-lb. Shim plates extending all around the fixture's perimeter maintain uniform specified gaps between the lid and flange.

From operational and handling considerations it was not feasible to fabricate the test lid to the actual closure thickness of 11 inches. Since the test was a verification of the seal performance under the actual temperatures and amounts of compression experienced in the package, seal and groove dimensions were identical to those in the package. The number of bolts was increased from 12 to 20 in the test only to minimize local deflection between bolts of the relatively thin 1-in. lid. This local deflection is absent in the actual package due to the considerably thicker and stiffer lid. The bolt torque in the test was less than actual (100 versus 235 ft-lb) but the seals are fully compressed in both the test and package configurations. Once the seals are fully compressed in their grooves and metal-to-metal contact is established, additional bolt torque produces no further compression of the seals. For the reduced compression predicted by the accident analysis, shims of a known thickness were inserted between the lid and its base to duplicate this effect.

Prior to testing, the small volume between the flange and lid is initially evacuated. When the test begins, this volume is filled with helium to atmospheric pressure. A second port located between the O-rings is continuously evacuated by the helium mass spectrometer leak detector, and the detector measures the helium leakage past the primary (inner) O-ring. The detector output is recorded by a conventional strip chart recorder.

For the tests carried out in the conditioning chamber, the fixture temperatures near the inner seal are measured by two thermocouples (Type T) and recorded.

C. Test Results

1. Honeycomb Impact Limiter Force Deflection Tests - Comparison of Test and Analytical Results.

The largest differences between the test and analytical results occur during the end crush. The test showed a higher initial crush load, dropping to the analytical value later in the crush. The higher load is partly attributable to the buckling of the impact-limiter-bolt guide tubes in the end of the impact limiter. To reduce this tube buckling load, GA changed the tube material from 1/8 hard Type 304 stainless steel with a yield of 90 ksi to 304 Stainless Steel annealed with a minimum yield of

30 ksi. In addition, the tube wall thickness was decreased from .07-inch-thick to .035-inch-thick. This change reduces the guide tube buckling loads to a level that is small compared to the honeycomb crush load, while still maintaining the necessary energy absorption capability during an end crush. The remaining difference between test and analytical results is ignored, since the design margins for the package and neutron shield structure for the 1-ft end drop are high.

The crush forces from the 30° test are lower than the data calculated in ILMOD. (ILMOD is a GA developed computer code to compute the load versus deflection curves for impact limiters with standard unidirectional honeycomb.) This is almost certainly due to the fact that this was the second test of the impact limiter and the first 75° test weakened the impact limiter. After the 75° test, the outer diameter of the impact limiter had grown from 22.48 in. to 23.06 in.

The remaining difference between test and analytical results is ignored since the design margins for the package and neutron shield structure for the 1-ft end drop are high.

2. Structural Tests

The performance of the half-scale model showed the GA-4 package design to be robust. No permanent deformation of the model body occurred except local dents from the puncture attacks. The fuel support structure remained in its keyway attachment to the liner. The cavities formed by the fuel support structure and cavity liner had no permanent deformation except a local indentation of 0.07 in. at the location of the puncture in Sequence 2, Test 2. The model held its full 80 psig (0.55 mPa) internal pressure and remained leaktight (helium leakage less than 1×10^{-7} std cm^3/s) throughout all seven tests.

(Note: The analyses show that the maximum neutron-shield-fluid pressure is 416 psi for a 30-ft end drop at the maximum temperature condition. This pressure includes the increase in pressure due to thermal expansion of the fluid. The analyses show that the ILSS top and bottom end plates can withstand this pressure and meet all design requirements. During a 30-ft drop, the effect of the fluid pressure is to act opposite to the impact limiter loads and thus will reduce the loads on the ILSS outer shell. Because it is less than the impact force, it may be conservatively neglected.)

i. <u>Side Drop</u>

- Sequence 1, Test 1; 30-ft Side Drop

  Using a mobile crane, the model was lifted to a height of 30 ft. above the pad with the package body oriented horizontally and the longitudinal edge of the package marked 225° facing the impact surface. The model released cleanly and did not rotate or yaw significantly during its descent. Time to impact from release for all 30-ft drops is 1.4 seconds.

- Sequence 1, Test 2; Puncture at Side of Closure

  This test was a puncture drop from a height of 40 in. with the model oriented horizontally and the punch striking the model package's structure adjacent to the corner of the closure end. The puncture pin itself was observed to be in good condition after the test. Some of the white paint was scraped off by the honeycomb, but no permanent deformation of the pin was visible.

ii. Slapdown Sequence

- Sequence 2, Test 1; 30-ft Slapdown

  For this test new impact limiters were installed and the model was rigged with its axis tilted 30° from the horizontal position. The model was dropped with the closure end striking first, and the flat side marked 90° facing the impact surface. The closure end impact limiter crushed a maximum of 7.5 in. The model rotated and crushed the bottom end impact limiter 7.9 in. Final orientation of the model on the pad was 3° from horizontal with the closure end sitting slightly higher than the bottom end.

- Sequence 2, Test 2; Puncture at Package Body Flat Side

  Since the puncture pin hit the body directly on the flat side, this event was a severe test of the package body integrity. Both the model and the pin experienced local permanent deformation. The dent on the package body was a maximum of 0.15 inches deep. An examination of the model interior revealed that a small amount of permanent deformation had been transferred to the cavity liner and locally to the edge of one plate of the fuel support structure. The deformation of the cavity was measured as 0.07 inches. No other deformation of the fuel support structure or liner was observed.

  An internal pressure of 80 psig was maintained for sequence 2. The helium leakage test showed that the seals maintained a leaktight condition for the sequence.

iii. CG-over-closure-corner sequence

- Sequence 3, Test 1; 30-ft Drop, CG-over-closure-corner.

This test was a CG-over-closure-corner drop from 30 ft. with the model axis tilted 12° from the vertical position and a longitudinal edge of the model facing the impact surface.

Both impact limiters stayed attached to the model package. The closure end had been fitted with a new impact limiter but the bottom end limiter was reused from a previous test.

- Sequence 3, Test 2; Puncture at Closure Bolt and Gas Sample Port.

  This test was 40-inch puncture drop with the model oriented 7° from the vertical and the puncture pin striking the closure in the vicinity of the gas sample port and closure bolts.

  The objective of this puncture test was to test the integrity of the closure bolts and the gas sample port and cover. The post-sequence examination showed that the puncture pin completely compressed the aluminum honeycomb in its path and dented the impact limiter housing. Some visible local deformation was transferred to the gas sample port cover, damaging the first few threads of the screwed on cover. Due to the damaged threads, the port cover had to be partially drilled out with a hole saw in order to be removed. With the cover removed, it was observed that the quick-connect nipple was undamaged. The post-sequence pressure check of the cavity and helium leakage test of the seals confirmed that this puncture attack did not cause a breach of the containment boundary.

- Sequence 3, Test 3; Puncture at a depleted uranium joint.

  The final test was another puncture drop with the model oriented horizontally, and the punch striking a longitudinal edge of the body near mid-length at the location of a joint between two depleted uranium rings.

  After the completion of the final test, the closure seals were leak tested to assess the condition of the liner and the model package body. Both the liner and the model package body were found to be leaktight.

3. Full-scale Closure Seal Tests

   Four tests were carried out with the ethylene propylene seals. One set of seals was used for the test at -40°F, and another set was used for the other three tests. The first two tests simulated normal conditions of transport, while the last two represented hypothetical accident conditions. For the latter conditions, the thermal and thermal stress analyses predict a maximum lid/flange gap of 0.024 inches, corresponding to a seal temperature of 240 °F, while 300 °F is the maximum seal temperature, corresponding to a zero gap. The conditions used in the test are therefore conservative.

GA tested the seal and found it to be leaktight (1×10^{-7} ref cc/sec as defined by American National Standards Institute (ANSI) N14.5, "Radioactive Materials - Leakage Tests on Packages for Shipment") at -42 °F, ambient (-75 °F), and 250 °F. The maximum calculated temperature of the seals is 135 °F in the closure and 143 °F for the drain.

For hypothetical accident conditions the analysis shows that the maximum primary seal temperature is 300 °F. This includes the closure seal and the seal for the gas sample and drain ports. Manufacturer's data indicate the seal material can withstand a temperature of 350 °F for 50 hours and 400 °F for several hours. Using conditions more severe than those predicted by the analysis, the applicant tested the seal at 380 °F, after heating for 1.5 hours above 350 °F, and determined it to be leaktight. Therefore, the seal will function during the hypothetical accident thermal event. For the post-accident steady-state condition the maximum temperature of any seal is 175 °F.

3. Model No. 2000 (Docket No. 71-9228)

The applicant used finite element and other calculation methods as part of the structural evaluation. A thermal test has been conducted on the Model 2000 transport package to verify the thermal analytical model by comparing the analytical results to those of the physical testing.

The information below was obtained from Chapter 4 of the application dated December 2000 (see ADAMS Accession No. ML063650011).

A. Tests performed

A thermal test has been conducted on a GE Model 2000 package. The objective of the test is to verify the thermal analytical model by comparing the analytical results to those of the physical testing. The testing is done at ambient conditions with 600 and 2000 watts heat input in the package cavity to simulate the maximum decay heat. The test also demonstrates the capability of the Model 2000 to safely dissipate 600 and 2000 watts of decay heat.

B. Description of the methods or analyses used in tests

The thermal test was performed by placing an electric heat source concentrically within the cavity. Thirteen temperature sensing devices were placed within the cavity, on the external surfaces of the package, and on the overpack surfaces. In the radial direction, temperatures measured included the package cavity air, internal package wall, external package wall, overpack inside surface, and the overpack external surface. In the axial direction, the temperatures measured included the package cavity floor, the bottom package surface, below the lower honeycomb pad, at the bottom of the overpack, at the top of the package cavity (under the lid), at the gasket lower surface, at the top of the lid, and at the top of the overpack. Three additional transducers recorded the external ambient temperature. The temperature data were recorded at 30 minute intervals during the transient until a steady state condition was reached. Temperatures remain significantly unchanged for a one hour period.

C. Test Results

Differences between the test and the model temperatures were 3% in the package cavity, 1% at the package outer wall, 5% at the overpack inner wall, and 1% at the overpack outer wall. Test data and the LIBRA predictions of cavity wall temperature versus time for the 2000 watt case were plotted in Figure 3.35: excellent correlation between test and analysis results is displayed. The physical test confirmed the adequacy and accuracy of the model used in the LIBRA finite element thermal analysis code for the 2000 package.

4. Model No. NAC-STC (Docket No. 71-9235)

The information below was obtained from Section 2.10.6 in NAC International's consolidated SAR (see ADAMS Accession No. ML112301077) for the impact limiter tests.

A. Tests Performed

The scale model test program for the directly loaded fuel configuration of the NAC-STC included: (1) quarter-scale model drop tests, and (2) eighth-scale model impact limiter quasi-static compression tests.

1. Dynamic Impact Limiter Tests

The objective of the quarter-scale model drop tests was to confirm the design of the NAC-STC packaging through a series of tests conducted at the Winfrith Technology Centre drop test facility located in the United Kingdom. The planned tests included:

- 9-meter (30-foot) top end drop,
- 9-meter (30-foot) top corner drop (24 degrees from the vertical),
- 9-meter (30-foot) side drop,
- 9-meter (30-foot) bottom end oblique drop (75 degrees from the vertical),
- 1-meter (40-inch) pin puncture drop at the cask axial mid-point, and
- 1-meter (40-inch) pin puncture at the center of the outer lid.

When testing commenced, the model employed aluminum shell impact limiters. After completing the first three tests above, testing was suspended to effect repairs to the model. Therefore, these three tests were identified as Phase 1. After repairs were completed, it was decided to perform the fifth and sixth puncture tests above and identify these tests as Phase 2. During the performance of these tests, however, the 24-inch long pin deformed excessively, to the extent that maximum damage was not inflicted on the cask body or the outer lid. Therefore, it was determined that the pin puncture tests would be re-performed using an 8-inch tall pin.

Testing resumed after replacing the aluminum shell impact limiters with stainless steel shell impact limiters. In addition, the planned tests to confirm the cask design was changed to the following:

- 9-meter (30-foot) side drop,
- 9-meter (30-foot) 75 degree oblique bottom end drop,
- 9-meter (30-foot) top corner drop (24 degree),
- 9-meter (30-foot) 75 degree oblique top end drop,
- 1-meter (40-inch) drop cask mid-point pin puncture, and
- 1-meter (40-inch) drop outer lid center pin puncture.

Performance of these six drop tests constituted Phase 3 of the drop test program. As a result of the side and 75 degree oblique bottom end drop tests performed in Phase 3, it was determined that the redwood in the overlap region of the impact limiter was not maintaining its original position and orientation. A design modification was added to the overlap region of the impact limiter to prevent the redwood in that region from changing its orientation during the side impact. After repairing pin puncture damage to the inner shell of the model cask body, both a 9-meter (30-foot) bottom oblique and 9-meter (30-foot) side drop test was performed. These two tests made up Phase 4 of the drop test program.

2. Static Impact Limiter Test Information

A series of quasi-static compression tests were performed to simulate an end impact, a corner impact, and a side impact using eighth-scale model redwood/balsa wood impact limiters. The static compression tests were performed to demonstrate that force-deformation curves are as predicted by analytical methods, energy storage (rebound) in the crushed redwood/balsa wood impact limiters is negligible, and the geometry of both the impact limiter and cask body effectively causes the impact limiter to stay attached to the cask.

Based on the results of the quasi-static tests, the force-deformation curve and the energy absorption capacity (area under the curve) of each model impact limiter was determined. The force-deformation curve is measured by compressing the model impact limiter and recording the deflections and loads applied to the limiter. The energy storage, or rebound, of the model impact limiter is shown by the load-deformation curve as the test machine is unloaded slowly. The model impact limiter presses against the test machine heads and applies a load proportional to the elastically stored energy. This extra energy component can be restored to a cask in a multiple-impact, oblique drop "slap down" scenario.

B. Description of the methods used in tests

1. Dynamic Impact Limiter Tests

The quarter-scale model packaging was an exact replica of the full-scale design with two exceptions: (1) O-rings in the inner and outer lids were not scaled; and (2) the neutron shield was not modeled, but the weight of the neutron shield was modeled by steel blocks welded to the outer shell. All aspects of the model can be used to reflect the strains, accelerations, and impact limiter crush strokes of the full-scale design. With respect to containment, the model represents the geometrical arrangement and materials used in the full scale design. However, the O-ring dimensions and the leak rate cannot be scaled. Therefore, the pressure measurements can only be used to indicate the condition of the seals and the adjacent seating surfaces. The initial scale-model design used impact limiters with aluminum shells. However, due to poor performance, the aluminum shells were replaced with stainless steel shells after the first several drop tests.

Ninety-degree tee-rosette strain gauges were mounted on the cask body for all of the drop tests. One gauge of each tee-rosette was positioned in the axial direction and another in the circumferential direction. These strain gauges allowed the axial and the hoop stresses to be determined. Later in the testing program, the 90-degree tee-rosettes at two locations were replaced with rosettes with three gauges at 45-degree orientations. This allowed the shear stresses to be determined at the surface. All gauges had at least a 50 kHz response time to ensure that the transient strains could be accurately recorded. Real-time recording was accomplished by a system of strain amplifiers, signal conditioners and a magnetic recording unit to store the data.

Accelerometers which could measure accelerations up to 20,000 g with an accuracy of 1 percent per 2,000 g were employed for the drop tests. Since an acceleration level of only 300 g was expected, the test accuracy was $\pm$ 0.5 g. The frequency response of the accelerometer was between 2 Hz to 15,000 Hz. which enveloped the frequency of the system. All accelerometer data were conditioned and stored on magnetic media for later processing, which included filtering and integrating to obtain impact velocities.

Two high-speed cameras were used to record the behavior of the quarter-scale model as it impacted the target surface. For the top end drop, both cameras were operated at 500 frames/sec. and the cameras were positioned 90 degrees apart (side and end views). For the final top corner drop, side drop, and oblique slap down drop, one camera was positioned to capture the overall motion of the cask at 500 frames/sec. and the other camera was set to obtain a close-up view of the crushing of the impact limiter at 1000 frames/sec.

2. Static Impact Limiter Test Information

Eighth-scale tests were performed with lower impact limiters primarily because the trunnion cutouts in an eighth-scale model upper impact limiter are extremely difficult to fabricate using the scaled shell thickness. This was determined to be acceptable because previous analyses, scale model compression tests, and scale model drop tests demonstrated that the trunnion cutout regions of the upper impact limiter do not significantly affect the energy absorption capability of the impact limiter. Initially, the model impact limiters were fabricated from aluminum alloy. However, during the quarter-scale drop test program, the model design was revised to use 0.031-inch stainless steel to fabricate the impact limiter shells. The impact limiter models used in the static tests represented the revised configuration. Like the full-scale design, the eighth-scale models used redwood/balsa wood as the energy absorbing materials.

The eighth-scale model impact limiters were crushed quasi-statically in a tensile test machine capable of also applying compressive loads. The tensile test machine capacity limited the maximum size of the test impact limiter to one-eighth scale. The eighth-scale model impact limiters were not attached mechanically to the cask-shaped test fixtures. Duct tape was used to hold the model impact limiter in place while the compressive test load was applied. The tape relaxed as successively higher loads were applied, demonstrating that the impact limiter geometry produces net crush forces that press the impact limiter against the cask body, regardless of the impact angle.

While each model impact limiter tested was being compressed, two calibrated linear variable differential transformers (LVDT) mechanically attached to test fixtures provided data to an X - Y recorder, which plotted crush force versus deformation. Deformation of the model impact limiter proceeded well into the compression lock-up range of the redwood. As the compression load on the model impact limiter was decreased after the test was stopped, force and deflection continued to be monitored, revealing the amount of elastically stored energy. The static force for each data point is multiplied by 1.06, a static to dynamic scaling factor, enabling a direct comparison with the analytically computed values. The dynamic scaling factor was determined from Figure 9 of NUREG/CR-0322, "The Effects of Temperature on the Energy-Absorbing Characteristics of Redwood," which is based on Sandia National Laboratories tests.

C. Test Results

1. Dynamic Impact Limiter Tests

The acceptance criteria for the cask body performance is that cavity pressure be maintained and that the fuel remain in a subcritical configuration. For the cask body, this requires that permanent deformation must not occur to the lids, the lid mating sealing surfaces, the fuel basket and the lid bolts (both inner and outer) after completion of the drop tests.

The impact limiters must limit the deceleration of the cask body and components during a cask drop event. Therefore, the impact limiter acceptance criteria requires that the crush stroke be limited to prevent the cask body from contacting the impact surface, the accelerations be limited to those used in the design analyses, and the impact limiters remain attached to the cask body and in position after the impact event.

To assess the model performance against the acceptance criteria, the following data was collected for each test:

- Metrology data - to assess the permanent deformation of the cask body, fuel basket, lid seating areas, or in the lids themselves. Measurements for all dimensions except the inner diameters in the lower portion of the model cask cavity were obtained in a lab before and after the tests. The tolerance for all measurements was $\pm$ 0.001 inch.

- Pressure and temperature data - to assess the retention of pressure by the cask primary containment boundary, the cask cavity was pressurized to 30 (+2,-0) psi using a pressure port located near the model cask midpoint. The pressure in the cask cavity was measured before and after each test. To assist in correlating the pressure change with a change in the cask temperature, the temperature of the cask body was also obtained by Chromel/Constantan thermocouples attached to the cask exterior near the pressure port used to pressurize the cavity.

- Strain data - strain time-histories were recorded for each of the 30-foot drops to determine the maximum amount of strain experienced by the cask body. Stress data were mathematically derived from the strain gauge data. Strain gauge data were only taken for the 30-foot drop tests. It was concluded that the strain gauge data were not needed for the pin puncture drops.

- Acceleration data - to determine the maximum accelerations to which the cask was subjected, two single-axis accelerometers were mounted on the cask body for each of the 30-foot drop tests. The directions were altered for each individual test to ensure that the vertical deceleration was measured. Acceleration data were only taken for the 30-foot drop tests. It was concluded that the accelerometer data were not needed for the pin puncture drop tests.

- Impact limiter deformation data - to evaluate the behavior of the impact limiters, the crush stroke for each orientation, and the condition of the limiter attachment to the cask body after each test, the limiters were inspected to determine the amount of deformation that had occurred after each test. In addition, the condition of the attachment rods and nuts was also determined. Photographs of the deformed limiters were taken to record the post-test condition of the limiters.

- High speed photography - to review and assess the actual angle of impact and the behavior of the cask body and impact limiters during the impact.

i. Thirty-Foot Top End Drop Using Impact Limiters with Aluminum Shells - Test No. 1 of Phase 1

This was the first drop test to be performed of the four phases of tests using the quarter-scale cask model. The impact limiters used the aluminum shell design, which weighs less than the stainless steel shell design used in later tests. The cask model at the time of the top end drop was within 0.5 percent of the design weight of 3906 pounds ($250{,}000/4^3$ pounds).

Essentially all of the crushing occurred within the backed region of the impact limiter. The crush deformation was 2.11 inches, corresponding to a crush strain of 23 percent. The maximum stress was 8.6 ksi. The hoop strain component was extremely small. All three gauge locations near the top showed similar behavior in the axial and the hoop direction. One of the strain gauges showed a maximum permanent strain of 0.0015 percent. Since the normal stresses are so low, this offset is not attributed to any yielding of the material. Two accelerometers were mounted 180 degrees apart at the top end of the cask model. The maximum acceleration obtained from a 1000 Hz filtering of the accelerometer trace was 247 g. This value corresponds to a full-scale acceleration of 62 g. The filter frequency was computed by considering the first longitudinal vibrational mode, f_l, of the model as determined using an expression from Blevins for a lump mass attached to a cantilevered beam. The pressure measured after the test showed a slight increase, which corresponds to the small increase in the cask body temperature. Since the temperature data cannot be expanded to determine the temperature of the cavity gas, an accurate calculation of the corresponding increase in the pressure cannot be made. The pressure measurements indicate that there was no loss of pressure. After the top end drop test, the basket was removed from the cask body to inspect for deformations. The fuel basket was removed without any interference and neither the fuel basket nor the dummy fuel assemblies had deformed. This indicates that out of plane buckling due to the vertical deceleration loads did not occur, and that buckling of the inner shell due to lead slump did not occur. As evidenced by the nearly uniform crush of the impact limiter and the data from the two axial accelerometers, the load on the lead was essentially uniform around the model circumference. Thus, yielding of the inner shell at one location on the circumference would have precipitated yielding of the larger shell around the entire circumference. However, there was no evidence of yielding in the lead, basket, lids, or bolts used in the model. This test satisfactorily verified the packaging design for the nine-meter (30-foot) end drop.

ii. Thirty-Foot Side Drop Using Impact Limiters with Aluminum Shells - Test 3 of Phase 1

The high speed film showed that the welds along the edges of the aluminum limiters failed immediately upon impact, which allowed the four steel blocks at the lower edge of the cask to strike the impact surface. The force to decelerate the cask model was concentrated at the four steel blocks. The energy was absorbed by the local deformation of the model cask body shells and the model fuel basket. Based on the high-speed film, the rebound of the cask body was small, indicating that essentially all of the energy was absorbed in the initial impact. As a result of the localized loading on the cask body shells, the top forging, which serves as the seat for the lids, was deformed and the internal cavity pressure was not maintained. However, the lids remained firmly attached to the cask body during and after the impact. This test served to describe the behavior of the cask body in a guillotine-type impact without a neutron shield.

Strain gauge data were recorded at nine locations. A permanent strain in both the hoop and axial directions was identified. Therefore, an equivalent plastic strain (e_{eq}) was computed based on the Von Mises and Prandtl-Ruess Flow Rule material representation for material yielding to assess the amount of work-hardening undergone by the material. The maximum value found was 1811 microstrains, or 0.18 percent, at the midpoint of the cask at the point nearest the impact plane. Two strain gauges were approximately 0.25 inches from the edge of the blocks, which were displaced into the outer shell. This implies that the strain gauges were able to reflect the maximum strains generated by the impact.

The maximum accelerations recorded were 996 g and 1190 g. The first six or seven milliseconds correspond to the crushing of the redwood, after which the large increase in the deceleration is due to the steel blocks striking the impact surface.

The radial deflection was greatest at the lid end of the cask. The inner radius was decreased by 0.126 inches at the point of impact (the 0-180 degree diameter). The impact also caused out-of-round deformation of the cask at other measured locations by approximately 0.06 inches to 0.09 inches on a radius.

The inner and outer lids were inspected for out-of-plane deformation, and the measurement of the out-of-plane dimensions showed that no deformation had occurred during the Phase 1 tests.

The deformation of the model cask body required that the fuel basket be partially disassembled in the cask cavity in order to remove the basket after the side drop test. Two support disks were damaged in the drop test. One support disk had been loaded by the steel blocks, and the other support disk was located at the axial center of the basket. The out-of-plane measurement for the support disk located at the axial center of the basket was 0.001 inch, which is the sensitivity limit of the equipment. In addition, the support disk at the axial center of the cask did not experience plastic deformation. For the support disk loaded by the steel blocks, the maximum variation in the direction perpendicular to the plane of the disk was 0.004 inch. The impact also caused deformation to the bottom of the support disk. The impact of the steel blocks into the cask body and basket resulted in the lateral movement of the lower four fuel assembly positions by 0.19 inches. In addition, none of the support disks exhibited any out-of-plane buckling.

The deformation of the fuel basket near the steel block is not classified as buckling deformation, but rather deformation imposed by the impact of the steel blocks. Before proceeding with further testing, damaged components were repaired to meet the model drawing specifications and a new basket was installed. Since the inner shell was subjected to a small degree of work-hardening, the yield strength changed slightly.

The change in the yield strength is estimated by the product of the tangent modulus (Et) and the e_{eq} strain determined from the drop test. From NUREG/CR-0481, "An Assessment of Stress-Strain Data Suitable for Finite-Element Elastic-Plastic Analysis of Shipping Containers," for Type 304 stainless steel, Et is 370,000 psi. Multiplying 370,000 by 0.18 percent, the yield strength increased by approximately 700 psi. The 700 psi change corresponds to a change of about 2 percent for Type 304 stainless steel at 70 °F. This change is considered to be insignificant. Although this test demonstrated that aluminum welds are inadequate to maintain the integrity of the impact limiter, this test clearly demonstrated the strength of the fuel basket design.

iii. Thirty-Foot Top Corner Drop Using Impact Limiter with Stainless Steel Shells - Test 3 of Phase 3

For the top corner drop test, the cask axial centerline was oriented 24 degrees from the vertical. This corresponded to the center of gravity of the cask being over the edge of the impact limiter.

The high speed film and the shape of the crushed impact limiter indicated that a small amount of impact limiter rotation occurred during the top corner drop. As the crushing was initiated, a force couple was applied to the impact limiter by the crushing force at the edge of the impact limiter and by a force due to the edge of the cask bottom moving into the impact limiter. This force couple resulted in rotation of the impact limiter away from the cask bottom, which produced the appearance of two crush faces on the bottom of the impact limiter. Initially, the crushed surface of the bottom of the impact limiter was at a 24-degree angle with respect to the uncrushed portion of the limiter (corresponding to the corner drop angle).

During the impact, the impact limiter shifted slightly and the angle became smaller. The maximum permissible deformation was assumed to be the distance from the edge of the redwood at the corner of the limiter to the edge of the limiter nearest the edge of the cask bottom. This was to ensure that the cask corner did not impact the unyielding surface.

The crush stroke for the corner drop was significantly larger than that for the end drop, since the crush area for the corner drop initially started out as a point and increased to the maximum area of 350 square inches. For the end drop, the crush area remained constant at 477 square inches. The decreased crush area and crush force in the corner drop resulted in a much larger crush stroke.

Although strain data were recorded for all locations, the data for some locations was inadvertently destroyed during post-processing. The maximum axial strain for the top end drop was 90 inch/inch, and maximum axial strain for the top corner drop was 63 inch/inch. With the exception of one axial strain measurement taken near the bottom of the cask, which was away from the point of impact, all axial strain measurements for the top end drop exceeded the axial strain measurements for the top corner end drop. Thus, the data confirm that the top end drop axial load envelopes the top corner drop axial loads. Therefore, the maximum stress computed for the data obtained in the top end drop, 8.6 ksi, envelopes the maximum stress that occurs in the top corner drop. The maximum acceleration was 127 g.

The trace reflects the gradual increase of the impact limiter crushing area. As the impact limiter crush area increases, the deceleration force also increases. Since the dynamic modes of deformation are similar to those for the end drop, the cut-off frequency used for the top end drop is applicable for the top corner drop. The data for this test were also filtered at 4000 Hz to demonstrate the effects of using higher filter frequencies. In comparing the top corner drop to the top end drop, the top corner drop produces significantly lower accelerations.

There are two reasons for this. First, the total crush area for the top end drop was 477 square inches while the top corner drop utilized 350 square inches. Additionally, the top corner drop does not subject the redwood to a uniform strain, but rather the top corner drop crush strain varies from a maximum value to zero. The cavity pressure measured after the top corner drop test showed a decrease of 0.2 percent due to a 1.5°F decrease in the cask body temperature. The pressure measurements indicate that the cavity pressure was maintained during the test. The lids, lid bolts, basket and fuel assemblies were removed and no damage to any component was observed. This test satisfactorily verified the packaging design for the nine-meter (30-foot) top corner drop.

iv. One-Meter Pin Puncture Drops - Tests 5 and 6 of Phase 3

In preparation for the pin puncture tests, the fuel basket was removed to prevent it from supporting the cask shells during the pin puncture tests. Bags of lead weights were placed in the model cask cavity to simulate the weight of the components that were removed.

A pin puncture test was performed at the axial midpoint of the cask, and a second pin puncture test was performed at the center of the outer lid. For the cask mid-point pin puncture test, the pre-test pressure was 3.007 bar and the post-test pressure was 3.002 bar. This pressure drop was attributed to the 7.2 degree drop between the pre-test and post-test cask temperature measurements.

In the cask outer lid pin puncture event, the cavity pressure valve cracked allowing the cavity pressure to decrease. (The cavity pressure valve in the model serves only as a convenient fixture to pressurize the model cavity, and is not a part of the full-scale NAC-STC design.) The cask was refitted with another valve and the cask was re-pressurized to 3.1997 bar. At the end of 10 minutes the pressure was still at 3.1995 bar, indicating that the closure lid system had performed satisfactorily.

The cask mid-point pin puncture resulted in an indentation of 0.33 inch in the outer shell. This did not result in penetration of the outer shell. The test, however, did result in deformation of the pin itself, but the effect of the deformation of the 8-inch long pin is considered to be negligible.

For the outer lid pin puncture test, the pin was found to have impacted at a location 2.53 inches away from the true center. This corresponds to approximately 10 percent of the diameter, and would produce essentially the same result as if it were at the exact center. The metrology data indicate no permanent deformation of the outer lid for the pin puncture condition at the off-center location. A pin puncture at the center would not be expected to result in permanent deformation of the closure lids either. Some minor scraping of the outer surface of the outer lid was also noted. Performance of these tests satisfactorily verified the packaging design for the pin puncture events.

v. Thirty-Foot Bottom Oblique Drop (Top End Slapdown) using Modified Impact Limiters with Stainless Steel Shells - Test No. 1 of Phase 4

In this test, the bottom of the cask impacts first causing the top end of the cask to rotate (and slap down). For a shallow angle oblique impact (near side impact), the slap down impact usually will result in a higher acceleration than for a side drop due to the angular momentum of the rotating cask. The high-speed film verified that the model orientation angle was 75 degrees from the vertical. It is required for the quarter-scale model that the maximum crush stroke be less than 3.22 inches to prevent the neutron shield from contacting the impact surface.

In the top end slap down, the maximum crush stroke occurred in the top limiter, which was subjected to the slap down effect and was determined to be 2.41 inches. The maximum stress occurred at the top end impact limiter at the 180 degree location. While some permanent strain was recorded, the level was significantly less than 0.2 percent. The maximum acceleration occurred at the top end of the cask and was approximately 10 percent greater than that at the lower end. The peak acceleration value, 225 g, was measured using a filter frequency of 750 Hz to avoid artificially inflating the acceleration levels because of higher frequency signals associated with the instrumentation.

The pressure measured before and after the test remained constant to within the accuracy of the instrumentation indicating that no loss of pressure occurred during the test. The metrology data indicate that, for a measurement tolerance of 0.01 inch, none of the diametral dimensions changed. No resistance was encountered when removing the basket. Performance of this test and Test No. 2 of Phase 4 described below, demonstrated that the modified impact limiter design prevents the neutron shield from contacting the impact surface.

vi. Thirty-Foot Side Drop - Test No. 2 of Phase 4

After rotating the limiter from the oblique drop test 180 degrees, the side drop was performed for Phase 4. In the slap down test, the loading is not uniform and tends to be concentrated towards the slap down end. In the side drop the loading tended to be uniformly distributed over the length of the cask and equally applied to each impact limiter.

The high-speed film verified that the cask was horizontal as it approached the impact surface. For the side drop test, the crush stroke was 2.16 inches for the bottom impact limiter and 2.04

inches for the top impact limiter. The crush data does reflect that a clearance will exist between the neutron shield and the impact plane after the side drop condition.

The maximum strain (135 in/in) and the maximum stresses (29.5 ksi axial and 17.5 ksi hoop) occurred at the 180 degree location of the cask midpoint for the side drop. The side drop stresses were larger than those for the oblique drop even though the oblique drop deceleration was 10 percent larger. The accelerometers measured accelerations of 208 g on one end of the cask and 204 g on the other end of the cask. This concurs with the crush stroke data.

The pressure measured before and after the test remained constant to within the accuracy of the instrumentation, indicating that no loss of pressure occurred during the test. The metrology data indicate that, for a measurement tolerance of 0.01 inch, none of the diametral dimensions changed. During basket removal, the basket was removed without resistance and no deformations was identified. Performance of this test and Test No. 1 of Phase 4 described above, demonstrated that the modified impact limiter design prevents the neutron shield from contacting the impact surface.

2. Static Impact Limiter Test Information

For the end impact case, the impact limiter compression forces from the quasi-static test are higher than the analytical values using the maximum tolerance cold temperature crush strength and the minimum tolerance hot temperature crush strength properties of redwood. This difference in compression forces can be attributed to the additional forces on the cask due to the redwood material's resistance to shearing along the periphery of the "backed" area of the cask. The calculated equivalent deceleration force of the full-scale cask, based on the quasi-static eighth-scale model impact limiter test, is 54.8 g for the end impact case. This force is greater than the analytically determined 44.6 g end drop deceleration force using the maximum tolerance cold temperature crush strength of redwood, but less than the analytically determined 56.1 g deceleration force obtained using the minimum tolerance hot temperature crush strength of redwood. The higher deceleration force obtained using the minimum tolerance hot temperature crush strength of redwood is a result of the larger deformation and the partial lock-up of the redwood that occurs before the cask is stopped. Based on the area under the dynamically scaled force-deformation curve for the end impact case, all of the energy of a one-eighth scale model of the NAC-STC for a 30-foot drop is absorbed when the impact limiter deformation reaches 1.58 inches (1.63 inches from the static force-deformation curve), which extrapolates to a 12.6-inch deformation for the full-scale NAC-STC impact limiter, or 42 percent of the depth of the impact limiter.

For the corner impact case, the calculated equivalent deceleration force of the full-scale cask, based on the quasi-static eighth-scale model impact limiter test, is 32.6 g. This compares with the analytically determined 44.0 g deceleration force using the maximum tolerance cold temperature crush strength of redwood, and with the analytically determined 49.3 g deceleration force calculated using the minimum tolerance hot temperature crush strength of redwood. Based on the area under the dynamically-scaled force-deformation curve presented for the corner impact case, all of the energy of a one-eighth scale model of the NAC-STC for a 30-foot drop is absorbed when the impact limiter deformation reaches 3.22 inches (3.30 inches for the static force-deformation curve), which extrapolates to a 25.76-inch deformation for the full-scale NAC-STC impact limiter, or 70 percent of the depth of the impact limiter.

For the side impact case, the calculated equivalent deceleration force for the full-scale cask, based on the quasi-static eighth-scale model impact limiter test, is 45.6 g. This force compares with the analytically determined 51.7 g deceleration force using the maximum tolerance cold temperature crush strength of redwood and with the analytically determined 51.3 g deceleration force using the minimum tolerance hot temperature crush strength of redwood. Based on the area under the dynamically-scaled force-deformation curve presented for the side impact case, all of the energy of a one-eighth scale model of the NAC-STC for a 30-foot drop is absorbed when the impact limiter deformation reaches 1.64 inches (1.70 inches for the static force-deformation curve), which extrapolates to a 13.12-inch deformation for the full-scale NAC-STC impact limiter, or 71 percent of the depth of the impact limiter.

The aluminum alloy shells split along the weld seams and came apart as compressive load was applied. The Type 304 stainless steel shells remained ductile and did not split along the weld seams. The eighth-scale model impact limiter compression tests showed that a maximum of 8.2 percent of the absorbed energy may be stored during crushing and later released. The results of the eighth-scale model NAC-STC impact limiter quasi-static compression tests clearly demonstrate that the NAC-STC impact limiter design provides the energy absorption capacity to decelerate the cask to a stop for a 30-foot drop accident for the various impact orientations while maintaining maximum compression forces that are less than the cask design values.

5. Model No. TN-FSV (Docket No. 71-9253)

The information below was obtained from Section 2.10.3 in Public Service Company of Colorado's SAR pages submitted with their response to a request for additional information by NRC staff (see ADAMS Accession No. ML063530662) for the impact limiter tests.

A. Tests performed

A series of static and dynamic tests was performed on three half-scale models of the TN-FSV impact limiters. The following two static crush tests were performed on a single half scale TN-FSV impact limiter at room temperature:

- Load applied radially on the side (0 degree) of the impact limiter, and
- Load applied on the corner at an angle of 10 degree to simulate the 80 degree corner drop.

The 0 degree orientation was selected because it has the highest transverse g loading at the center of gravity. The 80 degree corner crush test was selected because it has high axial g loadings, and higher expected impact limiter deformations than the end drop.

Two 30-foot drop tests were performed at low temperature to evaluate the adequacy of the impact limiter enclosure and attachments. The orientations were a 15 degree slap down (shallow angle side drop) and an 80 degree corner drop. The 15 degree slap down orientation was selected because it puts the highest stresses on the attachment bolts, and it is the orientation for which the highest impact force is expected for a side drop orientation. The 80 degree corner drop was selected to compare with the static test and because it is the orientation for which the largest limiter deformation is expected and for which significant decelerations are expected.

All the kinetic energy of the model package must be absorbed by a single limiter for the 80 degree corner test which is very nearly a center of gravity over corner test. In a 0 degree side impact, 50% of the total energy must be absorbed by each of the two limiters. However, analyses indicated roughly 68% of the energy is absorbed by the secondary impact limiter (i.e., the impact limiter hitting second) for a shallow angle (slap down) side drop. Therefore, since the impact of the second impact limiter is nearly the same as a 0 degree side drop, the 0 degree static crush test was continued to an energy level beyond 50% to approximate the second hit of the 15 degree side drop.

B. Description of the methods or analyses used in tests

For the static crush tests, the stainless steel impact limiter structure is steel shells closed off by flat plates and reinforced by six (6) radial gussets. The model and full-scale configurations are identical, but all linear dimensions in the model are half-scale. Although the balsa and redwood densities used in the model are consistent with that specified for fabrication of the full-scale impact limiters, the wood used in the models had densities on the high end of the specified range. The model contains the same number of wood blocks as the full-size impact limiters. The wood blocks are made up of a number of smaller pieces of wood glued together with phenol resorcinol adhesive, using the same procedure to be used on the full-size impact limiters. The attachment bolts are made from the material specified for the full-size limiters. Bolts (5/8-11-2A) with an undercut shank diameter of 0.511 in. were used on the models.

The testing was performed using a 1.2 x 10^6 lb. compression testing machine. The loading surface was maintained perpendicular to the direction of crushing and the massive impact limiter support fixture was restrained from shifting during loading. The deflection of the impact limiter was measured continuously during testing using a linear potentiometer mounted to the testing machine crosshead. The crush force versus vertical movement of the test plate was recorded continuously on an x-y plotter.

For the dynamic tests, the test model is a solid carbon steel test body with a front impact limiter on each end. The test body is 103.5 inches long with an outside diameter of 15.5 inches. The total length of the dynamic test model, including impact limiters, is 123.5 inches. The dynamic tests were performed in accordance with approved written procedures. The impact surface was an unyielding concrete pad weighing more than 250,000 lbs. resting on bedrock. A mild steel plate, 2 inches thick, was secured to the surface of the concrete pad. The test drop height was 30 feet + 1.0 in./-0.0 in. Each drop was photographed and videotaped.

For the slap down test, the impact limiter that impacts second was cooled more than 18 hours to a temperature of -20 °F prior to attachment to the test body. The ambient temperature was 9 °F during the drop test. Following the slap down test, the test model was rotated 180 degrees to allow the uncrushed side of the impact limiter used for the primary hit in the slap down case to be used for the 80 degree corner drop. As before, the ambient temperature was around 9 °F.

C. Test Results

The side crush static test was terminated when the deflection reached approximately 5 inches and 81% of the kinetic energy associated with a 30-foot package drop was absorbed (5 of the 6 attachment bolts were simultaneously broken at this point). Significant deformation of the impact limiter was evident after the test. The bolt tunnels above the heads of the attachment bolts collapsed so that the heads of the bolts were inaccessible. The weld seams between the outer cylindrical shell and the flat plates on the front edge of the impact limiter tore approximately 3-4 inches in length in two places directly below the impact surface. Similarly a 3-inch tear on the back edge of the impact limiter was evident. After the impact limiter was removed from the fixture, evidence of inside crushing of the impact limiter was noticed. The seam between the inner cylindrical shell and circular plate had split in the region closest to the impact.

After the slap down test, the impact limiter attachment bolts that hit first lost their torque due to deformation of the washers beneath the bolt heads as well as tightening of the impact limiter against the test body because of the drop. None of the bolts were broken. Two weld seams between the gussets and flat segments opened slightly. The primary impact limiter pushed inward toward the test body indicating inside crushing, and also crushed on the outside at a 15 degree angle. The crushing on the secondary impact limiter was similar to a 0 degree angle (perpendicular to the package axis) side drop crush. Therefore, dynamic test data from the secondary impact limiter is compared to load-versus-displacement curves generated from the side crush static test data. Analyses indicated, for a 15 degree side drop, that roughly 68% of the total kinetic energy associated with a 30-foot package drop is absorbed by the secondary impact limiter (rather than 50% for a true side drop). The permanent crush depth associated with the secondary impact limiter after the dynamic test was about 3.57 inches not including elastic springback and about 3.82 inches assuming an estimated 0.25 inches of springback from static crush test data. The corresponding deflection from the associated load-versus-displacement curve is 4.4 inches. The crush deformations, therefore, agree within about 14%.

The corner crush static test was terminated after the deflection reached 7 inches and a total of 113% of the required energy was absorbed. The bolt tunnels above where the bolts are installed had buckled and folded closing the bolt holes. Although the outer cylinder bulged near the top seam, no exterior seams opened as a result of the corner crush test. After the 80 degree corner crush test, one segment of the impact limiter was cut open. The wood was removed and examined. It was noted that glue joints between the individual pieces of wood in the blocks did not fail as the wood crushed. For the corner crush static test, with 100% of the kinetic energy associated with a 30-foot drop absorbed by the impact limiter, the measured deflection was 6.5 inches.

After the corner crush dynamic test, the attachment bolts also lost their torque. There was evidence of both inside and outside crushing. The side of the impact limiter closest to the edge hitting first pushed around the test body due to inside crushing of the limiter on one side. Total deflection for the 80 degree corner drop, without accounting for springback, was calculated to be 6.3 inches and 6.5 inches assuming an estimated 0.2 inches of springback from the static test data.

6. Model No. NUHOMS® MP187 Multi-Purpose Cask (Docket No. 71-9255)

The information below was obtained from Sections 2.10.11 and 2.10.12 in Transnuclear's NUHOMS®-MP187 Multi-Purpose Cask SAR (see ADAMS Accession No. ML063520505) for the dynamic and static impact limiter tests, respectively.

A. Tests performed

1. Pre-operational Tests

Prior to performing the certification test program a series of five pre-operational drop tests were performed, using two quarter-scale models. The pre-operational drops conducted with these limiters were:

- Test 1: 30 ft.-0 in. drop with package axis at 30 degrees to horizontal, impact onto small diagonal facet with slap down onto opposite limiter;
- Test 2: 30 ft.-0 in. drop with package axis at 30 degrees to horizontal, impact onto large flat facet with slap down onto opposite limiter;
- Test 3: 30 ft.-0 in. flat side drop with impact on large flat sides;
- Test 4: 30 ft.-0 in. vertical drop; and

- Test 5: 30 ft.-0 in. flat side drop with impact onto diagonal corners of the limiters to impart the maximum rotational energy to the test article.

2. Dynamic Impact Limiter Tests

   The dynamic tests were performed using two quarter-scale prototypical impact limiters, in multiple uses, in different drop orientations. The test sequence was as follows:

   - A 30-foot drop with the package longitudinal axis inclined at 30 degrees to the horizontal and the 90 degrees azimuth at the top. Primary impact was on the large flat facet at 270 degrees azimuth on impact limiter "a" with a secondary impact (slap down) onto impact limiter "b".

   - A 30-foot drop with the package's center of gravity over a corner and the package's axis inclined at 72 degrees to the horizontal and rotated so that impact will occur on a diagonal corner of impact limiter "a."

   - A 40-inch drop onto a mild steel quarter-scale, cylindrical puncture spike. The spike had a diameter of 1.50 inches, a top edge radius of 0.06 inches, and a projected length of 16.0 inches, or 28.0 inches above the surface of the drop pad. The package was oriented at 45 degrees to the vertical on limiter "b." The 40 inch distance was measured from the point of impact on the package to the top of the puncture spike.

   - The end puncture event was performed on impact limiter "a" using a quarter-scale, cylindrical puncture spike similar to that described for the immediately above.

3. Static Impact Limiter Test Information

   The static test program was developed using two quarter-scale model impact limiters for multiple tests in different loading orientations. The total crush deflections imposed for each test were larger than the maximum predicted deflection expected to occur under the bounding loading conditions. The static tests are numbered sequentially from test 5 to 9. Details of the test program, starting at Test Number 5 are:

   5. A static test with the load applied to the 45 degrees azimuth flat surface of the honeycomb portion of the limiter.

   6. A static test, on the same impact limiter, with the load applied to the 180 degrees azimuth large flat honeycomb facet.

   7. With a new impact limiter, the static load was applied at 30 degrees to the package longitudinal axis to the 180 degrees azimuth large flat side of the limiter.

8. A static load was applied at 30 degrees to the package longitudinal axis through the 40 degrees azimuth small corner facet.

9. A third static test was performed on the second impact limiter. For this test the load was applied at 60 degrees to the package longitudinal axis to the 325 degrees azimuth diagonal corner of the impact limiter.

B. Description of the methods or analyses used in tests

1. Pre-operational Tests

The pre-operational drop tests were performed to demonstrate that the equipment and digital acquisition system were working correctly. The pre-operational test models were identical to the certification limiters with the exception of the honeycomb material. For the pre-operational tests the honeycomb was replaced with foam of similar density and crush strength; all other details were identical.

2. Dynamic Impact Limiter Tests

The primary method of determining package deformations was high speed film. High-speed cameras provided 400 and 2,000 frames per second coverage during all four dynamic tests. The cameras were placed parallel and perpendicular to the package impact zone to provide optimum viewing for measurement of the dynamic deformation in the impact limiters. Photometric coverage also included real-time color video and still shots. Painted backboards 12 ft. wide by 8 ft high, with a 12-inch lined grid, were placed behind the drop pad to provide a contrasting background to assist in measurement of the dynamic deformation of the limiters from the film. Following film processing the total impact limiter deformations were determined using a film comparitor with the capability of resolving displacements to 1/10th of an inch.

As a backup to the film, eight accelerometers were mounted to the package to measure rigid body accelerations. The accelerometers were installed at the package center of gravity and at each end adjacent to the impact limiters, close to the location of the "critical" spacer discs for the full scale package.

3. Static Impact Limiter Test Information

The instrumentation used for these tests consisted of displacement and load transducers. The deflection of the specimen was measured with two wire potentiometers (wirepots), one on each side of the special loading plates installed under the universal testing machine head. The average of these two measurements was used to characterize the deflection corresponding to the applied load. Another displacement transducer was used to measure the deflection of the lower part of the impact limiter. In addition, the displacement of the universal testing machine head (stroke), measured via an internal transducer, was also recorded for data verification purposes. The applied load was measured by a pressure transducer installed inside the control panel of the universal testing machine. All instruments were calibrated prior to the testing program.

C. Test Results

1. Pre-operational Tests

Test 1: For the first test, the two impact limiters impacted at different times. The impact forces calculated for the first impact limiter were 57 g in the longitudinal direction and 187 g in the transverse direction. The impact forces measured for the first impact limiter were 66 g and 63 g in the longitudinal direction and 185 g and 90 g in the transverse direction. The impact forces calculated for the second impact limiter were 18 g in the longitudinal direction and 203 g in the transverse direction. The longitudinal direction impact forces measured for the second impact limiter were 29 g and 23 g. The transverse direction impact forces measured for the second impact limiter were 335 g and 225 g. These transverse direction impact forces, when adjusted by the dynamic load factor associated with the rigid body effect of the dummy package, which is discussed below, equal 210 g and 141 g.

Test 2: For the CG-over-corner drop test, the calculated longitudinal and transverse forces were 110 g and 40 g respectively. The measured longitudinal forces were 110 g. while the measured transverse forces were 52 g and 30 g.

Test 3: The 45 degrees drop puncture spike cleanly punched through the stainless steel impact limiter shell and foam until it impacted the reinforcing ring at the inner shell. The internal reinforcing ring deflected the pin to follow the inner cylindrical shell; no puncture of the inner shell occurred and the pin was stopped 1.5 inches short of the package attachment ring by a highly compacted wedge of honeycomb. The total penetration into the limiter was approximately 11.5 inches. With the vertical motion of the test article stopped by the limiter, the package toppled, and bent the pin through a 45 degrees angle without tearing any of the material from the limiter. After the opposite limiter contacted the ground, the test article rolled sideways and the pin was then bent approximately 90 degrees to the initial bend without any evidence of any material tearing from the limiter. Internal damage to the limiter was restricted to broken foam and crushed honeycomb material at the entry point. The maximum gap was 3 inches at the foam/honeycomb interface where the initial 45 degrees rotation of the pin occurred; there were no other damaged areas that have any impact on the integrity of the limiter.

Test 4: The puncture pin penetrated 13.0 inches to the inner steel shell while being deflected sideways approximately 4 inches to impact on the reinforcing ring at the outer edge of the inner shell. The slow-motion film showed that the test article stopped with the pin completely embedded in the limiter and then slowly toppled, bending the puncture pin at approximately 60 degrees. After the test article rotated about 60 degrees, the bottom edge of the impaled limiter contacted the test stand, the impact point on the bottom of the limiter was raised vertically, and the pin pulled up and out of the stand. Damage to the limiter was confined to the puncture entry point and the slightly curved path taken to the impact point on the inner shell. The stainless steel outer shell and foam had sufficient strength to totally restrain the pin and bend it through 60 degrees, plus extract it from the test stand.

Dummy Package Tests: Examination of both the pre-operational and certification acceleration test data showed a ringing signature at approximately 350 Hz for the flat side slap down drop. This response indicated that the test results included a significant non-rigid body response. Two additional drops of the dummy package without impact limiters were conducted to determine the natural response of the bare package. The package was instrumented the same as the certification tests and dropped from heights of 12 and 40 inches. These drops both measured a natural frequency at the same frequency as the ringing tone (approximately 350 Hz). Additional analyses were conducted to include the response of the dummy package and define the rigid body response.

A dynamic load factor was determined for a range of frequencies for a damping ratio of 6%. The damping ratio was calculated as the logarithmic decrement from the filtered drop acceleration time histories. The predicted rigid body response for the flat side slap down drop was used as the forcing function for the dynamic load factor calculation. This analysis produced a predicted dynamic load factor of 1.6. Applying the calculated dynamic load factor of 1.6 to the measured peak accelerations produced the rigid body response.

The pre-operational tests clearly demonstrated that the impact limiter attachment bolts, and package attachments, were robust and capable of sustaining the design basis drop loads and keep the limiters securely attached to the package for multiple design basis drops.

2. Dynamic Impact Limiter Tests

Test 5: The limit event for this test was a deflection of 5.5 inches. The specimen was loaded to a peak load of 287 kips, which produced a deflection of 5.61 inches. The deflection after removing the load was 4.69 inches.

Test 6: Test 6 occurred immediately following test 5. The limit event was defined to be a deflection of 5 inches. Loading was interrupted when the deflection was 3.86 inches (386 kips of load) because the loading fixture was too close to the test fixture. The displacement instrumentation was disconnected, and the position of the loading fixture under the universal testing machine head was moved to apply further deformation to the specimen. The displacement transducers were reconnected to the specimen, and the load reapplied.

Testing was stopped when the load was 603 kips and the displacement 4.81 inches because the loading fixture was again close to the test fixture. Local buckling and yielding were also observed in the test fixture beams supporting the triangular component of the test fixture. The localized failure of the test fixture permitted a small rotation of the test fixture and imparted a small deflection component to the measured results. As the honeycomb material had been crushed to a solid matrix, this deflection is not considered to be important. The local buckling failure of the test fixture is not loaded in subsequent tests and did not require repair before continuing the loading sequence. After removing the load, the permanent deflection of the specimen was (60% crush) 3.81 inches.

Test 7: The defined limit displacement was 4 inches. Loading was stopped when the deflection was 3.85 inches because the loading plate was too close to the test fixture. The maximum load applied to the specimen of 484 kips occurred at a displacement of 3.52 inches. A permanent deflection of 2.79 inches was measured after unloading the specimen. The wire of the potentiometer used to measure bottom displacement snapped at a displacement of 1 inch and 322 kips of load.

Test 8: The specimen was loaded to 339 kips, with a corresponding displacement of 5.67 inches. Following a visual examination of the impact limiter, it was decided to apply further loading. The peak load measured during the test was 373 kips with a corresponding deflection of 6.25 inches. The peak displacement measured was 6.44 inches, after the load had dropped to 353 kips. The permanent deflection after unloading was 5.15 inches.

Test 9: The limit event was a displacement of 9.6 inches. This displacement was achieved with a load of 484 kips. It was then decided to load the specimen further. The peak load achieved was 700 kips at a deflection of 10.74 inches. The peak deflection was 10.90 inches, measured after the test was paused, and the load had dropped to 648 kips. The permanent deflection after unloading was 9.06 inches.

7. Model No. HI-STAR 100 System (Docket No. 71-9261)

The information below was obtained from Appendix 2.A of the SAR for the HI-STAR 100 Cask System, Revision 15, dated October 11, 2010 (see ADAMS Accession No. ML102871079).

A. Tests performed

A series of static compression tests were performed on eighth-scale models of the impact limiter for the HI-STAR 100 package. These crush tests were performed to document the force-deflection and energy absorption characteristics of the honeycomb material used in the impact limiter. The tests were conducted at temperatures ranging from -30 °F to 120 °F with impact limiter orientations of 0 degrees (side), 30 and 60 degrees (oblique) and 90 degrees (end). The purpose of this test was to confirm the static force-crush prediction model, determine the effect of temperature on the impact limiter and access and confirm the performance of the impact limiter backbone.

A series of free drop tests were performed on quarter-scale models of the HI-STAR 100 package instrumented with five accelerometers. Thirty-foot drop test orientations included a vertical top end drop, top corner drop at an angle of 67.5 degrees from horizontal (center of gravity over corner), side drop, and a slap-down drop at an angle of 15 degrees from horizontal so that the top impact limiter hits the surface first followed by a higher velocity impact of the bottom impact limiter.

The purpose of these tests were to confirm the predictions of the finite element software LS-DYNA used to create a full scale model of the HI-STAR 100 package and the AL-STAR impact limiter, specifically, the deceleration of the package, impact duration, maximum crush depth of the impact limiter and performance of the attachment system. Following the confirmation of the model, three more drop tests were performed at 30, 45 and 60 degrees with content weight varying from 270,000 lbs. to 280,000 lbs. (range of weight allowed in the package) to confirm the impact orientation of maximum damage.

B. Description of the methods or analyses used in the drop tests

The package was instrumented with five accelerometers to determine maximum deceleration of the package and record the duration of the impact for the scale model. Three accelerometers were attached at three axial locations. The other two accelerometers were attached 120 degrees from the first three, and aligned with the top and bottom accelerometer.

C. Test Results

Data obtained from the tests consists of both qualitative information with respect to observations about the package and the limiter and quantitative data obtained from the recorders. The data for each test include measured impact limiter deformation, deceleration time history, and observations of the package and attachments.

1. Impact Limiter Force-Deflection Tests

The static compressions tests were performed on eighth-scale model impact limiters used in drop testing the quarter-scale model. Of the four compression testes performed, two (0 and 30 degrees) were unsatisfactory, because the backbone did not remain elastic. The other two orientations (60 and 90 degrees) showed close agreement of the numerical model. As a result, the backbone was redesigned, and three additional eighth-scale model tests were performed at 0 degrees (side), 67.5 degrees (center of gravity over corner) and 90 degrees (top end). The backbone remained elastic and the force-deflection results showed close agreement with those predicted by the numerical model. Additionally, the tests indicated that temperature (within the range tested and the regulatory requirements) has no effect on the performance of the impact limiter.

Following the quarter-scale model testing, three additional compression tests were performed on three eighth-scale models of the impact limiter. The orientations were 0 degrees (side) 67.5 degrees (center of gravity over corner) and 90 degrees (top end). Good agreement was observed between the theory and the test for the side and center of gravity over corner crush orientations. For the end drop, the test results suggested that there may be elastic behavior at the interface of the package and the impact limiter that the model was not capturing. The dynamic test results (described below) demonstrated that the predication of the peak deceleration, extent of crush and impact duration were not affected by these elastic behavior effects.

2. First Series of Drop Tests

The orientations used for the first series of drop tests included the top end, center of gravity over corner and the side drop. The peak deceleration of the top end drop was well above the design basis of 60g. The reasons for the discrepancy between the test and the model were determined to be the use of a low value for a dynamic multiplier assumed in the impact limiter design and the lack of pre-crush of the honeycomb material in the impact limiter. As a result, the impact limiter was revised with new crush strengths and redesigned and a new set of quarter-scale model impact limiters was manufactured.

3. Second Series of Drop Tests

The same orientations were used in the first phase of the second series of drop tests. In all cases, the deceleration values were within the design limit of 60g; however, the attachment system did not survive the side impact drop test. The attachment system was redesigned prior to the slap-down test, which is considered to be the most definitive test of the package/impact limiter attachment integrity. The bottom impact limiter remained in place following the slap-down test and the deceleration was within the design basis.

Table 2.A.3 and 2.A.4 of the SAR shows the comparison of the test date for the seven drop tests with the prediction of the LS-DYNA software model. The predicted deceleration values, impact duration and total crush depth compared well with the test data. In three cases, the test data exceeded the predicted value of the model. In these cases, the measured total crush depth was greater than the predicted crush depth, but in all cases, the actual total crush depth was less than 80% of the available depth. This means there was several inches of impact limiter available to crush before package would impact the surface.

The results of the compression tests and the drop tests confirmed the accuracy of the numerical model, which was used to simulate the one-foot drop required for the assessment of normal conditions of transport.

The simulation produced maximum decelerations less than the design basis.

8. Model No. UMS Universal Transport Cask Package (Docket No. 71-9270)

The information below was obtained from Section 2.10.3 of the SAR on the Model No. UMS Universal Transport Package (UMS), Revision 99A, dated June 1999 (see ADAMS Accession No. ML063480388).

A. Tests performed

A series of free drop tests were performed on a quarter-scale model of the UMS package. Thirty-foot drop test orientations included a vertical top end drop, center of gravity over top corner drop, and a side drop. The test data consisted of measurements of the deformations of the impact limiter, the package accelerations, and inspection of the retaining rods.

Two static crush tests were performed on quarter-scale models of the impact limiter for the UMS package. These crush tests were performed to confirm the design of the UMS impact limiters, specifically to identify any potential initial stiffness in the impact limiter and to indicate any effect of the thick walled screw tubes on the impact limiter crush force in the top end drop orientation. Additionally, the crush data were used to confirm the validity of the RBCUBED computer program analysis of the UMS impact limiter.

B. Description of the methods or analyses used in tests

The package was instrumented with 3 accelerometers for the top end drop and the top corner drop and 4 accelerometers for the side drop to determine the maximum deceleration and impact duration of each drop. Additionally, two high-speed cameras were used to record the behavior of the model as it impacted the target surface.

C. Test Results

Data obtained from the tests consist of both qualitative information with respect to observations about the package and the limiter and quantitative data obtained from the recorders. The data for each test include measured impact limiter deformation, acceleration time history and assessment of the angle of drop based on the high-speed camera.

1. Top End Drop

For the top end drop the impact limiter was sawn into quarters and the average crush was measured at 2.04 inches, which scaled up to 8.16 inches for a full scale package. During the removal of the impact limiter, which required two hydraulic jacks, it was observed that several of the retaining fords had broken due to plastic buckling. Based on the accelerometer time history, the maximum deceleration was measured at 51.8g and the impact duration was 12 milliseconds. The measured acceleration was bounded by the design basis value of 60g and the crush depth was less than that required for the package to impact the impact surface.

2. Static Crush Test for the End Drop Orientation

   To further confirm the design of the UMS impact Limiters, two static crush tests in a top end drop orientation were conducted to identify any potential initial stiffness in the impact limiter and to indicate any effect of the thick walled screw tubes on the impact limiter crush force. The results of the static crush test confirmed the response of the impact limiter in the top end drop impact, and confirmed the validity of the RBCUBED program analysis

3. Side Drop

   Following the side drop, it was observed that the three retaining rods nearest the impact had broken, but that the remaining 13 of 16 retaining rods were intact and still threaded into the model body. The scaled-up crush depth was 11.8 inches and 11.4 inches for the top and bottom impact limiters respectively, which compared well with and were bounded by the RBCUBED predicted values of 13.0 and 13.7 inches.

   The peak accelerations for the top and bottom impact limiters were 51.1g and 37.9g respectively, which were also bounded by the RBCUBED predicted values of 52.1g and 48.6g. The measured accelerations were all bounded by the design basis value of 60g and the crush depth was less than that required for the package to impact the impact surface.

4. Center of Gravity over Top Corner Drop

   Like the side drop test, following the center of gravity over top corner drop, it was also observed that the three retaining rods nearest the impact had broken, but that the remaining 13 of 16 retaining rods were intact and still threaded into the model body. The scaled-up crush depth, 12.8 inches, was significantly less than the RBCUBED predicted value of 27.7 inches. The RBCUBED value bounded the actual crush depth. The acceleration time history from the accelerometers produced a maximum deceleration of 30g that was also significantly less than the RBCUBED predicted value of 47.8g, both of which are less than the design basis of 60g.

   The test confirmed that the UMS impact limiters are able to provide an adequate design margin to limit the deceleration and the crush depth of the transport package for drop test.

9. Model No. FuelSolutions™ TS125 Transportation Package (Docket No. 71-9276)

The information below was obtained from revision 6 of the FuelSolutions™ SAR dated September 2006 (see ADAMS Accession No. ML070230678).

A. Tests performed

1. Development Testing

A series of specimen bench tests and sub-model tests was performed to support the development of the impact limiter design. Specimen bench tests were performed to determine the characteristic behavior of the individual energy absorbing aluminum honeycomb materials used to fabricate the impact limiter. The aluminum honeycomb specimen tests examined the effects of crush rate, temperature, and impact orientation on the crush properties of the aluminum honeycomb materials. In addition, sub-model specimen bench tests were performed to address specific impact limiter design and loading issues.

Quasi-static crush tests of aluminum honeycomb material having nominal crush strengths of 1,200 psi and 2,500 psi were performed for a range of crush orientations and temperatures. Quasi-static crush tests were conducted along each of the principal axes for both 1,200 psi and 2,500 psi material at room temperature. Additional quasi-static crush tests were performed at temperatures of –20 °F and 220 °F. Three separate specimens were tested for each condition.

2. Confirmatory Static Crush Testing

Quasi-static crush tests, using scaled impact limiter test articles, were performed to confirm the adequacy of the inputs used in the analytical tools and methodology used to calculate the impact limiter force-deflection relationships. These tests address the effects of impact limiter geometry and construction and the effects of backing on the force-displacement relationship for various impact orientations. In addition, the quasi-static crush tests confirm the adequacy of the attachments used to secure the impact limiters to the transportation package. The orientations for the quasi-static crush tests include the end, center of gravity over corner, and side crush orientations. All static crush tests were performed using either an eight-scale or quarter-scale replicas of the full-scale impact limiter design.

3. Confirmatory Drop Testing

The 9-meter (30-foot) free drop tests were performed using quarter-scale impact limiter test articles to confirm the adequacy of the analytical tools and methodology used to calculate the rigid-body response of the transportation package for the free drop conditions specified in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 71. The dynamic drop tests address the dynamic response of the impact limiter energy-absorbing materials for various impact orientations. In addition, the dynamic drop tests confirm the structural adequacy of the impact limiter shell assembly and the hardware used to attach the impact limiter to the ends of the transportation package. Four free drop orientations have been tested, including:

- **End Drop** – A free drop with the package longitudinal axis orientated perpendicular to the target (i.e., vertical).

- **Corner Drop** – A free drop with the package longitudinal axis oriented at an angle of 21 degrees with respect to vertical. This orientation places the package center of gravity directly over the center of the crush force.

- **Side Drop** – A free drop with the package longitudinal axis oriented parallel to the target (i.e., horizontal).

- **Slapdown** – A free drop with the package longitudinal axis initially oriented at an angle of 75 degrees from vertical (i.e., 15 degrees from horizontal). This free drop condition results in a primary impact and as subsequent secondary impact (i.e., slapdown).

B. Description of the methods or analyses used in tests

1. Development Testing

The tests were performed using blocks having a 4.00-inch by 4.00-inch cross-section and a thickness of 2.00 inches in the direction of crushing. Pre-crushed test specimens were used for the static crush tests. The pre-crushed specimens were created by cutting the blocks 2.25 inches thick, and crushing one side 0.25 inch to attain the final 2.00-inch thickness. Each specimen had 0.020-inch thick 5052 aluminum alloy sheets bonded to each of the 4.00-inch x 4.00-inch faces to help reduce the possibility of sample splitting during the test. This resulted in a nominal specimen thickness of 2.04 inches. All quasi-static crush tests were conducted using a head speed of 0.2 inch per minute.

2. Confirmatory Static Crush Testing

The eighth-scale end crush test was performed on a 300-ton capacity manual static testing machine at an approximate head speed of 0.2 inch per minute unless noted otherwise. The force measurements were manually recorded from the dial gauge on the Tinius-Olsen machine, and the displacements were manually recorded from a patriot gauge with a digital displacement indicator.

3. Confirmatory Dynamic Drop Testing

Data collection and reduction was performed by Sandia National Laboratories using the Mobile Instrumentation Data Acquisition System. The 44-foot long MIDAS trailer, which was developed by Sandia National Laboratories using the Mobile Instrumentation Data Acquisition System for the U.S. Department of Energy to provide efficient and accurate test data acquisition and analysis for radioactive material packages, was located at the drop test site.

The test assembly was instrumented with a total of 18 accelerometers attached at various locations on the outer surface of the test fixture, providing sufficient redundancy. The accelerometers used provide accurate measurement of response frequencies up to 20 kHz and accelerations exceeding 1,000g.

C. Test results

1. Development Testing

The results of the quasi-static crush tests are summarized in the table below. The average crush strengths shown represent an average over the flat response section of the stress-strain curves (i.e., neglecting initial rise and densification). The test results show that the average crush strength for 1,200 psi and 2,500 psi material at –20 °F is approximately 7.5% higher than the average crush strength at room temperature. The test results also show that the average crush strength for 1,200 psi and 2,500 psi material at 220 °F is approximately 11% lower than the average crush strength at room temperature. As discussed in the Hexcel CROSS-CORE® product literature, the crush strength is generally 7% to 10% higher at –20 °F and 9% to 11% lower at 220 °F than the crush strength at room temperature. Therefore, the results of the static specimen crush tests confirm the information contained in the product literature.

Core Type Designation Number)	Nominal Crush Strength (psi)	Test Temp (°F)	Test Direction	Average Crush Strength (psi)	Average Densification Strain (%)	Average Densification Modulus (ksi)
AL-CC-3/16-052-18.0 A8A00)	1200	-20	T1	1324	65.9	23.6
		70	T1	1239	65.8	26.1
		70	T2	1180	66.4	21.9
		70	W	179	NA	NA
		220	T1	1106	67.8	25.6
AL-CC-1/8-5052-27.0 (A8A06)	2500	-20	T1	2677	59.1	31.6
		70	T1	2417	60.6	20.5
		70	T2	2297	60.0	25.3
		70	W	814	NA	NA
		220	T1	2169	61.8	28.1

2. Confirmatory Static Crush Testing

i. Quasi-Static End Crush Test Results

The maximum crush depth (crosshead displacement) achieved in the eighth-scale quasi-static end crush test was over 5 inches, which is equivalent to a full-scale displacement exceeding 40 inches. This crush depth conservatively exceeds the 14 to 15 inches of crush (approximately 1.8 inches for eighth-scale) required to absorb all of the kinetic energy for the worst-case hot end drop condition. Thus, the available stroke exceeds that needed to assure that bottoming out under the worst-case conditions does not occur.

The resulting end crush force-deflection curve shows excellent agreement with the calculated static end crush force deflection curve. The general magnitude and slope of the test curve and calculated curve for the end crush test are within the expected experimental accuracy over the full range of interest. The only difference observed between the test results and the calculation is the initial response of the impact limiter. The test results show a gradual ramp-up over the first 0.2 inch of crush, whereas the calculated response shows an instantaneous rise. The difference results from the calculation assumption of fully effective backing through the entire crush range, which neglects initial shifting of the honeycomb segments within the impact limiter shell assembly. However, the ramped-up response for the end drop orientation is accounted for by adding a take-up deflection to the calculated acceleration time-history curves for the quarter-scale confirmatory end drop test and the full-scale drop load analysis, although this effect is not significant on the resulting loads. Therefore, it is concluded that the analytical tools, assumptions, and inputs used to develop the static force-deflection curve for the end crush are adequate. The resulting end crush force-deflection curve shows excellent agreement with the calculated static end crush force deflection curve.

ii. Quasi-Static Corner Crush Test Results

The crush depth achieved in the corner crush test was approximately 6 inches, which bounds the anticipated stroke required of the design. The calculated corner crush force-deflection curve was determined using the dimensions and crush strengths of the eighth-scale impact limiter, with the same analytical tools and assumptions applied to the full-scale impact limiter corner drop loads evaluation. The calculated force-deflection curve neglects the contribution of the impact limiter shell assembly, and assumes that all material is effectively backed and contributes to the crush force. The resulting corner crush force-deflection curve shows excellent agreement with the calculated static corner crush force deflection curve. The general magnitude and slope of the test curve and calculated curve for the corner crush test are within the expected experimental accuracy over the full range of interest. Therefore, it is concluded that the analytical tools, assumptions, and inputs used to develop the static force-deflection curve for the corner crush are adequate. The resulting corner crush force-deflection curve shows excellent agreement with the calculated static corner crush force deflection curve.

iii. Quasi-Static Side Crush Test Results

A 600-kip capacity Tinius Olson hydraulic quasi-static testing machine with a 12-inch stroke and 1.2-inch per second maximum displacement rate was used for the test. The static side crush results show that the measured force-deflection curve exceeds the pre-test prediction force-deflection curve for deflections of approximately 2.2 inches and higher. Based on the results of the post-test evaluation using the modified input parameters, it is concluded that the analytical tools, inputs, and modified input parameters accurately predict the static force-deflection response for the static side crush condition.

3. Confirmatory Dynamic Drop Testing

i. 30-Foot End Drop Testing and Results

The post-test measurements of the impact limiter show that the average crush distance resulting from the end drop is 3.5 inches, compared with the pre-test prediction of 3.2 inches. Upon post-test inspection following the end drop test, there was no noticeable failure of the impact limiter shell or the impact limiter attachment hardware. Therefore, the results of the 30-foot end drop test confirm the analytical tools, inputs, and assumptions used to determine the rigid-body response of the transportation package for the HAC end drop.

ii. 30-Foot Side Drop Testing and Results

The post-test measurements of the impact limiters showed that the maximum crush depth resulting from the confirmatory side drop test was 3.3 inches, compared to a pre-test prediction of 3.2 inches. In general, the test results show excellent agreement with the pre-test prediction.

iii 30-Foot Corner Drop Testing and Results

In general, the test results show excellent agreement with the pre-test predictions. The measured test result exhibits the same general shape, pulse duration, and peak acceleration magnitude as the pre-test prediction. The post-test measurements of the impact limiter show that the crush distance resulting from the corner drop was approximately 6.4 to 6.8 inches, compared with the pre-test prediction range of 6.8 to 7.3 inches. Upon post-test inspection following the corner drop test, there was no noticeable failure of the impact limiter shell or the impact limiter attachment hardware. Therefore, the results of the 30-foot corner drop test confirm the analytical tools, inputs, and assumptions used to determine the rigid-body response of the transportation package for the hypothetical accident conditions corner drop.

iv. 30-Foot Slapdown Drop Testing and Results

In general, the test results show excellent agreement with the pre-test prediction for the primary impact and are slightly higher than the test prediction for the slapdown impact. The post-test measurements of the impact limiter show that the crush distance resulting from the slapdown drop primary impact is 2.9 inches, compared with the pre-test prediction of 4.6 inches. Similarly, the measured crush depth in the secondary impact limiter is approximately 3.0 inches, compared with the pre-test prediction of 3.6 inches. Upon post-test inspection following the slapdown drop test, there was no noticeable failure of the impact limiter shell. During the test, two of the twelve attachment studs located on the crushed side of the impact limiter on the secondary impact end failed. However, the impact limiter remained attached to the test fixture throughout the duration of the test and all other attachment studs remained intact. Thus, the impact limiter attachment studs performed their intended function during the quarter-scale slapdown drop test.

10. Model No. TN-68 Transport Package (Docket No. 71-9293)

The information below was obtained from Section 2.10.9 in Transnuclear Inc., application dated May 19, 1999 (see ADAMS Accession Nos. ML063340727 and ML063340703).

A. Tests performed

A series of dynamic tests were performed on one-third scale models of the TN-68 impact limiters. The tests were performed to evaluate the effect of the 30-foot free drop hypothetical accident defined 10 CFR 71.73(c)(1). The objectives of the test program were to:

- Demonstrate that the inertia G values and forces calculated for the TN-68 package and basket are acceptable.
- Verify the adequacy of the impact limiter tie rods and bolts.
- Demonstrate the adequacy of the impact limiter enclosure.

The drop configurations examined are the 15° slap down, 90° end drop, 0° side drop, and 90° end drop for puncture.

B. Description of the methods or analyses used in tests

Accelerometers (12 possible in total) were mounted to brackets around the exterior of the test body at 0°, 90°, 180°, and 270° orientations at the approximate center of gravity location and adjacent to each impact limiter. An inclinometer will be placed on the test body to measure the initial angle (± 1) of its longitudinal axis with respect to the target (i.e., impact surface). Data were collected by accelerometers capable of measuring data at a minimum frequency response of 6,000 Hz per channel.

C. Results

The results of the 15° slap down showed that the adequacy of the attachment design of the impact limiters, and that the impact limiters did not bottom out and the trunnions would not impact the target. The 90° end drop resulted in both impact limiters remaining intact and crush values comparing well to predicted values. The 0° side drop indicated good correlation between measured and predicted crush depths for the side drop event. 90° end drop for puncture resulted in impact limiters staying attached to the package. In all, measured and predicted performance agreed very well.

11. Model No. NUHOMS®-MP197, NUHOMS®-MP197HB (Docket No. 71-9302)

The information below was obtained from Section 2.10. 9, in Transnuclear Inc's., application dated May 2, 2001, pre-application meeting slides, and Appendix A.2.13.8.1 (see ADAMS Accession Nos. ML063190444, ML090420165, and ML112640444.

A. Tests performed

One-third-scale tests were performed for the MP197 package to evaluate the effect of the 9 m free drop hypothetical accident defined in 10 CFR 71.73(c)(1). The unyielding drop surface consisted of a 2-inch thick steel plate secured to the surface of a concrete pad. The test model was a solid steel third-scale mockup of the package body with impact limiter. The steel body was designed to scale the weight and the center of gravity of the package.

For the MP197 package, the objectives of the package impact limiter tests were to:

- demonstrate that the inertia g values and forces used in the analyses are conservative,
- demonstrate that the extent of the crush depths are acceptable (i.e., the neutron shield does not impact the target), and
- demonstrate the adequacy of the impact limiter enclosures.

The tests performed consisted of the following:

- A 0° side drop because this orientation generates the highest transverse acceleration as well as significant deformation. The 0° side drop also provides a reasonable estimate of the likelihood of the neutron shield impacting the target.
- A 20° slap down drop because the 20° orientation puts the highest load on the impact limiter attachment bolts, and stainless steel shell.
- The 90° end drop orientation was chosen because it causes the highest axial deceleration.
- The 40-inch drop onto a 1/3 scale 6-inch diameter puncture bar was performed in accordance with 10 CFR 71.73(c)(3).

Test Number	Drop Orientation	Drop Height	Impact Limiter Number	Location of impact limiter
1	0° side drop	30 ft	1 2	Top Bottom
2	20° slap down	30 ft	3 2	Top, 2nd impact Bottom, 1st impact
3	90° end drop	30 ft	3 4	Top Bottom-impact end
4	90° end drop	40 in	3 4	Top Bottom- puncture end

B. Description of the methods or analyses used in tests

An inclinometer was placed on the test body to measure the initial angle (±1°) of its axis with respect to the drop pad (impact surface). The impact surface was an unyielding horizontal surface, weighing more than 250,000 lb. versus 9,750 lb. for the weight of the 1/3 scale dummy.

Accelerometers were used to measure the inertial g load during impact for the three 30-foot drops performed. At least 10 accelerometers were used during each 30-foot drop.

Four, third-scale impact limiters were constructed for the drop testing.

The following data were measured and recorded before, during and after each drop test:

1. Prior to each drop test

- Torque of the impact limiter bolts,
- Impact limiter dimensions,
- Height from test article to drop pad,
- Angular orientation of the test article to the impact surface, and
- Atmospheric condition data (i.e., ambient temperature, wind speed, immediately and prior to the release of the test article).

2. During each drop test

- Test article behavior on videotape,
- Date and time of test,
- Observations of damage or unexpected behavior of the test article, and
- Impact acceleration time histories and frequency responses (excluding the puncture drop test).

3. Following each drop test:

- Observations of the damage to the test article on features other than the limiters (i.e., attachment bolts).
- Measurements of deformation to each impact limiter to fully describe the extent of the damage, including:
  - Depth of internal and external crush of the impact limiter
  - Overall thickness of each impact limiter after each test.
  - Dimensions of impact footprint

C. Test Results

The four drop tests were performed without any unusual observations. The impact limiters contained the wood during the drop tests, and none of the attachment bolts failed.

No openings in the stainless steel impact limiter shell were evident and no welds in the shell failed.

The puncture bar sheared a circular section of the outer shell of the bottom impact limiter. No other sections of the impact limiter were damaged and no welds on the impact limiter shell were broken. The puncture bar did not penetrate the inner stainless steel shell of the impact limiter or the aluminum thermal shield. Both impact limiters remained attached to the package during the puncture drop event and no additional impact limiter attachment bolts were damaged.

The predicted performance of the impact limiters in terms of decelerations and crush depths agrees well with the measured data.

Results of the Third-scale benchmark LS-DYNA drop test vs. analysis

Test Conditions	Parameter	Drop Test Results	LS-DYNA Analysis Results
90° End Drop (-20 °F)	Acceleration	65g	65.1g
	Impact Duration	0.010 sec.	0.012 sec.
	Wood Crush Depth	2.5"	2.8"
0° Side Drop (Room Temperature)	Acceleration	61g	65.6g
	Impact Duration	0.012 sec.	0.013 sec.
	Wood Crush Depth	2.69"-2.75"	2.7"-2.9"
20° Slap Down 1st Impact (Room Temperature)	Acceleration at Center of Package	17g	20.8g
	Acceleration at Bottom of Package	36g	40.1g
	Impact Duration	0.016 sec.	0.018 sec.
	Wood Crush Depth Bottom Limiter	4.92"	4.9"
20° Slap Down 2nd Impact (Room Temperature)	Acceleration at Center of Package	32g	36.3g
	Acceleration at Top of Package	73g	72.2g
	Impact Duration	0.009 sec.	0.010 sec.
	Wood Crush Depth Upper Limiter	4.72"	2.8"

Calculated decelerations (maximum value and time duration) are close to or bound the measured drop test decelerations. It is therefore concluded that the methodology, material models, and material properties are properly benchmarked.

The results of the tests demonstrate that:

- The crush depths do not result in lockup of the wood in the impact limiters,
- The crush depths for the 0° side drop case would not result in the neutron shield impacting the target,
- The predicted performance of the impact limiters in terms of decelerations and crush depths agrees well with the measured data,
- The impact limiter enclosure is structurally adequate in that it successfully confines the wood inside the steel shell,

- The impact limiter attachment design is structurally adequate in that the attachment bolts hold the impact limiters on the ends of the package during all drop orientations, and
- The effects of low temperature (-20 °F) on the crush strength of the impact limiters is minor, and is bounded by the conservative accelerations and forces used in the analysis.

A 40-inch drop onto a scaled 6-inch diameter puncture bar, as per 10 CFR 71.73(c)-(3), does not significantly destroy the impact limiter. The impact limiter and attachment remain firmly secured to the package and the impact limiter wood is confined.

12. Model No. TN-40 (Docket No. 71-9313)

The information below was obtained from Section 2.10.9 in AREVA's application dated August 31, 2006 (see ADAMS Accession No. ML070750136).

A. Tests performed

Tests performed on a third-scale impact limiters and a dummy package included three 30' drops (side drop, slapdown and end drop) and one 40" drop onto a puncture pin. Accelerometers were placed on the dummy package.

B. Description of the methods or analyses used in tests

The test goals were:

1. validation of calculated acceleration values,

2. demonstration that the crush depths are acceptable,

3. demonstration of the adequacy of the impact limiter enclosure and attachment design,

4. evaluation of the effects of low temperature (-20 °F) on dynamic performance of the impact limiters, and

5. evaluation of the effects (puncture depth and shell damage) of a 40-inch drop onto a scaled 6-inch diameter puncture bar on a previously crushed impact limiter.

The side drop orientation was chosen to generate the highest transverse acceleration as well as a significant deformation.

The slap down orientation puts the highest load on the impact limiter attachment bolts, tie rods, and stainless steel shell.

The end drop orientation causes the highest axial acceleration, while the pin drop orientation was chosen because it assures that the puncture impact absorbs 100% of the drop energy.

The test package consisted of a third-scale model of the TN-40 transport package with impact limiters on each end. The impact limiters are attached to each other by thirteen, 0.5-inch-diameter tie rods, snug tight, and to the package with four, 0.5-inch bolts. The test package weighs approximately 10,100 lb. and has maximum dimensions of approximately 87.0 inches long by 48.0 inches in diameter.

Lifting and dropping the test article was accomplished using a mobile crane. A quick release mechanism was used to initiate the drop. It consisted of a hydraulic piston that loaded a bolt to failure releasing a shackle supporting the test article via a rigging system.

An inclinometer was used to measure the initial angle (± 1°) of the test body longitudinal axis with respect to the drop pad (i.e., impact surface). A measured line, 30 feet long (+ 3.0, -0.0 inches), was attached to the lowest point on the test package in order to assure the proper drop height.

The impact surface was a 2-inch thick steel plate attached to a concrete block weighing approximately 250,000 lb. resting on bedrock. This configuration can be considered as an essentially unyielding surface.

A puncture bar made of cold-rolled steel was welded to the impact surface for the 40 inch puncture drop. The pin was scaled to match the test article resulting in a 2-inch-diameter pin with the upper end edges rounded to a radius of approximately 0.083 inches.

Accelerometers were used to measure the impact *g* load for all drops performed.

Twelve PCB Piezotronics 353B18 accelerometers were attached to aluminum blocks that were bolted to the test body at 0°, 90°, 180°, and 270° orientations at three elevations.

Drop Test Sequence

Test Number	Drop Orientation	Drop Height	Impact Limiter Number	Impact Sequence	Comments
1	0° Side Drop	30 feet	1	-	Limiters 1 and 2 installed.
			2	-	
2	64° CG Over Corner drop	30 feet	1	1st	The 1 and 2 impact limiters were rotated 180° so the undamaged portion of the impact limiters face the pad.
			2	2nd	
3	0° Side Drop (2nd test)	30 feet	1	-	The test body and limiters were rotated 90° so that an undamaged portion of the impact limiters faced the pad.
			2	-	
4	20° Slap Down	30 feet	3	1st	Limiters 1 and 2 were removed and replaced with limiters 3 and 4.
			4	2nd	
5	90° End Drop	30 feet	3	1st	Limiter 3 was removed and chilled at -20 °F for 48 hours before being re-installed on the test body.
			4	-	
6	90° End Drop (Puncture Test)	40 inches	3	1st	Drop onto 2 inch diameter puncture bar.
			4	-	
7	90° End Drop (2nd test)	30 feet	2	1st	Limiters 2 and 4 were used. The center portion of limiter 2 was relatively undamaged by previous drops and thus provided a useable crush volume.
			4	-	

The following data were measured and recorded before, during and after each drop test listed in the table above.

1. Prior to each drop test

   - Torque of the impact limiter bolts.
   - Impact limiter dimensions.
   - Height from test article to drop pad.
   - Angular orientation of the test article to the impact surface.
   - Atmospheric condition data (i.e., ambient temperature, wind speed, immediately and prior to the release of the test article).

2. During each drop test

    - Test article behavior on videotape.
    - Date and time of test.
    - Observations of damage or unexpected behavior of the test article.
    - Impact acceleration time histories (excluding the puncture drop test).

3. Following each drop test

    - Observations of the damage to the test article on features other than the limiters (i.e., attachment bolts).
    - Measurements of deformation to each impact limiter to fully describe the extent of the damage include depth of internal and external crush of the impact limiter, overall thickness of each impact limiter after each test, and dimensions of impact footprint.

C. Test Results

The performance of the test program allowed to:

- Verify the impact limiters are not dislodged from the package as a result of the drop.
- Demonstrate the effectiveness of the impact limiter tie rods, attachment bolts, and stainless steel covers.
- Provide data on the deformation of the impact limiters due to the drop.
- Provide data on the acceleration experienced by the test package during impact.
- Provide data on the impact limiter damage caused by a 40-inch drop on a 2 in. diameter puncture bar.

As an example, the following table shows the maximum transverse accelerations measured by the accelerometers during the second 0° side drop (converted to full scale), as well as the maximum acceleration predicted by a computer program.

Accelerometer Location	Measured Transverse Acceleration (gs) (converted to full scale)	Average Measured Transverse Acceleration (gs)	Predicted Maximum Transverse Acceleration (gs) (Appendix 2.10.8 in SAR)
Top (2)	68	57	51
Center of Gravity	50		
Bottom (10)	55		

The following table summarizes the measured and predicted crush depths for the bottom impact limiter. A spring back of 0.50 inches is assumed (based on previous crush tests).

	Impact Limiter Number 1	Impact Limiter Number 2
Maximum Inside Crush Depth (in.)	1.44	1.50
Maximum Outside Crush Depth (in.)	0.75	0.75
Spring Back (in.)	0.50	0.50
Total Crush Depth (in.)	2.69	2.75
Predicted Crush Depth x 1/3 (in.)	4.52	

From the above table it can be seen that the measured crush depths are slightly less that those predicted by the computer program. It should also be noted that neither the neutron shield nor the trunnions would contact the impact surface during the impact. The distance between the outer diameter of the neutron shield and the outside diameter of the impact limiter is 7.16 in. Therefore, a clearance of 7.16 - 2.75 = 4.41 in. would remain between the impact surface and the neutron shield, based on the measured crush depth. Similarly, a distance of 3.84 in. would remain between a trunnion and the impact surface.

Both impact limiters remained attached to the package during and after the side drop impact. All of the tie rods and tie rod brackets remained intact, thus preventing separation of the impact limiters from the package. In addition, the impact limiter attachment bolts remained in place, in spite of damage to two of the eight bolting brackets. Only a single small opening in the stainless steel shell of each of the impact limiters was evident. Both openings consisted of a tear along the weld between two of the outer flat plates of the impact limiter. The tears were roughly 4 inches long. Despite these tears, all impact limiter wood remained completely confined within the shell.

The results of the tests demonstrate that:

- The loadings used in the basket and fuel rod cladding structural analyses bound the dynamic measured data.
- Based on the applied loading and factor of safety, the package can withstand much higher loads than those resulting from the dynamic measured data.
- The crush depths do not result in lockup of the wood in the limiters.
- The crush depths for all the drop cases would not result in the neutron shield or trunnions impacting the target.
- The impact limiter enclosure is structurally adequate in that it successfully confines the wood inside the steel shell.
- The impact limiter attachment design is structurally adequate in that the impact limiters remain on the ends of the test dummy during and after all drop orientations.
- The effect of low temperature (-20 °F) on the impact limiter wood is not available due to lost test data. However, based on a similar design (TN-

68) chilling the impact limiter wood (-20 °F) will increase the g load roughly by 15% to 20%.

- An increase of 20% in the accelerations for both axial and transverse directions is acceptable based on applied loading and resulting factors of safety shown in the analyses.
- A 40-inch drop onto a scaled 6-inch-diameter puncture bar, as required by 10 CFR 71.73(c)-(3), does not significantly damage the impact limiter, nor are there any indications of damage to the test dummy.
- The impact limiters remain firmly secured to the test dummy, and the impact limiter wood is confined.

13. Model No. HI-STAR 180 (Docket No. 71-9325)

The information below was obtained from Section 2.7 in the Holtec International consolidated application dated May 29, 2009 (see ADAMS Accession No. ML14114A178).

A. Tests performed

No tests were specifically performed for the HI-STAR 180. For casks in this list that note "no tests," 10 CFR 71.41 permits the use of analyses to demonstrate compliance with 10 CFR 71.73 test requirements.

B. Description of the methods or analyses used in tests

The applicant derived an LS-DYNA model of the HI-STAR 180 impact limiters consistent with the previously HI-STAR 100 benchmarked model. The HI-STAR 100 benchmarked analysis was revised only to properly account for the compressive and shear strength properties of the honeycomb material in three directions.

The structural qualification of the HI-STAR 180 package relies only on transient LS-DYNA analyses and static ANSYS analyses. LS-DYNA is used to predict peak rigid body decelerations and impact limiter crush behavior. ANSYS is used to determine stress/strain levels in the package components by applying peak decelerations from LS-DYNA. Conservative upper and lower bound strength properties were analyzed where physical test data was not available for a particular crush/shear direction.

C. Test Results

No tests were performed for the HI-STAR 180.

14. Model No. HI-STAR 60 (Docket No. 71-9336)

The information below was obtained from Section 2.7 in the Holtec International consolidated application dated May 29, 2009 (see ADAMS Accession No. ML091540454).

A. Tests performed

No tests were performed specifically for the HI-STAR 60.

B. Description of the methods or analyses used in tests

The certification basis for the HI-STAR 60 was predicated on analysis only and relied on HI-STAR 100 test data solely to the analytical modeling capabilities with respect to rigid body dynamics. This preliminary analysis was designated as a benchmark study and consisted of simulation of HI-STAR 100 quarter-scale drop tests conducted by the applicant on the HI-STAR 100. The benchmarking of the HI-STAR 100 drop test provides additional assurance that the deceleration and gross deformation results obtained from the HI-STAR 60 evaluation are reasonably accurate and conservative.

C. Test Results

No tests were performed specifically for the HI-STAR 60.

15. Model No. BEA Research Reactor (BRR) Package (Docket No. 71-9341)

The information below was obtained from Section 2.12.2 and 2.1 in AREVA Federal Services, LLC, consolidated application dated May 29, 2009 (see ADAMS Accession No. ML112640462).

A. Tests performed

A series of free drop tests were performed on a half-scale model of the BRR package. The SAR presents test results, including time-history deceleration package body response traces and photograph records of deformations and damaged attachments of the impact limiters, for the initial series of three, 30-ft free drop tests and five puncture drop tests. The specific drop tests were:

- 30-foot end drop
- 40-inch puncture test oblique drop onto thicker end plate
- 30-foot slapdown drop, with the longitudinal axis 15 degrees from vertical
- 30-foot drop with CG-over-corner
- 40-inch puncture where the puncture bar strikes the inside edge of the slapdown primary-end damage from 2nd drop test, above,
- 40-inch puncture where the puncture bar impacts the end of the damage impact limiter from 2nd drop test, above,
- 40-inch puncture test with CG-over-corner drop, with impact on the thinner conical shell material, and
- 40-inch puncture test with the puncture bar striking the center of the slapdown secondary damage.

B. Description of the methods or analyses used in tests

The test articles were chilled generally between -10 °F and -20 °F. The impact limiter performance is demonstrated by tests of the half-scale, prototypical units and a dummy package body.

The primary means of recording the results of the tests was physical measurements and observations of the test packages (including impact limiters) before and after testing. Each free drop impact was recorded using accelerometers.

C. Test Results

Data obtained from the tests consist of both qualitative information with respect to observations about the package and the limiter and quantitative data obtained from the recorders. The data for each test include measured impact limiter deformation, strain gauge data and stress calculations (for the end drop and side drop only), and observations of the package and attachments. The strain gauge data were only presented for the end drop and the side drop since the loads developed in those tests are the most severe from an overall structural consideration. The end drop corresponds to the maximum axial loading condition, while the side drop developed the maximum lateral loading on the overall package body.

Five puncture drops were performed on the half-scale certification test unit to demonstrate structural adequacy of the package. The tests showed that the puncture bar would neither penetrate beyond the impact limiter shell located on the flat bottom nor create a significant exposure of foam adjacent to the package to the containment seal. The impact limiters would remain attached to the package ends. Also shown in one of the tests was that the puncture bar would not enter the impact limiter through a side impact on the limiter shell and rip open a large area. For the package thermal evaluation, the tests provided bounding impact limiter damaged configurations for fire event modeling consideration.

The tests demonstrated that the impact limiters were capable of limiting the package body deceleration to the design basis of 120 g applicable to all package drop orientations. The application notes that, although the impact limiters were damaged with some exposure of the foam, they remained attached to the package body. Also noted is that the damaged configuration is included in the thermal model for the HAC fire event analysis.

Therefore, the value of 120 g was utilized as the bounding value to calculate and evaluate package body structural performance for all drop orientations for all the tests.

16. Model No. TN-LC, (Docket No. 71-9358)

The information below was obtained from Sections 2.6 and 2.7 in AREVA Inc., consolidated application dated November 30, 2012 (see ADAMS Accession Nos. ML12340A309 and ML12340A310).

A. Tests performed

No tests were performed specifically for the TN-LC.

B. Description of the methods or analyses used in tests

The applicant demonstrates the structural capabilities of the package by analyses using ANSYS code. A third-scale model drop testing of a structurally similar packaging is used (Model No. NUHOMS MP197 transportation package), to benchmark the impact limiter finite element analysis model, which was subsequently adapted, for determining bounding deceleration g-loads for the package structural evaluation by analysis.

C. Test Results

No tests were performed specifically for the TN-LC.

17. HI-STAR 180D (Docket No. 71-9367)

The information below was obtained from the NRC staff's safety evaluation report dated August 5, 2015 (see ADAMS Package Accession No. ML14211A015), that used Holtec International's SAR dated July 16, 2014 (see ADAMS Package Accession No. ML14203A285), as its basis.

A. Tests performed

There were no tests performed specifically for the HI-STAR 180D.

B. Description of the methods or analyses used in tests

The licensing basis for the Model No. HI-STAR 180D package structural performance is predicated on analytical modeling rather than experimental testing. The LS-DYNA modeling approach had been determined to be adequately benchmarked for certifying the Model No. HI-STAR 100 package in computing cask rigid body decelerations for the free-drop accident conditions.

C. Test Results

There were no tests performed specifically for the HI-STAR 180D.

Senator INHOFE. Thank you very much, Chairman Burns.
Commissioner Svinicki.

STATEMENT OF KRISTINE SVINICKI, COMMISSIONER, U.S. NUCLEAR REGULATORY COMMISSION

Ms. SVINICKI. Thank you, Chairman Inhofe, Ranking Member Boxer, and distinguished members of the committee, for the opportunity to appear before you today at this hearing to examine policy and management issues pertaining to the NRC.

The Commission's Chairman, Stephen Burns, in his statement on behalf of the Commission, has provided an overview of the agency's current activities, as well as a description of some key agency accomplishments and challenges in carrying out the NRC's work of protecting public health and safety, and promoting the common defense and security of our Nation.

The NRC continues to implement safety significant lessons learned from the Fukushima accident in accordance with agency processes and procedures while also maintaining our focus on ensuring the safe and secure operation of nuclear facilities and use of nuclear materials across the country. Concurrent with this, the NRC is undertaking a comprehensive reevaluation of our agency's structure and processes under the Project Aim initiative. This initiative has engaged—and continues to solicit the input of—all agency employees, as well as interested stakeholder groups.

I appreciate the opportunity to appear before you today and look forward to your questions. Thank you.

[Ms. Svinicki's responses to questions for the record follow:]

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

COMMISSIONER

December 21, 2015

The Honorable James M. Inhofe
Chairman, Committee on
Environment and Public Works
United States Senate
Washington, D.C. 20510

Dear Chairman Inhofe:

I appeared before the Committee on Environment and Public Works on October 7, 2015, at a hearing entitled, "Oversight of the Nuclear Regulatory Commission," along with my colleagues on the Commission. In response to the Committee's letter of November 2, 2015, enclosed please find my response to a question for the record, directed to me, from that hearing. In addition, I have enclosed my individual response to a question for the record posed by Senator Capito to NRC Chairman Steven Burns, for his response on behalf of the Commission (specifically, Question 9, subparts d. and e.), where my views differ from those transmitted to you by Chairman Burns.

If I can be of further assistance, please do not hesitate to contact me.

Sincerely,

Kristine L. Svinicki

Enclosure: As stated

Identical letter sent to the Honorable Barbara Boxer.

The Honorable Edward Markey

QUESTION 1. **In 2001, in response to the terrorist attacks of 9/11, the NRC issued cyber-security Orders for nuclear reactors, which were turned into regulations in 2009. A report was recently released that concluded that nuclear power plants are in a "culture of denial" over hacking risks worldwide. The report pointed out the rapidly evolving and increasing sophistication of the cyber-threat. Would you be willing to require a fresh look at these old regulations to determine whether updates are needed to address new threats or vulnerabilities? If not, why not?**

RESPONSE.
The Chatham House report to which your question refers addresses the nuclear industry globally and is not focused on NRC's cybersecurity requirements for U.S. nuclear power plants. NRC's current regulations under 10 CFR Part 73 require licensees to monitor their programs and cyber plans to ensure that they take into account system changes, operating experience, and evolving changes to the cyber threat landscape. These performance-based requirements are intended to make sure that licensees have dynamic cybersecurity programs that respond to new threats and vulnerabilities. The NRC's security analysts, working with their counterparts throughout the U.S. government, also monitor current and emerging threat vectors and keep the Commission currently informed. These ongoing processes render it unnecessary, at present, to revise the NRC's cybersecurity regulations.

The Honorable Shelley Moore Capito

QUESTION 9. **On June 11, 2014, the NRC Inspection Manual Chapter [308 Attachment 3][1], Appendix M, *"Technical Basis for the Significance Determination Process SDP Using Qualitative Criteria,"* was revised to: "...provide a technical basis for using qualitative criteria in determining the safety significance of an inspection finding."**

d. Does the revision incorporate additional subjectivity into the SDP and the Reactor Oversight Process?

e. Should the Commission's direction in response to SECY 14-0087 apply to modifications of the Reactor Oversight Process?

Commissioner Svinicki's Individual Response.

d. Yes, additional subjectivity has been introduced with the issuance of IMC 0308, Attachment 3, Appendix M because it provides a more qualitative approach to assess the safety significance of findings.

e. Yes. The Commission's direction in its Staff Requirements Memorandum for SECY 14-0087 states that "[t]he appropriate degree of weight of application of qualitative factors in regulatory decision making ultimately lies with the Commission." Significance determination is a fundamental component of regulatory decision making in the oversight process and the documents discussed in this question meet this criterion.

[1] The title of the referenced document is corrected by the brackets.

Senator INHOFE. Thank you, Commissioner Svinicki.

Commissioner Ostendorff.

STATEMENT OF WILLIAM OSTENDORFF, COMMISSIONER, U.S. NUCLEAR REGULATORY COMMISSION

Mr. OSTENDORFF. Chairman Inhofe, Ranking Member Boxer, and distinguished members of the committee, thank you for the opportunity to be here today.

I am in complete alignment with the chairman's testimony. I will expand very briefly on two topics: post-Fukushima safety and Project Aim.

The Commission recently approved what I consider to be the capstone of our response to Fukushima, the Mitigation of the Beyond Design Basis Event rulemaking. This rulemaking codifies significant enhancements for station blackout, spent fuel pool safety, onsite emergency preparedness responsibilities, and other command and control aspects.

I look at Senator Carper and note an exchange we had in this committee hearing 4 years ago on the half-dozen, and I believe that this rulemaking codifies the bulk of that half-dozen we exchange comments on in 2011.

Seeing a light at the end of the tunnel, the Commission also directed staff to provide a plan and schedule for resolving all remaining Fukushima action items. That is due to us the end of this month.

Project Aim is a real opportunity for this agency to take a fresh look at how we operate and see where we can improve our efficiency and effectiveness in executing our mission. This fresh look has new faces leading the change. As Senator Inhofe mentioned, we have Victor McCree now leading as the Executive Director for Operations. We also announced a number of other significant management changes. I have the utmost confidence in these leaders.

In closing, I appreciate the opportunity to be here today and look forward to your questions.

[Mr. Ostendorff's response to a question for the record follows:]

Response to Senator Markey

Question:

In 2001, in response to the terrorist attacks of 9/11, the NRC issued cyber-security Orders for nuclear reactors, which were turned into regulations in 2009. A report was recently released that concluded that nuclear power plants are in a "culture of denial" over hacking risks worldwide. The report pointed out the rapidly evolving and increasing sophistication of the cyber-threat. Would you be willing to require a fresh look at these old regulations to determine whether updates are needed to address new threats or vulnerabilities? If not, why not?

Response:

The NRC's cybersecurity regulations are in Title 10 of the *Code of Federal Regulations (CFR)*, Part 73, "Physical Protection of Plants and Materials." Among these regulations is 10 CFR 73.54, "Protection of Digital Computer and Communication Systems and Networks," which the NRC issued in 2009. Under this regulation, NRC licensees must establish cybersecurity programs, develop cybersecurity plans, and implement these plans. NRC licensees must also monitor their programs and plans to ensure that they take into account system changes, operating experience, and evolving changes to the cyber threat landscape. These requirements ensure that NRC licensees have dynamic cybersecurity programs that respond to new threats and vulnerabilities, and they help protect against the NRC's regulations becoming outdated.

Since the NRC issued 10 CFR 73.54 in 2009, it has taken further action to ensure cyber protections.

- In January 2010, the NRC published Regulatory Guide 5.71, "Cyber Security Programs for Nuclear Facilities." This guide has obtained recognition from other United States agencies, international organizations, and foreign government regulators for its comprehensiveness and forward-looking approach. This guide and a 2010 Nuclear Energy Institute guidance document, NEI-08-09, "Cyber Security Plan for Nuclear Reactors," have been determined acceptable by the NRC for use in the development of licensees' cyber security plans.
- The NRC recently published a Cyber Security Event Notification rule, 10 CFR 73.77, which requires licensees to report cyber attacks within specific timeframes.

The NRC coordinates with other government organizations to promote cybersecurity. For example, the NRC currently chairs the Cybersecurity Forum for Independent and Executive Branch Regulators, a working group chartered to increase the overall effectiveness and consistency of regulatory authorities' cybersecurity efforts pertaining to the United States' critical infrastructure. In addition, the Commission is provided with a classified briefing on a semiannual basis to provide the latest information on the cyber threat environment.

Based on the dynamic nature of licensee cybersecurity programs as required by NRC regulations, as well as the specific requirement that NRC licensees evaluate their current programs based on evolving threats and technology, I do not currently see a need to update these regulations.

While the findings in the recent report from Chatham House are generally consistent with the operating experience and lessons learned in the United States, the report addresses the nuclear industry globally (rather than by country) and does not adequately capture all the NRC's efforts and regulatory requirements described above. I would further note that in a recent report, the Department of Homeland Security found that a cyber attack executed through the internet would not be expected to cause a nuclear power reactor to malfunction and breach containment.

Senator INHOFE. Thank you, Commissioner Ostendorff.
Commissioner Baran.

STATEMENT OF JEFFREY BARAN, COMMISSIONER, U.S. NUCLEAR REGULATORY COMMISSION

Mr. BARAN. Chairman Inhofe, Ranking Member Boxer, and members of the committee, thank you for the opportunity to testify today.

Chairman Burns provided an overview of the agency's current activities. I would like to highlight just a few of those efforts.

NRC continues to address post-Fukushima safety enhancements and lessons learned. Progress has been made in several areas, but a lot of work is still underway. Later this month, as Commissioner Ostendorff mentioned, the NRC staff will be sending the Commission a plan for how to proceed on the remaining Tier 2 and Tier 3 items. There are some significant safety issues in these categories, so we will need to do some careful thinking about how to best address them.

The staff has begun work on a rulemaking for decommissioning reactors. This rulemaking offers an opportunity to take a fresh look at a range of decommissioning issues with the benefit of public comment. It is also a chance to move away from the current approach of regulation by exemption, which is inefficient for both NRC and its licensees.

The Commission has been working to resolve the policy issues raised by the expected applications for small modular reactors. Earlier this year, we decided to proceed with a rulemaking to establish a variable fee structure for small modular reactors which will provide regulatory certainty and transparency for potential applicants.

In addition, the Commission recently approved a rulemaking related to the size of emergency planning zones for small modular reactors. This will allow the agency to examine novel emergency planning issues in a way that engages potential applicants and other interested stakeholders.

As you have already heard, the agency is working to increase its efficiency and agility, while remaining focused on our core mission of protecting public health and safety. Through our Project Aim rebaselining prioritization efforts, we will strive to implement NRC's existing scope of work more efficiently, identify any outdated and unnecessary initiatives, and adjust to declining workloads in some areas. Project Aim is not about relaxing regulatory oversight of licensee performance and safety; it is about more efficiently focusing on the right safety priorities.

Thank you, and I look forward to your questions.

[Mr. Baran's responses to questions for the record follow:]

Responses from Commissioner Baran
Senate Environment and Public Works Committee
October 7, 2015

Questions from Senator Markey

1. After the Fukushima meltdowns, NRC inspected each reactor in order to see how they used their *voluntary* severe accident management guidelines. What did these inspections reveal?

When NRC inspectors evaluated the status of SAMGs after Fukushima, their findings were not reassuring. For example, in Region I, inspectors found that SAMG triggers at one site had not been revised since initial issuance in 1998 even though they were directly impacted by license basis changes over the years, such as power uprates. Inspectors concluded that the SAMGS were therefore not "maintained" at that site. At another site in Region II, 8 of the 33 emergency response organization members were not qualified on SAMGs and 2 of the 4 Site Emergency Directors did not have SAMG training. In Region III, one site had not fully implemented the initial owners group SAMGs from the 1990s or any of the subsequent revisions. Inspectors also found that no exercise or drill had been conducted at the site since 1998 and that ongoing training on SAMGs was not occurring. The Near Term Task Force therefore recommended making SAMGs mandatory and the NRC staff agreed.

2. In response to these inspections, the NRC staff recommended that these procedures should be *mandatory* – but you were the only Commissioner to vote to support the NRC staff. In your opinion, was this proposed requirement an example of a requirement that would have been burdensome to the nuclear industry? Why or why not?

No, such a requirement would not be burdensome to licensees. During a public Commission meeting on the Mitigating Beyond Design Basis Events rulemaking, an industry representative confirmed that requiring SAMGs would not be burdensome because it would not require licensees to do anything they were not already planning to do. The key difference between a voluntary initiative and a regulatory requirement is enforceability.

3. In 2001, in response to the terrorist attacks of 9/11, the NRC issued cyber-security Orders for nuclear reactors, which were turned into regulations in 2009. A report was recently released that concluded that nuclear power plants are in a "culture of denial" over hacking risks worldwide. The report pointed out the rapidly evolving and increasing sophistication of the cyber-threat. Would you be willing to require a fresh look at these old regulations to determine whether updates are needed to address new threats or vulnerabilities? If not, why not?

The recent Chatham House report addresses the nuclear industry globally and is not focused on NRC's cybersecurity requirements for U.S. nuclear power plants. NRC's current regulations under Part 73 require licensees to monitor their programs and cyber plans to ensure that they take into account system changes, operating experience, and evolving changes to the cyber threat landscape. These performance-based requirements are intended to make sure that licensees have dynamic cybersecurity programs that respond to new threats and vulnerabilities. There is no question that the cybersecurity threat environment is evolving with increasingly sophisticated threats. It is important for NRC to keep abreast of and consider new information and analyses that may inform our cybersecurity efforts.

Senator INHOFE. OK, thank you. Thank you all, commissioners.

The NRC proposes to spend $91 million on research in 2016, which is 9 percent of the total budget. Now, three times, including the last meeting that we had, I have asked for a list of all ongoing research projects. I understand that that is one reason that some are saying that the amount of money in my opening statement that I talked about should be looked at is going to research projects, in writing and once personally with you, Chairman Burns, when we met in my office.

Now, late last night I finally received the list. So that has been several weeks ago, and then we get it right before the meeting, which makes it very difficult to analyze. But it still doesn't have, according to those who have read it, all of the cost information or the risk reduction information that we asked for.

So, commissioners, how do you develop a budget and meet your responsibility to be good stewards of taxpayer dollar and license fees if it takes 6 months and three oversight requests to produce a list of what projects this $91 million will be spent on? Any one of you want to respond to that, why it should take that long? Because it did.

Mr. BURNS. Senator, I will take that, and my colleagues can add.

I think the difficulty that we had in terms of the way that the agency tracks some of the research projects and its accounting, and our accounting is responsible; it meets management requirements. We assure within our process that projects are identified, have a user need; they are reviewed by management and are undertaken. So we try to do the responsible thing.

But what I have asked our EDO and our CFO to do is to tell me how can we, in effect, track some of the data in a way that I think we have gotten a request from your staff. So I don't think this is a matter that we are irresponsible. I think we are quite responsible in terms of how we plan the research of the agency, how we account for it, and how we carry it out. But there are ways we could make it, perhaps, more transparent for you.

Senator INHOFE. Well, do you disagree with the staff's first analysis of the document that we received last night is not complete, is not as thorough as it should be?

Mr. BURNS. I think it has the projects that are there. What I understand is what we don't have is the granularity at the individual project level. I think that is what it is. That is what I have asked our EDO and CFO to look at in terms of going forward and we have a process in terms of how we bin the data that can meet that.

Senator INHOFE. Well, other members are going to have specific questions about that. I would observe that in April I asked about the 2005 IG finding that the NRC needed to update its budget formulation procedure, and you indicated that the revised procedure was complete. Was the 2017 budget that we referred to developed using this procedure?

Mr. BURNS. I think, Senator, my understanding is what we have was we have a set of management directives that would come to the Commission for its review, given its policy, and I think by the end of this year, for our approval. Our budget, as I understand it, has been developed in accordance with procedures that the agency has in place and are consistent with the standards that OMB expects as we develop a budget.

Senator INHOFE. Wouldn't a thorough updated budget formulation procedure establish some discipline that there has been criticism of before and prevent the sort of thing that we are seeing in the Office of Research?

Mr. BURNS. I think the updated procedure can help us improve our processes, and I think that is one of the outcomes that we are looking for.

Senator INHOFE. Do you think Senator Alexander, when he was making his analysis, is accurate in most of his assertions?

Mr. BURNS. I am sorry; I didn't hear that.

Senator INHOFE. On the budget, looking at it from an appropriator's perspective, Lamar Alexander made recommendations and criticism. Well, let's do this. For the record, why don't you respond to his criticism. Would you do that?

Mr. BURNS. Yes, we will.

Senator INHOFE. OK.

Senator Boxer.

Senator BOXER. Thanks very much, Mr. Chairman.

Mr. Burns, I was perplexed by the Commission's decision to approve exemptions from emergency response planning requirements at the San Onofre Nuclear Generating Station. I am sure you know millions of people live around it. And the plant has been permanently shut down, but significant amounts of spent fuel remain at the site. I know you know that as well. They are in spent fuel pools.

I don't understand. Why did you do that? Why did the Commission decide it was wise to exempt the plan from emergency response planning requirements?

Mr. BURNS. Thank you, Senator. The current framework for plants under decommissioning relies, for better or worse, in terms of a construct that includes both looking at amendments to the license, as well as exemptions. And the exemptions are from rules that applied during operations, when there is fuel in the reactor, when the reactor may be operating.

The judgment with respect to emergency planning and the exemptions from certain emergency planning requirements was based on the staff's analysis that the risks with respect to the spent fuel pool are not such that it requires the full emergency planning complement. That is the basis for it.

Senator BOXER. OK, so let me understand. So if something were to happen, God forbid, because, as you know, there is a lot of storage right there, your answer to the people who are exposed to these materials would be, oh, we didn't do it because you weren't operational; this happened after you closed down? That makes no sense to me.

Now, I am introducing legislation, or I actually have done it, to prohibit emergency planning exemption at decommissioning reactors until all the spent fuel has been moved into safer drier cask storage. And I understand that NRC is developing a rule to address decommissioning issues.

Will you take another look at this issue or is this your final decision? Once a plant is decommissioned, you don't care how much spent fuel is there, they don't need a plan? You have to be kidding. Are you going to look at this again when you do that rule, in terms of decommissioning?

Mr. BURNS. I believe that within the scope of the decommissioning rule, we would look at the processes for what requirements would remain place and what time frequency.

Senator BOXER. OK. Well, I am going to talk to you further about this, all of you, and make the point. If you are exposed to nuclear materials, it is very serious; and people don't care if the plant was operational and there was an accident or the plant was decommissioned and there is an accident. They get just as sick.

I don't know how many of you have been there. Have all of you visited the plant? Can you nod? All of you? One hasn't, three have.

I spoke to the sheriff there and I said, what is the plan in case there is an evacuation, and she kind of shrugged her shoulders and she pointed to the road, which was backed up 24/7. That is the way people get away from there. So, please, your decision is dangerous, is wrong.

Now, Mr. Burns, will you commit to respond to me with specific timelines for implementation of all the task force's recommendations? You did send a good letter and had some deadlines, but you left out others. Will you get back to me on what the deadlines will be?

Mr. BURNS. Yes. I can look at the gaps that are there and make sure we understand what they are and what you are looking for. I would be pleased to do that.

Senator BOXER. OK.

Mr. Baran, recently, the Commission decided to ignore the recommendations of NRC staff and remove safety requirements from a proposed rulemaking that were opposed by the nuclear industry. In a press release, the Nuclear Energy Institute said, "The measures were not justified using quantitative measures."

What are the limitations of relying solely on quantitative measures to justify new safety enhancements?

Mr. BARAN. Well, I think a purely quantitative approach isn't going to do a good job of addressing low probability, high consequence events. A Fukushima style or Fukushima type event is a very low probability of occurring. So when you run the numbers, that makes it difficult for even common sense steps to pass a cost-benefit test that looks only at quantified benefits.

In fact, I think it is unlikely that any of the major post-Fukushima requirements that were instituted by the Nuclear Regulatory Commission with broad support would have passed a purely quantitative test. The Commission required flex equipment and hardened vents both as necessary for adequate protection of public health and safety, which is an exemption to the back-fit rule. Spent fuel pool instrumentation was required under the rule.

Senator BOXER. OK, I am going to interrupt you. I agree with you, but I am running out of time. Are there any other rules that don't look at quantitative only, in your knowledge? Do they all have to pass that quantitative test? Obviously, the staff didn't agree with that.

Mr. BARAN. Well, when you are doing a cost-benefit analysis, you need to examine both quantitative factors and factors that you can't quantify.

Senator BOXER. I agree.

Mr. BARAN. So all the costs, all the benefits. You need to look at them all. If you can quantify them, that is great; if you can't, you do need to still examine them.

Senator BOXER. You have to examine the worst that could happen, is that the point?

Mr. BARAN. Some benefits are not easy to quantify, but you still need to consider them when you are making decisions about weighing the pros and cons of whether to proceed with the requirement.

Senator BOXER. Thank you. I agree with you completely. Thank you.

Senator INHOFE. Thank you, Senator Boxer.

Senator Rounds.

Senator ROUNDS. Thank you, Mr. Chairman.

Chairman Burns, five reactors have shut down in recent years, and more closures are possible. I think in your written testimony you indicate an expectation for Oyster Bay to be shut down in 2019. My understanding is that it takes more resources to oversee the operating reactors than it does for those that have been permanently closed. In spite of this, the budget of the Office of Nuclear Reactor Regulation has grown about 42 percent, if our calculations are correct, since 2012, including a $32 million increase in corporate support costs.

Chairman Burns, do you think it is sustainable to continue increasing this section of the budget while the size of our reactor fleet continues to shrink? The reason why I am asking, it looks to me, while we focus on the safety side of things and we understand, as you have heard right here, there is a concern on that end of it, the dollars and cents side of it is an important part of the oversight as well. I think it is a fair question when we start looking at, if we have a shrinking number, how do we react to that in terms of the size of the entity that oversees these operations.

Mr. BURNS. Well, I would agree with you, Senator, that the size of the operation should meet the resource commitments or the projects that we would expect to come in. I would note in the operating reactor area, though we expect, for example, the Oyster Creek Plant in New Jersey, which this has been a longstanding plan, to cease operation in, I think, 2019, and there may be some others, we also, in the area of the operating reactors, we expect the Watts Bar 2 Plant to come online sometime next year. We are taking steps to work off the licensing backlog and to finish the Fukushima requirements. So those are things that I think, responsibly, that we need to budget for.

I agree with the principle that the resources should reflect the type of work that we have, and it may shift. It may shift. As you get out to 2020 in terms of operating reactors, the forecast would be you have four additional units online between the Vogtle and the Summer plants.

Senator ROUNDS. Let me just continue on a little bit. In both 2014 and in 2015 the fee recovery rules, the NRC has accounted for the reactor closures so far and the resulting loss of those fees by simply billing the remaining reactors more, on a per reactor basis to make up the difference.

For example, the NRC stated in their 2015 fee recovery rule, the permanent shutdown of the Vermont Yankee reactor decreases the fleet of operating reactors, which subsequently increases the annual fees for the rest of the fleet. As I say, now you have Oyster Bay, which is planned for decommissioning in 2019.

This is for all of you. Do you believe that this is a fair way, an appropriate way to structure the fee collection, to drive up the fees on the operating reactors because of a closure of a plant currently in existence today? Is this the right way to do it or should we be looking at another alternative?

Ms. SVINICKI. If I might jump in, Senator Rounds. Not speaking to whether or not it is fair, as long as the legal requirement exists for NRC to recover 90 percent of its budget, by virtue of mathematics, if there are fewer reactors in the United States, the fixed costs of our activities will be allocated across a smaller number of reactors with, again, the mathematical result that the fee would increase. So I think there is likely some minimum number of reactors where that would become unsupportable, and at that point perhaps Congress would then look at options for a different fee allocation.

Senator ROUNDS. Do you have any recommendations for this committee?

Ms. SVINICKI. I do not, but if I might respond for the record, please.

Senator ROUNDS. That would be appropriate. Thank you.

Thank you, Mr. Chairman.

Senator INHOFE. Thank you, Senator Rounds.

Senator Cardin.

Senator CARDIN. Thank you, Mr. Chairman.

Let me thank the commissioners for being here and for your service to our country. The mission of the agency is critically important to this country. The amount of energy met by nuclear electricity is significant, particularly when you look at the carbon-free generation. And your mission on safety, as we have already talked about several times, is very important to the public health of people of this country, not only the design and operation, but, as Senator Boxer said, the handling of spent fuels. All that is a critically important mission.

I want to talk about the workforce for one moment.

Your agency consistently ranks among the top as a best place to work. I mention that because I am sure that is because of your headquarters location in Maryland. But I want to talk about the impact that may have moving forward.

You have a highly skilled workforce. You are looking at Project Aim, with the realities of the reductions in the number of applications that you have received. You look at the demographics of your workforce and you see a significant number, over 20 percent now, are eligible for retirement, and that number is going to escalate pretty dramatically in the next few years. You look at the average age of your workforce, and that is increasing pretty dramatically.

So as you are looking to rebalance and you are looking at the realities of budget here in Washington, what game plan do you have to be able to recruit young talent that is needed in the agency, maintain expertise so that the mission of your agency moving forward can maintain that excellence?

Mr. BURNS. Thank you, Senator. One of the things I think we continue to do is have a robust entry level program for technical staff, and there is still a lot of excitement about that. I have had the opportunity in the last few months to go to Penn State University, which does some research for us but also has a large nuclear engineering department. They say they have an excellent interest in nuclear engineering there.

We support, through our budget, a grant program that goes out to not only universities, but also some craft and trade schools that help throughout. So, again, I think what we can do is leverage off being a great place to work, having an exciting mission that jumps around. That is what kept me there and kept me in Maryland for 34 years at the NRC before I left and then came back.

But it is an important area because there is a generational shift there, and there are fewer of us folks who were there in the late 1970s and early 1980s, and we need to make sure we have the next generation and we are able to also transfer knowledge to them. So we work at that.

Senator CARDIN. But as you are looking at Project Aim 2020 and rebalancing, which in many cases is code for downsizing, do you have a concern that young people may not see the future of the agency and that you may not be able to recruit? Also, downsizing numbers. You are going to get hit on both sides, it seems to me, retaining the expertise you need, but recruiting the new people. Is there any help you need? Any tools that you need in order to be able to get this done?

Mr. BURNS. I think we have the tools that we need. What I agree with you with is part of it is our communication, because what it is, although we are getting smaller, we need to retain critical disciplines. Those are our highly skilled workers.

But we also need lawyers, we need administrative staff, we need IT people, and communicating that out so that while we are shifting around we expect ourselves to be somewhat smaller, again, communicating those opportunities. That communication piece is important. I think we have the tools we need to recruit and do those types of things.

Senator CARDIN. Thank you, Mr. Chairman.

Senator INHOFE. Thank you.

Senator Boozman.

Senator BOOZMAN. Thank you, Mr. Chairman.

And thank you all so much for being here. The work of the NRC is so very, very important, and we need a Commission that is responsive to Congress, collegial, and thorough. The Commission must be science-based and quantitative analysis of benefits and costs, and it must be focused on the right priorities. We need to budget for these priorities.

First of all, I want to acknowledge the hard work and dedication of your staff in Arkansas. We are very proud of Arkansas Nuclear One. There was an industrial accident at the plant in 2013 that involved contract work that was performed onsite in a non-radiation area. This was a very serious and tragic accident, but it involved no risk to public health or safety.

The NRC has been very active over the last 2 years, reviewing safety measures at the plant. In the meantime, the Commission has determined that the plant remains extremely safe to operate, and we appreciate the work that has gone into fixing issues that were identified.

Our nuclear plant provides nearly 1,000 really good jobs in Arkansas, which is a huge boost to the economy of the city of Russellville and the area. In addition to those permanent jobs, hundreds of additional contractors regularly work onsite and invest in the community.

The plant has the capacity of over 1,800 megawatts. Our nuclear plant truly keeps the lights on in Arkansas, and it keeps our industry and manufacturers going. It is the largest producer of emissions-free energy in Arkansas by far. In fact, each year this plant reduces air emissions by over 13,000 tons of sulfur dioxide, it eliminates nearly 10,000 tons of nitrogen oxide emissions, and it cuts almost 8.5 million tons of carbon emissions. For all these reasons, we are very glad to have Arkansas Nuclear One.

So, again, we are very proud of our nuclear plant. We appreciate the potential and all that nuclear energy does.

Chairman Burns, the NRC's corporate overhead costs have risen significantly over the last decade, reaching $422 million, or 41 percent of NRC's total budget authority, according to the NRC's fiscal year 2015 fee recovery schedule. I am told that the NRC is considering an accounting recommendation that would allow some overhead costs, such as the human resources and financial management, to be reclassified within the NRC's business lines in order to make the costs attributed to corporate overhead appear smaller.

I guess the question is does the NRC plan to adopt what I would call almost an accounting gimmick, or is the Commission planning to find ways to actually reduce corporate overhead costs, rather than simply placing them in such a way in the business line budget that it is harder to get to?

Mr. BURNS. Well, thank you, Senator. We need to be transparent in terms of how costs are allocated and where they are. We do, as part of Project Aim, we are taking seriously looking at efficiencies in terms of the corporate support costs, as well as overhead costs in our activities. As directed by the Congress in the last appropriation bill or in the report on the bill, we used the consultant services of EY, formerly Ernst & Young, to look at corporate support.

My understanding is that we are generally aligned with other agencies. But this is an area we are focused on in Project Aim to try to reach a better balance and efficiencies in how we do it.

Senator BOOZMAN. So I guess the question is, are you going to do that. Are you going to, again, make it such that you reclassify some of your costs that shifted away from the overhead costs?

Mr. BURNS. I believe that the way we are portraying some of the costs will include overhead costs, yes. And I think in doing that, again, the idea is not to hide them, we want to be transparent about it, but a direct effort of a technical person does require some overhead in terms of office space, other types of support activities and the like, so that overhead. But we want to do it in an appropriate way.

I fully agree with your principal. This is not sort of hide the peanut, move a shell game here. We want to be responsible about it.

Senator BOOZMAN. Good. Thank you, and thank you all for being here.

Thank you, Mr. Chairman.

Senator INHOFE. Thank you.

Senator Gillibrand.

Senator GILLIBRAND. Thank you, Mr. Chairman.

I would like to talk a little bit about Indian Point, which is one of our reactors in New York. Following the May 8th transformer fire at Indian Point, which resulted in oil leaking into the Hudson River, I wrote to you expressing concerns about the incident and the number of incidents involving transformers over the past 8 years, including fires in 2007 and 2010. In our correspondence following the incident, we discussed the Commission's decision to not require an aging management plan for transformers as part of the licensing renewal and instead continue to monitor them as part of NRC's ongoing oversight inspection and maintenance activities.

Can you please explain, any of you who have looked at this, why, given multiple incidents involving transformers at Indian Point over the past 8 years, the Commission believes that the current monitoring regime for transformers is sufficient?

Mr. BURNS. Thank you, Senator Gillibrand. As I think we discussed when I met with you, I did not participate in the Commission's adjudicatory decision related to that because I am disqualified from doing that. I think the general principal is that in looking at license renewal, the focus is on the aging of long-lived passive components, which a transformer generally is not considered. I think there is oversight and monitoring that the licensee is expected to do through its maintenance programs that we monitor. I think that is the basic dichotomy.

Senator GILLIBRAND. OK. Despite the fact that Indian Point experienced four unplanned shutdowns earlier this year, including a shutdown that was a result of the transformer fire, the mid-cycle assessment states that NRC plans to conduct baseline inspections at Indian Point. What are the criteria for a baseline inspection versus other levels of inspection? And when making a decision on the level of inspection that a plant will be subject to, do you look at the previous violations in a cumulative way, or do you only look at a specific period of time?

Mr. BURNS. I would like to be able to provide you more detailed information for the record. The general approach is we do look at a history of operation or performance during the time. I have to say I am a little fuzzy in terms of how the things will line up, but I would be pleased to provide that for you for the record.

Senator GILLIBRAND. OK. And for the record, if there is a number, if there is a number of incidents or violations within a certain period of time that NRC would then require a different level of inspection above baseline inspection, please let us know.

Mr. BURNS. Yes. Because there is generally, through our reactor oversight process, and I just don't have the details in my head, in terms of how the levels of inspection and expectations are. So we will make sure we get that to you.

Senator GILLIBRAND. OK.

Mr. OSTENDORFF. Senator, if I may just make a brief comment here on your question.

Senator GILLIBRAND. Sure.

Mr. OSTENDORFF. One of the concerns on tying plant shutdowns or trips to performance evaluations is, it could send a signal to a licensee that there is going to be a penalty to pay if they shut down. And in many cases our licensees will take the conservative safety step of shutting down.

Senator GILLIBRAND. Right.

Mr. OSTENDORFF. We do not want to send a different incentive to that licensee.

Senator GILLIBRAND. OK.

On December 12th the license for Indian Point Unit 3 will expire. As you know, the license for Unit 2 expired in 2013. The reactor has been operating with an unrenewed license for the past 2 years in what is called a "timely renewal period." Is Unit 3 also expected to enter into a timely renewable period when its license expires in December? Have there been previous instances where multiple reactors at the same plant were both operating without a renewed license? What impact do you think this will have on the plant and the NRC's inspection process for Indian Point?

Mr. BURNS. I would expect, given the status of the adjudicatory proceeding on renewal, that the other unit would go into so-called timely renewal. That is a provision under the Administrative Procedure Act that is incorporated in our regulations.

What I understand is that the licensee, Entergy, will implement the enhancements to the license that are expected that have come through the process of staff review. They would continue to have the oversight by the NRC. They are still expected to follow the license. In a sense, the open item is the conclusion, the proceeding on license renewal, but our oversight would remain and our ability to do that remains the same.

Senator GILLIBRAND. Thank you.

With my remaining 5 seconds, will you just submit for the record an analysis about the Fitzpatrick Plant? Because we have hear from Entergy that they may shut it down, and I just want to know what NRC's role, if any, in being part of these decisions, whether you are notified of plans, whether you have any input. Because it is a huge community issue right now, and I would love to know what your perspective is and if you do involve in these decisions on any level.

Mr. BURNS. I will certainly do that. We don't have a role in the decision with respect to operation, but if a plant decides not to continue operation, there are processes, and we can provide you information on that.

Senator GILLIBRAND. Thank you.

Senator INHOFE. Thank you, Senator Gillibrand.

Senator Capito.

Senator CAPITO. Thank you, Mr. Chairman.

I want to thank all of you all for being here today. I would like to ask some questions along some of the same lines as my colleagues have. I also would like to mention that I do not have a nuclear facility in my State, but I was able, by the courtesies of AEP, to visit the Cook Plant in Michigan, which has just had a 20-year extension, I believe, on their license. So I learned quite a bit there.

But as I understand it, when companies need to modify their plants or alter their procedures, the NRC has to approve that. Correct?

Mr. BURNS. For many things. There are provisions in our regulations, and perhaps also in our licenses, that allow certain types of changes to be made if the licensee does the analysis and concludes, for example, under one of our regulations, that there is no unreviewed safety question. So they have some flexibilities themselves.

Senator CAPITO. All right, good. Thanks for that clarification. And you budget for about 900 reviews a year. Am I correct in assuming that you have stated that you prioritized the licensing actions based on safety significance? So the ones that would have more impact on safety obviously are going to rise to the top? Is that how you prioritize 900 reviews a year?

Mr. BURNS. I think that is generally true. Part of that also comes in discussion with licensees who apply for the amendments or other types of licenses.

Senator CAPITO. OK. So we are going to put the chart back up that the chairman used. The first point I would like to call your attention to is the number of operating reactors has gone down due to economic challenges. So we have gone from 104 to 100 reactors. But resources for the agency have gone up 15 percent over that same time period.

I learned just today, more specifically, that Project Aim 2020 is aimed at probably that discrepancy, but the second thing I would like you to notice is how the workload is down, but there is still a backlog in reviewing licensing actions on time. So I would say since the NRC prioritizes reviews based on safety, which we pretty much just established, any licensing action that companies are pursuing for economic reasons but do not have a safety nexus, are they the ones that are more likely in this backlog? Do you understand my question?

Mr. BURNS. No, I understand the question. I think I would have to look at that in terms of the record.

Senator CAPITO. Let's talk about the backlog a little bit. How extensive is it and what kind of time periods are allowed for backlogs? Is there a stop dead date where you can no longer be in a backlog, when you have to have a decision made?

Mr. BURNS. Essentially what the objective is, I think, is to work through license amendment or licensing action type of requests from licensees within a year, and what happened over the last few years, particularly after the Fukushima accident, is a backlog grew as we focused on the safety significant Fukushima enhancements. So that grew.

What I give credit and credit mostly goes to, I think, our current Director of Nuclear Reactor Regulation, Bill Dean, and his team in terms of they have been taking steps that are working down that backlog, and I think their objective is that we basically have it down to zero by fiscal year 2017.

Senator CAPITO. OK. And I think the chairman mentioned the document the NRC gave to appropriators, I am on an appropriations committee, NRC Fiscal Year High Level Impacts of Further Reductions. In that document, it indicates that the NRC would delay domestic licensing actions prior to suspending the review of foreign reactor design for construction in a foreign company. How do you justify giving foreign work a higher priority than a domestic licensee's operational needs?

Am I understanding that correctly, the statement that you made in that document?

Mr. BURNS. The document that the chairman referred to was developed at looking at potential impacts of rolling significant cuts to our budget request.

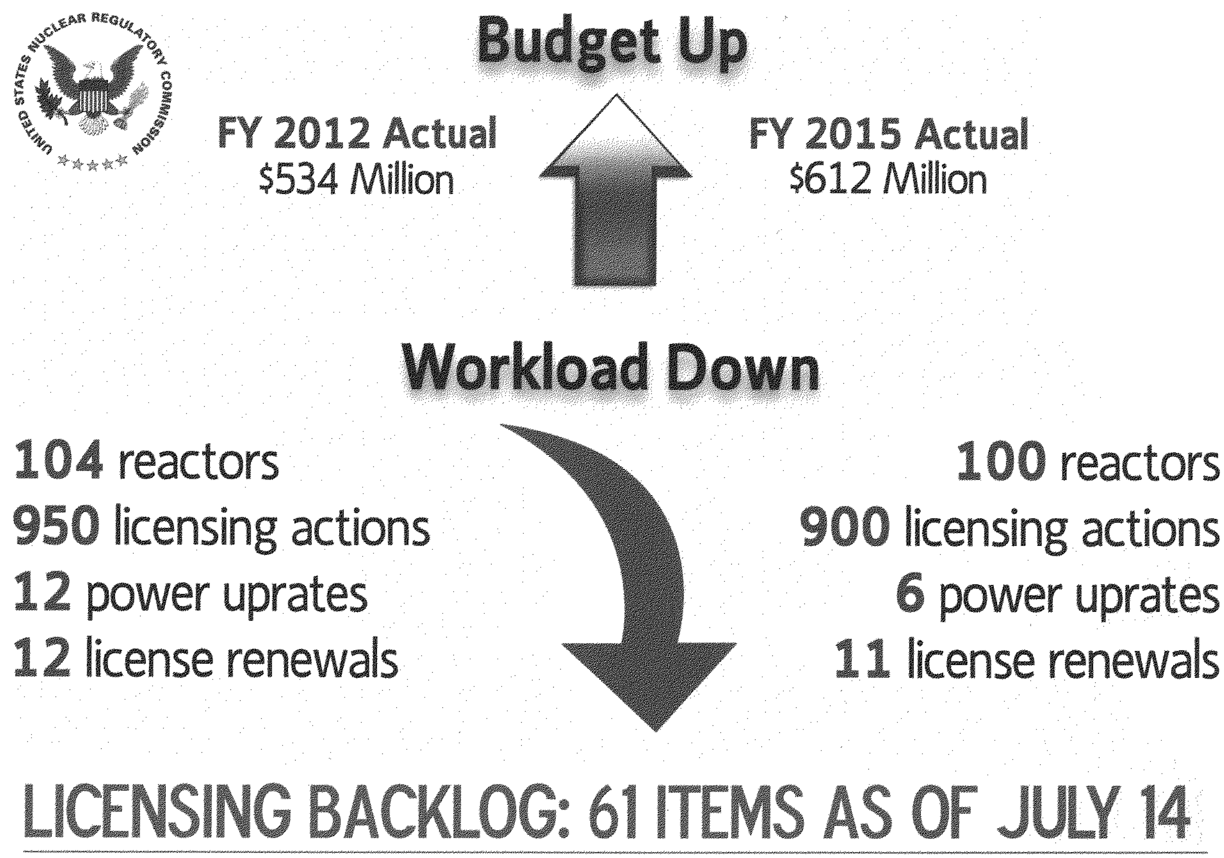
Senator CAPITO. Right.

Mr. BURNS. And in one of them, yes, it does identify the Korean design certification that is under review. I think what we look at in terms of if we have cuts that go along those levels, or certain actions that we would have to go to look at in terms of the relative priority. I think that when it comes to the actual decision, the Commission would look at the priority of the particular items and things that are under review. Like, for example, on the backlog it may be a question of stretching out, again, the review versus saying we are not going to undertake that review.

Senator CAPITO. And I guess the point of my question is I would think, just on the face of it, that one of the priorities that we would certainly like to see, and Senator Boxer has talked about this in terms of her State, is a domestic influence here, or not influence, a domestic priority over what might be occurring around the rest of the world.

Anyway, I thank you for that and I thank you for the response.

[The referenced documents follow:]


UNITED STATES NUCLEAR REGULATORY COMMISSION
Budget Up
FY 2012 Actual
$534 Million
FY 2015 Actual
$612 Million
Workload Down
104 reactors
950 licensing actions
12 power uprates
12 license renewals
100 reactors
900 licensing actions
6 power uprates
11 license renewals
LICENSING BACKLOG: 61 ITEMS AS OF JULY 14

NRC FY 2016 High Level Impacts of Further Reductions

The U.S. Nuclear Regulatory Commission (NRC) recognizes the importance of reducing its budget and number of employees. At the same time, we need to assure that regulated entities are operating safely and securely in conformance with NRC requirements and that licensing actions are processed in a timely manner. Recently, the Commission directed the staff to undertake a number of the strategies included in the Project AIM 2020 report that will result in the agency operating more efficiently.

The potential impacts of the budget reductions at each level discussed below are cumulative, and would adversely affect the agency's regulatory program and licensees. Achieving the FTE levels in the budget scenarios would likely require the agency to implement a reduction-in-force (RIF). With the loss of NRC employees, potential restart of these programs in the future would be delayed by at least 3-5 years in order to rebuild lost capacity. All levels assume $15 million in funding for the Integrated University Grants Program.

Level 1: Reduce $30 million and 140 FTE

- Delay operating reactor license amendment reviews with less safety or security significance, resulting in a decrease in licensing actions by 5%. This would impact licensees' operational needs such as outage planning and reliability improvements.

- Terminate remaining Tier 2 and 3 Fukushima work, which would truncate NRC's evaluation of whether additional regulatory action is warranted on issues such as evaluations of containment vents for ice condensers, hydrogen control, severe accident instrumentation capabilities, and periodic seismic and flooding hazard reevaluation.

- Reduce by 25% NRC investigations of alleged criminal wrongdoing, reducing oversight and prosecution of criminal violations under the Atomic Energy Act.

- Eliminate work on medium priority rulemakings, such as digital instrumentation and control and financial qualifications for new reactor licensees, impacting the agency's effectiveness and efficiency in future licensing reviews, adversely affecting the ability of merchant plants to obtain a construction permit or operating license, and hindering the NRC and industry from more fully adopting risk informed approaches to regulatory requirements.

- Reduce confirmatory research in support of the operation of power reactors beyond 60 years and refinement of tools for reactor risk assessment. These reductions would eliminate improvements in assessing reactor safety risk and risk informing NRC's regulatory framework.

- Terminate the integrated response planning events with the Federal Bureau of Investigation and the Department of Homeland Security, impacting future capability for government response to threats at nuclear power plants.

- Terminate the program to supply potassium iodide (KI) tablets to States in support of nuclear power plant emergency preparedness programs.

- Reduce international travel by 50%, impacting NRC participation and influence at international meetings and in global safety and security initiatives.

- Eliminate grants to minority serving institutions.

- Eliminate improvements to the search capability of the agency's document management system (ADAMS) impacting the public's ability to find documents quickly and easily.

- Eliminate funding to maintain the FOIA Express system impacting NRC's ability to comply with statutorily mandated FOIA response times.

Level 2: Reduce $40 million and 180 FTE

- Suspend work on the design certification review of the Korean reactor (APR-1400 KHNP) and associated research activities. Staff and contractors with critical skills would be lost. It would take considerable cost and time to re-hire and re-establish the expertise and disciplines that currently support design certification reviews.

Level 3: Reduce $50 million and 185 FTE

- Defer licensing reviews of new applications for medical radioisotope production facilities impacting the ability of the U.S. to meet the domestic demand for such isotopes.

- Reduce confirmatory research on thermal hydraulic experiments to enhance understanding of severe reactor accidents and make improvements to risk assessment tools.

- Terminate revisions to NRC requirements for low-level waste disposal, leaving unresolved issues with disposal of depleted uranium.

- Reduce by half the training and travel of State employees in Agreement States, which would require Agreement States to fund training and travel to maintain the competency of personnel and the quality and effectiveness of State programs.

- Eliminate annual information exchanges with licensees, international counterparts, and public, including the Regulatory Information Conference, Fuel Cycle Information Exchange, and the Spent Fuel Regulatory Conference.

- Eliminate efforts to comply with new requirements to reinvestigate employees every 5 years instead of 10 years, increasing the backlog of personnel security reinvestigations and increasing agency vulnerability to insider threats.

Senator INHOFE. Thank you, Senator.

Senator Whitehouse.

Senator WHITEHOUSE. Thank you.

I would like to follow up on Senator Capito's questions about the backlog with some questions about what you might call the frontlog.

People have been talking about modular nuclear reactors for decades. So far, not a single one has ever been approved by the NRC. I believe that the first likely one is the NuScale project coming up next year. There have been significant advances in nuclear next generation technology, the traveling wave technology. TerraPower is, to a large extent, Bill Gates' company. He is no idiot. He has not been able to develop that technology beyond the experimental. Not even beyond the experimental, beyond the theoretical stage in America. Instead, he has signed contracts with China's nuclear commission.

And we are looking at, at a time when carbon pollution is probably going to be the disgrace of our generation, 4.2 gigawatts of carbon-free power lost just in the last 2 years to decommissioning. Now, some of those decommissionings may have been necessary for safety purposes. It is obviously a case-by-case scenario. I know our ranking member is very concerned about a plant in her State. But to the extent that these are viable plants that are providing carbon-free power and they are being decommissioned on economic grounds because nobody has bothered to figure out a way to price the carbon savings that they provide, we are losing a big piece of our fleet.

So if you look at those three emerging things, the modular power, the next generation power, and the decommissioning that we are seeing, it doesn't look to me like you guys even have a windshield. You are living looking in your rearview mirrors at problems of the past, and I don't get why we seem to be behind or not paying attention in all of those three frontlog areas.

Now, I am probably exaggerating for effect, but I feel some real frustration when American technologies get developed in China instead of here. I feel some real frustration when strategies for modular, which is basically still light water, it is not even a new technology, that we have talked about for decades, are still backed up; and we are looking at the very first certifications a year from now, after decades, and when we see these plants being decommissioned with no evident review as to the significance of their carbon savings.

So, great on the backlog. How about the frontlog?

Mr. BURNS. Thanks, Senator. I think we are looking forward and we are looking forward in some of those areas. We have to ensure the safety, obviously, of the existing fleet. We have to ensure that the plants that go into decommissioning are handled safely. But there are initiatives and there is work that we are doing with respect to both small modular and also advanced technology.

Let me describe that a little bit, but one thing let me point out is that with respect to our ability to review or take, in effect, licensing type action on those new technologies, they have to come in with a sponsor who is ready to pay, basically pay the fees as we are required to collect under that. That is some of the challenge.

I have had some discussions with the Department of Energy, because they have a role, too, in terms of the R&D part. We are the safety regulator; we have to give judgment to say are these types of concepts going forward.

We recently had a very good workshop with the Department of Energy where we invited in people who are looking at this type of innovation, and there are things we can do with DOE, staying in our appropriate roles, that look at what are the types of safety issues that are different than the light water technology, and we are doing some of that.

Senator WHITEHOUSE. Would you agree as a general proposition that regulatory agencies have ways to preadapt regulatory processes to emerging technologies so that the emerging technology doesn't have to face a regulatory regime that was developed for an old technology but, rather, a more welcoming, equally rigorous, but welcoming in terms of fitting the new technology? I would love to know what steps you have taken to change the manner in which modular reactors will be certified in advance of this clearly oncoming means of giving us some clean power.

Mr. BURNS. Well, let me make two quick points. First, with respect to the NuScale design, they are coming in under what I will call the design certification process, and there has been a dialog with them as they prepare to submit the application to make sure that both sides' expectations meet. So that is one thing.

The second thing I would say, and this is an item that came out of that workshop, is whether we are prepared to do more. While we are not giving the final license, if you will, the final certification, I think we can be responsible about making step-wise decisions that signal and indicate to developers and investors that we have looked at this aspect of the technology, we have issued a topical report or review on it, and that that looks OK, you can go to this step. That is the type of thing that they are looking for. I think within our framework we can do that because I would agree with you, we need to be adaptable.

Senator WHITEHOUSE. Thank you, Chairman.

Senator INHOFE. Thank you, Senator Whitehouse.

Senator Fischer.

Senator FISCHER. Thank you, Mr. Chairman.

On July 15th I joined in a letter with Chairman Inhofe and other members of this committee to the Commission expressing concerns based largely on defense of NRC's existing backfit rule. This rule provides that before a new requirement can be added to an existing license facility, the NRC must demonstrate that the new requirement would result in a substantial increase in the protection of public health and safety, and that the direct and indirect costs of implementation for that facility are justified in view of this increased production.

Commissioner Ostendorff, what policies or procedures are in place at the Commission level to ensure that the backfit rule is consistently applied in staff analysis and recommendations?

Mr. OSTENDORFF. Senator Fischer, thank you for the question. If I may, let me address this in the context of a recent Commission decision I think that is very important. I referred to it in my opening statement, and that is the Mitigation of the Beyond Design

Basis Event rulemaking, which brings together in one rule a large number of Fukushima-related action items.

Our regulatory framework is predicated upon two essential notions. One, adequate protection. If something is required for adequate protection, then we don't take cost into account, period. And I wanted to say that because I know there was an exchange earlier Commissioner Baran had on this topic with Senator Boxer. Added protection, no costs are considered.

If it is a lower safety issue, such as it does not rise to adequate protection, then it becomes under the backfit rule; is there a substantial safety enhancement that passes a cost-benefit analysis. In the Mitigation for the Beyond Design Basis Event rulemaking, which overall the Commission approved that rule, there is one small part of it that the majority of the Commission did not approve because it did not pass the cost-benefit analysis test using quantitative analyses, which were available, and that is the requirement for severe accident management guidelines.

So I would say that the staff made a recommendation to the Commission in the spirit of an open collaborative work environment. We do not want to stifle the staff coming forward with a recommendation. At the end of the day, when it comes to the backfit rule, it is the Commission that makes the final decision. That is what we have done.

Senator FISCHER. OK. Thank you.

Commissioner Baran, I see you nodding. Did you have comments you wanted to add to that?

Mr. BARAN. I don't think so. Commissioner Ostendorff mentioned severe accident management guidelines, and that was a situation where I disagreed with my colleagues. I thought the staff's analysis was the right one there. What we heard from both the staff and from our advisory committee on reactor safeguards was that the staff's quantitative analysis wasn't a complete picture of all the safety benefits of requiring SAMGS, as they are called. In other words, the staff didn't have all of the tools they would need to do a complete quantitative analysis that captured all the safety benefits.

So, from my point of view, the staff, therefore, appropriately did a qualitative analysis to supplement the limited quantitative analysis, and when they did that analysis they found that it was a substantial safety enhancement. But as Commissioner Ostendorff pointed out, and I completely agree with this, it is ultimately a Commission decision about whether or not to accept that analysis, accept that recommendation. The staff's job is to lay out all of their analysis in a way that is transparent and understandable for decisionmakers and for stakeholders, and I think they did that here, and then the Commission made a decision about it.

Senator FISCHER. Commissioner Burns, as we look at the rulemaking process, I think really a critical first step in addressing the impacts when we look at a new regulatory requirement to be verified is to be safety significant and cost justified, and that is required by the NRC's backfit rule. But we have seen the NRC staff proposals that fall short of that. In fact, the NRC IG has noted, "The agency may be vulnerable to errors, delays, wasted effort, and flawed decisionmaking because of the limited experience of its cost estimators. It also increases the potential to make less than optimal rulemaking decisions because the NRC Commission uses regulatory analysis to determine whether to move forward with rulemaking."

Do you agree that the Commission should, I guess, more closely scrutinize rulemaking initiation and how those rulemaking processes are prioritized so that you can better use staff time and resources on proposals that are brought forward by the staff?

Mr. BURNS. There is certainly an important role for the Commission in rulemaking, and one of the things I have done, we are expecting a paper from the staff very shortly, is taking a look at steps to assure greater involvement at more critical points in time of the Commission and rulemaking. So we will be deliberating on that over the next few months. But I would agree with you, Senator, it is important for our leadership role to assure that we take as a Commission a hard look at rules that we propose to impose.

Senator FISCHER. Well, I thank you for that, and I agree with you. I think it is especially important that the Commission provide scrutiny at the initiation of the rulemaking process. So thank you very much.

Thank you, Mr. Chairman.

Senator INHOFE. Thank you, Senator.

Senator Markey.

Senator MARKEY. Thank you, Mr. Chairman, very much.

More than a year ago Senator Sanders and I wrote the Commission about why NRC's economists were improperly prevented by their supervisors from asking Entergy questions about whether Entergy had the financial resources to, if needed, deal with the safe operation of its reactors. In the Commission's response to us, NRC maintained that there was no "direct link between safety and finances." It is time to revisit that statement.

The Pilgrim Nuclear Power Station in Plymouth, Massachusetts, was recently placed in NRC's least safe operating reactor category because of repeated unplanned shutdowns and other safety problems. There are only three reactors in that category, and every single one of them is run by Entergy. In fact, of the 10 reactors Entergy operates, only 4 are currently rated as being in NRC's safest categories.

Moreover, financial analysts are openly saying that it isn't economical for Entergy to continue to operate Pilgrim and other reactors.

Do any of you disagree that if NRC staff wants to renew their request to you so that they can receive detailed financial information from Entergy in order to determine whether Entergy has the money needed to safely operate its reactors, that they should not be allowed to do so? Mr. Chairman.

Mr. BURNS. There may be an appropriate circumstance in which we would do that. I would say on a day-to-day basis I want our inspectors in the plant looking at how activities are being carried out at the plant. I think that, for us, is the primary way to do it.

I am not particularly familiar with the letter you and Senator Sanders sent, but, again, if there is an appropriate basis for us to do so, certainly we could do so.

Senator MARKEY. I think this is a very suspicious situation, Mr. Chairman, when Entergy has three reactors in the same category and every single one of them is an Entergy plant, in this lowest category, and that analysts are wondering whether or not Entergy has the financial capacity to run the Pilgrim plant, that we give to the NRC staff the ability to be able to make that determination as to whether or not the financing capacity is there. Would you agree that that makes some sense?

Mr. BURNS. Again, I think there are circumstances in which it may be appropriate to do that. Whether that is here or not, I won't say.

Senator MARKEY. OK. Commissioner Baran.

Mr. BURNS. But I want our inspectors on the ground.

Senator MARKEY. OK. Commissioner Baran.

Mr. BARAN. Well, if the NRC staff thinks there is a nexus between underinvestment at a plant and safety problems at that plant, I think they should get the information they need to address that issue.

Senator MARKEY. I agree with you.

Do any of the other commissioners disagree with that?

Mr. OSTENDORFF. Senator, just a comment. I had a chance, in June of this year, to visit Pilgrim, and I appreciate that one of your staff from Massachusetts attended that visit with me, and we spent a lot of time with the licensee looking at their operating performance. Subsequent to that visit, 2 months later, our staff made the recommendation to place them in column 4, as you noted in your comments. I would just observe that having spent quite a bit of time, along with other commissioners and senior staff, looking at this particular issue at Pilgrim, we have not assessed that there is a nexus between plant investment and operating performance.

Senator MARKEY. Commissioner Baran, every time a reactor gets placed in a lower safety category by NRC, it gets subjected to more inspections and requirements, and those cost the industry money. There is currently a proposal in front of the Commission that would basically allow reactors to experience more safety problems before they fail into NRC's second worst safety category for operating reactors. Is that your read of the new proposal?

Mr. BARAN. The Commission is currently deliberating on whether to increase the number of white findings, or low to moderate significance findings, in the same cornerstone necessary to put a plant in column 3, so the proposal is to increase that from two findings to three findings, which would raise the bar for column 3.

Senator MARKEY. My experience with nuclear power plants is that they age, and what has happened here is that each one of these plants keeps requesting an extension so that they can continue to operate longer and longer. But the older the plants get, the more problems they have; and the industry historically has tried to avoid having to make the additional investment in safety, because that is cost for them that they don't want to have to have factored in, the lifetime cost of keeping these plants safe.

So, from my perspective, I just think that the NRC should listen to their staff, they should allow them to do the financial analysis of whether or not the actual overall financial well being of Entergy is in any way inhibiting their investment in the safety procedures that are needed, given the fact that Entergy has such a high percentage of the plants in America that are considered to be the least safe operating reactors in America.

So that is my request to the Commission. I think you should give them permission, and I think we will get the answer we need. This linkage between financial viability of a corporation and the investment they make in safety. It is pretty clear here it is an issue that has to be answered, and soon.

Senator INHOFE. Thank you, Senator Markey.

Senator Barrasso.

Senator BARRASSO. Thank you very much, Mr. Chairman.

Chairman Burns, the EPA has proposed a rule to set forth groundwater protection standards for uranium recovery facilities. I believe the EPA proposal ignores the successful 40-year history of in situ recovery projects. It imposes numerous overly stringent standards that would jeopardize the future of the uranium recovery industry in the United States. I believe the EPA is once again asserting power over another area of the economy, even though they are not the primary agency that Congress created to manage and oversee uranium production. That role belongs to the NRC.

So while I recognize EPA has some standard setting authority under the Uranium Mill Tailings Radiation Control Act, it is my understanding the NRC is charged with determining how to implement these standards, and the question is has the NRC adequately looked at this issue.

Mr. BURNS. Thank you, Senator. I think we have looked at the issue with respect to the proposed changes to the EPA regulations, I think in 40 CFR part 192, and our general counsel has submitted commentary with respect to that.

Senator BARRASSO. Do you feel the NRC was adequately consulted on the rulemaking?

Mr. BURNS. I think we had an opportunity to provide input, which we did, on it. That is what the general counsel's letter does.

Senator BARRASSO. Any other members want to jump in on that, whether the NRC was adequately consulted?

Mr. OSTENDORFF. I would just add that I think the NRC and EPA have a very solid ongoing working relationship. We have, however, as an agency, identified concerns with perhaps their regulatory footprint going into our jurisdictional issues in dictating how certain methods are to be used by our licensees, and that causes us concern. But I think we understand the EPA will be talking to us about our concerns here in the near future.

Senator BARRASSO. Because I know the NRC has indicated in a July 28th letter to the EPA that the proposed rule "may encroach upon the NRC's authority." So I wonder has the NRC met with the EPA specifically to discuss the concerns. You said you are going to meet with them in the near future? What is the plan on that based on that July 28th letter?

Mr. BURNS. My understanding from our general counsel is that we met on preliminary basis, but there is the intention to have future meetings on the subject.

Senator BARRASSO. Because a 2009 NRC memo from staff entitled Staff Assessment of Groundwater Impacts from Previously Licensed In Situ Uranium Recovery Facilities states that the staff is unaware of any situation indicating that the quality of groundwater at a nearby water supply well has been degraded, any situation where the use of a water supply well has been discontinued, or any situation where a well has been relocated because of impacts attributed to an ISR facility. So the question is has there ever been a leak that you know of from uranium in situ recovery facility that impacted drinking water?

Mr. BURNS. Not that I am aware. I could check with our staff.

Senator BARRASSO. OK. That is the recent staff report from a couple of years ago.

So, Chairman Burns, in April I asked you about the length of time that it should take to review an application for a new uranium recovery facility, and your response you concluded was I think this is an area I am willing to look at and see. We are trying to do a better job. And I agree with you.

This is what we found from information that we requested from the NRC. By our math, the agency takes an average of 3 years to review an application for a new facility; one application took 5 years. I mean, that is longer that it took for the NRC to issue the licenses for the new nuclear plants in Georgia and South Carolina.

So uranium recovery licenses are for 10 years, and there has to be a reapplication for a renewable. We found the NRC sometimes spends 5 years deciding whether to grant the 10-year license extension. So a company spends about half of its time paying for license reviews.

Is a uranium recovery facility as complicated as a nuclear power plant? And if not, why should it take a comparable, if not longer, amount of time to review a license application than it does for a nuclear power plant?

Mr. BURNS. I think, Senator, in some of the circumstances the requirements on consultation under the National Historic Preservation Act, those requirements, because of the consultations, have to be done with local tribes, those have been extensive.

What I understand from talking to our staff, a couple areas where I think we have seen some improvement in that area is, one, encouraging the license applicant to have dialog with local community. Second, we have been focused also on improving our processes with request to this consultation process. We issued recently a tribal protocol in terms of helping our communications. I think that is going to help in that area, but it is something I think we can continue to work on.

Senator BARRASSO. So finally, then, would a longer license duration, rather than the 10 years, a longer duration, help the NRC manage its workload better?

Mr. BURNS. That is a possibility. We would have to take a look at that.

Senator BARRASSO. Thank you.

Thank you, Mr. Chairman.

Senator INHOFE. Thank you, Senator Barrasso.

Senator Boxer, I think you want to submit something for the record.

Senator BOXER. Yes. I just wanted to thank you for this hearing and thank the Commission and all of our colleagues.

I ask unanimous consent to place in the record an explanation of the rulemaking that Senator Barrasso talked about. We want to make sure that the water is safe when you have this uranium mining. I think the EPA could go either way; they could do a rule under the Uranium Mill Tailing Radiation Control Act or under the Safe Drinking Water Act. So I just want to put that in the record.

Senator INHOFE. Thank you. Without objection, it will be in the record.

[The referenced information was not received at time of print.]

Senator INHOFE. Let me just say to the four commissioners, first of all, thank you for being here. You are doing a good job in some areas, but the big concern that gave birth to this hearing is that when you are looking at operating reactors dropping down from 105 to 99, licensed action going down from 1,500 to 900, material licensees 4,500 to 3,200, licensed renewals 43 percent down at the same time, there should be cuts in the budget commensurate with this lighter workload.

I know that Project Aim is supposed to be helping us to do that, but I don't think anyone on our side over here is satisfied with the progress that we have made so far, and I want to make sure that you leave with that message and that you continue on this and come up in a very short period of time with better results that respond to what we refer to as the workload and financial concern. And I thank you very much for the hearing today. Thank you, Chairman Burns.

Mr. BURNS. Thank you.

Senator INHOFE. We are adjourned.

[Whereupon, at 10:58 a.m., the committee was adjourned.]

[An additional statement submitted for the record follows:]

STATEMENT OF HON. BERNARD SANDERS,
U.S. SENATOR FROM THE STATE OF VERMONT

Although this hearing will cover a range of important issues, the issue that is of the greatest concern to me is the Vermont Yankee nuclear power plant in Vernon, Vermont, and the process involved in decommissioning the plant.

Since the plant's operator, Entergy, announced its plan to close Vermont Yankee in August 2013, we have turned our attention to employing the safest measures for decommissioning and to protecting the livelihood of Vermont Yankee's employees and the communities in which they live. There has been a strong desire on the part of Vermonters to participate in the decommissioning process to ensure a complete, full, and safe decommissioning.

I have voiced my very serious concerns regarding the lack of input communities in Vermont have in the decommissioning process. I have said time and again that these communities deserve a seat at the table, and have introduced legislation that would allow States and cities to provide meaningful feedback during the decommissioning process.

It is my understanding that most of the conversation surrounding Vermont Yankee's decommissioning plan has been between Nuclear Regulatory Commission staff and members of the Nuclear Energy Institute, which includes Entergy—to the exclusion of the State government and local communities that house the plant. This is simply unacceptable. I believe that it is unconscionable that the NRC would deny Vermonters the opportunity to participate in the decommissioning plan—a plan that will affect their day to day lives and livelihoods for generations.

Even more unsettling than the lack of communication with local stakeholders is that Entergy has been allowed to take money out of its decommissioning fund for spent fuel management, even though NRC regulations expressly disallow the use of decommissioning funds for spent fuel management. [1] To my mind, this is a misuse

[1] "Amounts are based on activities related to the definition of 'Decommission' in § 50.2 of this part and do not include the cost of removal and disposal of spent fuel or of nonradioactive struc-

of funds and could potentially put Vermonters on the hook if Vermont Yankee is unable to cover the costs remaining for full decommissioning. If the company goes bankrupt in the process of decommissioning the plant, then Vermonters will be responsible for finishing the job, which includes at this point, managing spent nuclear fuel.

There is little assurance that Entergy itself will still be around 60 years from now or that the cost of decommissioning won't skyrocket between now and then. Vermont is counting on the NRC to play the role it needs to play and look out for our communities' interests, not just Vermont Yankee's. I intend to hold them accountable.

The NRC rules allowing for a 60-year decommissioning process were adopted many years ago. We now know more about nuclear plant safety and degradation issues than we did when the regulations were first promulgated. Yet, unfortunately, the Commission's regulations have not addressed those issues. I look forward to working with the committee to ensure the NRC's decommissioning process addresses the full span of the NRC's obligation to protect communities across the country.

I encourage my colleagues to cosponsor and support my bill (the Nuclear Plant Decommissioning Act) that would give States a seat at the table and to require the NRC to represent the interests of the communities that are dealing with decommissioning plants like Vermont Yankee.

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tures and materials beyond that necessary to terminate the license." 10 CFR 50.75(h)(1)(iv) n. 1.