UNITED STATES

NUCLEAR WASTE TECHNICAL REVIEW BOARD

Meeting of the Panel on the Repository

Disposal of aluminum-clad, highly enriched, DOE-owned spent fuel; vitrified high-level waste; and immobilized weapons-grade plutonium in a repository

December 17, 1997

Radisson Riverfront Hotel Augusta
Augusta, Georgia

NWTRB BOARD MEMBERS PRESENT

Dr. Daniel Bullen, Meeting Chair
Dr. John Arendt
Dr. Paul Craig
Dr. Alberto Sagηϑs
Dr. Jeffrey Wong

NWTRB STAFF

Dr. Carl Di Bella, Senior Professional Staff
Ms. Linda Hiatt, Management Assistant

I N D E X

PROCEEDINGS

8:30 a.m.

BULLEN: Could we all have a seat, please?

We have a very full agenda today, so I think we need to get started on time. My name is Dan Bullen, and I'm a member of the U. S. Nuclear Waste Technical Review Board, and I will be chairing today's meeting.

It's actually been several years since the Board paid a visit to Savannah River, so I'd like to take a few minutes outlining what the Board is, and after that, I'll make a few introductions and give a little background and the ground rules for today's meeting.

In 1982, Congress enacted the Nuclear Waste Policy Act. The law created the Office of Civilian Radioactive Waste Management, or OCRWM, within the Department of Energy, and charged OCRWM to develop repositories for final disposal of the nation's spent nuclear fuel and high-level radioactive waste.

Five years later, Congress amended the 1982 Act to limit OCRWM to characterizing a single site for final disposal, that being the Yucca Mountain site, which is a site 300 meters below Yucca Mountain on the western edge of the Nevada Test Site.

In the same 1987 amendment, Congress created the U. S. Nuclear Waste Technical Review Board as an independent agency to review the technical validity of OCRWM's program, and to periodically furnish the Board's findings, conclusions and recommendations to the Secretary of Energy, to Congress, and to the public.

The U. S. President appoints our board members from a list of nominees submitted by the National Academy of Sciences. Each nominee shall be imminent in his or her field of science or engineering and shall be selected solely on his established record of distinguished service. A full board consists of eleven members.

As you see by the overhead here, our member come from a wide range of organizations and institutions, and I'll go through their backgrounds briefly. As I do so, I will ask the ones present today to either stand up or raise their hand.

Our Chair is Jared Cohon. He's the President of Carnegie Mellon University, and holds expertise in environmental systems analysis and hydrology. He is a Registered Civil Engineer.

John Arendt--John, would you recognize yourself--began his career in the Manhattan project in 1945, and works in nuclear materials transportation and nuclear materials facilities, their quality assurance, quality control, and inspection. He is a chemical engineer, and as I noted, he is here with us today.

My name is Dan Bullen, and I coordinate the Nuclear Engineering Program within the Department of Mechanical Engineering at Iowa State University. My expertise is in nuclear waste management, performance assessment modelling, and material science. I'm a nuclear engineer by training.

Norm Christensen, who is not here today, is the Dean of the Nicholas School of the Environment at Duke University, and is an expert in biology and the ecology.

Paul Craig is Professor of Engineering Emeritus at U.C. Davis, with expertise in energy policy issues associated with global environmental change.

Debra Knopman, who is not here today, directs the Center for Innovation in the Environment at the Progressive Foundation in Washington, D.C., and has expertise in hydrology, environmental and natural resource policy, and systems analysis.

Priscilla Nelson, who actually chairs the Repository Panel which is hosting this panel meeting, is also not with us today. I'm acting on her behalf as chair of this meeting. She is the Program Director of the Directorate for Engineering at the National Science Foundation. Her expertise is actually in rock engineering and underground construction.

Dick Parizek is Professor of Geology at Penn State and specializes in hydrogeology and environmental geology.

Alberto Sagηϑs is Professor of Materials Engineering within the Civil Engineering Department at the University of South Florida. His expertise lies in corrosion and materials engineering, physical metallurgy and scientific instrumentation.

Jeff Wong is a toxicologist at Cal. EPA, specializing in risk assessment and scientific team management.

The Board is supported by ten professional staff and six clerical staff. Most of the staff are full-time federal employees, but some are part-time. Except for Priscilla Nelson, who is a full-time federal employee, Board members are part-time federal employees.

Also sitting at the front table today is Dr. Carl Di Bella. Carl, would you acknowledge yourself here? Carl is a chemical engineer. He is one of the members of the Board's professional staff, and has done a lot of the behind the scenes work necessary for today's technical program.

Dr. Dan Metlay also did a great deal of work, and unfortunately will not be able to join us today. He's another staff member from our Washington office.

I would like to acknowledge and thank another staff member who does a Herculean effort, Linda Hiatt in the back of the room. Would you raise your hand, please, Linda? She's responsible for putting together the logistics of today's meeting, and more importantly, she would like everyone to sign in so she can make you a name tag. Is that not correct?

One of the very important ways our Board goes about obtaining information needed to accomplish our mission is by holding public information gathering meetings. We hold three full Board meetings and some five to ten panel meetings yearly, and this is one of those panel meetings.

The full Board meetings are generally two days long, covering a variety of issues, and are attended by the entire board. Panel meetings are shorter, more focused, and are attended by Board members with specific interests in one or more of the topics to be covered. As a result of these meetings and our access to other relevant information, the Board makes assessments and recommendations about how to improve the technical and scientific aspects of the Waste Management Program.

A bit more information about the Board may actually be found on our web site, and this is a commercial for that site, the HTML, or the address for our web site is www.nwtrb.gov., and all of our reports are summarized and all of our calendar is there, so you can actually look at our calendar to see upcoming Board meetings.

Today's meeting is a meeting of the Board's panel on the repository. The purview of the panel includes everything that would be engineered at Yucca Mountain, including the surface and underground aspects of the repository, the waste packages that would go into the repository, and the waste form that the packages would contain.

There are a number of wastes at Savannah River that are destined for disposal at Yucca Mountain should the site be found suitable for the development as a repository. These wastes includes spent foreign research reactor fuel and vitrified high-level waste, among other things. We understand that the process, development and manufacture of vitrified surplus weapons plutonium will also be done at Savannah River. This waste again will be destined for disposal at the repository.

Now, if there's one single triggering reason for the meeting, it's the aluminum-clad, highly enriched uranium spent fuel, most if not all of which comes from research reactors. I'm the facility director for the Iowa State University research reactor, which has aluminum-clad spent fuel, and when I first saw the designs that were proposed by the OCRWM program for the direct disposal of spent fuel, I had three reactions, which were all caused by a combination of the material's high enrichment, coupled with its limited engineering stability of the cladding for long-term disposal.

The first is that the material clearly presents a heightened long-term criticality control issue within the repository. The second is that the material probably has a relatively high monetary value. It wasn't cheap to make highly enriched uranium. And, finally, both the monetary value and the potential utility for nuclear explosives of this material would make it attractive for future generations to retrieve, with motivations that are not necessarily coincident with those of the United States. These essentially could compromise, or these issues could compromise the performance of the entire repository long-term.

Now, this morning, we will focus on DOE-owned spent fuel, with a particular focus on the aluminum-clad, highly enriched spent fuel. This afternoon, we will move on to some of the other wastes that will be disposed of in the repository, specifically vitrified high-level waste from reprocessing and vitrified surplus weapons plutonium.

Now, just a brief background on the ground rules for today. I want to alert the speakers that brief questions from the front table, the panel members and staff, will be allowed during the course of the presentation. This is a little bit different than we have in our formal board meetings.

We keep it a little bit more informal to have interaction between the board or panel and the speakers. However, there are either 10, 15 or 20 minute time periods at the end that will be saved for essentially longer questions, and if time permits and we've exhausted questions from the front table, we would be happy to take questions from the audience during that time period. So please identify yourself and we'll recognize you for questions.

Now, I also want to point out that there is a public comment period at the end of the day, and Linda Hiatt, who we introduced in the back, has a special sign-up sheet for public commenters. Please sign up with her if you wish to make a statement or ask a question at the end of the day. We ask that the statements be limited to five minutes or less, and we will try to take the statements in order of sign-up on the sheet.

In all cases throughout the day, people asking the questions or making statements should use a microphone and should identify themselves, and that includes Board members.

I also want to say a few words about the Board's positions and the Board's pronouncement and member statements. The NWTRB, this Board, has a very important role. What it says is taken seriously by policy makers and the members of the public. The Board generally conveys its findings, conclusions and recommendations in written form in the form of formal reports, letters to Congress and/or the Secretary of Energy, or to the Director of OCRWM, and in written Congressional testimony.

Of course, the Board consists of several individuals, each of which has his or her own style. Each is free to say whatever they choose. But comments by individual Board members, including me, are just that, Board member comments.

On occasion, one of us, especially Chairman Cohon, will make statements on behalf of the Board that that person explicitly states to be Board positions. Otherwise, when we make individual comments, they are no more than that. Whether comments by Board members eventually become Board position, only time will tell. But, of course, a Board member's thinking is relevant. In effect, at these meetings when we make statements and ask questions, we are thinking out loud as a Board. Our thoughts and comments do not represent Board positions unless we indicate so, but they may be on their way to becoming positions. And if they do, we will convey them in writing.

Now it's time to get on with the meeting. Our first speaker is Howard Eckert, who is going to introduce the topic of DOE spent fuel. Before he starts, however, I want to particularly thank him for his efforts in putting together the technical program on the meeting today. It's not been a simple undertaking because the speakers in today's meeting come from many parts of DOE, and the organization is very diverse and it's a real difficult task to pull those together.

So without further ado, I will give you Howard

Eckert, and he will give us an overview.

ECKERT: Good morning. As Dan said, I'm going to briefly give you an overview of DOE-owned spent fuel, and this includes actually commercial fuel, not just the fuel produced by DOE's production and research reactors.

What I'll cover today is our current inventory, how much we have, where it is, what fuel will be coming into our inventory from operating reactors--we're still generating fuel today--some characteristics of the fuel, how we can distinguish or describe different types of fuel, the Department's strategy for dealing with the fuel, storage, transportation, eventual disposal, and then some ongoing efforts between the Office of Environmental Management and the Office of Civilian Radioactive Waste Management, cooperative efforts to move the fuel along the path toward disposal.

We use three metrics to quantify our fuel. Metric tons heavy metal is the traditional metric incorporated in some of our Congressional legislation. Of more relevance to storage would be the volume of fuel we have to deal with, cubic meters. For handling, the number of assemblies, or piece count terminology, is more relevant.

Most of our fuel, and we have roughly 2,500 metric tons heavy metal, or 1,200 cubic meters, most of this is stored at three sites, Hanford, Idaho and Savannah River.

Depending upon which metric one uses, one could say that Hanford has most of the fuel if the use the metric tons heavy metal, at 86 per cent. If you talk about volume, then Idaho has almost half of it, depending upon which argument you're trying to make.

This is the current inventory as of this month. It does not include fuel that we've made a decision to process. We are processing or reprocessing fuel at Savannah River.

There's some additional fuel at West Valley, commercial fuel, that's in storage, a small amount at Oak Ridge in the high flux isotope reactor spent fuel storage pool, and we've taken ownership of the fuel at the Fort St. Vrain facility in Colorado.

For planning purposes, we've blocked out a period of time of roughly 40 years, to the year 2035, which is about the time, for planning purposes, the repository is going to stop accepting fuel. During this period of time, we expect to receive fuel from up to 41 research reactors. Actually, they'll be fewer, since not all of the original participants are going to ship us fuel. We have operating reactors, as I said, at Oak Ridge, the high flux isotope reactor. We have a reactor, which I believe is not operating at the moment, but may again, at Brookhaven National Laboratory, the high flux beam reactor, and we have the advanced test reactor at Idaho.

There are about 30 research reactors at various universities around the United States, and the Navy of course is generating a good deal of our spent fuel, preponderance of the HEU.

I'm not going to go into all of the numbers, but for Savannah River, if we just look at the amount of fuel that we plan to receive at this site, if you look at volume, it turns out to be approximately a third of the fuel that will be shipped to one of the two sites. If you look at I guess the number of assemblies, it's probably more like two-thirds. These are the only two sites, Savannah River and Idaho, that we'll be receiving fuel in the future.

Foreign research reactor fuel is and has been coming to Savannah River. We expect next year, it will be shipped also to Idaho. Hanford will basically retain or keep the fuel it has in its possession, possibly shipping some of the fuel to Idaho for treatment. But all the other sites where we have fuel, we'll eventually consolidate that at Hanford, Idaho and Savannah River.

These are the characteristics. The first one is geometry. Geometry, the fuel meats, or the actual fuel contained within the cladding, the enrichment, the condition of the fuel, and what we call special case fuel, something particularly different about it; these are the general characteristics that we can use to describe fuel and to group it or put it into different categories.

For example, we have 250 different types of fuel within our inventory. To work with that fuel, that is, to analyze it and decide what to do with the fuel, how to treat it, how to store it, how it's going to behave in the repository, we group it into approximately a dozen different groups or categories so we can handle it more easily.

And what I've done is just given different representative fuel types that exhibit one or more of the characteristics I've shown here. For example, the N-Reactor fuel is the first one on the list. It consists of two concentric cylinders. It's a zircaloy-clad fuel element. I think it's roughly two and a half feet long. It has a uranium metal fuel meat.

This particular fuel, as most all of our production fuel, was designed and handled to be readily processed to obtain materials for weapons production. It was not meant to be stored long-term in our water basins. So this has experienced significant corrosion. It represents what we have under here condition, the degraded fuel.

The TRIGA fuel, the nomenclature, something about test reactor, I forget what the "I" is for--but it's made by--isotopes--General Atomic. This fuel is used in a number of our foreign and domestic research reactors. It's a cylindrical fuel. The cladding can be either zirconium, aluminum or stainless steel. One of the fuel meats is the zirconium hydride, and this is roughly two and a half feet long, generally in good condition, and the fuel we're taking back from foreign research reactors.

The Advanced Test Reactor is an aluminum-based fuel, aluminum clad with an aluminum alloy meat. It has curved plate elements. Again, I believe this is roughly two and a half, three feet long. This is from our reactor in Idaho.

Some of the fuel I'm not going to discuss. It will be discussed by Mark Barlow later in the morning. But this is the high flux isotope reactor, an aluminum fuel.

The Fort St. Vrain, I guess has a unique shape, it's a hexagonal block of graphite with holes drilled to accept the fuel rods, also to allow the coolant to flow through it. The fuel is called a fuel compact. It's actually a solid rod composed of uranium carbide pellets centered with carbon, and these are inserted into the holes within the block of graphite. These are in storage both at Idaho and at a dry storage facility in Colorado that I said the Department has taken ownership of. We are also now the licensee of that facility with the Nuclear Regulatory Commission.

The materials test reactor fuel is another element that Mark will discuss in greater detail, since these are coming into the Savannah River plant, and this is the bulk of the fuel assemblies we have to deal with at the Savannah River site.

The experimental breeder reactor is shut down. The fuel for the most part, or I guess all of the fuel is actually at Idaho. These are fairly long rods, about five feet long, about a half inch in diameter, stainless steel clad. And what's unique about this fuel is it has a sodium bond. There's sodium contained between the cladding and the fuel. We have about 60 metric tons, I believe, in total of sodium bonded fuel, not all EBR-II fuel. The blanket fuel contains depleted uranium.

Also what I guess I've neglected to mention earlier was the enrichment of most of this fuel. If we go back, the N-Reactor fuel has--

BULLEN: Excuse me. Could I interrupt you for just a second?

ECKERT: Sure.

BULLEN: We have a distinguished visitor in the back, Secretary Pena.

PENA: I heard you were having a meeting, so I thought I'd just come on down.

Please sit down. Please relax, Ladies and Gentlemen. I normally don't barge in on people's meetings. Let me just say to the Board members how much I appreciate your very hard work.

I spent yesterday touring Savannah River and looking at the tritium facility and waste processing facility, and drove by the canyons and other things. So as all of you know, this is a very important group of citizens who have volunteered their time to give us some guidance in the Congress on how we can think of complex-wide issues, and we very much appreciate their recommendations, your ideas on how we can deal with Yucca Mountain, which we have responded to, and suggestions of doing an east-west cut, and other things.

So I just wanted to drop by and say thank you very much. I happened to be talking to some business people next door and heard you were here, so I wanted to come by and thank you for taking the time to be here and to do this kind of work. So I appreciate it.

Thank you. Happy Holidays to all of you.

BULLEN: Thank you, sir.

ECKERT: What I was going to do is recap the enrichment, which I had neglected to talk about. The N-Reactor fuel is very low enriched, on the order of 1 per cent. TRIGA varies from--we still call low enriched uranium anything up to 20 per cent, and TRIGA starts at about 20 and goes up to 93 per cent enrichment. The advanced test reactor is 93 per cent. Fort St. Vrain is 93 per cent, and EBR-II is, I'm not sure, I believe that's also HEU.

Well, let me finish up with this one. The EBR-II fuel, as I said, was sodium bonded, and we're looking at means for removing the sodium. The one treatment technology under development at Argonne National Laboratory, both East and West, is the electrometallurgical processing technology, and that is actually undergoing a demonstration today and it shows great promise for processing our more difficult fuel, such as the EBR-II fuel.

The EBR-II fuel is low enriched. Thank you.

Okay, what else might I have missed. The Three Mile Island Unit II core debris is in canisters at Idaho. And I think I've covered the other examples here.

Now, we have roughly 2,500 metric tons of fuel on site, more coming in. Dealing with this fuel, both the storage, transportation and ultimate disposal, is the primary activity of the group that I work for, Office of Spent Fuel Management, in cooperation with other offices within the Department.

Several years ago, in '93, the Office of Environmental Health and Safety at DOE conducted a vulnerability assessment and published their working group report, which listed 105 vulnerabilities with the storage of spent fuel throughout the complex. As part of our effort, we put together a plan of action to resolve these vulnerabilities, and particularly those that were of higher priority, such as the fuel in storage, the N-Reactor fuel in storage at the K-Basins. Approximately half of these vulnerabilities have been completely resolved. The others are taking much longer, due to the cost and time required to deal with the fuel.

One of the means for resolving the vulnerabilities, besides putting it into safer, longer term storage, dry storage, is to process it, and the production reactor targets and fuel at Savannah River are being reprocessed. Actually, the targets have been completed. The step ongoing now with RW is to understand what will be required to prepare that fuel for ultimate disposition.

To understand better what's required, we have decided we were going to work more closely with RW, and we have. Over the past two years, we've been able to--actually, the memorandum of agreement, the standard contract that RW has with the commercial utilities is an agreement we have about ready to be signed. It's completed. It's just waiting for some final concurrences. And this describes the conditions under which RW will accept DOE-owned fuel.

RW and the contractors have actually done some of the analyses, criticality analyses on DOE-owned fuel. We felt although we could have done it, it was expedient initially to have the RW/M&O contractors do it, since they know exactly what has to be done, and we work closely with them.

The key documents coming up in this year, the Viability Assessment and the Total System Life Cycle Cost Evaluation, will explicitly contain information on DOE-owned fuel. There are a number of requirements documents that go into greater detail on what DOE has to do with the fuel to have it accepted by RW.

There is an interface control document now in review that gives the sort of bounds perhaps on geometry, for example, that we must meet for the equipment and storage envisioned by RW in the repository. And to better understand all of these items, learn where RW is, where EM is, we've been holding semiannual strategy meetings of several days where we go into greater detail about what's in the documents, better understanding of how we're going to meet those requirements, and planning what we have to do as a group in the next six months in order to move firmly along that path toward disposal.

As I say, DOE-owned fuel is primarily owned within DOE by the Office of Environmental Management. I guess the Navy owns the next largest chunk. The Office of Nuclear Energy owns some, particularly that which is being stored at the operating reactors, HFIR, HFBR. But as the reactors get shut down, the operating entities within DOE want to put it into a caretaker mode, and that's where EM is taking more and more of the fuel.

That's all I have. Are there questions?

BULLEN: Thank you, Howard. Questions from the Board? Carl Di Bella?

DI BELLA: You showed on your overhead for the N-Reactor fuel, that it has zircaloy cladding.

ECKERT: Yes.

DI BELLA: And yet a substantial amount of that fuel is degraded, I understand.

ECKERT: Yes.

DI BELLA: Zircaloy really shouldn't degrade in the kind of chemical environment that that fuel has been exposed to. Do you know what has caused the degradation? Is it a corrosion phenomena of the zircaloy?

ECKERT: It is, but not primarily the zircaloy, but of the uranium meat within. When they unloaded the N-Reactor fuel, it was, as I understand it, just basically tumbled down a chute into bins to be collected, so it wasn't handled carefully. The cladding was damaged, breached, and where the uranium metal is exposed to water, it naturally corrodes rapidly.

You might be able to see some corrosion. This isn't one of the worst ones, but the cladding would tend to peel away from the meat as the corrosion products expand. I guess I didn't bring another one that had a more graphic description, but these are the concentric cylinders within a canister in which it was stored in the N-Reactor spent fuel storage pool.

BULLEN: Jeff Wong, Board?

WONG: I just have kind of a general question. We've been waiting to hear about the DOE's waste isolation strategy. What role does the waste isolation strategy have in your planning of your management of your fuel forms, or what input do you have into the waste isolation strategy in dealing with your fuel forms?

ECKERT: Waste isolation means you're referring to the ultimate disposal. I guess our input is to work with RW to make it reasonably cost effective to dispose of our fuel. We certainly are not going to dictate a disposal criteria. That's their job. We are almost in the role of a commercial utility. We have fuel that we need to dispose of, so we have to live within the RW requirements. But as they evolve, the documents that are written, we get to review and comment on, so that we can suggest changes perhaps that would make it easier and perhaps more economical, safer for us to dispose of the fuel.

BULLEN: Any other questions from the Board?

(No response.)

BULLEN: Thank you very much, Howard. We're right on schedule.

Our next speaker is David Curtis. He is the Director of the Reactor Materials Division in Naval Reactors. In that capacity, he is actually responsible for the program's expended core facility at the INEEL, where Naval reactor cores are examined and where Naval spent fuel will be prepared for ultimate disposal. And David will speak to us about Navy spent fuel.

CURTIS: Thank you. I'm here today to talk about Naval spent fuel and how it fits into the DOE program. Some of the introductory material I'm going to talk about is going to be very similar to the material that Rich Guida from the Naval Reactors Program presented to a panel of the Board about a month ago. He focused mostly on transportation. I'm going to focus mostly on the repository aspects of it.

Details of the Naval spent nuclear fuel are classified, but the results of the analyses and tests that we have done on this are unclassified and can be discussed in a public forum such as this. More detail than I can present in the 20 minutes that I have today is provided in an unclassified basis to the appropriate parts of the various organizations involved, and detailed technical backup information is presented to people that have clearances, such as John Arendt from the Board, shortly hopefully Dan Bullen from the Board, Carl Di Bella of the Board staff, and various regulatory agencies, such as the Nuclear Regulatory

Commission.

Let me take a moment to introduce the Naval Nuclear Propulsion Program. The Naval Nuclear Propulsion Program is a joint program of the Department of Energy and the Navy. We are a part of both agencies, and the ships that we provide reactors for are a key part of the Navy's defense mission.

CNN regularly tells the story of these ships. As we're sitting here today, the Nimitz and the George Washington, two nuclear powered aircraft carriers, are in the Persian Gulf. Both of them sprinted to the Persian Gulf recently to provide a military presence. The Nimitz came from Hong Kong. The George Washington was in Haifa. And being able to make the high speed transits and arrive at station ready for the mission is a key part of the Navy's story.

The submarines that we have don't get the CNN coverage, but submarines are in most of the world's trouble spots also.

The Naval Nuclear Propulsion Program is about 40 per cent of the Navy's principal combatants. We have about the same number of total reactors as the civilian commercial nuclear power industry. We have between 110 and 120 now. I think the commercial nuclear industry has a little fewer than 110.

We have 4,800 reactor years of safe operation without a reactor accident. That's about twice the--actually more than twice the accumulated experience of the civilian nuclear industry. I think the civilian nuclear industry now is between 2,000 and 2,500. The fact that ours is without a reactor accident is really one of the keys as to why we can take our ships with these nuclear powered reactors into over 150 ports in over 50 countries worldwide.

This slide, which is in the pass-out, gives a few more statistics. I'm not going to spend time dwelling on it. You can review the statistics later; just a few more statistics about the program.

I would like to get on to talking about what Naval spent nuclear fuel is. Naval spent nuclear fuel is solid. It's metallic. It's not flammable. It's not hazardous. We have done a TCLP test and gotten EPA agreement that it is not hazardous. We have that certification.

Naval spent nuclear fuel is--or Naval fuel is built and designed to be operated on war ships, and as such, it needs to be very rugged. The design requirements for the fuel are well in excess of 50 g's, which is considerably more than the other types of spent nuclear fuel that you're talking about here. The Naval fuel fully contains all of the fission products and the long-lived radioactivity, and it operates for very long periods of time.

Most of the ships that we have now have cores that are operating for about 20 years between refuelings, some longer. I'll get to that in a minute. But they operate at power for long periods of time and they operate in very close proximity to the crew. Sailors live, eat, sleep literally within feet of these operating reactors. The reactors have to be designed to be able to take rapid power trangents. The commanding officer of the ship may decide he wants to stop, sprint, et cetera, so the power trangents that they're designed for are much, much faster than commercial power trangent. So the whole design philosophy of the Naval fuel drives us to come up with very, very rugged reactor cores.

We periodically take ships and do shock tests of the ships. Let me move this up on the screen a little bit. This is a picture of a shock test of the Theodore Roosevelt, one of our carriers. What this represents is tens of thousands of pounds of high explosive being detonated about a ship length away from the ship. That rocks the ship pretty well. The reactor cores come through this kind of an experience very straightforwardly, very easily.

Key to what we say about the behavior of fuel in a repository is our knowledge of the fuel. We have very detailed knowledge of the fuel through the manufacturing process. After the cores are made, they go through a design and a manufacturing certification. Then they go through detailed acceptance testing. The testing is followed in detail, and the operation through core lifetime is followed in a great deal of detail, so that we know very well what each and every reactor core is, how it was made, how it was designed, how it's operating, how it's behaving.

After we're done with the cores, after they get removed from the plants, from the ships or the prototype reactors, we then take them all to the expended core facility that Dr. Bullen mentioned at the Idaho National Engineering and Environmental Lab, INEEL, where we examine each of the cores to basically confirm that the performance of the core was in fact what we expected. On some limited number of cores, we do destructive, more detailed examinations to understand more.

In addition to that, we have an extensive radiation test program, and have had for many years. We've used MTR, ETR. We're currently using ATR, where we thoroughly explore the failure modes of our fuel. We think we know pretty well what causes them to fail, where the limits of mother nature are, what doesn't cause them to fail, et cetera.

The key to much of what we know and think about the behavior of our cores is based on the examinations that we do in Idaho of the spent cores.

There's a very powerful economic and a very powerful operational incentive to have long lifetime cores, but we couldn't do it unless we were confident technically that the cores could last. Our first core, which was put into the Nautilus in the early Fifties operated for two years. Our newest ship, the Seawolf, which was commissioned in July of this year, has a core in it that we expect to last the 30 year life of the ship, will never be refueled, will have only one core, will be the core load for that ship. That's a big improvement, both in terms of just the availability on line, the cost and economics in terms of buying the cores and maintaining the cores, and also in having the reduced number of cores to worry about for reprocessing.

Before 1992, after we had examined the cores at the expended core facility, they were sent to the Idaho Chemical Processing plant, where the unused uranium 235 was recovered. In 1992, reprocessing of our cores at ICPP was stopped, or the decision was made. Since then, we're in kind of a transition phase. We've been preparing our fuel for repository disposal. We've got some construction projects underway to facilitate that in Idaho.

What we intend to be doing is we intend to basically canisterize the fuel. In '96, we had an EIS that was published where we looked at various kinds of container systems. We had a Record of Decision in December of 1996. We selected a Dual Purpose Canister. What we intend to do is to put the fuel in canisters in Idaho. We will be storing it temporarily in Idaho until either an interim storage site or a repository is open for it. Then we will be responsible for transporting it to either the repository or an interim storage site when one is open.

The DPC system is currently in a design stage, and we're working on that. It's our goal, it's our intention that once these canisters arrive at a repository, only the canister needs to be handled. The fuel won't need to be handled. We can't really make that claim on the record until the acceptance criteria for the repository get established, but we expect and it's our intent to try to make the DPC sufficient for emplacement in the repository.

Let me talk some about the amount of Naval spent fuel that we're talking about here. As Howard mentioned, there are a variety of different metrics. There are a variety of different coins of the realm. The one most frequently used is the metric tons of heavy metal.

We currently have 14 metric tons of heavy metal of Naval spent fuel. By 2035, we are projecting 65 metric tons of heavy metal. I have on this slide also the DOE non-naval fuel now and in 2035, and an estimate of the commercial spent nuclear fuel now and in 2035. The small amount of Naval fuel is really largely because we have fairly small reactors, and very infrequent refuelings.

So ultimately, by the coin of the realm, we'll have about a tenth of a per cent of the spent nuclear fuel in the country, about a tenth of a per cent of the amount of spent nuclear fuel that is intended for the first repository.

Other possible metrics are total weight or volume or piece count, as Howard mentioned. The 65 metric tons represents volume of canisters, or weight of canisters. These number over here are volume and weight of the waste form within the canisters. We'll have about 300 canisters of Naval spent fuel, which is I think about 3 per cent of the number of canisters, the piece count of canisters for the repository. Each canister will be about 66 inches in diameter, some slight variation in lengths, but the longest ones will be 212 inches.

We've at this point calculated several aspects of the performance of Naval fuel in a repository. The next several slides summarize some of the key results that we've come up with.

Because the design and manufacturing of the fuel drives us to build very, very rugged fuel, we calculate on a best estimate basis that the fuel will remain substantially intact in a repository environment for in excess of a million years. In that million year period of time, the radioactivity in the fuel has decayed by more than four orders of magnitude.

Specifically, the best estimate prediction is that the cladding will not be penetrated by corrosion on any of the elements; that the best estimate prediction is there will be zero corrosion through the cladding in a million years in any of the elements. Now, this doesn't assume that the canister provides a whole lot of protection. We're assuming that 10 per cent of the canisters are degraded and ineffective at about 3,000 years, and the rest of them are degraded and ineffective at 10,000 years.

We further conclude that the basic fuel assembly geometry is maintained intact for in excess of a million years.

As a result of this, the only radioisotopes that we expect to be released to the drift are from the crud layer that's on the fuel elements, and small amounts of impurities and activated elements in the cladding that corrode, and as soon as the cladding does corrode, to the extent that it does corrode, we assume that all of that material is released to the drift.

The peak release rate is much less than a curie per year. It is carbon-14 and it occurs relatively early in the repository life.

Now, although we don't anticipate it, we have calculated so far two cases that are hypothetical cases where if there is something that happens to expose the fuel material itself, what would the consequences be. In one case, what we calculated was a mechanical damage, such as from a rock fall, that physically breaks the canister, or what's left of the canister, breaks any of the internal support structure, and shears through our fuel.

The other case, what we assumed is if we had enough corrosion so that it corrodes through the fuel by whatever reason, and that's, you know, beyond the 99.98 percentile on this, in this case, we would be exposing 1,000 elements, in this case, we'd be exposing a couple hundred elements. In neither case are we contributing significant uranium, or for that matter, significant fission products to the drift.

We've also done nuclear criticality calculations for a number of cases. We do use highly enriched uranium in our Naval reactor cores. The fact that we use highly enriched uranium is not a show stopper, we think. It turns out the amount of U-235 per container is about the same amount as the amount of U-235 per container of commercial spent nuclear fuel. So we're not putting a lot more U-235 in there.

As a matter of fact, the fact that we don't have all the 238 at the beginning of life of our reactor cores means that to a substantial extent, we don't need to worry about plutonium. So we do have what we think is a tractable amount of uranium-235.

In addition, though, we are fixing with each of our fuel assemblies hafnium control rods that are permanently attached to the assembly that will provide a nuclear poison for a long time. Hafnium corrodes even less than zircaloy. We think the hafnium control rods are going to be a nuclear shut-down presence for far in excess of the million years.

We've analyzed a number of cases with intact containers, degraded containers, degraded structural supports within the containers, damaged containers, a variety of assumptions on flooding, partially flooded, fully flooded, preferentially flooded, et cetera, et cetera, and in all of these cases, we think we can show that as long as we maintain the basic geometry of our fuel assemblies, that we have adequate nuclear shut-down from a criticality point of view. And we think that the basic geometry of the assemblies will be maintained for in excess of the one million year time that we're calculating here.

DI BELLA: A quick question, please, Carl Di Bella here, on the hafnium control rods. Are they the control rods that are actually part of the core, or are these new hafnium control rods that you put in?

CURTIS: The answer is some of both, probably, a mixture. We will use hafnium control rods where we have hafnium control rods. We will put some new hafnium in where we don't.

DI BELLA: How much does hafnium cost?

CURTIS: I'd hesitate to answer that off the top of my head. I can get you an answer. I guess I would say that this whole adventure is not cheap, but it's part of the price of doing business.

Finally, let me talk about decay heat. Decay heat is just not a problem. The decay heat per container for us is about half of the decay heat of the spent fuel of a typical commercial spent fuel container.

The peak repository heat load in any year from all of the Naval containers is about 700 kilowatts. That's about 500 to 600 of those hair dryers that are in the hotel rooms up here. So it's not a tremendous decay heat load.

For about three years now, we've been interacting very closely with RW. It kicked off in November of '94. Some of us had a meeting with Dan Dreyfus and Lake Barrett, and since then, we've had a very close relationship with RW and with EM in terms of making sure that the things that the DOE is doing for the repository are in fact compatible with incorporating Naval spent nuclear fuel.

I'd like to highlight just a couple of things on this slide. The previous four slides, I gave some summary results of corrosion and degraded conditions and nuclear criticality and decay heat. All of that, and more, was provided in a document about an inch thick in July to the Yucca Mountain Site Characterization Office with detailed calculations for use in their EIS. So that information has been presented. It's unclassified. Copies of that were made available to the Board and the Board staff.

The second thing, as Howard mentioned for EM, we are also working on a memorandum of understanding with RW. What that will do is that will nail down the details of the interfaces between us and RW to make sure that all aspects of controlling Naval spent nuclear fuel are adequately controlled as we go to the repository.

The third thing I would mention is that our principal prime contractor that supports us on this, the Bettis Atomic Power Laboratory, has opened a field office in the Yucca Mountain Site Characterization Office/M&O facility, so that we have an on-site presence in Las Vegas working closely with the M&O to make sure that we have coordination, cooperation, communication, et cetera, et cetera.

We also, it was on that slide and I glossed over it, but we will provide the detailed analyses of Naval spent fuel to the DOE so that DOE can use those in the license applications to the Nuclear Regulatory Commission for the repository licensing process.

In summary, I guess my points are that the same attributes that drive us to design and build reactors that are rugged for operation in extreme conditions, maneuvering, battle shock, et cetera, et cetera, that are suitable for long-term operation in an enclosed environment like a submarine, with the crew in very close proximity, living in close proximity for months at a time, those same characteristics of the fuel, although it wasn't our motivation in designing it this way, turns out make the fuel almost uniquely--maybe I shouldn't say uniquely, but at least very suitable for repository emplacement.

Furthermore, our knowledge of the design of the reactor, the manufacturing of the reactor, the operation of the reactor, the post-irradiation examination of the reactor, and our knowledge of the behavior of the fuel system from our radiation test data lead us to believe that we have a solid basis for predicting the performance of our fuel in a long-term environment.

As I mentioned, we've got the detailed design of a number of things going in progress. We've got tests and analyses that are in progress right now. But we're confident at this point that the defense in depth that we have in our fuel and in our understanding of the fuel will basically confirm the situation that we think we have now where the strength and the integrity of the fuel will be maintained in excess of a million years. We believe we will show that.

The hafnium will remain in excess of a million years and will in fact provide adequate nuclear shut-down, and that the total releases of fission products and U-235 from Naval fuel will present a very insignificant contribution to the total dose rate coming from the repository.

That's all I have. Are there any questions?

BULLEN: Thank you, David. Questions from the Board? John Arendt?

ARENDT: David, I think you showed the corrosion--or you didn't show the corrosion rate of zircaloy, but I believe you have corrosion data for some 30 or 40 years that you're basing your conclusions on, aren't you?

CURTIS: Yes. And as a matter of fact, yesterday and the day before, one of our materials scientists from the Bettis Atomic Power Laboratory was presenting a paper on zircaloy corrosion--I'm not sure which of those days, he didn't do it both days--but on one of those days, there's an expert elicitation panel this week somewhere on the West Coast, and we're presenting a paper on zircaloy corrosion at that meeting.

Basically, what we have is a lot of experience out of in pile irradiated corrosion. We have tens of thousands of samples that we have irradiated, most of them in the advanced test reactor, for periods of time up to--well, ATR is now almost 30 years old, it's about 30 years old. There are some of those specimens that have been in almost from the inception. We have specimens that have been in for about 30 years in autoclaves at the Bettis Atomic Power Laboratory, and we have some--I mentioned the expended core facility--some of our earliest cores we've been keeping around as library samples just because there are unique sorts of things. A few years ago, we took out a core that had been sitting in the water pits at ECF for 28 years with no appreciable degradation of that.

But we have what I would characterize as good reliable engineering data at about the 30 year point, both in pile and out of pile. What we believe is that for the repository situation, the out of pile is probably more pertinent.

SAGγIS: Sagηϑs, Board. On the corrosion matter again, the kind of environment that you would expect in the repository would be significantly different perhaps from some of the environments for which you have long-term experience. The accumulation or potential accumulation of chloride ions, for example, in the water, the concentrations that may exceed by several orders of magnitude where you may have experience would be an example of that.

Indeed, in a number of highly corrosion resistant materials, including super alloys and so on, have been looked at for the outer waste package, and for those materials, estimates of penetration after a million years would be considered very optimistic.

I just wanted to know a little bit more about how they arrive at those estimates of, for example, one million years of penetration time. And you mentioned some percentages when you showed the transparency, the one on calculation results. Maybe you want to put it up again. That was the million years, like 1 per cent or 10 per cent.

CURTIS: This one?

SAGγIS: Yes.

CURTIS: I think perhaps what I mentioned was on this one, I mentioned the 99.98 percentile there, in the case of the accelerated corrosion. Let me address first your point on the fact that the environment in the repository is not likely to be the environment that we have our long-term data on. You're absolutely correct on that. We've looked very carefully at the environment that may be there.

We just recently started some tests at Livermore in the J-13 water. We're doing some tests in concentrated J-13 water, but those have literally just started. So we don't have any real data on those.

We've looked at the types of chemical species that can be there. We've looked at the way that that could impinge on our waste form, and we think, and this gets into a little bit of the classified aspects of it, so I can't discuss this in great detail here, but we think that the bottom line is still going to be true. Even if there is some local penetration because of a persistent drip or something like that, we think that is going to be very well bounded by the types of analyses that we've done that assume much more than very localized penetration in one spot.

SAGγIS: And that is estimated at 99.98 percentile for how many years?

CURTIS: This is for a million years. This is projected out to a million years.

SAGγIS: That's for a million years.

CURTIS: And the first penetration of that at that rate comes through at about 300,000 years, and so there is corrosion from the 300,000 year point to the million year point in this. There is exposure of the fuel material to the drift environment for 700,000 years there.

SAGγIS: And that is ascribed primarily to the corrosion resistance of the zircaloy that would be surrounding the elements?

CURTIS: Substantially, yes.

SAGγIS: Substantially. And what kind of thickness is that?

CURTIS: That's a classified number I'm afraid I can't discuss. I would say that we have provided that information to John Arendt and Carl Di Bella.

BULLEN: Paul Craig?

CRAIG: Craig, Board. Yes, I'm struck by the difference between your million year time frame and the 10,000 year type of time frame from the DOE designs for the external canisters. This has to do with the materials that you're using. Can you tell us something about the nature of the materials that gives this two order of magnitude longer time frame for corrosion?

CURTIS: Well, zircaloy, which is the cladding that we use and is heavily used in the structural material of our reactors, was designed specifically as a non-neutron absorbing, corrosion resistant material. As a matter of fact, zircaloy was invented at Bettis Atomic Power Laboratory back in the late Forties or early Fifties at the start of the Naval Nuclear Propulsion Program, and it was designed and it's been tweaked through the years to make it more and more corrosion resistant.

Zircaloy 4 is what's used today in industry substantially, and zircaloy 4 is better than the earlier versions of zircaloy.

One of the characteristics of zircaloy corrosion is that the corrosion film tends to be very tightly adherent to the base metal, and tends to form a protective oxide film on the base metal. So that actually for a substantial period of time, the zircaloy corrosion protects the underlying base metal.

After a while, zircaloy goes through transition, and then you get a steady state linear corrosion rate with time that's very well characterized. We've done a thorough analyses, electron microscopy, ta-da, ta-da, ta-da, to study just how that layer builds up, how it's structured, why it behaves as it does. The details of that really are substantially in this paper that I mentioned as being presented yesterday or the day before out on the West Coast.

That paper, I haven't seen a final published copy of that paper. Once there is a final published copy of that paper, I have already told Carl Di Bella I would make copies available to the Board.

BULLEN: Di Bella, Board?

DI BELLA: Switching subjects a little bit, you mentioned you used to, until 1992, reprocess spent Naval fuel. Where did the uranium go that was recovered from that reprocessing operation? Do you know? And were any of the other actinides recovered for any purpose, as far as you know?

CURTIS: I don't know the answer to that directly because Naval Reactors wasn't responsible for operation of ICPP, so I don't know the answer about the actinides specifically. I could try to find out the answer.

Before 1992, I was in a different job than the one I'm in now, and so I didn't even have the contact with ICPP that I have now.

In terms of the ultimate use of the U-235 that was recovered, my understanding was that was provided to the DOE for the weapons production reactors, I believe principally Savannah River, but some may have also been provided to other DOE reactors. But the ICPP is run by a different part of DOE than Naval Reactors.

Before 1992 when we gave them our fuel, we thought we were done with it. Now, it turns out there is some inventory of Naval fuel at ICPP, and on one of the slides I showed, it talks about a second Record of Decision in April of 1997. We have decided and have DOE concurrence that we will bring that Naval fuel back to the expended core facility and we will canisterize it and prepare it for repository disposal. So we are retaking possession of that fuel that was in the back log when the decision to stop reprocessing was made.

BULLEN: Bullen, Board. One last quick question. You talk about the 300 canisters and their size and their dimensions. Do you have any data or calculations on the surface radiation dose rate that you expect those canisters to have?

CURTIS: For transportation, there is a transportation overpack, and I don't have that number off the top of my head. If I can ask for a voice from the audience, is there a number?

Less than 10 milligrams per hour at 2 meters with a transportation overpack.

BULLEN: That's the transportation. Now, what you're assuming is that these containers would then be overpacked again at the repository?

CURTIS: We're assuming that there will be some sort of repository overpack or something for handling at the surface facility and to transport them into the drift. And in the October Board meeting, there was a discussion of how they would then get unpacked from the overpacks once they're down into the drift, so that as they sit in the drift, they wouldn't have that repository overpack there, is my understanding.

BULLEN: Well, that's the follow-on question, is do you know the radiation dose in repository? Has that been calculated?

CURTIS: It has been calculated. It will be high around the sides of the container. It won't be high on the ends of the container. The ends of the container will be shielded. They'll be welded. And so there's enough shielding built into the conceptual container design so that we can go do the welding. So it won't be high at the ends, but it will be high at the side, and I can't off the top of my head give you a number. I can get that number and give it to you later, but I don't have it off the top of my head.

BULLEN: I guess the key to the calculation would be the spectra that you'd expect to see. Primarily the cesium and strontium are the dominant gamuts, but after a few half lives, they're gone and you end up with the neutron spectra, which is going to be significantly different than the neutron spectra you'll get from commercial fuel. And so I'm interested in the concept of self-shielded packages, which means that you could have worker access in drift, and so that's why I'm asking the question.

A follow-on question essentially would be in dry storage, have you also done the calculation on the radiation associated with the near-field and any potential radiolysis effects and degradation products that may end up accelerating the degradation of the clad if you're making nitric acid at the surface?

CURTIS: In dry storage, we expect them to be dry. So we don't expect to be making nitric acid at the surface.

BULLEN: Zero per cent relative humidity dry storage; is that--

CURTIS: We are going to back fill them with inert gas, yes.

BULLEN: Okay.

CURTIS: We will dry them and back fill them with inert gas.

BULLEN: I guess in the long-term analysis then, if your package does breach and you have moist air in there, do you still expect the million year performance of the clad, is kind of the bottom line question?

CURTIS: Yes, the bottom line question is yes, we assume that the package will breach. We assume that the package will breach in the repository. We assume that the package will breach. We assume that when the package breaches, we are putting stuff, which is other than nice pure water, into the proximity with the fuel assemblies, and we expect that we will have the million year performance under those conditions.

BULLEN: I look forward to seeing those analyses if I ever get cleared.

CURTIS: Yes, sir. We look forward to presenting it to you.

BULLEN: In an effort to keep us on schedule today, I think what we'll do is move right on to our next speaker.

Our next speaker is Mark Barlow from the Westinghouse Savannah River Corporation, and Mark will speak to us about the aluminum clad, highly-enriched spent fuel.

BARLOW: Good morning. I'm the manager of the Alternate Technology Program for Westinghouse and DOE Savannah River. My job involves the ongoing activities in performing tests, analyses and engineering studies for preparing the spent fuel, specifically the aluminum clad spent fuel, for interim dry storage at Savannah River and ultimate disposal in a geologic repository.

There are two other speakers who will be speaking after me this morning on topics related to the engineering and the science behind alternate technology. My objective with my presentation is to introduce in a little bit more detail than Howard did a description of the aluminum clad fuel, its corrosion characteristics. I'll also describe to you the quantities and locations of the spent fuel today, and the projections for the future, and I'll also address the management strategy that we have developed for, again, the interim storage and ultimate disposal of this fuel.

The diagram to your left are two examples of typical configurations of the majority of the spent fuel that we refer to as research reactor spent nuclear fuel. 80 per cent of the fuel assemblies that we will be dispositioning are constructed of flat or curved or involute plates, as shown in that diagram. The box type shown on the left side of that diagram is the most common configuration. We refer to it as the material and test reactor, or MTR assembly.

Typically, the majority of them are a little over a meter in length, about 8 centimeters in width, and consist of somewhere in the neighborhood of 25 flat or curved plates. The typical loading of uranium in assembly as shown there is about kilograms. The enrichment can vary anywhere from low enrich, below 20 per cent, but more commonly for the majority of that fuel type, they are highly enriched, in many cases, above 90 per cent enrichment U-235.

The involute type that's shown to the right, the majority of that type is what Howard referred to earlier as the high flux isotope reactor, or HFIR cores that have come from the Oak Ridge HFIR reactor.

The length of that element is a little bit less than a meter in length, a diameter of about 43 centimeters, and it is a much heavier element with about 10 kilograms of uranium in the uranium oxide core.

The spent fuel core material for the aluminum fuel is a two-phased alloy comprised of a dispersement of uranium aluminide, oxide or silicide and uranium pure aluminum matrix.

Typically the core plate thickness is about a half of a millimeter and is enclosed in cladding that's about three-eighths of a millimeter thick. Generally the fuel that we receive from the various research reactors is in very good condition, and in Dr. Iyer's talk, he will show some pictures of the condition the fuel is in.

There are instances of aluminum fuel having corrosion difficulty. Howard referred to those earlier. Those in our basins generally are the production fuel related targets or fuel assemblies that came out of our production reactors at Savannah River site, and those are being dispositioned, as Howard said, through the canyon reprocessing process.

I've already mentioned that the enrichments for most of those fuel types are highly enriched, and in many cases, more than 90 per cent.

The time in the reactor varies greatly. Some of the fuels, essentially fresh fuel, there are some instances of fuel failure and removal of the fuel and put into temporary storage, but typical burn-up is in the range of about 30 per cent.

The thermal out put for the fuel assemblies also covers a fairly broad range, but in general, it's very low, and a typical number you can put it on a hook is about 10 watts per assembly.

Because it is a two-phased aluminum alloy, the corrosion behavior is very similar to standard aluminum. The onset of corrosion is very much a function of the environment of which it is stored, and aluminum by nature is very susceptible to humidity and the temperature of the storage environment. However, we have found by experience that after the onset of corrosion, there's a formation of a thin oxide layer of the order of 10 to 50 microns in thickness, and that essentially provides a protective coating, that as long as it is maintained, it significantly reduces ongoing corrosion rate.

When the cladding is breached and the fuel material is exposed, the release rate of the fission products is, as compared to commercial fuel, a much lower rate, and that has to do with the fact that the fission products are bound in solution with the aluminum matrix as opposed to commercial fuel where it may be more in a gas state.

In my talk in just a few moments, and then later in Dr. Iyer's talk, I will describe for you the program that we have going on at the Savannah River site that includes material testing and analysis, and is built upon the operational experience that we have here at Savannah River. I'll next describe our current program and the future program for managing this fuel.

With this diagram, what I'm intending to describe is how we will be managing the fuel at the Savannah River site and its final end state disposition. On the left-hand side, we have grouped the fuel by its either location or its relative stability with regards to environmental, safety, health implications.

The disposition path is shown from left to right. The boxes that are--it may be difficult to see, the shading is a little bit light--but the shaded boxes pertain to the disposition path of the research reactor spent fuel, and I'll walk through that in a moment. The solid boxes here represent either existing facilities or activities within the Savannah River site. Those that are in a dashed box are planned future facilities. And over here, of course is the ultimate end state for disposal of the material.

Down here in the legend, and it's hopefully more legible in your hand-out, although it's a small copy, are a list of the various NEPA actions that pertain to dispositioning the materials here that I'll be describing. And I will not, for the sake of time, describe each of those in detail.

Across the top line, these are the materials that, as I mentioned, have come from our production reactors. They do include some research reactor material. For example, the Taiwanese Research Reactor was deemed to be an at risk material that should be dispositioned through the F&H canyons at the Savannah River site, so they are being reprocessed and they will be disposed of through the vitrified waste form from DWPF.

Generally, the topic for this morning pertains to the materials here in the category referred to as the stable aluminum fuels and targets, and there are some stainless and zirconium clad fuel which are in our receiving basis for off-site facility, off-site fuels facility. The stainless and zirconium fuels are destined to be shipped at sometime, probably in the 2010 time frame, to INEEL, and I do not plan in my presentation this morning to be discussing the disposition of those fuels any further.

Future receipts of domestic and foreign research reactor fuel will be received into our receiving basin for off-site fuels, and I believe that on the tour tomorrow, you'll be visiting that facility. We also, because of space limitations, we are also storing fuel in the basin, the wet storage basin that's in proximity to our L-reactor.

Our plan, as Howard alluded to a few moments ago, is for those fuels that are not deemed to be at risk and would, therefore, be stabilized in the canyons. Otherwise, they will be transitioned out of wet storage into a new yet to be constructed dry storage facility at the Savannah River site, and then prepared for shipping to a geologic repository.

BULLEN: This is Bullen, Board, just a quick question. That's predicated on the fact that the repository will accept the fuel in that form? If it doesn't accept it in that form, then it goes back for some other stabilization?

BARLOW: Well, the objective of this facility is to have a road ready waste form, that we can demonstrate that it will meet the repository waste acceptance criteria.

BULLEN: So you'd really like to have Number 7 come before that box where you've got the repository, EIS that says yes, we'll take this waste form before you put it in treatment and interim storage? I'm just trying to figure out your sequence.

BARLOW: Yeah, relative to the timing. Well, the reason it's reflected that way is based on the current schedule of activities.

BULLEN: Right, I understand that.

BARLOW: It's hard to predict. We would like to get out of wet storage into dry storage. If as a result of the EIS and the licensing process for the repository additional treatment is required, then yeah, we'd have to cycle back through this treatment.

BULLEN: Okay.

BARLOW: On the left-hand side, I've just presented a graphic of our plan for transitioning from wet storage into dry storage. Our current anticipated start-up for this new facility is in the year 2005. And as you'll see, these inventories here to the far left represent materials that are being stabilized through the canyons, and those basins were projected to be the inventory around the year 2000.

L-Basin and RBOF would be de-inventoried in about 2009 or 2011. So, again, to answer your question, Dan, our intention is to get out of these basins, reduce the operating costs associated with maintaining those basins.

For this graphic, I'm just attempting to describe by three different measures what the current and future or projected inventory of spent fuel will be. This does not necessarily represent the total inventory that will be in dry storage, depending upon the availability of the repository. But at least it's the throughput for this new facility that's destined for the repository.

The quantities that are shown here either in terms of the number of assemblies or cubic meters, volume or mass, in terms of metric tons of heavy metal, these quantifies do include the 7,000 assemblies of INEEL fuel just for completeness. It also includes some material which are deemed to be potentially, but has not been decided through NEPA action, potential candidates for reprocessing. And they constitute a big percentage of the total mass, such that if they are reprocessed, ultimately what would be sent to the repository would be about 20 metric tons below the quantity that's shown here.

These materials, the reason they're considered candidates for reprocessing is that they include the EBR-II material, which at Savannah River has been de-clad, and because of it's uranium metal condition is deemed potentially not a good candidate for direct disposal at the repository.

Likewise, there are some particulate target residues from medical isotope production which are now in about 900 cans and don't constitute a very large mass.

As I mentioned, assuming that these candidates are reprocessed, then in the end, there will be about 300 cubic meters of material that would be prepared for disposal in the repository. That is about 24 metric tons, as I said, lower than what's shown here, about 70 per cent of which, unless it's treated and diluted, would be highly enriched uranium.

A question had come up in an earlier talk about the number of canisters. We have not made the final selection of the technology for preparing for disposition of this material, and I'll discuss that in a moment, but depending upon the technology, the number of canisters that would be in this facility containing fuel in a dry storage environment range--our best estimates at this point range from somewhere between 400 to over 1,000, maybe 1,400 canisters. Those canisters would be then shipped to the repository and loaded in a waste package at the repository. By volume, this material represents less than 1 per cent of the total inventory that's projected to be placed in the repository.

SAGγIS: Excuse me. Sagηϑs, Board. Again, the ordinate in that graph--

BARLOW: Oh, I'm sorry, it's the number of assemblies.

SAGγIS: Number of assemblies?

BARLOW: Number of assemblies.

SAGγIS; Thank you.

BARLOW: As I mentioned with do have a program for making determination of the best technology for dispositioning and preparing this material, and that's what I'll be describing next.

In late 1995, DOE assembled a team of experts from within the DOE complex for people who have had working experience with these materials, and gave them the mandate to come up with a strategy for safe interim storage and disposal in a geological repository. This task team conducted its review over a several month time frame, and included a number of interactions with what's referred to as our technology champions, folks who thought that they had the best technology to disposition this material and prepare it for storage and disposal.

This charter came as a result of decisions and announcements that had been made by the Secretary of Energy to discontinue operation of the reprocessing alternative. And so this task team's job, by its terminology, was to come up with alternatives to the reprocessing technology.

In the task team report, they do show, for comparison purposes, costs and schedules associated with the reprocessing option, and as I indicated earlier, it is currently the Department's policy that those materials that are a potential health and safety risk would be reprocessed if there is an imminent concern.

In your hand-out, the following two slides contain a list of the alternate technologies that the task team spent a number of hours and days and weeks evaluating and deliberating between themselves. Again, in the interest of time, my intention is not to present all of those, but rather describe their conclusions and the recommendations, and then in a little bit more specificity, the two technologies that we're pursuing today.

In evaluating the alternative technologies that are listed in your package, the task team used a multi-attribute analysis, and came up with a consensus among the team as to what were the conclusions of evaluating against the criteria that are listed here.

The first criteria was a judgment on their part of what they thought the confidence or the likelihood of success of being able to implement the technology, and within the established cost and schedule and the technical performance parameters as best they were understood at the time. This particular criteria was given a weight of 30 per cent.

Second criteria was cost, a comparison both of ten year cost and life cycle cost, and that cost was to include competing technology development, design, construction, start-up, operations of the new facility that I just mentioned a moment ago, as well as projection of the waste disposal costs which obviously are related to the number of canisters and the volume of material to be disposed of. This was also given a weight of 30 per cent.

Third criteria were the technical merits. Each technology was judged with respect to its conformance with environmental safety and health standards, waste form compatibility with repository requirements, again, as well as they were defined almost two years ago, and some consideration of potential proliferation issues. And that was given a weight of 20 per cent.

And last, but certainly not least, in DOE's priority was timeliness of implementation of the operation and the beginning of transitioning out of wet storage into dry storage. That was also given a weight of 20 per cent.

The next slide in your hand-out is a table that shows the results of their evaluations and the scoring that they came up with, and that's been about two years. Since that time, some of the input and some of the information they had available has evolved. We have a better understanding of the costs than we then had, a better understanding of some of the performance requirements. And so today, I don't imagine the scoring would come out the same exactly, but they did perform and we have looked back at their criteria, and in the end, we think that their conclusions still stand relative among the technologies.

BULLEN: Bullen, Board, with a quick question. Are you going to put up the table?

BARLOW: I can if you'd like to talk about it.

BULLEN: Actually, just a quick question to ask about that. I notice that in picking your weighing, and everyone is always going to criticize how you pick your numbers, and this is not a criticism, this is just a comment, that if costs weren't weighted so highly, if you switched cost and technical suitability as 30 per cent, 20 per cent, the other way around, would you expect a significant change? I mean, the cost heavily weights toward direct co-disposal, according to your inventory there. If it were only 20 per cent and the technical suitability of melt/dilute or press/dilute or whatever, which has a higher technical--and obviously we're the technical review board, which is why we look at those things--do you think it would change? And particularly in your second choice, because if you look at the overall score, the difference between 66, 63, 62 and 60 isn't very big.

BARLOW: That's right.

BULLEN: And so were costs not so heavily weighted and technical suitability given a little bit more weight, would that skew the results a little bit differently?

BARLOW: Well, as I said, they did do a sensitivity analysis, and I didn't prepare that graph. The way they did the sensitivity analysis was to take out, or zero out one category, and then determine the ranking. I don't recall exactly how the sensitivity result came out when the cost was zeroed out. But I think I would agree with your perception that most likely, at least between these two highest scores, they would come much closer, if not switch. And, you're right, because of cost and timeliness, the direct co-disposal and direct disposal came out a little bit higher than the others.

BULLEN: I think from my personal perspective, I'd like to see technical suitability be more heavily emphasized as opposed to cost, because I'm not sure we have as firm a handle on cost right now as we're going to have in the time frame for disposal. And so cost usually is one of those things that drives things early on, and maybe falsely so, and so you really worry about giving a lot of emphasis to cost. I mean, fully 30 per cent to cost is a strong emphasis there, whereas I think from a technical perspective, we'd like to see more weighing in that area.

BARLOW: An evaluation similar to this, but our intention is to conduct it a little bit differently, an evaluation similar to this will need to be done to narrow down between those that we are now implementing or developing, which are the direct co-disposal and melt and dilute. That evaluation may or may not have the same kind of weighing.

I would also mention that most recent cost estimates, in fact there was one just published last Friday and, among other groups who were interested, the National

Academy of Science was provided a copy, show that the difference in cost between direct co-disposal and melt and dilute is essentially zero, at least within the range of accuracy. So even with that weight, the scoring would change.

DI BELLA: While you've got the table up there, this is Carl Di Bella, I'd like to make--I'm not sure if this is going to come out as a comment or a question--the ultimate objection here is to get rid of this stuff, to dispose of it permanently, and it would seem to me that your criteria ought to include some sort of performance metric so that decision makers could look at the trade-offs between the various factors and performance. It appears right now all you have is like meets a minimum performance standard. I'm talking about performance in a repository meaning something like the dose or something of that sort. Am I correct in interpreting your table that you really don't have a performance goodness metric there?

BARLOW: Again, what I'm referring to here is an evaluation that was done a couple of years ago. And in the category called technical suitability, there was an attempt to evaluate the performance against waste acceptance criteria as it was understood. And since I was not directly involved in it, there may be someone who would correct me in the audience, but my understanding of it is that there wasn't much of an attempt to give too much weight to going beyond acceptable. In other words, there wasn't a good, better, best grading that played very heavily into that evaluation. That, of course, is very subjective and it's somewhat difficult to quantify, but our intention in the evaluation that I'm going to describe in a few moments that will be happening here in about six months would be to give a little bit broader look to that consideration. And your input and the National Academy's input would be certainly valuable to that, among other stakeholders.

As the table showing the scores indicated, the simpler or less complex options generally came out ahead of the more complex technologies. And those that came out near the top, as we just mentioned, were direct co-disposal and melt or press and dilute. We talked about because of cost and timeliness, that direct co-disposal tended to be a little bit higher, but taking that out of consideration and looking at how you would rate melt and dilute, its advantages and technical suitability, it came out higher in that particular category.

Electrometallurgical treatment was acknowledged by the task team as an ongoing development activity that had its own funding, and ought to be kept in the wings, as it were, as a diverse, more advanced technical backup. And, in fact, part of the assessment or presentation and deliberations that we had with the National Academy of Sciences two weeks ago included the description of ongoing effort related to aluminum fuel and electrometallurgical treatment.

However, within the Savannah River site and the program that I'm managing, we are not actively involved in that particular technology, but we are with direct co-disposal and melt and dilute. We have not continued the development of press and dilute primarily because, again as was pointed out on that graphic, press and dilute accomplishes some but not all of the, if you will, technical advantages of melt and dilute, and in our evaluation, the melt and dilute does not pose that much additional cost or technical complexity as compared to melt and dilute. So we are not pursuing that option at this time.

Subsequent to the Research Reactor Task Team completing its work and publishing its report, which is documented in two volumes, the first of which is a volume that describes just what I did in much more detail, and a second volume of appendices that provides primarily cost, but also some critical analysis and some other data that backs up this report. And if the Board does not have it, certainly we can provide a copy.

Shortly after this report was published, the Department of Energy directed Westinghouse Savannah River Company to implement the recommendations that are contained within that program, and that's why I'm here. For about a year and a half now, we've been conducting the evaluations both on the technology side, and we've begun the engineering and project work for the facility. And though describing that project activity is not part of my discussion this morning, I'm available to answer questions about where we are with that.

But what we will spend some time on, and I'll describe here briefly, is the technology program, and later I guess the next speaker, Dr. Iyer, will go into more depth than I will here. Generally, I would describe the goals of our program to be three-fold; first, to conduct and ultimately complete waste form qualification studies on both the direct co-disposal form as well as a melt and dilute product form. Those studies are being done for the environment, both for dry storage in an interim basis at Savannah River, and for repository disposal, such that, again, we're looking at those environments and the variations associated with those.

These studies and the technical basis, which I'll mention in a moment, will become the input to a technology decision which we are now scheduled to make in the fall of next year. In the summer of next year, Dr. Iyer and our team will be preparing a report that will be an assessment of these technologies, these two alternatives, and provide a recommendation to the Department as to which one, or perhaps even a combination of the two, we would recommend for implementation.

The technical data package of course is the necessary documentation that needs to be provided, both as a basis for design and construction of a facility at Savannah River, but likewise for the design, the environment impact study and the licensing of the repository.

Discussions have been going on and have increased in frequency, and sometimes intensity, between ourselves and DOE RW in sharing documentation and in face to face meetings on what this data package needs to consist of. And in fact, yesterday all day, and I think all day today, we have engineering and technical folks meeting with DOE RW discussing what the data needs are and how that data is going to be used so that we can be sure that the information that we're generating is compatible with their needs.

And thirdly, as I mentioned a moment ago, we are defining those functional requirements necessary to build the facility that may or may not include treatment, but certainly would include certain conditioning steps to prepare the fuel not only for dry storage, but ultimately for disposal.

In this past little over twelve months, we have made excellent progress, we think, and as I mentioned, not to steal Dr. Iyer's thunder too much, let me just hit some of the highlights of what we have accomplished in the technology side of our program.

Within the direct co-disposal arena, we have documented a set of specifications for drying of the aluminum fuel, and this is based upon vacuum drying tests that we've conducted at the site. We have also, both within the site organization and in conjunction with other DOE laboratories, and DOE RW's M&O, have conducted a number of performance studies that include thermal analysis, air and vapor corrosion testing and criticality analysis, which will be part of the topic following Dr. Iyer.

We've also installed, or are in the process of installing a test canister to be used out in the facility with real irradiated spent fuel for the purposes of monitoring and validating its performance against the models that we have been creating and are refining as we go through our qualification studies. That test canister should be in place by March of this coming year and be ready to have fuel installed into it.

Within the melt and dilute area, most of the focus has been in the process development side of things. That includes looking at the various crucible materials and the performance of those materials during the melting process. We have been looking at furnace design, and one of the primary concerns or considerations for moving forward with this technology is how to deal with the by products that come as a result of melting these spent fuel elements in a furnace. And so the off gas system is an important component of that process that we are spending a lot of effort addressing.

All of this work has been documented in reports, many of which have been provided to the National Academy of Science and to the U. S. Nuclear Regulatory Commission for their information, as well as the Department of Energy, and they're available to the Board at your request.

Looking ahead at least for the next six to twelve months as we anticipate this technology decision, our focus on direct disposal is really to do more expanded and detailed analyses than those that we've been able to accomplish in the past several months. And likewise, within the melt and dilute process, we envision, or we have committed to and have schedules for starting up prototype facilities and apparatus within the laboratory to demonstrate the process works, and we will do that to form various compositions of alloys. And, again, Dr. Iyer will discuss that for you in a little bit more detail in a few minutes.

I'd like to switch gears now at the Board's request and take just a few minutes and describe the activities and the dialogue that are taking place now with the U. S. Nuclear Regulatory Commission.

A few months ago, the Department of Energy and the USNRC agreed to an exchange of technical information regarding this program, and specifically what DOE has requested is that the NRC staff, technical staff, provide comments regarding the suitability of the various technologies as it pertains to repository acceptance, and some of the issues that may relate or pertain to that.

Since this memorandum was signed, we have provided, as I mentioned a moment ago, to the NRC the documents, some of the primary documents that describe the work that's been completed and the work that's planned. Tomorrow, we will be having really the first working meeting that we've had with the NRC to make plans for activities between now and at least six, twelve months from now. So we do not yet have feedback from the NRC that I can relate to you today.

In fact, if they haven't introduced themselves, I think that at least two of the folks that we'll be meeting with from the NRC are here this morning.

Finally, as I mentioned, and that you're aware, the National Academy of Sciences has been asked to assist DOE by conducting a review of this program, and that has begun. Specifically within the National Research Council's Board on Radioactive Waste Management, a principal investigator has been assigned, and a panel of experts has been assembled to conduct a review in anticipation of this upcoming decision that needs to be made.

The National Academy of Sciences has met with us twice. First, just a small group with the principal investigator, Milt Levenson, and their study director, Kevin Crowley, for the purpose really of kind of scoping out the effort, understanding what information is available.

Then the second meeting took place two weeks ago in this building down the hall, and it was a day and a half of--really a full day of presentations made by myself and my staff, as well as individuals from DOE RW's M&O conducted a number of analyses and studies that we are using and will be described for you a little bit later. Their schedule is to provide DOE with a report in March. It's a very compressed time frame by their standards, and they've expressed that a number of times, that concern, but they have asked their panel of experts to provide their written input by Christmas, and presuming that they're all going to do that, they are committed to providing the Department with a report. That report and its recommendations, its comments and observations will certainly be factored into our assessment and ultimately the Department's decision.

That concludes my presentation, and I thank you for the opportunity to present it.

BULLEN: Thank you, Mark. This is Bullen, Board.

As a followup to your last viewgraph, I've been in contact with Dr. Levenson, and have agreed to provide him everything that we learn in this meeting to further augment his studies, and he has also agreed to provide us with his input, actually his drafts as they come out. So we are very interested and are very aware of this study and look forward to seeing it.

Any questions for Mark from the panel?

(No response.)

BULLEN: Okay, I'm going to defer questions from the audience in light of keeping us on schedule. We will take a break for how about 13 minutes, and reconvene at 10:45.

(Whereupon, a brief recess was taken.)

BULLEN: Could we reconvene, please, have at least the Board members come back and have their seat at the table? And could everyone grab their coffee and pull up a chair?

Our next speaker is Dr. Natraj Iyer. He manages the technical activities at the Savannah River Site Spent Nuclear Facility Technology Development Program, which includes aluminum spent nuclear fuel, alternative treatment technology. He also manages the Materials Application and Corrosion Technology group at the Savannah River Technology Center. And Dr. Iyer will speak to us about the treatment options for aluminum clad, highly enriched uranium spent nuclear fuel disposal.

IYER: Good morning. I want to thank the Board for this opportunity to talk about the alternate treatment technology program that's currently underway at SRS. As Dr. Bullen mentioned, my name is Natraj Iyer and I'm from the Technology Center at SRS, and I'm going to be talking to the activities that are currently underway in the context of the alternate technology program.

What I'm going to do is give kind of a background. I know Mark has covered some of the aspects of aluminum fuel and its characteristics, and what I'm going to try to do is try to go into some more detail and give some background on aluminum fuel, show you a little bit about its corrosion performance in basin storage or wet storage, which could be one of the conditions the fuel is in before it's put into a road ready package either for direct disposal or melt/dilute, also show what the corrosion performance is in dry storage. Again, those are kind of the starting conditions. And then really share with you the road map we have for the technical activities related to the options that we are currently pursuing.

I should point out that the charter that was given to us as part of this program was to look at both these options. As the task team looked at all the different options, we were not necessarily champions or defectors for any of the options. So what we hope to do and attempt to do is look at both these options objectively in the context of the decision drivers that our leaders have laid out, and see how things play out.

As most of you know, there are really three different kinds of aluminum fuel that are being considered as part of the MTR aluminum, the foreign research and domestic research reactor. Primarily, the UAlx type and what that consists of, the UAlx phase, the aluminite phase and aluminum matrix, and it's primarily made--some are made by casting extrusion technology, but most of it's made by primarily the rolling or casting technology.

Another kind of fuel is the U3O8 element of fuel, where again you have a U3O8 phase and an element of matrix. And then the low end rich fuel, a lot of it is the silicite fuel, which is kind of the more recent fuel that came out of the RRTR program, the Reduced Research and Test Reactor Program, and what this consists of is primarily the silicite phase, again in an aluminum matrix. As I said, typically most of these fuels are made by power metallurgy followed by some kind of metal working, be it rolling or extrusion. Some of it is made by casting, but that's a fairly small fraction.

Obviously what we are more interested in is what do these fuel microstructures look like after irradiation. And what you see for all the three types of fuel, obviously you're going to see the fission cavities or the fission porosities in these fuels, and both in the UAlx fuel and the

U3O8 fuel, a lot of it is converted to the UAlx phase. So what you see in the oxide fuel, for example, is, depending on the burn-up, a lot of UAlx phase, a little bit of oxide, and then primarily a aluminum matrix. And, again, with the silicite fuel, it's primarily silicite and aluminum.

So if you're talking either dry storage, direct disposal, this is kind of the condition of the fuel that we're looking at as we get it ready to be packaged.

I just wanted to share with you a little bit about corrosion performance. There has been a lot of discussion, both in the technical community and in the operations area in terms of performance of aluminum fuel. The fact is aluminum is reactive, but what's unique about aluminum is it's reactive, so it forms the initial oxide layer very quickly. And once it forms that oxide layer, it passivates very readily. So in a sense, as long as the integrity of the oxide layer is maintained and the environment is maintained, aluminum fuel is very corrosion resistant, and that's been borne out again and again in a lot of our field experiences, both in wet storage and dry storage.

Typically, what happens with most of these fuels is in reactor service, you very rapidly for a typical irradiation, anywhere from 50 to 70 per cent burn-up, you build up a bromide layers of about 50 microns, and as long as the handling is typical, that is, you don't initiate scratches on the fuel, the bromide layer is very resistant to any kind of corrosion. And so when we take this into the basins, and we looked at both basins, and by basins of origin, I mean basins across the world in a lot of foreign countries which are not necessarily either technically or economically built up to the extent the U. S. is, so we looked at the fuel in the basins of origin and then of course the basins at SRS.

What you find is typically unless you breach the oxide surface somehow, you see very good corrosion performance for long periods of time. Over the last three years, we have had an extensive foreign research reactor off-site inspection program, where our folks have gone and actually visually looked at fuel and fuel plates in a lot of these foreign countries, Brazil, Venezuela, Uruguay, and a lot of those other countries, and what we have found is what you see on the right is kind of typical of over 95 per cent of the fuel that we see as what is categorized as FRR fuel, and that we expect back at SRS. That is, it's in near pristine condition. Between 2 to 5 per cent of the fuel has some degree of corrosion, and what I've shown on the left-hand side there is the absolute worst fuel that we've seen so far in our inspection of a lot of different foreign basins. And even that particular fuel where you see these oxide build-up, or the corrosion product build-up, that particular fuel, if you stick it in the basin water by itself and do a leach test, or what we call a SIP test, which is one of the accepted tests, the leaching of cesium 137 is almost non-detectable. It's less than about 8 or 9 nanocuries per hour, but it kind of gets into the range of sensitive--and things like that. But it's almost non-detectable. And as I said, that is the absolute worst case fuel that we've seen.

A big difference between the leaching characteristics in aqueous environment, and I'm talking about a typical basin aqueous environment as opposed to the repository environment. What happens is even though you initiate a pit, which is a primary form of degradation, it passivates very quickly and you form an oxide layer over the pit. So typically unlike commercial fuel, where if you have a breach, it pretty much releases all the fission products because of the interconnected porosity, in this case, the release is really driven by diffusional processes, which are time, temperature, diffusion coefficient driven, because all the fission products are really attached in the core.

So that's the big difference between alloy fuels and what we see in commercial fuels, and because of that, in typical basin aqueous environments, we don't really see--or I would say see insignificant leaching, where we get into the so-called non-detectable or near non-detectable uncertainty range. But that again only goes to the 5 per cent of the total inventory or typical FRR fuel that we'd be receiving back is more what you see on the right side.

That particular fuel, for example, was stored in wet storage for 25 years in Brazil, and a wide range of chemistries over its life storage history, and that's typical of a lot of the foreign research reactor basins.

At SRS, most of our basins are maintained to 1 to 2 micromole semens per centimeter, if that's the right unit. So most of our basins maintain very pure conditions, and again, unless you see a breach of the oxide layer itself, we haven't seen any corrosion.

We have an extensive corrosion surveillance coupon program where we periodically remove corrosion samples and actually also look at our fuels visually to see if corrosion has been initiated, and if it has proceeded.

I didn't bring the slide on dry storage, but then talked a little bit about dry storage. We've done extensive work, which could be another adversive condition for some of these fuels. What we've seen also in our foreign inspection is that dry storage under a wide range of conditions, from fairly humid conditions in Brazil where they actually had some water leak on some of these fuels, to controlled conditions, I would say more controlled conditions, again the condition of the fuel that we've seen in dry storage under wide ranging conditions has been fairly pristine, as long as the oxide layer hasn't been breached.

Now, within SRS, we have done extensive work on dry storage to basically develop the storage criteria. But what I was trying to point out was really the range of experience that we observed in receiving these fuels from the different basins across the world.

I'm going to switch gears now, having described the condition of the fuel that we're going to be receiving for some kind of alternative treatment, I'm going to talk a little bit about the program that Mark alluded to. It's referred to as the alternate technology program, and it was initiated in FY97, so we are a little over a year underway. And the purpose of this program, our charter was really implement the recommendations of the Research Reactor Task Team that DOE headquarters had instituted, and the primary recommendation was to pursue direct co-disposal and as a backup, melt/dilute.

As the program got started, our approach has been we are pursuing direct co-disposal and melt/dilute on equal par in terms of the way we are approaching the issues, and as I said, we're looking at it in the context of the number of decision drivers, and just a year from now, we're going to see how things play out.

What direct or co-disposal is is primarily putting a spent fuel canister in a waste package which consists of

DWPF glass canisters. And the melt/dilute option is taking the fuel, melting it, adding depleted uranium to dilute the enrichment, solidifying the fuel directly in a canister, and thus flexibility in terms of what that canister material could be. For example, it could be in a canister, or we could remove it from the canister and put it in the spent fuel canister. But one option is, for example, if titanium alloy or some other hastelloy is a good engineered barrier, then the option does exist to cast it in a crucible of that material, seal it, and then put it in the spent fuel canister. So it does provide a lot of flexibility in terms of how we would process it.

The primary drivers for direct disposal, as you saw from the report, was both the cost and the timeliness, and of course the issues and challenges are to demonstrate criticality control and demonstrate that to the satisfaction of all the stakeholders. That was one of the primary challenges. The other big challenge is the characterization requirements.

I failed to mention that as these fuels come from across the world, they have different pedigrees in terms of the way they've been characterized or the way the characterization data has been assimilated in all these different countries. And so the question is how do you reconcile that versus what the RW requirements may be for characterizations. So the characterization requirements could be quite significant for direct disposal. That's still something, you know, will evolve through discussions, but it could be significant given that they all have different pedigrees as they come from across the world.

The dilution option has the benefit that we basically erase the bulk of the history of the fuel, so the characterization requirements are fairly limited in context of direct disposal. You do have the benefit of the dilution, which helps with the criticality control.

And, finally, as we have done the work, and you'll see later, it does have a significant volume reduction potential, up to 70 per cent, and I'm going to get into that a little later.

So the way the program was laid out is our primary decision drivers were the spent fuel performance, and by that we mean both performance preclosure and postclosure in the repository, the characterization requirements, costs, schedule, licensability, stakeholder acceptance. And when you look at those decision drivers in terms of what needs to be done technically, we basically can categorize in three boxes. One is the spent fuel form development, and I'll get into what that means a little later. The other one is performance, and then another issue unique to alloy fuels is the fact that you need to be very careful how you interpret the performance results. And so you need test protocols. For glass, there's an ASTM standard, which is commonly used for commercial fuels. Again, there are standard techniques that have been used. No such technique exists for aluminum alloy fuel. And so that particular activity is just as important if you're going to get good scientific data which can be input into the PA, and other data needs requirements.

So the way the program was organized for both these options was to look at these three major issues so that we could provide the information both to our management and all the stakeholders in terms of all the decision attributes.

Just getting into it in a little more detail, what we mean by spent fuel form development is for direct disposal, it's really coming up with the road ready storage criteria in the context, that is, what is the--we already had a dry storage criteria for aluminum fuel established here at SRS, and what we are doing there is building that and taking it to the next step in the context of what we know currently is the waste acceptance criteria, and to come up with what we call the road ready package criteria for aluminum fuel. So that's really what it means in the context of direct and co-disposal.

On the other hand, in the context of melt/dilute process, it means what's the right composition of your melt/dilute form. What is the dilution level, what's the uranium aluminum ratio, and what is the result in microstructure and what are the characteristics. So it's really the bench scale process development leading to some kind of what we call the optimum microstructure to drive the performance.

The performance in both cases, in direct disposal, it's really performance of the fuel microstructure. I showed you earlier the irradiated microstructure. In the context of melt/dilute, it's really performance where you don't really have those fission pores. It's a fairly homogeneous microstructure with uranium-aluminum ratios. So it's really looking at performance of those kind of microstructures.

And the reason we are doing this is really to feed two critical data needs. One is to feed the process requirements so that the SRS site can start developing functional requirements and start planning towards the feasibility, and the other one is to address the data needs that RW has as they get into PAs. Right now, most of the information, for example, that's being used in the preliminary performance assessments are assumptions of aluminum dissolution rates and pure uranium dissolution rates. And as this program evolves and as we start generating dissolution rates for the exact microstructures, those will be fed into RW.

And then the last block at the bottom, the test protocols, is the activity I mentioned earlier, is to make sure that the way we measure the performance characteristics and the way we interpret it is consistent and has consensus to ASTM or some other national consensus board.

What I'm going to do next is kind of show you a road map of the technical program for both of these options, and then just highlight some of the accomplishments. My intent here is to just give you an idea of the kind of activities, and then as I said earlier, tomorrow as we go through the tour, if you'd like more details on any one specific activity or all of them, we'd be glad to provide it as we walk through the tour, since I can't condense everything in a half an hour.

But the direct disposal area, as I said, there are several activities, and those are these major blocks leading to basically what we call the technical basis for direct disposal. That is how that body of information that would help the decision makers make the right technical decision. The first block is the spent fuel form definition, the first two blocks, and the form development and development of the storage criteria. What that is is basically building on the dry storage criteria, looking at the environmental conditions that are required in the road ready package canister so that you basically don't have degradation preclosure or very limited degradation within acceptable limits up to the preclosure stage. And that has been defined and we have documented that, at least the preliminary road ready storage criteria.

The next block is basically a better definition of the road ready package. In this context, we are primarily working with the national program as it defines the various canister configurations.

The performance analysis, a big aspect of direct and co-disposal, is obviously the criticality analysis, since we are dealing with high enriched uranium, and the next speaker is going to talk to that, but we're working with RW M&Os as they perform the early criticality analysis for some of these fuel types to see where we stand, just so that again we get the body of information that we need to make a wise and right decision.

One major aspect of criticality performance obviously with high enriched uranium is the response and the reconfiguration of materials. That's just as important as just normal degradation. How do they reconfigure and reconstitute through the geological time. So that is a major activity.

Obviously, a lot of the criticality analysis is already underway, and what we hope to do with the information we generate is go back and either validate the assumptions or fine tune the assumptions, and that's the intent.

And at the end, what we expect to have is for each of the attributes in the waste acceptance criteria, basically have a technical basis, for example, for reactivity, pyrophoricity, compatibility, the dissolution rates, and have that package of information for both these options, in this case, I'm showing it for direct disposal, so that come next fall, we would have the same set of information for both these options in the context of repository performance, and at the same time, also have the functional requirements for both these options so that we can make the decision.

What I have in these blocks on the side are primarily some of the specific activities that have been going on in fiscal '97 and what's currently underway in fiscal '98. A lot of the activities in fiscal '97 focused on the definition of the road ready storage criteria, and what we have currently underway, as Mark said, is basically a lead surveillance program that we're trying to go validate the corrosion models that were developed for the road ready package.

The focus of the '98 activities is really to look at the materials response reconfiguration issue, and also start generating the information in the repository environment, and those tests are underway so that we can get some realistic dissolution rates. And I'm going to get to that a little later.

The next slide I'm just going to highlight. This by no means is a summary of all the activities. But I'm just going to highlight some of the activities that had been underway. The first block on the left is the road ready package. As I said, we have established the drying criteria and the backfill criteria, and we're actually conducting field vacuum drying tests for these MTR assemblies in the canister to make sure we have enough information to issue the drying specs so that we would have reasonable functional requirements for the storage facility.

At the same time, we also have what we call--where we initiate the lead surveillance canister, and the purpose of that is really to validate our storage criteria and our corrosion models, our degradation models. And what we are doing in there is we have a highly instrumented canister, which is shown in the photograph on the second block on the right. We have a highly instrumented canister which has all the bells and whistles in terms of temperature measurement, gas pressure measurement, everything else, so that we can validate both our thermal models and our degradation models as we take a canister to the extremes of the safety and operating envelope.

We have developed degradation models, and these have been done parametrically initially using typical cold samples, including gamma radiolysis effect, and then the models are being validated actually using hot samples. So that's the approach we've taken, is we have done a very extensive parametric analysis using cold samples in gamma radiation with radiolysis effect, and then we go and pick key spots and then do a validation using hot samples. That's the approach we've been using, and found it to be very cost effective.

And the last block on criticality analysis, the next speaker is going to talk to that, so I'm not going to say anything at this time.

I'm going to switch gears and talk a little bit about the melt/dilution process. And as I said, you'll see a lot more detail tomorrow as you walk through the lab specifically on this process, because we will show you our apparatus and what we've been doing. But the key blocks again are development of the melt/dilute form, and by that, what I mean is to make sure, number one, we know we can cast these alloys because that's how we made the fuel at SRS for a number of years, but make sure that we can get the right kind of microstructures to make sure we understand what those microstructures mean in terms of performance, and also make sure some of the process issues in terms of crucible mold interactions and the various options we have, for example, do we pour it, do we cast in crucible. If you cast in crucible, does it make sense to cast in a carbon steel crucible, or is there something to be gained casting it in an engineered barrier kind of material and sealing it. So those are the kinds of issues we are looking at.

The other major issue is the fission product release, and this is in the context of the functional requirements for a feasibility at SRS. So we have done extensive analysis of what the total fission inventory is for all the 20,000 elements coming back, what would be the inventory, how would we process it through a melt/dilute furnace, and what the primary actors are and what kind of off gas system do we need to treat those.

Then the next block is small scale validation. A lot of the up-front work was doing bench scale. And by bench scale, I mean hockey puck kind of samples. And what the small scale validation is aiming to do is primarily taking full-scale MTRs and taking it through a melt/dilute process to show and convince ourselves that we are making a homogeneous product. And where that stands is during FY98--well, during FY97, we actually did a full-scale MTR melt/dilute experiments, and during FY98, we actually are developing a facility which would be more prototypic, that is, switching from a resistance furnace to an induction furnace with induction storing so that we have a more homogeneous microstructure. And the plan is then to take that to FY99, actually do a--

And then the last block is the form assessment, which is looking at the waste acceptance criteria attributes, such as reactivity, pyrophoricity, corrosion resistance, gas generation, et cetera, and make sure we have a technical basis for those, and develop that. And that obviously, a lot of it is common to direct disposal, except for the microstructure dependents.

BULLEN: Bullen, Board with a quick question before you leave this. As you've done the melt/dilute, then you don't have to worry about the criticality analysis or the criticality--

IYER: No, one of the activities, and I don't know if it's in that block, but one of the activities we have underway in FY98 is the criticality analysis.

BULLEN: Okay. It's in the far right block way at the bottom, yeah. I see it now.

IYER: Okay. And that's currently underway, at least the scoping analysis is currently underway in-house, and then taking it to the next step through the RW/M&O, we haven't resorted to that extent yet.

BULLEN: So it's sort of dependent upon the geometry of the package as you put it in, and then how it reconstitutes again from the waste form as it dissolves?

IYER: Right.

BULLEN: And I guess that's the follow-on question which you're trying to address.

IYER: Yes. This viewgraph just highlights again some of the accomplishments in melt/dilute. I've kind of spoken to a lot of that. The extreme left block, again just shows you the kind of variables we looked at in our bench scale work, and most of them are process issues as opposed to waste--well, we have looked at waste form microstructure dependency on those process issue.

What the right block shows is basically the volume reduction potential. The typical nominal numbers that we've been using for all our analysis is about 1,400 canisters for direct co-disposal. And in that context, if you go to 20 per cent dilute, depending on the alloy composition, you typically get in the 400 ballpark range, 400 canisters ballpark range for the melt/dilute process.

The block at the left bottom kind of shows just a schematic of the off gas system. And I'm going to get to the radionuclide inventory in a bit, but just shows that you have to worry about the cesium, iodine and krypton, they are major players, and we currently have detailed work going on both bench scale and then taking it to full scale in terms of what that off gas system would look like.

And the extreme right block at the bottom is just a view of the furnace that we have, which you'll see tomorrow when we do the melt demonstration, melt/dilute demonstration of a full scale MTR.

I'll talk a little bit about form testing because as I said earlier, this is just as critical for these alloy forms as the rest of the work. And the main reason is for the glass waste form, as I said, there's been a lot of work done in the aqueous environment, and also for commercial waste form.

SAGγIS: Excuse me. Sagηϑs, Board.

Are we now back into the initial form, or--

IYER: This is common--

SAGγIS: Not the melt/dilute form any more?

IYER: Okay, in terms of this particular activity, what we are trying to do is make sure we understand as a function of the aluminum uranium ratio, for example, in the microstructure, how that affects performance. So this particular activity is common to both forms, and one particular point, for example, the typical nominal 19 per cent uranium alloy, which is a standard MTR, that would be the point which applies to direct co-disposal. Some of the other compositions would apply to the melt/dilute. So it's common.

SAGγIS: Thank you.

IYER: And this particular activity is more understanding dissolution performance of such alloy waste forms because after all, they are heterogeneous dissolution, unlike the typical UO2 or glass. What we have here, and I'm going to get to that at the bottom, but what we see is heterogeneous dissolution. So as we do dissolution tests and get data, it's very important that we understand its dependence to microstructure as we interpret the results. So that's really what we're trying to do here.

And the approach we've taken is we are developing what we said is a test protocol, and what that is is just an assembly of tests, the kinds of tests you need to do to define degradation performance, and we are working through the ASTMC 26 committee to get a consensus on that methodology.

The tests we are looking at are the--and we are working with BNNL and INEEL in this activity.

DI BELLA: Excuse me. Di Bella. Are you testing the feed to the melt/dilute process, or the product, or both?

IYER: In this case, it will be the products.

DI BELLA: Thank you.

IYER: In this activity, we are primarily focused on the performance in the repository environment. So this is in the J-13 and the modified J-13 environments. And the kind of tests we are looking at are the flow test, drip test. Being an alloy form, it does offer--we have the flexibility of doing also chemical tests, especially for long-term predictability, and also vapor phase test.

But what I wanted to show here was the microstructure dependence of these on dissolution. That is, you take a typical MTR element, and what you see is aluminite particles, UAl4 typically, and what you see here is aluminum matrix with a eutectic microstructure. So what you are looking at here is kind of an off eutectic microstructure in the uranium aluminum phase diagram.

What happens in dissolution is people tend to relate dissolution of these fuels to aluminum dissolution rates, and that's debatable at this point because we know that a lot of the fission products we know are tied in with the core. In terms of the partitioning of the fission product between the aluminum matrix and the aluminite phases, we are trying to understand that. What is the partitioning of fission products? One could speculate a lot of it is tied up with the uranium, but we are trying to understand that at this point. And depending on that partitioning, what happens typically in the dissolution tests, or in J-13 environments, is the aluminum will start dissolving and the aluminite particles pretty much stay intact. And we've tried to kind of show that here where your matrix is kind of dissolving, and what you end up with is aluminite particles.

Now, for the aluminite particle, it's typically reasonably stable in the context of the aluminum, and so what is the release from those particles or the fission product associated with the particles are at the particle boundary. And so what we are trying to do in this activity is make sure we can come up with a test and a methodology so that we can interpret the results correctly.

SAGγIS: Okay, Sagηϑs, Board again. Just to make sure of the terminology, we are talking about now dissolution in the aqueous phase of an alloy that may or may not be a dilute alloy.

IYER: Right.

SAGγIS: I'm trying to understand the difference between dilution and dissolution here. Now, those microstructures that you're showing over there are for which waste form? Is that the initial waste form or the dilute?

IYER: No, this would be one of the diluted products.

SAGγIS: Diluted like in which ratio again?

IYER: This is uranium, 19 per cent alloy, and it's an off eutectic microstructure. So what you end up with is a eutectic phase in that aluminum matrix.

SAGγIS: Right. But the overall composition in there would be approximately what?

IYER: 19 per cent uranium.

SAGγIS: 19 per cent uranium? And the rest?

IYER: Aluminum.

SAGγIS: And the rest aluminum. And this is after dilution?

IYER: Yes, this is after dilution, and typically if you're looking at MTR aluminum irradiated at 50 per cent burn-up, what you see is somewhat similar. That is, you primarily see the UAl4 phase in aluminum matrix.

SAGγIS: But the initial fuel, what composition would it have again? The initial fuel, what alloy composition would it have?

IYER: The typical MTR averages to about 19 per cent, but then it does go through irradiation, so there are changes.

SAGγIS: Okay. Maybe we can talk afterwards. Somehow I get into the dilution process, may be confused here.

IYER: Tomorrow in the lab, we have posters with the phase diagram and all the different microstructures. We can show you, or I can show you. I'll be glad to pursue it. As I said, we have people--sometimes there's confusion, at least when I look at the phase diagram, in terms of weight per cent, and then the dilution levels. And so we have a matrix, and maybe that will help.

SAGγIS: Okay.

IYER: I just wanted to kind of summarize the last slide again showing the aluminum form characteristics for both these options. I just wanted to summarize by showing some of the waste form characteristics for these options. The direct disposal, as I said, the microstructure, you do see porosity, and typically what you see are those three phases, aluminum plus UAl3, plus UAl4, with melt/dilute, and what we've shown already with some early work in J-13 environments is that the corrosion resistance of aluminum plus UAl4 is better than the mixed microstructure, which is aluminum plus UAl3 plus UAl4. Now, how much incrementally better and what that does to performance assessment, that's a different question. But it is better.

And so with direct disposal you basically have the microstructure you get in hand. With melt/dilute, you do have the flexibility of tailoring the microstructure so that you can end up with an aluminum plus UAl4 microstructure. When you get to criticality, we haven't done the criticality analysis, but the presumption is poisons, we know poisons are necessary for direct disposal, you'll hear that in your next talk. Poisons are probably going to be necessary for melt/dilute too. If necessary, though, we can make that part of the melt/dilute process so that the form will be integral to the microstructure again, so that you basically have uniform degradation as part of that. You basically form a UAlbx, or whatever the poison is, compound.

In terms of radionuclide release for direct disposal, you have fission gases in the pores, fission and activation products, although we're not sure how it's partitioned between those three phases. In melt/dilute, you pretty much, all the cesium is gone, the iodine is 90 per cent gone, krypton is gone, technetium is still there, and you do reduce the total inventory, so even if you do dissolution tests and you get the dissolution rate, you would expect that your actual release is going to be lower.

And then finally, proliferation resistance, which is depending on which party you talk to, which can be a big driver. You do have isotopic dilution. That was one of the attributes of this process when we initially looked at press/dilute, we did do a paper analysis on the press/dilute process and we found that we really can't get a homogeneous product and it actually created more problems in terms of drying and storage, because we had to insert DU plates and roll it, and it didn't make sense.

So in the context of melt/dilute, we have isotopic dilution and as long as we stay in the sub-liquidous or liquidous range in the phase diagram, which is what our plan is, and so there's no separation of U-235.

So this kind of gives you an idea of the technical activities that are underway, and our current plan is to, as I said, to have this body of information by next fall so that it will help in the decision making. In the meantime, the way this program is translating to do, primarily it will be in the validation stage next. That is, going from cold and some hot work.

BULLEN: Thank you, Natraj. Questions from the panel?

SAGγIS: Yes, Sagηϑs, Board again.

Do I understand--so you're conducting right now corrosion rate measurements in the dilute form of the fuel. Do I understand then that there was very little corrosion rate information on this before now?

IYER: Yes. Primarily what's been used before now is the uranium metal corrosion rate, or the element of metal corrosion rate. There's been very little corrosion rate information of mixed microstructure, like UAlx in matrix. There's a lot of corrosion rate information in aqueous environment that is pure water or typical basin water. We have a lot of that, and there's a lot of information again in good quality aqueous vapor phase environment, because we have gathered a lot of that information in the context of dry storage. But in the J-13 repository environment and the modified J-13, there's I would say what we are generating is probably the first pieces of data.

SAGγIS: I see. And this data began to be developed you would say like in the last year or so?

IYER: In the last eight months. And so those tests are still underway and we don't really have the data coming in. I would say really in the last six months.

SAGγIS: So we're going to have corrosion rate information over a very short time period, at least for coupons, and then of course--

IYER: You should have some preliminary data by spring--late spring to early summer, preliminary data. And a lot of this data, by the way, is as I said in conjunction with BNNL.

SAGγIS: All of this is of course very, very preliminary kind of information when we're talking about extremely long-term durability?

IYER: Correct. Let me qualify that. The corrosion rate information will be for the direct disposal form, that is, the fuel that we have. As far as the melt/dilute, until we go through the hot demo, we won't be testing that hot. But we will have qualitative information based on how these phases dissolve in these environments.

BULLEN: John Arendt?

ARENDT: Thank you. I want to make sure I understand these last three papers, or even the next one. Assuming that the aluminum clad fuel cannot be disposed of directly in a repository, the treatment options that you're looking at would be conducted in lieu of direct disposal; is that right? And if that is right, are you going to have to characterize each group of fuel, or are you going to have to inspect all the fuel? Or once you decide on a treatment technology, are you going to treat all the fuel in the same manner? Am I on the right track, or did I get lost?

IYER: Let me try to clarify that. What we are doing right now is we're not saying aluminum fuel cannot go to the repository. We are looking from the context of performance assessment; can we put aluminum fuel in a direct disposal form in the repository. And if for some reason that runs into problems, be it criticality or stakeholder acceptance or whatever that may be, we're looking at the backup option, the option of diluting the fuel, and then packaging that the same way you would package any other fuel. That is, once you get a diluted product, it still goes through the road ready package, and then disposing of that. So that's really what we're looking at. But at this point, there's a lot of work, and the next presentation is going to talk to this, which is looking at direct disposal of element of fuel in a repository, and there's some preliminary PA work done, which is I believe what the next speaker is going to talk about.

ARENDT: I guess what I wonder then if you, once you decide on the optimum treatment, will all the fuel be treated in the same way?

IYER: That's really--there's a lot of issues there. For example, the low enriched, less than 20 per cent salicite fuel, it will probably be a cost benefits kind of a driver, that is, can we dispose of that direct or in a co-disposal package, or does it make sense to take that to melt/dilute. So those are cost benefits issues.

ARENDT: I'm only talking about the highly enriched.

IYER: As far as highly enriched, it will either be the program we currently have underway, you're only looking at two options, which is direct or co-disposal as one, and dilution as the other one for all the high enriched aluminum fuel.

ARENDT: Right. But are you going to have to characterize or are you going to have to inspect? Will some of the fuel be--maybe it's too early yet, but will some of the fuel be able to be disposed of directly in a repository and some will have to be treated? And you're going to have to look at all the elements then to decide what the treatment is going to be, I assume.

IYER: I'm kind of giving you an answer off the top of my head. If we decide we're going to treat aluminum high enriched fuel, that is, the UAlx aluminum type fuels, my presumption is we'll probably treat all the fuel. That would make more sense, rather than trying to license two different forms. But as I said, there's a lot of work going on in trying to qualify direct co-disposal fuel, and at this point at least, we haven't heard anything in terms of the technical results that are coming back that would necessarily warrant that option. So we're looking at both.

BULLEN: Other questions from the panel? Carl Di Bella?

DI BELLA: I have a two-part question. What is the temperature of your melter?

IYER: Oh, the melter temperature is very high, but the treatment process temperature we're looking at is below 1000 C. It's around 850. But we're using an induction melter, so we could go very high if we had to.

ARENDT: At 1000 degrees C., you're going to volatilize a number of the fission products, that's correct, and apparently you're going to collect them. And where do they go?

IYER: Right. They're going to be--basically gets into different high level waste streams. The total inventory of the--when you look at the total 20,000 elements and look at the inventory, the total inventory of the cesium, iodine, et cetera, in the context of FRR is still very small in the context of all the high level waste we have. But they become part of our high level waste stream.

DI BELLA: Okay. They will come out in the HEPA filters?

IYER: Yes.

DI BELLA: And then how do you put HEPA filters into the high level waste stream?

IYER: Well, there's kind of a detailed washing process which Lee Hyder can talk to, and we'll show you that tomorrow.

DI BELLA: Okay.

IYER: But we have a scheme as to how we would basically wash out the cesium and other radionuclides.

BULLEN: Thank you very much, Natraj.

Moving on to our last speaker before lunch, not to put any pressure on Dave to be on time, we have Dave Haught. He is an engineer with the Yucca Mountain Site Characterization Project Office. He's responsible for the oversight of the development of waste package design, materials testing and modelling, waste form testing and modelling program, and work performed in support of the National Spent Fuel Program and the Office of Civilian--excuse me--Fissile Material Disposition. And Dave will speak to us today about the disposal of aluminum clad, highly enriched uranium spent nuclear fuels.

HAUGHT: As Dr. Bullen mentioned, I am David Haught, and I work at the Yucca Mountain Site Characterization Project. I'm not going to repeat all the stuff that he said.

One thing I would like to do is acknowledge some of the help that I've gotten in putting this together from Peter Gottlieb and Jim Doogood primarily, but others.

This is an overview of what I'm going to talk about today, the waste package design, performance assessment and the criticality analyses we've done to date on the aluminum clad HEU fuels. I'm also going to get into some of the process that we go through in performing these analyses.

That portion of it will not be repeated for the fissile materials disposition because the process is basically the same.

I put this up. Here's the waste package design that we have conceptualized to date. Based on the input from Savannah River, we have looked only at the MIT and the Oak Ridge Research Reactor Fuels, and so the design that you see here is for them. The concept we believe works across the board, but what you're going to see, because this is all we've looked at so far, I make no claim that MIT or Oak Ridge is bounding in any case. We haven't done that yet.

You've already heard a lot of this this morning. We are looking at a co-disposal concept with high level waste, and we are putting long term criticality control features in the canister.

I'm sure many of you have seen this picture before. In fact, I think you've even seen it earlier today. This is the co-disposal concept. You will occasionally hear it called the five pack. There may be an instance or two where we have a four pack, and the five refers to the number of the vitrified high level waste canisters that are in there.

Going right into performance assessment, we did a sensitivity analysis for DOE unspent fuel this past year, and besides just doing the sensitivity analysis for DOE unspent fuel, we took advantage of some of the things we have learned since we did TSPA-95, and I'm going to go through some of these.

We have updated the percolation flux rate. These are the ranges. And if you looked across the repository footprint, that average worked out to about 6.2 millimeters per year. We looked at some various quantities of waste packages that would see drips, and the impact on that. We've updated the diffusion properties of the rock.

This is the same, but it bears mentioning because there have been a number of TSPAs done with different aerial mass loadings. The results I'm going to show you, or findings I'm going to show you today are based on 83 MTU per acre, centered-in-drift emplacement.

We've updated the near-field thermohydrologic calculations. And this last one, at least on this page, this last one is very significant. We have received some evidence and data that indicates that the solubility of neptune is about two orders of magnitude less than what we previously have been using.

Continuing on with some of the changes to the TSPA-95 bases, we have updated the waste package degradation studies. I had in here a slightly busier version of this slide that I deleted to try to avoid confusion, but I'm going to mention it anyway and take a chance.

Part of the updated waste package degradation studies is we categorized waste packages into eight different failure groups. One of those groups was kind of--it was the most at risk group. We basically ignored any discussion or consideration of galvanic protection, and the results that you will see here for the aluminum based fuels are based on all the waste packages are in that group. So it's a fairly conservative curve.

We updated the saturation flux and the porosity. We have not included any climate cycles in this, which simplified things considerably. And at the bottom here is here are the amounts of materials that we have considered in this sensitivity analysis.

The findings: dose at the accessible environment. The peak dose is roughly equivalent to commercial spent fuel in both of the types of fuel that we looked at, the uranium aluminum alloys and uranium silicide fuels. We used those as basically two different groups.

Now, what you will see from the dose from the uranium aluminum alloy is that there is a peak earlier and around the 15,000 year time frame, due mostly to releases due to technetium and iodine. The curves I'm going to show you are going to be a combined release from all radionuclides. But typically what you see at that time frame is these are the major players.

It is less than an order of magnitude difference for the aluminum--I'm going to call it the technetium/iodine peak and commercial spent fuel, and then it is also less than the peak dose, which occurs later in time.

Now, here's the graph for the comparison of the uranium aluminum alloy fuels versus an equivalent amount of commercial spent fuel, and here's that peak, the technetium/iodine peak that I was referring to before. But as you can see, the actual peak dose occurs out into this time frame, and they're roughly equivalent.

And in the case of the uranium silicide fuels, they are basically bounded, or roughly bounded by commercial spent fuel pretty much in all cases throughout time.

BULLEN: Bullen, Board. Just a quick question.

HAUGHT: Yes.

BULLEN: Kind of a point of order here, when you do the melt/dilute, which probably wasn't included in this because you're doing co-disposal, the iodine goes away, so you're going to drop that peak maybe a factor of two on order of magnitude, because there's no iodine?

HAUGHT: Yeah, you still have the technetium.

BULLEN: Right.

HAUGHT: But you're right. We have only looked at the direct disposal option.

BULLEN: Okay.

DI BELLA: While we're here, Di Bella, define equivalency, please.

HAUGHT: An equivalent amount metric tons heavy metal.

DI BELLA: Okay.

HAUGHT: Okay, I'm going to move on to the criticality analyses, start off with kind of some ground rules that apply across the board. I've already mentioned it's based on the two fuels, and these are the enrichment characteristics of those fuels. We have analyzed the criticality potential with MCNP, and we have looked at some alternate neutron absorber materials and we have some conservative assumptions that we have applied to all the analyses. We are assuming fresh fuel and we have an optimum moderation in clay.

We're pursuing a phased analysis approach, and this is a description of Phases 1 and 2. There is a Phase 3. Phase 1 is the intact configuration, in which case we also identify a conceptual waste package, because the criticality analysis tends to be somewhat geometry dependent, so we have to kind of take these on somewhat of a case by case basis.

Phase 2, we look at degraded configurations within the waste package, and we have used EQ 3/6 to analyze the geochemistry. And we have varied some of the environmental parameters and corrosion and degradation rates.

Phase 3 is a cumulative analysis, and basically the Phase 3 is we look at the configurations outside of the waste package. So these are some of the things that we're going to do.

One of the things that I had been asked to talk about today was the estimate of probability and consequences, since we are advocating a risk based approach, and Phase 3 is where that is done. The bad news is is I'm not prepared to talk about Phase 3 today. That's planned for FY99.

Let me mention this, because I believe I need to talk about it later. When you look at the external configurations to the waste package, there are kind of three predominant deposition mechanisms that are of concern. That's absorption of the fissile material on clays or zeolites, the presence of a reducing zone for either organic or hydrothermal upwelling of hydrogen sulfide, which would tend to have the fissile material kind of collect there, and then there's a general chemical reaction with the host rock. And that is something that--a thought that needs to be carried on to the discussion about plutonium.

BULLEN: Bullen, Board. Just a quick question.

You said that the Phase 3 analyses won't be done until FY99. Does that mean that these data won't be available as sensitivity analyses for the TSPA/VA that's going to be done? Or will there be some VA analysis that will include these kinds of sensitivities?

HAUGHT: No, I don't believe that we will have any of this done as the sensitivity analysis for TSPA/VA for the aluminum, or really any of the DOE spent fuels.

BULLEN: Okay. So the criticality analysis for VA will just be spent nuclear fuel and defense high level waste cans; right?

HAUGHT: Correct.

BULLEN: Okay.

HAUGHT: For external.

BULLEN: Right.

HAUGHT: Now, I apologize somewhat for this chart. It's very difficult to read, but there is a hand-out.

This is a configuration generator for how we get to the different--well, the degradation scenarios for how we reach the various configurations that we have evaluated. And there are four basic configurations that we have looked at. This is one of the places where the probabilities, we haven't assigned the probabilities yet for these boxes, and this is one of the places where the probabilities come into play.

The first configuration is basically a homogeneous mix of the clay and the degraded fuel within the waste package. I'll go through these quickly. We have a possible configuration where we have the fissile material stratified on the bottom.

SAGγIS: Excuse me. Sagηϑs, Board.

HAUGHT: Yes, sir.

SAGγIS: Can you explain a little bit the meaning of clay in your diagrams?

HAUGHT: It is the resultant degradation of the vitrified high level waste as it, you know, as it is altered and corroded, it tends to form a sort of clay-like mixture.

SAGγIS: Okay. In other words, the clay would be a corrosion product?

HAUGHT: Of the high level waste, yes.

SAGγIS: And the corrosion product of the composition of the ceramic--

HAUGHT: Right.

Another configuration that's possible is a stratified on the top, and finally, there is the possibility that we have some extreme stratification within the canister that contained the aluminum clad fuel. In other words, the canister is still roughly intact, so all of the fissile material is in here. It is not mixing with the clay, and it's still inside the waste package.

I'm going to go through some of the findings on criticality now. In the case of the Oak Ridge SNF, if that canister contains a carbon steel basket having a borated stainless steel between-layer separator, our evaluations are showing that that can remain subcritical in all configurations.

In the case of the MIT, it's a little more complicated. The SNF in an intact basket--now, in this case, the fuel has degraded but the basket is still intact--we require approximately a kilogram of either boron or gadolinium distributed in an absorber plate.

Going further down, if you degrade both the basket and the MIT fuel, now you're looking at you have to have .25 kilograms of gadolinium homogeneously distributed within the soup that's in there, if a stainless steel basket is used, but .12 kilogram if a carbon steel basket is used. The reason for that is we're considering the fact that as the carbon steel degrades, you have all the rust, and it takes up more space that otherwise water could fill.

More findings on MIT. Configurations external to the canister but internal to the waste package, stratified on top, we need .2 kilograms of gadolinium homogeneously distributed with the SNF. In this case, it's not homogeneous throughout the waste package; it's just in that top. And that is ignoring any contribution due to the iron.

Stratified on the bottom is .1 kilogram of gadolinium. And then the homogeneous mixed in clay, our evaluations are showing that that remains subcritical.

Current status is that Phase 1 is complete. Phase 2 is in review. And Phase 3 is planned for FY99.

In summary, the co-disposal concept appears workable. We're showing a small impact to repository performance. I need to caveat that somewhat that in both cases of the aluminum clad fuel and commercial spent fuel, we are assuming no credit for cladding. If we were able to develop a technical basis for taking credit for cladding in the commercial spent fuel, that may change somewhat. But given the situation we're in right now, they look equivalent.

The internal configurations can be maintained at subcritical levels. And then here's the work that we have yet to do.

That's all I have. It looks like you have a question.

BULLEN: Thank you, Dave. This is Bullen, Board. I applaud your ability to solve the problem that was posed, but the question I have is what criteria do you use, or how do you make a decision that maybe it's not such a good idea to dispose of or directly dispose of aluminum clad fuel, and you go back to the melt/dilute as an option? I see the analysis and I see that yes, indeed, if we so chose and we could get the homogeneous mixtures, we could design it so it wouldn't go critical. When or where does the program plan to make a decision as to do we co-dispose or direct disposed, or do we melt/dilute?

HAUGHT: I view that decision as not being RW's to make. That is Savannah River's to make. They have asked us to help them assess the feasibility of direct disposal, and that's the work that we're doing.

Now, I would hope that we're given an input into that decision, but I would still say that is Savannah River's decision.

BULLEN: Do you seriously think that you could just set a waste acceptance criteria that says I don't take this stuff? I mean, that's RW's decision; that's not EM's decision.

HAUGHT: The waste acceptance criteria, that's correct.

BULLEN: Right. And so if you set a waste acceptance criteria that said it's too hard to make this not critical, let's make you guys dilute and then melt it, don't you think that's something that RW could do?

HAUGHT: I think, yeah, I believe yeah, we have the purview to do that. We could set an impossibly high acceptance criteria for these fuels that would basically force Savannah River into a melt and dilute option.

BULLEN: I guess the follow-on question is it's going to cost money to add gadolinium, it's going to cost money to make sure the boron, stainless steel plates are there. It might be a whole lot cheaper for you to just say the waste form acceptance criteria is, boom, and you're done. And then the cost benefit essentially comes back to here, where you take a look at the waste form that's coming out of the processing facilities that you accept from.

I guess the question there would be how do you communicate the cost differences? I mean, if you're going to do a cost benefit analysis, it's got to be the total system.

HAUGHT: I think with regard to our terms, and I'm really speaking off the cuff now, within some reason, whatever those acceptance criteria are is somewhat immaterial to us. You know, obviously, we don't want to throw things out to extremes, but I think within the acceptance criteria that I could imagine that would apply to a direct dispose versus a melt and dilute, the cost to RW to license that waste package is probably roughly the same, and given the quantity of waste packages that we're building, gets somewhat lost in the noise.

But, you know, again, I have kind of a range on what I envision those acceptance criteria would look like, and I'm assuming--I see that Mark just grabbed a microphone. I think he wants to weigh in.

BULLEN: Other questions from the panel first, and then we'll do that. Alberto?

SAGγIS: Sagηϑs, Board. So if I understand the diagrams, most of the reaction products are not going to be expansive. They're still being considered in these scenarios, the volume of the resulting product after interaction with the ingressed water?

HAUGHT: Yeah, there is some expansion of volume. I don't think there's enough to exceed the capacity of the waste package itself. And I might add that there were three configurations that I didn't provide to you here in this package, primarily because they're not very interesting, one of which is where the materials have been flushed out of the waste package. And then there's the fully intact, and then, you know, the intact basket with the degraded, because in both cases, the criticality control features are in place.

But, Jim, do we have enough expansion to actually split the waste package open, or Peter?

GOTTLIEB: I'm Peter Gottlieb with the CRW M&O.

The waste package is one container. We could have the criticality because there's water in the waste package. There's obviously penetration of the waste package. Whether the waste package retains water or not once it's penetrated is a question that we analyzed both ways. We could have criticality even if the waste package doesn't retain water.

Was that what the question was?

SAGγIS: No, really what I meant was simply assume a number of reactions in the system must come in and creates products that have a certain molar volume, and then from there, you get the final volume of whatever was inside after interaction with water and oxygen and whatever else that may be in the package. So I was wondering if in these scenarios, the initial versus the final volume was considered, or if you're considering other things?

GOTTLIEB: Well, let me put it this way. The geochemistry code that was used, EQ 3/6, did a water balance for the water that's coming in and the water that's flowing out, whether the water is flowing out through holes in the bottom or overflowing through holes in the top. But there was a water balance which also included the water that would go into deposited minerals as a result of the reaction.

SAGγIS: So, anyway, do I understand then that the overall result of this was that the volume of the final product is not larger than the initial volume inside the container?

GOTTLIEB: Well, but the point is the initial volume inside the container is not necessarily the controlling parameter because you have water coming in.

SAGγIS: Right. Go I guess the question is after--

GOTTLIEB: So you could end up in a situation, although it's pretty extreme, you could end up in a situation where the whole waste package is filled with clay and other degradation products. More likely, much of that will have overflowed, and so the volume in the waste package won't be more than what was there originally. But it would never--you always have sufficient holes in the waste package for the water to get in, such that whatever was--any reaction products which increased the size would overflow the waste package. It wouldn't burst it.

SAGγIS: But of course you encounter a whole bunch of situations in which if you allowed something inside the material, there is enough process to allow for the water to come in, but the reaction product has a physical makeup such that it does not allow for it to come out through the same holes through which the water came in.

GOTTLIEB: Well, if you were to get very rapid reactions, that is possible. These are very slow reactions. Everything that we are modelling with the geochemistry code indicates these are reactions that take place over thousands or tens of thousands of years.

SAGγIS: Okay, thank you.

BULLEN: Other questions from the panel? Carl?

DI BELLA: Carl Di Bella. After hearing the last two presentations, I have a question for Mark Barlow, if he's still in the audience. And that is why wasn't the dissolve and dilute technology included in the alternatives studied?

BARLOW: The dissolve and dilute option was evaluated by the Research Reactor Task Team and in fact it was I think, and I'm going by memory now because I wasn't directly involved in that, but my recollection in reading the report was it was judged unfavorably compared to the other dissolution options. So it was not carried forward to the final grading, but it was considered.

BULLEN: Any other questions? I have one final one that maybe I didn't get answered when I asked it originally. Is there a mechanism whereby either RW or EM can decide not to direct dispose? And if so, how does that happen?

HAUGHT: There is a mechanism, and in fact one of the ways--in fact there are mechanisms, when you get right down to it--one of the ways that that could be done is, as you said, we could give Savannah River an acceptance criteria that they couldn't meet. Before we did that, I think we would have to get together and make certain that such a decision would be in the interests of both of our programs in total, not that we would have to research a consensus on that, but that, you know, you would have a net benefit to it.

Other than that, it's really just a matter of working closely with them, telling them the kind of things that will have to occur in the design of their canister in order for an acceptance criteria to be met for direct disposal. I mean, one person from Idaho has described to me that, you know, right now what we've got--we have an acceptance criteria right now, and that's commercial spent fuel o borosilicate glass, and we could do that. But we are working with the National Spent Fuel Program to try to come up with some other different categories along with it, and we believe that that's probably of the greatest benefit to both of our programs.

BULLEN: Mark Barlow?

BARLOW: Mark Barlow from Westinghouse.

Westinghouse of course won't be making the decision, but we will be making recommendation that will reflect the waste acceptance criteria and the input and the feedback we've gotten both from RW and USNRC, National Academy of Science.

The decision that the Department will make right now regarding technology would be reflected in a Record of Decision to what was referred to as the Site Specific EIS. The current anticipated date for that Record of Decision is I think September of October of next year. And along with that decision will be a decision regarding this treatment and storage facility at the Savannah River site, and which fuels would be reprocessed and what kind of treatment would be involved.

BULLEN: Bullen, Board. So that would provide the input from your National Research Council review, from the NRC, from DOE, and then potentially any comments that we might have with respect to that?

BARLOW: And any stakeholder who reviews the draft EIS would have input that would be considered in that Record of Decision.

BULLEN: Okay, thank you.

At the risk of finishing four minutes early, and I'll thank Dave for that, we are now adjourned until 1:25, when we will reconvene with the presentation about Savannah River Site.

Thank you.

(Whereupon, the lunch recess was taken.)

AFTERNOON SESSION

BULLEN: Good afternoon and welcome back. Could I ask the Board members to come up and take their seats at the table, and everyone else to grab their cup of coffee and have a seat, please?

In our afternoon session, we're going to shift gears a little bit away from the highly mixed uranium and aluminum clad spent fuel and talk about, first, a little background on the Savannah River Site and the Defense Waste Processing Facility and then some of the characteristics of the vitrified high-level waste and the disposition of surplus weapons plutonium.

Our first speaker is Charlie Anderson. Charlie Anderson is going to give us an introduction to the Savannah River Site. He is currently the director of reactors and spent fuel division and responsible for the management and direction of the reactor programs and the spent nuclear fuel program.

Charlie?

ANDERSON: I'm going to begin with a map because I could tell from discussions during lunch with several people that the size of Savannah River is probably something that a lot of people have a hard time visualizing. Savannah River Site is 310 square miles, thereabouts, about 190,000 acres, and right here is the administration area, what's commonly referred to as the 700-Area and includes the materials manufacturing facility. But, as a flow of material during the days when Savannah River was in production reactors, the fuel was fabricated here in M-Area. We had five production reactors which are basically--here's C, K, L--the colors don't come out here--R, here and here; so a kind of semicircle around the center of the site with the Canyon Separations Facilities here in F-Area and in H-Area where also the tank farms that are associated with each of the Canyon Separations Facilities. Then, DWPF right here, and Saltstone at this point here. There's some other facilities and I'll show a few slides here, some of those as we get through, but the 400-D Area and the pump house and this area right here. I was trying to think, I'm not sure. The dimension across here is right around 15 to 20 miles. It is 20 miles to give you some idea then of the dimension. And, of course, the Savannah River runs right along in through here just to give a feel and a flavor for the size of the site.

It was established in 1950 by the Atomic Energy Commission; its purpose to produce nuclear materials for national defense. In 1972, it was designated as the nation's first environmental research park. The original facilities, the fuel and target fabrication facilities, I'm going to show some of these here. This is what is in M-Area where the fuel fabrication facilities are. This is a picture of one of the reactors. This is K-Reactor. In this picture here, this is H Canyon, and then as you look in the background here, this is DWPF, the glass waste storage building. Over to the left right in this area just to the left would be the tritium facilities, and right here is the first part of the tank farms. In fact, in this area right here, were the first four tanks in H-Area and the tank farm would extend that in this particular area here. The term "Canyon Facility", if you were to look inside this building here, particularly during a construction area the way it's arranged, it looks like canyons where the process vessels fit within the shielded construction.

Another picture gives a little different view that shows--actually, if you put these two together, you would see quite a bit. This is the tritium facilities here with this being DWPF and Saltstone right here in the back. These are Saltstone vaults with the Saltstone Processing Facility right here. And, this is another part of the tank farm right here where the new style Type 3 tanks which are full secondary containment type tanks.

Also, in this area, this is RBOF, the Receiving Basin for Offsite Fuels. This is an interior shot of RBOF showing where we receive and have been receiving for some time the fuels from research reactors across the world and domestic research reactors in the United States, too. This is the basin then where they are received and a good bit of them are now stored. The reactors, all five, pretty much, you know, are very similar as far as when you look at them. There would be different outbuildings and that type of thing.

It was originally operated for 40 years, 40 plus years, by the DuPont Corporation which is the difference at Savannah River Site as compared to the other sites in the DOE complex. There were a lot of contractor changes over the years at a lot of the sites and Savannah River for 40 years had the same contractor. Since that time, Westinghouse since 1989 has been the operating contractor for Savannah River Site.

Of particular notice at Savannah River Site is that there's been a number of facilities that have been brought on line recently in the last few years. The Defense Waste Processing Facility which Neil Brosee is going to talk more about here shortly, the Saltstone Facility, In-Tank Precipitation which are feed preparation facilities for the Defense Waste Processing Facility. A lot of people refer to Defense Waste Processing Facility as the glass plant, the vitrification facility. The Consolidated Incinerator Facility has been brought on in the last, oh, over a year and a half, I guess, right now--about a year, right at a year ago. Replacement Tritium Facility and Low-Level Waste Vaults, a significant change in how we dispose of low-level waste. In fact, that was something that I wasn't able to get a real good picture of, but looking at the site. I'll go back here; I meant to mention that. Right in this area right here is where a lot of the solid waste disposal facilities are, both the vaults and some of the true waste type interim disposal.

Major facilities restarted, F and H Canyons, heavy water purification, and then tank farms have been in operation for quite a few years. Even in those areas since they never really stopped operation, there's been some major milestones that have been accomplished there, too; the first tank closure in the DOE complex and, of course, a lot of infrastructure and construction in that area, as far as being able to support the retrieval of the waste and feed to the Defense Waste Processing Facility and cleaning out of some of the older style waste tanks.

Spent fuel basins, a lot of times people said my division's title is reactors and spent fuel division and none of the reactors are operating now. There's been a major change in the drive for the operating staff there that instead of operating a reactor, they are becoming much more as basin managers and looking, of course, at managing the special nuclear materials of the spent fuel in the basins and also how the enriched uranium and heavy water and other materials that we do have on-site that we're managing until the ultimate disposition of them is determined or carried out, depending. Some of them where there is a path forward and others where we're still doing some studying to determine the final disposition.

With that in mind for the facilities and all, the primary missions at Savannah River Site, high-level waste processing to glass vitrification, environmental restoration, making a lot of progress in areas such as seepage basins and a lot of other areas across the site where we're going back and closing and returning things back to a stable condition. Stabilizing the nuclear materials left over from weapons production programs, a lot of emphasis there in order to put them in a stabilized form. Spent nuclear fuel receipts, obviously, the Foreign Research Reactor Program which is a 13 year program. The main intent was to recover the highly enriched uranium that was provided by the United States to foreign countries under the Atoms for Peace Program. Research and development programs, there was a lot of research and development over the years at Savannah River Site for production missions. There's a lot of research and development now, particularly in the last 10 years and it's still heavy now as far as environmental restoration and material stabilization in order to achieve the other missions at Savannah River Site. Then, the economic development efforts. Trying to turn over defense capabilities and defense attributes at the site and see if they can't be developed for peace time or turned into something that can help the area as far as economic development.

Operating philosophy always has been an increased emphasis for safety, disciplined operations, cost effectiveness for all of us that look at our budgets and how the cost of doing business has definitely risen. We've had to take a hard look and try to make that more effective, try to cut down on our costs, continuous improvement, and a teamwork that values people. When we look at the budget here, one of the biggest portions of the budget are the people at Savannah River Site.

Current staffing. Primary contractor, it says Westinghouse and partners. Westinghouse had a five year contract of its own before, and when that contract was renegotiated, they teamed up with several other contractors also. So, that's the reference here to Westinghouse and partners. It includes Babcock and Wilcox and BNFL and a number of other smaller contractors, about 13,000 people. Wackenhut which is the security contractor for Savannah River Site which is 750, our other contractors around 500 with a DOE staff at about 550 at this point, Department of Energy. So, that brings the total staffing a little less than 15,000 people at the Savannah River Site. With the budget as shown here--and, even though these carry out to four significant figures, I really intended to revise this chart to be a little--because it does vary up and down. There's lots of discussions about budgets around a $1.4 billion range.

Future of Savannah River Site. Ongoing missions, obviously not only the accomplishment of environmental restoration, but demonstration of new technologies and methods of doing environmental restoration. High-level waste processing to glass, still in this country the only vitrification facility for high-level waste. Stabilization of legacy nuclear materials determining both the path forward and actually implementing the stabilization of those materials. Ongoing economic development efforts to leverage Cold War technologies and capabilities into the local economies.

Also, there's a lot of--as far as new missions for Savannah River Site. Most people have heard accelerator production of tritium; still one of the big goals. Additional nuclear material stabilization missions. As DOE as a complex looks at the various materials that they have across the complex, where they are, where they can be treated, there's a lot of discussion and looking at some of the facilities and the capabilities at Savannah River Site for those type of missions. Planned new facilities, actinide storage vault and spent fuel transfer and storage facility which is--the spent fuel and transfer and storage facility is one that's looking at dry storage and preparation of nuclear material for the disposition into the repository. I apologize for having to leave a little early this morning, but I assume you all probably covered a lot of that information on the spent fuel this morning.

I meant for this to be fairly brief, but if there are any questions or anything, I can help. If not, I was going to let Neil--we'll get into the processes for DWPF.

BULLEN: Questions from the Panel? Actually, I have one and then Carl Di Bella.

You noted that Westinghouse had been the prime contractor for almost nine or 10 years now. Is there a change in the offing with acquisition of NBC? Does it look like there's going to be a change in the prime contractor? Or CBS, I'm sorry, get the right network here. Is there a change in the contracting that's going--

ANDERSON: In the name change, and as far as the contract is concerned, there is no changes that are planned as far as the contract part, no.

BULLEN: That would be foreseen, okay.

Carl?

DI BELLA: I'm curious. What were the materials that the materials that the materials production reactor made or perhaps concealed?

ANDERSON: Tritium and plutonium.

DI BELLA: Weapons grade plutonium?

ANDERSON: Yes. I'm not sure--it is weapons grade, but I mean, if you're going to get into a whole lot more detail than preliminary, I'm probably going to have to call in some help here.

BULLEN: Any other questions from the Panel?

(No response.)

BULLEN: Thank you very much. That was a very nice overview.

Our next speaker is Neil Brosee. After a distinguished career in the military and in civilian nuclear power, he joined the Savannah River Site or Westinghouse Savannah River Company in 1993 as a deputy manager of the Defense Waste Processing Facility. While deputy manager, he successfully completed both the code runs and the waste qualification runs for the DWPF, and just recently in August of '96 became the manager of the DWPF facility. He will be speaking to us about the DWPF.

BROSEE: Good afternoon. I'm going to cover the facility itself in an overview. The first thing to notice in high-level division is extremely unique in all of the areas on the Savannah River Site. The main reason is it's integrated. The Defense Waste Processing Facility cannot operate without every other facility in this division also operating. That's very unique in this area. It's not like a canyon or a reactor. I have to have receipt of sludge, precipitant. I also have to have the tank farm to receipt for my recycled waste. So, the main point to gain is that the only way that this facility runs is with every other facility in the high-level waste division also operating.

Size-wise, going from Charlie's overview of the Savannah River Site, the waste right now is about 34 million gallons of waste stored in 51 underground tanks. I actually should say 49. We have closed the first two tanks in the complex. The first tank was actually closed in July of this year, and we just finished closing the second tank this month. So, it was the first two tanks; there are now 49 underground tanks. The extended sludge processing section of the tank farm provides the sludge for my facility. Likewise, the other part of the in-tank precipitant and the late wash provides the precipitant to the facility, and I'll go through the process. Likewise, I send the recycle back to the tank farm. The canisters are moved and stored in the safe interim storage of the glass waste storage building.

Just to capture some of the time frame since you were last here, in 1993, we completed our cold chemical runs where we actually proved the process with the cold chemical feed. In 1994, we initiated melter heatup. We initiated melter feeding, and we poured our first non-radioactive canister. Later in 1995, we completed the waste qualification runs which we're going to provide an overview of the details of what that included, but that provides a test of the extreme range of the glass that we can make and its acceptability. We completed the DOE ORR, operational readiness review, and on March 12, 1996, we commenced radioactive sludge operation.

The mission is very similar to, as Charlie pointed out, the one item of note that I'm very proud of is the first one. First of all, in October of this year, the site was awarded the ISO 14001 certification. We've also gone through the DOE's review of the VPP program for our safety program, as well as the DOE review of our integrated safety management system, both Phase 1 and Phase 2. So, we provided an integrated safety program that performs this evolution of vitrification.

This is a very simplified view of the entire process. I'll walk through the entire process and then I'll over each area in detail. As I mentioned, we receive the sludge from the tank farm into the chemical processing cell. Two main areas right now are the SRAT which is the Sludge Receipt and Adjustment Tank and the SME, Slurry Mix Evaporator. There are several other tanks involved, but those are the two of concern. The feed is then prepared. It is then transferred to the melter which has been continuously energized since its initial heatup. In the melter, we then melt the glass or the frit slurry and then we pour into stainless steel canisters and then the canisters are then deconned, welded for final closure, and then moved one at a time for interim storage in the glass waste storage building.

Presently the heart of the melter feed preparation is the chemical processing cell. We receive the sludge into the SRAT which is the Sludge Receipt and Adjustment Tank. It is chemically adjusted, and for the present, we are using simulated PHA for the precipitant side. It is then processed and transferred to the Slurry Mix Evaporator. At this point is where the glass frit is added to the Slurry Mix Evaporator to start the blending process. It is then transferred to the melter feed tank which is an interim tank before it is fed to the melter. Likewise, this shows the connections back to other systems which I need to have functional in order to operate.

In the melter which is operating approximately 1150 degrees Centigrade, the feed is fed into the top and it forms a melt pool right here. There are electrodes actually sending current through the glass. It's a jewel heated melter and that actually melts the frit slurry to make molten glass. The frit slurry that has not yet melted forms a crust or a co-path that floats on the top of that melted pool. The offgas is taken from the top of the vapor space and we have the vapor heaters or dome heaters, as we call them. The glass is then moved up through the pour spout and poured into the stainless steel canisters. The stainless steel canisters are approximately 10 foot tall, two feet in diameter, and they're three-eighths of an inch thick stainless steel. At this point when the canister is full, there is a temporary tapered plug that's inserted into the top of that canister. We then perform a helium leak check in order to show that it is airtight. We need that before we send it on into the decontamination process. If it fails that leak test, we then reprocess that can back through in order to have it pass before we send it into the decontamination chamber.

In the chamber itself, one-by-one, the canister is grappled by its flange and with the top section it is rotated and moved up and down while it's being blasted with a frit slurry compound. Think of it as sandblasting. That removes any of the surface oxide or contamination on the outside of that can so the outside of the can is meeting all of the DOT transportation requirements. That frit slurry that we used at decon is now sent back and is used into the Slurry Mix Evaporator for the next batch. So, that's one way that we reduce that type of waste. We just reuse it into the next batch that we process.

At that point, it is moved out of the chamber. It is smeared to show that it has been cleaned or it is re-decontaminated and it is sent to the weld test cell. At that point, there is a hydraulic ram that pushes the tapered plug into the throat of the canister and puts the actual final welded cap in place. The cap itself is actually larger than the flange and this is an upset welding; in fact, it's the largest upset welder in the world. We press that top into the flange while we're applying 240,000 amps for one and a half seconds with about 80,000 pounds of force. That then provides a weld that's as strong as the three-eighths thick canister itself. It is then done a final smear and then transported to the glass waste storage building.

The glass waste storage building is a seismically reinforced concrete block below grade. The SCT is a unique device. It alone weighs close to 120 tons. It handles each canister at a time. It has a shielded container that pulls the canister up inside. It moves; its top speed is about five miles per hour. We only use two miles per hour. We transport it from the vitrification building to the glass waste storage building. The machine is powered by two diesels for reliability. It lifts the plug from the floor and that's about a four foot thick plug. It then moves the trolley forward, puts the canister in its slot, then rotates the trolley back and puts the plug back in the hole, and then goes back and picks up the next one. There's over 2200 slots in this first glass waste storage building for storage. It does have forced ventilation, but the natural air flow is designed to come in. The air arises around the canister and actually will cool itself.

For the three years since March 12 that we've been running radioactive sludge, the first year we had a set goal of 60 and we beat it. The next year, we had a set goal of 150 more and we beat that. And, right now, you can add five more to that. We just finished the 39th can today. So, for a brand new one-of-a-kind facility, that's the first three years of production. We did pour the one millionth pound of vitrified glass on the 3rd of November with the 14th canister this year.

Now, the waste acceptance process is just as involved as the process itself. We have established the waste acceptance product specifications that we meet. These basically have five sections. There's three technical sections, there's a QA section, quality assurance, and then the requirement for documentation and administration. The way in which we meet that at the Defense Waste Processing Facility is through four steps. We have the waste form compliance plan which is a general description of the process and the methods by which we meet these specifications. We then have the waste form qualification report which is at least 13 volumes of all the tests that we have performed and the results that show the compliance with these specifications.

We now have the production records which we use for every can and every batch to show that we have complied during the production with this program that we have established. And then, we have the final storage and shipping records in order to show while it's in the glass waste storage building its storage conditions.

Two basic inputs form the production records. First of all, for every canister there's a canister wallet which shows the canister that we use, the supply, the fill height, and various data points in the process as we measure according to our compliance plan for every canister. So, for 272 canisters, we have 272 canister wallets. Every procedure that was used showing all of the data. Likewise, we have a batch wallet for the chemistry side of the process so that every SME batch, Slurry Mix Evaporator, we can show the chemical composition, the batch acceptance, and the radionuclide inventory. So, there's two types of wallets that form information to the production records.

Now, I want to look at the physical process with these controls in mind. First of all, we have taken a sample of the sludge in the large tank in the tank farm. That's what we call the macro-batch. We have a complete analysis to show the chemical and radionuclides used for that macro-batch. The tank we're now using is about 500,000 gallons. We go through the various processes and down at the Slurry

Mix Evaporator is our hold point. We sample every Slurry Mix Evaporator batch, and if it does not meet acceptable standards, it is then remediated. So, we do not process it further until it passes this hold point. We then feed it into the melter feed tank and go on into the physical process. Likewise, we sample the melter feed tank for every batch and periodically we take a glass stream sample from the pour stream again to show compliance on every canister and every batch.

That's basically the overview of the process and the control process.

BULLEN: Thank you. Questions from the Board? Carl?

DI BELLA: Can you tell me how many canisters one melter feed batch tank is equivalent to?

BROSEE: Oh, let me answer that from the Slurry Mix Evaporator. Every batch we move forward is about six to eight cans. The melter feed tank is kept continuously varying level because the batch is the control point, is the Slurry Mix Evaporator. We process about six to eight cans in every chemical batch.

BULLEN: I have a quick one. You mentioned that you check it for a leak rate parameter on the cans after you've poured them when you put the conical slug in the stop. If it doesn't pass the leak rate, how do you remediate? Do you go back and repair the can or do you have to re-pour it into a new container?

BROSEE: There's two things that we can check for first. The first thing we do is we check the gasket of the test device itself against a known flange because the gasket we use wears. If that's the failure, then we replace the gasket and retest the can for acceptability. If the can itself has failed, then we have what is known as a repair plug. We actually push the tapered plug and sleeve into the throat of the canister and we insert the repair plug which is a straight plug into the can and we re-leak test it. From there, it passes on.

BULLEN: So, most of your failures are in the neck area. They're not seam welds or side welds on the can?

BROSEE: Absolutely. That's correct.

BULLEN: If there were a failure of a side weld, you'd have to go back and chop the can up and start over or how would you remediate that one?

BROSEE: We have not had that condition occur. The process would be put aside and we would have to build a repair plan for that canister. Any canister that does not go through the waste form compliance plan as we have written it has to have a unique repair plan that is approved by DOE before we can use that repair. And, that would be one that we would have to do that with.

BULLEN: Okay. As a followup to that question with respect to your storage in the glass waste storage building, do you monitor can degradation? Do you periodically inspect or do you have a plan that you will inspect prior to shipment of how the cans are going to degrade in the environment? It's a pretty aggressive environment to pour 1150 degree C glass into a stainless steel container, and after you've cleaned the surface, it's going to basically be re-oxidized, but you'll also have some potential for pitting corrosion and the like in the moist air environment that you have in that storage building. Do you inspect and how will you document those kinds of issues?

BROSEE: There are three answers to your question. First of all, we have done several studies just on the initial design of the building itself. The thickness of the 304 L stainless steel has various corrosion rates and we already know all that information. That has gone into the design of the glass waste storage building. The second part of the answer is we have put several canisters of non-radioactive glass from the previous test periods in the glass waste storage building in various locations. So, periodically, we'll pull the non-radioactive cans out and do tests on them to see if our calculations, assumptions, and projections are, in fact, true. Then, before shipment, we will have to do some verification prior to the final closure of the storage record.

BULLEN: As a followup to that, the cans that you've poured, thus far, have not had a very high radionuclide in loading; so, they're not extremely radioactive?

BROSEE: That is correct. They are not.

BULLEN: So, surface radiation dose is actually an area that I'm interested in. What comes to mind is the Climax Mine test where they put fuel assemblies in 304 L in the ground and they had alternating heaters, fuel assemblies, heaters, fuel assemblies, and when they pulled them out, the radiolytic decomposition of the moist air nearby the stainless steel had enhanced corrosion greatly around those while the ones that had heaters in them were as pristine as when they went in. And so, the question I have is in the ventilated building that you have, as your glass becomes hotter--meaning more radioactive, not thermally hotter--as the glass becomes more radioactive, you have the potential for in a moist air environment in South Carolina, as we are, for enhanced radiolysis and enhanced corrosion of those cans. I just wondered if there were plans--and you mentioned there were--to monitor this, but to monitor actual pour canisters as opposed to your test canisters?

BROSEE: We would not monitor the pour canisters unless the non-radioactive canister shows some reason why we should.

BULLEN: Okay. But, they're in enough of the radiation field to see the same kinds of effects?

BROSEE: That's correct.

BULLEN: Okay. Carl, you had another question?

DI BELLA: Yes, I do. You mentioned you measure the glass coming out of the melter periodically. How does that--what is that period and how do the composition results agree with your melter batch tank composition?

BROSEE: One of the things that we'll cover a little bit more in detail--of the actual chemistry of the glass. We take on a requirement one glass sample per macro-batch. We are taking a little bit more frequent than that right now. We are taking a melter feed tank sample on every batch, every batch that we move forward from the Slurry Mix Evaporator. And, we have the comparison between the MFT and the SME which is matching very well. We have not completed all of the information for the actual radioactive glass pour stream samples yet. That is still ongoing.

DI BELLA: I don't understand the units that you're using. How many samples have you taken of the glass in--

BROSEE: For instance, right now, we are on the 60th batch from the Slurry Mix Evaporator. We have 60 samples from the Slurry Mix Evaporator.

DI BELLA: Okay.

BROSEE: We have also 60 samples from the melter feed tank. Right now, we have approximately seven from the pour streams themselves.

DI BELLA: Okay.

BROSEE: The requirement is one per macro-batch.

DI BELLA: Okay. My other question was you said you're using a surrogate for that PHA stream?

BROSEE: Precipitous hydrolysis aqueous, yes.

DI BELLA: I guess the real PHA stream, you're not yet feeding to the melter; is that it? And, is that a significant stream?

BROSEE: Right now, because of ITP, the in-tank precipitant chemical process, the actual radioactive precipitant side of this is not in service. We did test the precipitant side during the waste quarrel runs with again non-radioactive similar to its full degree. So, the salt cell has been tested to the full range of glass production that we have written into the waste form compliance plan. Right now, we're adding acid in order to make the simulant look like it's receiving the precipitant in the SRAT.

BULLEN: Alberto?

SAGγIS: Yes. So, what is the fate of the water in the waste tank then? Like, where does most of the water go after this process?

BROSEE: Where does the waste go?

SAGγIS: The water?

BROSEE: The water goes back to the tanks in order to be re-evaporated and then reprocessed again if it's high enough level waste. It's through underground transfer lines that we send back to that tank farm that's on the other side of the canyon.

SAGγIS: I see. So, no water goes out the stack then?

BROSEE: No, sir, it does not. Any of the water is trained from these things and sent back through underground inter-area transfer lines to the tank farm.

SAGγIS: Uh-huh. So, they--after basically zero humidity?

BROSEE: That's correct. We do have a sand filter. It has roughly a DF around 200, seismically qualified and again under-grade.

SAGγIS: So, the water will end up basically in the Saltstone?

BROSEE: Well, the water--if it's high-level--in other words, if it's mixed back, evaporation will actually just have it removed. If it is part of the filtrate, it will end up going to Saltstone as the low-level waste, grout, or saltstone.

BULLEN: Any other questions from the Panel?

(No response.)

BULLEN: I actually have one more. I may have missed it, but you're on target, on schedule, and you've got 49 tanks left. When is the job one and how does that coordinate with the opening and closing of the repository? I guess that's kind of a loaded question, but when is the job done is the bottom line?

BROSEE: Right now based on the available existing waste in the tanks, we project around 6,000 canisters to handle the existing tanks. Right now, our accelerated cleanup plan is based on the repository starting to take shipments in the year 2015.

BULLEN: We have a little bit of time. Are there any questions from the audience?

(No response.)

BULLEN: Well, if not, we'll forge ahead. Thank you very much.

We'll get to more meat and potatoes with our next presentation which is going to be on the characteristics of the vitrified waste form. Our presenter is Sharon Marra. She is the manager of the chemical processing and analytical division within the DWPF engineering department, and she has been responsible for the development and implementation of plans and programs to insure the acceptance of the waste form as produced.

MARRA: Well, good afternoon and welcome to all of you to the Savannah River area.

Before I start, I wanted to give a little perspective. This Board, the Nuclear Waste Technical Review Board visited us in February of 1992, and at that time, DWPF was finishing construction and getting ready for the startup test program. I realize many of the members have changed since then, but at the time we presented to the Board our strategy for how we were going to assure an acceptable glass product and how we were going to collect the records to be able to prove in the future that we made an acceptable product. And, just to give a little perspective, when I was preparing for this presentation, I went back and looked at what we presented to the Board at the time. And, while we don't plan on getting in as much technical detail today, the program has really stayed fairly stable over that time. I'll talk a little bit about our startup test program and how we proved the program was adequate and how we're using it today. I just wanted to point that out. There's always changes in implementation and operating procedures and things like that, but over that time period, we've really had a fairly stable approach and it has proved successful.

Neil talked about the waste acceptance process and the waste acceptance product specifications. I want to just give a quick overview from the glass perspective of what those require us to report and I'll get into a little bit of data later on. We're required to report the chemical composition of the glass on an oxide basis. We're given a limit of elements that are present at greater than .5 wt%. We're also required to report the radionuclide content.

Those radionuclides that contribute to greater than .05% on a curie basis to the total inventory at any time up to 1100 years after production, and they also have to have a half-life of greater than 10 years. We also report the uranium and plutonium isotope content to meet International Atomic Energy Agency requirements. And, finally, and we'll talk a little bit more about this report, Product Consistency Test results and I'll talk about what that is. That's really the limit on insuring glass acceptability. We have to verify in our records, as Neil talked about, that we have made an acceptable product.

There's a couple other requirements I'll just mention that we're required to report at the time of shipment. Again, the previous ones were at the time of production. In other words, today when we finish the can today and these will be whenever we ship a particular canister. And, those are dose rate and heat generation. You can see the limits there. A limit of 1500 watts per canister and I'll talk in a little while about where we are compared to that limit. And then, the dose rates, the gamma and neutron dose rate limits.

Jumping back a little bit really where Neil finished off on glass product control, the requirements that we have require that we control our process so that our glass is better than a benchmark glass. And, that benchmark glass referred to here is the environmental assessment glass. It was a glass that was used in the initial environmental assessment prior to the construction of DWPF. Our limit is based on a product consistency test which is an ASTM certified test. It's a crushed glass leach test. We use that to insure that we produce an acceptable product and a consistent product time after time. The way the test works is the glass is crushed, put into ASTM type water, heated at 90 degrees C for seven days, and then we analyze the leachate. We look for the elements of lithium, sodium, and boron. And, previous work over the years has shown that these three elements leach out the fastest and are a good representation of what the quality of the glass is.

In order to meet that requirement, DWPF developed this glass product control program and that's the program I mentioned earlier that we spoke to you about five years ago or so and that we demonstrated through out startup test program. What it basically entails--and Neil talked a little bit about this--is that we control our feed composition at the last feed preparation vessel; in other words, the last vessel where we can make changes to it. That's the Slurry Mix Evaporator that you saw in Neil's slides. The program also requires that we provide the documented evidence that we have done this and again that gets into the wallets and the production records. Also, that the program be robust enough to handle changes and I'll have a couple of slides on this of how we demonstrated that during our startup testing in our waste qualification runs.

Just very quickly not to get into a lot of detail, we talk about controlling the process and controlling the product. We have three extremes; the frit itself, the PHA which as Neil said we're simulating right now, and then the sludge. We have to make a feed material that will meet all of our constraints. One of our constraints is this product consistency test durability. We insure that by controlling the chemical composition and we have a correlation of that test to the chemical composition. We also have constraints for processing prospective viscosity. And, again, those are based on chemical composition and there's also some glass solubility limits. When you take all those limits together and you take these three feed streams, you get a region which represented here by that dark region of--if we're anywhere in that region, we're going to meet all these constraints and be able to process it and produce an acceptable glass.

Now, because we analyze the composition, we recognize that there are uncertainties associated with that composition; variability in our laboratory, things like that. So, we make the window a little bit smaller by taking into account those analytical uncertainties so we have a very high confidence that our product will be able to be processed through the system, as well as meet the durability requirements.

Now, I mentioned these waste qualification runs. Just to give you a quick idea of what we did during those runs, this FA-13 designated our initial chemical operations. We prior to that used startup frit in the melter to get the melter heated up and we produced two batches of feed at that time and fed those to the melter and flushed it through the melter. The feed that we used was representative of really all the waste in the tank farm. We called it a composite feed. These were simulants. This was a simulated sludge and a simulated PHA that we used at the time.

Once we did that, we got all the systems ready. We had the startup flushed out of the melter and we began these waste qualification runs. And, during that campaign, we produced 56 canisters of simulated waste glass. The majority of these canisters were destructively examined and I'll show you some of the results of that. What we did during each of these campaigns, as we called them, is we buried the feed drastically. We started out with again a composite feed, but instead we doped it with a neodymium tracer. The purpose of this was to track that neodymium through our process and through the melter to determine whether we had a plug flow situation in the melter where you'd have step changes in composition or whether we really had a well-stirred tank which is what the expectation was. As it turns out, we had a well-stirred tank, a slow increase in the neodymium content in our glass product.

The next campaign--and I show here this transition from a composite type, sort of run-of-the-mill feed material to a high iron feed which led to a low viscosity feed material. We were focusing on viscosity here because viscosity was the one parameter that we wanted to make sure that we wouldn't get any segregation of the feed. We wanted to make sure that we could make these transitions from a baseline to a low viscosity and vent it back to a high viscosity and we could still control our product, this program still worked, and we could still make an acceptable glass. After that high iron feed, we jumped to a high aluminum feed which was a high viscosity feed. Again, that was our most severe transition. During that time, we ran through the whole process, as Neil explained it, watched melter behavior, and then destructively examined the glass to make sure that we were still in compliance.

Then, the final campaign was back to a blend type feed that was similar to what we would be producing during the initial radioactive operations. So, we transitioned again from a high viscosity back to a baseline type feed. As I said, we produced 56 canisters during this campaign. These canisters were sectioned, had windows removed from the wall of the can and glass samples were taken. Tomorrow on the tour, you'll see the bottom portion of the section of one of these cans in the lobby when we walk you through the building. All these glass samples that were taken were analyzed for this product consistency test I mentioned, as well as chemical composition.

Carl asked the question about how our prediction was comparing to the actual result. This line here shows the predicted PCT result, Product Consistency Test, on a gram per liter basis for boron. These are all the campaigns that we produced during that waste qualification runs. And, you can see it's fairly stable. It jumps up a little bit and jumps down, but really is fairly stable. And, we're down in the less than one region. Our limit is way up here. It would be off the chart. If I drew this to scale, it is 16.7. So, you can see how far away we are from our limit. The other important thing to point out is this line here is the predicted PCT value based on our batch analysis. And, you can see the straight lines indicating that this particular batch made roughly seven canisters; the next batch maybe made five canisters. So, you can see that jump up there. And, they really track fairly well and we did a lot of statistical analysis on it to show that we were well in control and that what we predicted was represented by what we actually tested in the glass.

Just to give you a feel for what this glass is made of and what it looks like on an oxide basis, this is a range for these major components of the glass of what we expect our glass to look like over time. This range is based on analyses that have been done of the sludge in our waste tanks. Neil mentioned our planning process. We do have a planning process where we designate what sludge batch will come next, but this range represents what we expect to see over the lifetime of the facility. As you can see, iron is a major component of the glass. There's some sodium in there, aluminum, things like that.

Now, jumping to radioactive operations, where are we today? Again, back to this Product Consistency Test, this particular column here--and I got the results for the first 43 radioactive feed batches--this is the average of what we predicted for each of these feed batches, roughly .7 grams per liter. As Neil mentioned, we pulled about seven pour stream samples, tested four of them for Product Consistency Test, and the actual results are approximately .9 gram per liter; again very, very close within statistical significance. Again, you can see the limit there; far, far above where we're operating our facility.

Now, over on the radionuclide inventory side, there was some questions on where we are now. On a curie content per canister, this represents the major isotopes, the DWPF design basis glass. This design basis glass is--you can think of it as a binding type case. It was used for some of the design of the DWPF facility from the perspective of shielding and environmental considerations. As you can see, the curie contribution there, fairly evenly split between strontium 90 and cesium 137. Then, if you look at where we are today, this Tank 51, that's the tank we're feeding out of now. So, this is information of the glass we're producing today. You can see the strontium 90 is the higher contributor for curie content. That's because we don't have this precipitant feed.

Yes?

DI BELLA: Could you clarify what Tank 51 means? Is this the batch melter feed tank?

MARRA: No, I'm sorry. This is the waste tank in the tank farm, that 500,000 gallons that Neil mentioned.

DI BELLA: Okay.

MARRA: That's the tank that we happen to be feeding from today. The melter feed tank batches are--we've made approximately 50 of those to date. Yeah, as I mentioned, the strontium 90 is the higher curie contributor because we don't have that precipitant feed on line. The precipitant feed will contribute most of the cesium 137 to our glass.

BULLEN: Before you leave that one, all the radionuclides that are listed on the bottom, are those that are required because they are greater than the .05% of the curie concentration with half-lives greater than 10 years?

MARRA: That's correct.

BULLEN: Okay.

MARRA: That's correct. Yeah, I just listed the major ones here, but these are the other ones that we're required to report, as well.

I believe there was a question earlier about heat generation. Where are these canisters in comparison to design basis and limit? The DWPF design basis glass again led to 750 watts per can. Again, that's a bounding case extreme for waste at Savannah River Site. As you can see, the canisters we're making now, as we mentioned earlier, are on the cool side, only about 4 watts a can. This next batch of sludge we're receiving at Tank 42 indicates the large tank in the waste farms jumps up a little bit, but still very much on the low side. The limit of 1500 watts per canister, the SRS DWPF canisters, unless some unexpected new waste stream comes in, wouldn't approach that limit, at all. So, typically, once we get our precipitant on line and we get processing, we'll probably be in the 200 to 300 watts per canister range.

I thought I'd just finish up with a slide that's a graph that's representative of one of our waste qualification documents on decay rates on a curie per canister and a watts per canister basis up to 1100 years after production. This is the design basis glass. So, this is the higher heat generating waste stream. But, the canisters we're making now, although they would start out much lower, would follow the same type of decay pattern.

So, that's all I have. If there's any other questions, I'd be happy to answer them.

BULLEN: Questions from the Board?

ARENDT: Are you subject to IAEA inspections?

MARRA: I'm not sure I can answer that. Bill? No, not yet.

ARENDT: Process control measurements versus quality control measurements. First off, are you in Neil's organization or what organization are you in?

MARRA: Neil is our program manager at DWPF. I'm with the engineering organization within DWPF, and we support Neil.

ARENDT: What I'm trying to get at is differentiating between process control measurements and quality control measurements. Are they one in the same or--

BROSEE: There's a different program associated with both. Besides the engineering evaluation of the glass samples and the product consistency, we also have a quality assurance group who independently look at my production. They look at the wallets. They are actually an independent arm under the quality assurance program for RW0333P.

ARENDT: Okay. On the canisters, I suppose the canisters are your design. Do you have any vendor inspection or do you inspect the canisters at the vendor? Do you do any in-house inspection of the canisters prior to use?

BROSEE: There's all of the above. First of all, we do have an inspection program that is done by the vendor at the vendor. And, we also have an independent quality assurance program that looks at his program while he's doing that. We then do a receipt inspection to show that there was no damage or change during the transportation from the fabrication location to our site. Before we actually move them into the canyon--and you'll see this tomorrow on the tour--we actually have a canister receipt area where we inspect for cleanliness, foreign material, and various inspections that we go through before we even put them into the canyon for processing.

ARENDT: And, I was going to ask earlier, but your final closure weld, do you do any inspection on the final closure weld?

BROSEE: We do visual only. We have done various burst testing on test canisters and also on test nozzles which have proven the parameters of our upset welding. Then, periodically, we do current tests in order to show that nothing has changed in that process.

ARENDT: Okay. Thank you.

BULLEN: Other questions from the Panel?

DI BELLA: From a repository point of view, I think it would be somewhat important to know on any given canister how well the radionuclide composition reported in the canister wallet matches with what's actually in the canister. I don't quite see from what's been presented how that is established. Could you elaborate?

MANNA: Yeah. What we're doing right now, these glass pour stream samples that we've pulled, obviously we couldn't test that when we were in startup testing like we could the composition. These glass pour stream samples that we've pulled now are being extensively analyzed for radionuclide content at the Savannah River Technology Center at SRS. We are comparing those results back to the melter feed--what we would report from the melter feed to make sure that we're in synch. And, we have a commitment to update our waste acceptance documentation to reflect any uncertainties or errors associated with that result. But, we couldn't do that until we got into initial radioactive operations.

BROSEE: One additional item. The production record that actually is shipped to the repository will have the information of the batch it came from, the radionuclides, as well as the chemical composition, the results of the Product Consistency Test, as well as if it had a pour stream sample, the results of the pour stream sample, in that document.

BULLEN: As a followup to that one, you haven't noted in any of your analyses that there is selective segregation of any of the specific radionuclides. It's pretty much well-mixed and always coming through or do you see spiked changes within the same batch of, I don't know, neptunium 237 shows up in a higher concentration in one than the other?

MARRA: So far, we haven't seen that. We've seen a very homogenous product.

BROSEE: In fact, what we did during the WP-14 that Sharon mentioned, we actually tested for that using the dope feed to show not only their stay time in the melt pool for the melter life, but also for the consistency as we would make that transition from FA-13 to WP-14 test material.

SAGγIS: I guess this is a two part question. First, any idea as to what would be expected rate of generation of gasses like hydrogen isotopes or helium and the like? And, the other would be how much of a dead space is left at, say at the top of the canister when you load it?

MARRA: Let me answer your second question first because I may need a little help from the audience on the first one. We fill the canisters, our target is approximately 96 inches. The container is 118 inches tall. That is equivalent on a volume basis of 90 percent of the volume of the canister.

BROSEE: The total volume of the canister is 26 cubic feet. We fill to 25 cubic feet and we need that extra volume for when we push the tapered plug and sleeve in for the final closure at the weld cell.

MARRA: And, as far as gas generation, we've done some studies on that and I'm not sure I could quote numbers. I don't know if Ned Bibler could help me if he's out there somewhere. As far as gas generation inside a canister, I assume you're referring to?

SAGγIS: Yeah.

MARRA: I'm going to let Ned help me out on that one.

BULLEN: Please, identify yourself?

BIBLER: Ned Bibler, Savannah River Technology Center. The only gas that's produced of any significant quantity in the canister is helium from the alpha particle neutralization. Tests have shown from helium implantation and from dope class with curium 244 that all of that helium that's produced will remain within the interstitials of the glass and not go up to the open space and no hydrogen. And, there's no water, very little water.

MARRA: Thanks, Ned.

DI BELLA: You showed that the canisters you've been making, so far, are very low wattage implying very low radionuclide loading. Will you be able to catch up and sort of recover for this or does this mean that many more canisters are going to go to the repository than originally had been planned?

BROSEE: The original 6,000 canister projection was based on the sludge which is a controlling item in the mix feed of the coupled operations. We do not expect to see a loading problem, although that is one of the areas that we will have to look at as we then bring on the coupled operations. But, the original projection from the existing waste was based on the sludge volume, not the salt volume.

MARRA: And, we're still putting into our canister 700 to 800 gallons of sludge per canister. It's just that this happens to be older sludge in addition to what Neil said.

BULLEN: This is a final question, I think, before the break. I guess it's a question of interfacing between the people at the Mountain who are doing performance assessment analysis and the types of inventories that you're producing. How do you provide communication? I know your wallets and your notebooks and all that are going to go in association with this. But, how do you provide communication to people now in the TSPA-VA and those kind of things with respect to inventories and potential release rates and mechanisms associated with that? Is there good communication or could there be room for improvement?

MARRA: Well, let me comment on one thing. Some of our requirements that I didn't talk about, in addition to all those reporting requirements, we have a requirement for projecting what we expect to produce. We've done that and presented that in our waste qualification documentation. That was extensively reviewed by DOE and DOE is working together to provide that information to the repository folks. So, someone else might want to comment on that, but I think we've provided the information and the communications are fairly good in that area.

BULLEN: Any other questions from the Board or Panel?

(No response.)

BULLEN: Questions from the audience?

(No response.)

BULLEN: Okay. Now, I'm going to take a risk here. It's a very nice day and there's a nice river walk right outside the door there. I want everyone to promise to be back here at 3:00 o'clock which is about 22 minutes from now, and we will reconvene.

Thank you.

(Whereupon, a brief recess was taken.)

BULLEN: Could everybody take their seats, please, and Ill ask the Board members to come back up to the front so we can get started for the final session of today.

Our final two presentations deal with the issue of the immobilization of surplus weapons-grade plutonium. We're going to have a talk from Bill Danker, first. Bill is the plutonium immobilization project lead for the DOE's Office of Fissile Materials Disposition, and he's going to talk about the immobilization of surplus weapons-grade plutonium.

DANKER: Good afternoon. I'm pleased to be here to talk about one track of this nation's dual-track strategy for securing surplus plutonium. With me today is Tom Gould who is at Lawrence Livermore National Laboratory and who heads up the immobilization research and development team. Tom will describe the immobilized form and how we're supporting repository analyses by the Office of Civilian Radioactive Waste Management and working closely with them. If you sense I'm not getting to an area where you have a question, please interrupt. I'm here to address topics which are of issue or concern to you.

This slide takes care of the toastmaster's requirement to tell how much you're going to tell them. I'll briefly review why we're trying to make big, heavy, radioactive objects. I'll describe progress since I last briefed members of the Board in January '96. Then, I'll give you a quick overview of where the immobilization project stands today. As I mentioned, Tom will close with a closer look at the form.

Four years ago, a number of reports, most notably one by the National Academy of Sciences, focused on the proliferation danger posed by plutonium being removed from warheads at an unexpected rate. The good news was arms control was working; the bad news was that Russia and the United States quickly needed long-term plans to secure huge quantities of weapons and usable plutonium. The United States established the Office of Fissile Materials Disposition later that year to focus on reducing the global nuclear danger posed by this material. I'll describe in a minute earlier this year the President and the responsible Government agencies decided that this country should use both existing reactors and immobilization to secure this material.

I hope this is right because I carved up a slide that had both plutonium and uranium on it. It should be accurate. The United States has declared about 50 metric tons of plutonium surplus to national defense needs, the bulk if which is at Pantex, Rocky Flats, and Hanford. This slide is a reminder of the plutonium quantities and locations. Tom, I think, will show you one a little later that gives you a little bit of a different cut at it in different categories, and I've got a backup slide that will show how you rationalize the 52.5 shown here and the 50 that he has. But, the bottom line is it's about 50 metric tons. While I haven't seen a comparable slide released by Russian authorities, I understand the declaration of total surplus quantities is due shortly.

I have to say I appreciate the Panel not scheduling in a blizzard in D.C. this week. The last time I briefed the Board, the snow complicated things. Since that time, the Administration has decided on the two-track approach for plutonium disposition. This decision was supported by three principal legs; technical, environmental, and nonproliferation analyses. I chickened out on lugging all of the reports with me, but the references here are available upon request from our office. Also, earlier this year, we issued a notice of intent, I think, in May to prepare the next set of environmental documents and therein said for immobilization we prefer to use the canister technology of the Savannah River Site. Principal attraction in doing it here is that the Defense Waste Processing Facility exists, as you just heard, and is producing high-level waste canisters. I don't think I'll dwell on this slide given the focus of this meeting. Note that it clearly shows that both plutonium disposition tracks result in forms which are intended to go to the repository which this Panel reviews.

The principal project driver is nonproliferation. We're trying to make this plutonium as unattractive and as inaccessible as that in commercial spent fuel. Making it big, heavy, and radioactive is one way to do that. I have to say this as an aside and as the son of a minister that some of the more theological discussions I've ever gotten into is on the concept of the spent fuel standard which is, in fact, not a standard, but a perspective. But, the words I gave you right now are the best take on what we mean by that measure of proliferation resistance; make this plutonium as unattractive and inaccessible as that in commercial spent fuel.

A related driver is urgency. I consider the 2005 date that you see here an aggressive schedule. While we have to develop a process that results in reliable, quality, cost-effective production, a key part of our development program is focused on providing characterization data to support acceptance for eventual disposal in the repository.

Pictures make it easier to understand what the can and canister looks like. If anyone hasn't seen a DWPF canister and you probably have given the previous briefings, that's the one in the middle between Leonard and Gene. Currently, we expect to put 28 cans, three inches in diameter, about 21 inches high inside each canister. These would be supported at--I'm sorry?

SPEAKER: 28 cans?

DANKER: 28 cans, yeah. 28 cans, three inches in diameter, 21 inches high inside each canister. These would be supported at four levels, seven cans in a circular ray given the current configuration. The picture at the right actually is one from one of the cold pours back January of 1996 and actually shows a spiral array of about eight cans.

Since the January record of decision retained the option of all 50 metric tons of plutonium being immobilized, this project for now is using that as a planning assumption. There's a range of feed material from converted pits to stabilized impure oxides to fuel. So, material conversion is important. We're assuming that there are enough canisters with enough fission products to support the mission. And, again, as an aside, we don't mind that they're running on sludge at the moment. As an example, if each can contained roughly a kilogram of plutonium, then about 175 canisters a year would be needed for the full 50 metric ton case. In a hybrid case, you have 18 metric tons going to immobilization. It would require about 60 canisters a year for the plutonium mission. We're currently expecting that the spent fuel standard requires 100r/hr 30 years after the canisters are poured. As the next speakers will discuss, we'll need to qualify for repository disposal.

DI BELLA: How much plutonium per can did you say?

DANKER: Current guess is about a kilogram per can--

DI BELLA: Okay. A can is a little can?

DANKER: Right.

DI BELLA: Okay.

DANKER: So, 28--right, 28 kilograms per big canister. We try and stick--we're not always successful, but we try and stick to the nomenclature of cans being the small ones.

It's a busy slide, but I'll stick to a few basic impressions I'd like to leave with you on this schedule. There are essentially two long poles in the tent. One is focused on activities needed to qualify for repository disposal which are shown in red at the bottom. I noted with interest Neil's reference earlier to the waste acceptance process being just as involved as the process itself, and that's probably a good perspective for us. Based on my time in that office, it is a complicated activity. The other is composed of activities needed to define the process and construct and start the facility. We're currently our way through the tasks identified in the latest immobilization plan describing development and characterization work needed to start up in 2005.

We might want to leave that one up, and if you could switch over--yeah. It's dangerous. It might stimulate questions. We're currently focusing our work on ceramic forms. In July and August of this year, data that had been gathered on both glass and ceramic forms were evaluated in an intensive process described by this busy slide. A technical evaluation panel identified discriminators between the forms using a set of technical criteria, but didn't try and pick the best form. Livermore, as head of the development team, then completed an integrated assessment and drafted a recommendation recommending ceramics. An independent peer review panel examined those reports, met with the experts, and issued a letter report confirming that both forms could do the job and that ceramic had some advantages. Since September, our office has focused on the ceramic form.

More than a year of work into producing data to support this downselect process. As noted here, advantages for ceramics were identified in proliferation resistance in that basically less known about ceramics. Perhaps, Tom when he gets to this discussion of forms may touch on some of the other aspects of that. Worker dose was a discriminator and cost-effectiveness. A large part of that had to do with the higher density for the ceramics requiring fewer canisters and one compliment of that is saving on costs for repository disposal.

Since we're here in the neighborhood of the Savannah River Site, this is a reminder of why we've identified it as our preferred site for immobilization. As I noted earlier, DWPF is the key reason. In the future, the APSF also offers potential for synergy, storage, and so on. Westinghouse Savannah River Company is a key player on the immobilization team. Process experience here will help us expedite demonstration and startup. Right now, we're planning to produce prototypic cans. Current schedule 2000 and, hopefully, a canister or two in 2001. Clearly, this is tied to our qualification process.

I'm going to allow enough time for Tom to get into form, process for producing the form, and discussion on our work in support of RW's analyses, but as a segue to Tom, this chart highlights the current project team structure. I'll resist the temptation to dive into detail here and turn it over to Tom to provide more detail. I might mention he is sort of a walking example of technology transfer in that he's at Livermore and heads the development team, but is on loan from Westinghouse Savannah River Company which is where we prefer to transfer this technology.

BULLEN: Any questions at this point or should we proceed?

CRAIG: Let me ask a couple. Maybe Tom is the one that wants to answer them, but I'd like to understand how--what is the source of radioactivity? How does it compare with waste after 1,000 years and what about retrieval since the plutonium is presumably still there as some form of plutonium that you'll tell us? Is it plutonium oxide?

DANKER: Okay. Let me read your question back to you. Your question is on the source of the radioactivity and also questions about retrieval of the plutonium form from the repository. Is that correct?

CRAIG: Yeah, and how the radioactivity compares with reactor waste after 1,000 years time frame rather than the shorter time frame that you mentioned?

DANKER: Okay. That brings to mind decay curves for--yeah. Let me try this and then hand it off. The bottom line, one driver to go to can and canister was to simplify the recipe, if you will. Isolating the external barrier simply implies you're doing the immobilization of plutonium in small cans and you're relying on the high-level waste that was discussed earlier by Neil and others within the canister to provide your radiation barrier. So, it's a pretty easy answer to the first question in that the radiation barrier comes from the high-level waste glass and we simply provide a first stage immobilization ram in a framework within the canister and then put it under the DWPF melter and do a pour. One of the differences is you have volume displacement. So, it ends up with additional canisters. But, the source of the radioactivity is, in effect, the high-level waste glass.

In terms of retrieval, I may need help on the decay curves, but, Tom, are you going to be getting to that one?

GOULD: No, but I can cover it.

DANKER: Yeah.

GOULD: When I talk about the form.

DANKER: Yeah. The spent fuel standard again is--I think, Sharon had some decay curves before for the high-level waste glass that you can use to pick off the curies per canister and so on that would give you the decay curve. But, I can tell you that kicks you back to the spent fuel standard. 100r/hr, 30 years after fabrication, is a fairly arbitrary point and is tied to a range of considerations including what is considered self-protecting in NRC and IAEA regimes and so on, but it's an art and not a science. I danced a little on that last one. So, you may want to hit Tom when he stands up here.

CRAIG: Yeah, it was a longer time frame than I was focusing on.

DANKER: Yeah, okay.

GOULD: I'll try to cover that.

BULLEN: Good. Any other questions at this point?

DI BELLA: Do you think the glass they're making right now would be radioactive enough to provide this protection that you're talking about?

DANKER: No, no.

DI BELLA: Okay.

DANKER: Sludge. What they're currently pouring is sludge.

GOULD: Yeah. No, it would need a little more cesium content in there to provide a level of protection comparable to spent nuclear fuel.

BULLEN: Right now, the cesium doesn't come in because they're just putting strontium is essentially--so, you need both cesium and strontium there to get you the high gamma field to self-protect?

GOULD: Yeah, the cesium is what's really going to protect you over the next several decades.

BULLEN: 300 years, yeah.

GOULD: Yeah, 300 years.

BULLEN: Okay. Paul, did you have any more questions or do you want to move right into--

CRAIG: No.

BULLEN: The segue got a little disrupted, but we'll let Tom step in and follow up on the waste form.

DANKER: By the way, he has with him a surrogate ceramic puck, 20 or 21 of which go into the cans. If he leaves any of these with you, don't do what I did at the airport. When it comes up, you know, and they ask you what it is, don't get complicated on the discussion. Don't say it's a surrogate ceramic for the plutonium disposition because the word they pick out of that sentence is plutonium.

GOULD: Actually, paperweight works well at the airports.

What I want to do right now is to give you sort of a brief overview of what the form is, very briefly on how we're going to make it, and talk a little bit about the development program we have in place primarily focusing on providing information to the repository analysis.

Let me start and at least summarize some of the materials that we're going to need to incorporate within the ceramic form which will then go into the can and canister larger form. We will be focusing primarily on what are called the impure plutonium materials that are coming both from the weapons program, as well as from the fast reactor testing program in this country. In the pure dual-track approach, the clean material, primarily material returned directly from weapons, would be converted to mix oxide reactor fuel and that fuel then radiated in existing power reactors. That comprises about 32 to 33 metric tons of plutonium and it's very pure plutonium with just a little bit of gallium in it.

The other materials would require significant purification processing in order to convert that material to an acceptable oxide for fuel and this is one of the advantages of the immobilization process is, in effect, we're not doing any purification. We will be converting all of these materials, if they are not now oxides, into an oxide feed for the ceramic form. They include various alloys, as well as uranium, plutonium, alloys, and oxides, some impure plutonium oxides that are predominately plutonium but contain residues from the various processing steps that we'll use in the weapons program.

It also includes fuel that was used in the zipper reactor at Idaho, Argonne West in the fast reactor program, as well as un-irradiated fuel that had been prepared for radiation in the Fast Flux Test reactor at Hanford. Coming with the plutonium in this fuel is going to be approximately 17 metric tons of predominately uranium 238.

The plutonium form itself, the ceramic form, is based on titanate minerals. The early work in developing the form was performed in Australia by Ringwood in developing the so-called sin rock form for high-level waste. These minerals primarily that we're using are going to by pyrochlore and zirconolite and also a little bit of brannerite. The chemical formula for these is basically $A^{(+2)B(+4)}Ti_2O_7$. Basically, the plutonium, the uranium, and the hafnium will substitute indirectly to the B site. This is normally occupied by zirconium. Long-lived mineral phases, pyrochlore and zirconolite, have been around for a billion years or so. They have contained thorium, as well as uranium. So, there's some long-range data on these forms. The A site is occupied primarily by calcium. Gadolinium that we want to put in there again is another neutron absorber with hafnium. It's a +3 element and it will partition between the A and the B sites.

This is the ratio of the different oxides in the primary form. There are a lot of impurity materials, cations such as iron, chromium. There will be a little bit of moly, some aluminum. There's also going to be a little bit of silica coming in some of the plutonium materials. These will be contained in the form at percentages that are less than two percent by weight. Some of these will substitute directly into the titanate base phases and in other cases, such as silica, there will be a silicate phase that forms.

We've got some preliminary data indicating that the thermodynamics at the formation of the ceramic pretty much force the plutonium and the uranium, as well as hafnium and gadolinium, into the primary titanate base phases. So, we really are doing a pretty good job at this point of associating with the fissile materials and appropriate quantity of neutron absorbers. We're putting in hafnium at about a one-to-one atomic ratio to plutonium. U-238 is going to be in there at about two-to-one to plutonium, and gadolinium, a little less than one-to-one with plutonium.

The can and canister form itself will be comprised of, as Bill indicated, 28 cans of--they're about 21 inch high cans; oh, about three inch OD cans containing 20 of these ceramic pellets. The pellets are about one inch thick and two and a half inches--2.6 inches in diameter. Each pellet contains about 50 grams of plutonium. So, there's going to be about 1kg of plutonium per can. The cans will be distributed in sort of a circular array, four layers of seven columns. The volume occupied by the cans and the support structure for the cans is estimated right now at about 12 percent of the free volume within the canister. We have gotten some preliminary results on two early pours done during the startup of the Defense Waste Processing Facility indicating that the glass fills effectively the entire volume with a slightly different configuration. We had a 20 can configuration that we poured. We are now doing analytical modeling for this configuration that indicates that we should get a complete occupancy of high-level waste glass within the canister. Later, we will be doing some pours with surrogate materials to confirm that.

In terms of the number of canisters that will be affected for the so-called 17 metric ton case which will probably involve about 18-1/2 metric tons of plutonium, for that case--and that will be all of the impure plutonium materials--we're going to occupy 635 canisters of glass and there will be 77 extra canisters that will have to be generated as a consequence of the volume displacement. In the case of all of the materials coming to immobilization, the 50 metric ton case, that would occupy a little over 1700 canisters and there would have to be produced an additional 210 canisters of high-level waste to accommodate the volume loss.

The process for making the canistered forms really can be considered in three parts or three stages. The first is feed materials characterization. Here, all of the various feed materials will be converted to an acceptable oxide feed for the ceramic formation process. This head end will be accommodated within the same facility as what we're calling first stage immobilization.

In the first stage of immobilization, we basically are going to be performing a MOX like fabrication process with some small changes, but primarily a cold press and centering operation. The centering temperature is 1350 degrees Centigrade for the ceramic form. At that temperature, you actually get the chemical reaction that takes place primarily in the solid state among all of the oxides occurring in the mixture. It requires basically a milling and granulation step, cold pressing, followed by centering, and we will be doing some individual nondestructive analysis on each of the pellets and then those will be basically stacked within the cans.

The cans will be brought out of the glove boxes in an operation that seals the cans. The cans then will be loaded in the second stage immobilization process, or at least leading to it, into basically an empty DWPF waste canister in a rack. These operations will be performed basically either in a new facility that will be joined to the actinide packaging and storage facility that is being constructed at the Savannah River Site in F-Area or the operations will be performed basically in areas of 221-F. Those are two facility options that the Department of Energy is currently evaluating.

From the F-Area, the canisters loaded with plutonium will be transported to the DWPF facility. Those canisters will be taken into the processing canyon facility just like the other canisters except for a little added security and appropriate safeguards. The glass will then be poured into the canisters and the whole process is just like described earlier.

During the past year, we've gone through, as Bill had indicated, a laboratory scale development program in developing the ceramic form, as well as the glass form, looking at what the processing conditions would be doing some rough engineering analyses of the production process leading to a decision on which of the forms we wanted to develop further for the final production facility. At this point, we've got some preliminary information basically on the characteristics of the ceramic form. We've gotten some preliminary durability testing data, and over the next two to three years, we're going to be focusing basically on fine tuning the compositional aspects of the plutonium form itself, developing the processing condition envelope for the production system, and we will be doing some prototype equipment testing both with and without plutonium depending on the critical nature of the equipment and whether it's plutonium dependent. And, we are also going to finalize on the design of the canister form.

What I want to do at this point is talk a little bit more in detail about what we specifically want to do to support RW so that they have adequate data to provide an analysis of how the form will perform in the repository over a long period of time. And, I just remembered your question, Paul, and I apologize for--let me pick it up at this point and then I'll talk about the repository aspect of our program.

One of the reasons that ceramic was chosen was that the plutonium is tied up in the titanate lattice in such a manner that normal processing that is used for plutonium materials in Russia, as well as the United States, doesn't work very well at extracting it from the crystalline lattice. As a matter of fact, you have to go to a different processing scheme entirely. So, it makes it a little more costly and more difficult to extract plutonium. Around the plutonium, of course, we're putting it in this high-level waste glass which provides a proliferation barrier at least for the period that we're going to--you know, between generating the canisters and emplacing them in the high-level waste repository.

Subsequent to emplacement, of course, the fission products are going to decay in a few hundred years. Just like with spent fuel, you're going to have a plutonium mine.

Okay? That's the result of decisions made in this country. In the case of spent fuel, the plutonium is going to be incorporated in a ceramic material with uranium. In the case of this form, it's going to be incorporated in a ceramic material that's going to be surrounded by glass that will be surrounded by a canister. The concentration of plutonium in the whole high-level waste canister is going to be, oh, a little bit over a percent; somewhere between 1 and 2 wt%. That's, more or less, comparable to what you're seeing in spent nuclear fuel. So, from the standpoint of meeting the spent nuclear fuel standard, it's roughly comparable. Trying to extract it from the high-level waste glass presents different kinds of processing problems than one would have for spent nuclear fuel. You can argue which would be more difficult. I guess, it depends on the types of facilities that you have or would have to build.

So, for the most part, I think we have satisfied the spent nuclear fuel standard. You know, the longer range question is, you know, we have created plutonium mine for the future generations, but that's the nature of our program.

One of the focus areas for our program is trying to provide necessary and sufficient data that would, number one, allow us to understand the mechanisms of degradation that this form will undergo in the repository environment, as well as providing some data for a variety of different repository conditions, so that we can compare the behavior of this form with, say, glass and other forms that have been characterized.

We have a series of corrosion tests that have been set up at Argonne National Laboratory, at Pacific Northwest National Laboratory, as well as Livermore, and Savannah River. Both tests under static conditions, such as the PCT tests that Ms. Marra talked about, MCC-1 tests. We're also doing a variety of single path flow unsaturated tests at both Argonne and PNL and Livermore. Argonne is also doing vapor hydration tests and other tests to look at accelerated leaching conditions associated with these forms. A great deal of effort, especially at Argonne, is being spent to look at the nature of the secondary phases that are formed as the material degrades and the nature of these phases so we can understand how does the fissile material and the neutron absorbers, hafnium and gadolinium, partition into not only stay with the primary phases, pyrochlore and zirconolite, but also in the degradation phases how do they partition into those phases, hopefully, showing that indeed there will be enough neutron absorbers homogeneously distributed in any of the degradation phases that we have no concerns, whatsoever, about long-term criticality.

We are also developing basic thermodynamic data on hafnium and gadolinium so that we can input some of the modeling work being done by the RW contractors. We're developing an analytical model to predict the degradation behavior long-range of the form itself. A lot of these tests that we're performing with the form are done with the high-level waste canister materials basically in the soup, so to speak, when we're testing that.

And then, finally, one of the issues that has been raised with the ceramic form has been the fact that the alpha damage that is caused over a period of approximately 1,000 years will cause the ceramic to go from basically a crystalline form to a form that is more amorphous in structure; otherwise, becoming metamict. And so, we anticipate at this point with some of the existing data that we would probably see maybe an order of magnitude increase of the leach rate associated with the metamict form, but we're providing specific tests using PU-238 doping, as well as Ned Bibler is going to be doing some work with ion implantation to look at the effects of radiation. This is going to be like a four to five year program.

So, these are the things that we're going to be doing. In subsequent briefings to the TRB, what we'd like to do is some back and be more results specific. This is just to provide you with sort of an overview of what we're trying to accomplish in the program and I think, Bill, you took the schedule, but we really have from the repository analysis standpoint some key milestones. Coming up in July of '99 is basically providing RW with input for the licensing application, and we will be updating that annually over the next several years as we learn more and more about the behavior of the form under simulated repository conditions and analyses of those conditions.

BULLEN: Thank you, Tom. Questions from the Panel?

DI BELLA: Just in the last thing that you said, "providing RW with information for the license application", I am almost positive their current program plan does not call for disposal of this material in the repository; not to say that it couldn't be, but I just don't think it's within their plan. Has their plan changed?

DANKER: Yes, it is in the process of being changed.

DI BELLA: Okay.

DANKER: Yeah, let me try. It's a timely question. As we speak--and I think Jim Brazee was alluding to it--there are changes to their technical baseline ongoing. So, it's active to get plutonium on their radar screen. I think, if I'm not mistaken, there's a meeting this Friday of their change control board to formalize the change to their technical baseline.

BULLEN: Any other questions from the Panel?

DI BELLA: Now, to a technical question. What size range do you have to mill the particles to to get the centering at a reasonable time period?

GOULD: Basically, it's reactive centering and we've found that if we get the particle size down to 10 microns that we get pretty much a full dissolution of the plutonium oxide into the relevant mineral phases. We notice that if we're above 20 microns that we tend to have some small ions of plutonium oxide that exist in the centered form.

DI BELLA: Is there experience with milling plutonium that fine within glove boxes and what it does to your dust--

GOULD: Yes. As a matter of fact, that's typical of the mix oxide fuel fabrication business. As a matter of fact, our baseline flow sheet right now and choice of equipment mirrors quite a bit what BNFL is doing in their new Sellefield plant.

BULLEN: One last question. You mentioned the 10-fold degradation due to radiation damage and I guess that's in comparison when you made the selection of the ceramic waste form. Is similar radiation damage expected for a borosilicate glass waste form?

GOULD: I think actually borosilicate glass, we wouldn't expect to lose as--

BULLEN: Right. It's amorphous to begin with.

GOULD: Yeah, it's amorphous to begin with and I think Ned can--if Ned Bibler is still here, he can speak more authoritatively on this subject, but I don't think we would anticipate that the leach rate to change as much for the glass waste form. The one we were looking at was a lanthanide borosilicate glass, a very high melting temperature glass at about 1500 degrees Centigrade, in order to get significant quantities of plutonium into the glass matrix. Our leach tests on actual samples containing plutonium showed both static and flow-through tests. If you looked at just sort of a range of the results, it showed the ceramic being a factor of 100 to 10^4 more durable than the glass.

BULLEN: So, even a 10-fold decrease in leachability--

GOULD: It still should be at least comparable with any glass form, but probably better.

BULLEN: Thank you.

Well, we come to the cleanup position again and I didn't realize that both times it was going to be Dave Haught, but Dave is going to close out the afternoon session as he did the morning session speaking about the disposal of vitrified high-level waste and immobilized weapons-grade plutonium from the DOE perspective or the Yucca Mountain Site Characterization Office perspective. Dave is still an engineer at the Yucca Mountain Site Characterization unless he got promoted since noon. He's responsible for the oversight and development of waste package design, materials testing, and modeling programs.

HAUGHT: I'd like to start--and this feels like a cardinal sin of presenting something--with an apology. There is a hard copy of this presentation. It is making its way here. It has not arrived. It was Fed-Ex'd yesterday and I had hopes of it actually being here by noon today, but that didn't happen.

I'm going to talk to you about the disposal of high-level waste and immobilized plutonium. Actually, as I get into this, I'm going to talk more about the immobilized plutonium than the high-level waste because it's a more interesting topic. When I show you some of the performance assessment curves that we have, you'll see why there is very little issue with vitrified high-level waste, but we do have to address criticality in the case of the immobilized plutonium because there is some fissile content.

The waste package design for both the vitrified high-level waste and immobilized plutonium is either a four or a five pack with the four high-level waste canisters per waste package. The containment barriers here are current design. Reference design is 10 centimeters of carbon steel and two centimeters of alloy 625. I note that we are evaluating C-22 as a replacement for the 625 and that is in review as we speak. The M&O may have actually worked that through by Christmas time, but as of today, it's still 625.

The only difference when you get to the immobilized plutonium is you would swap out some of the vitrified high-level waste canisters with one or two plutonium containing canisters per package. Now, the basis of what I am telling you today is based on the old formulations for the ceramic. The new formulation is kind of moving in a safer direction and so these numbers here may go up somewhat, but we have not done that analysis yet.

BULLEN: Just a quick question, Dave. Those one or two are limited by criticality issues within the waste package itself?

HAUGHT: That's correct.

And, you have seen the five pack. The five pack without co-disposal would just be a five pack without the canister down the center and the four pack looks like this.

Performance assessment, I'm going to show you one curve and that is the sensitivity analysis for DOE and SNF which includes allowing for vitrified high-level waste and our current basis for vitrified high-level waste is DWPF canisters. The one thing that I would like to note and it has gotten a fair amount of attention recently is that the dose history curves I'm going to show you do not consider the colloidal transport of plutonium. It is planned for TSPA-VA, though.

Our findings are that vitrified high-level waste does not significantly impact the dose at the accessible environment. In comparison to commercial spent fuel, it's about two orders of magnitude less. The sensitivity analyses that we've done on immobilized plutonium show that its performance is similar to vitrified high-level waste. So, at least, our current understanding--again based on the old formulation--is that, you know, the vitrified high-level waste and immobilized plutonium curves are going to look just about the same.

This is a curve. The red line here is the--well, as you can see, it's 8,745 metric tons of heavy metal of high-level waste and the peaks are tracking about two orders of magnitude less than commercial spent fuel and is included in the black line. Actually, I misspoke; that's the base case, not just commercial spent fuel. But, given the very small contribution, it pretty much is just commercial spent fuel.

Now, what is different about the immobilized plutonium is that we do have to look at criticality and here's some of the bases that--I've gone through the analysis process, the Phased Analysis Process before. So, I'm not going to discuss that, but I will talk about here are the basic assumptions that we've brought into the thing. That is of the plutonium, it is 93 percent plutonium-239. The plutonium is immobilized in glass. We have done the immobilized plutonium in glass for intact degraded and external configurations. We have done the old formulation of the ceramic for intact and internal degraded configurations.

Now, to try to relate what we've done to what you've heard from the folks from MD's program is I've got this comparison here of the old versus new. Some of this, you have already heard from both Tom and Bill. But, as you can see, the amount of plutonium per canister is going down, and more importantly, the amount of hafnium--well, as important--the amount of hafnium is going up. So, we have a better waste form.

Just like in the case of the aluminum clad fuels, I have an I chart for you on the scenario generation of how we arrive at the configurations. You'll see that's a Step 1. There is a Step 2 which gets into a lot more detail into some of the chemical processes that are going up in this round. But, just for simplicity sake of saying how we get to the configurations, I'm just going to show this one.

Now, here's how we kind of start as the degradation process begins. This is showing a four pack with two plutonium containing canisters in it and, as you can see, you've got some water, you've got some clays. The canisters are beginning to degrade; this particular one is breached in some of the cans. The plutonium containing cans are degrading.

As you go a bit further into it, we have a couple different configurations. The one on the left here is that the glass and the plutonium are kind of degrading at approximately the same rate. On the right side is we have what would be a fairly severe configuration where we have the--the glass is degrading out of the plutonium containing canister and leaving the cans behind. In addition, you have in this case these canisters on the bottom are remaining intact enough such that the canisters on the top are supported above any of the clays or any of the neutron absorbers that might have been leached out. So, that is kind of a most severe configuration. We believe that it's fairly unlikely because it requires a period of a low pH in order to get the separation of the plutonium from the absorber. But, if you have that situation, you probably would not have these on the bottom still intact, but we have considered it.

And, just another look, this was inside a canister, the different configurations you could have in there. You could have a case where the glass is kind of turned into kind of a clay-like mixture and you still have some intact cans and then the other case of you have kind of a soup, if you will.

Here are some findings. Again, this is on the old formulation. An internal criticality can be prevented with a mass limit of 50 kilograms of plutonium-239 per waste package. We did make a recommendation that hafnium would provide some additional defense and depth, and as you've seen, the amount of hafnium has gone up in the hockey pucks. And then, in the case of external configurations, there's a couple of ways that we can have the re-disposition of the fissile material. And, as we have run the codes, this is a configuration, a worst case configuration, that we believe can happen and that is that we have five kilograms of fissile material within a 15 cubic meter area under the footprint of the waste package. The K effective of that is below critical. In fact, I think, the highest we were able to get it is .95.

In the far field, it's well-known that zeolites are abundant in Yucca Mountain. We believe that the maximum uranium--and this is then after the plutonium is decayed into uranium--absorption is about 0.17 percent by weight in the zeolite and that's insufficient to accumulate a critical mass. Another mechanism would be a reducing environment and we just have not seen in the Yucca Mountain environment any more than trace quantities of this. So, we believe that that's a low probability of the re-precipitation of uranium by any reducing environment.

Now, we have done a consequence analysis for the external configurations. Now, in the case that I gave you of what I thought our worst case configuration was, it wasn't critical. So, the consequence is nothing. So, what we have done is we have hypothetically put together a case where the plutonium in a single waste package would go critical and that's if we could get six kilograms of plutonium-239 to condense into a one cubic meter block and the result would be we'd have 500 watts of power generated for approximately 4,000 years and we'd have about a 14 percent increase in the radioactivity of that package's plutonium.

Our current statuses, we are planning to update our analysis of the intact and internal degraded configurations using the new formulations of the ceramic during the course of FY-98. And, we will finish the analysis of the external configurations and the probabilities and consequences again in FY-99. That may sound like I've said it before and it has and we have deferred all of the external configuration analyses to later on because it is a cumulative effect, and it kind of doesn't make sense to really do it. It's not as efficient to do them on a case-by-case basis.

In summary, we believe that the impact of total system performance is small for most vitrified high-level waste and immobilized plutonium. Internal configurations of immobilized plutonium can be maintained at some critical levels. The disposal of immobilized plutonium appears workable from a technical point of view. I would like to reiterate what Bill said, although the plutonium is not currently in our baseline, it is in our--the BCP has made it through?

SPEAKER: Right.

HAUGHT: Okay. It's a recent development. That's really all I have.

BULLEN: Thank you, Dave.

Questions from the Panel?

DI BELLA: You said that five kilograms of plutonium-239 in a 15 meter volume under the waste package would be the worst case and that wouldn't be critical. Can you explain how you know that or how you think that is the worst case?

HAUGHT: Can I explain it? No, I'd like to call on Peter to come here and help me with that.

GOTTLIEB: Peter Gottlieb, M&O. The analysis was using the same geochemistry code that was mentioned for the internal criticality this morning. It was used in a different mode, but it was interpreted so that we could get the maximum distribution of deposits of plutonium and uranium in fractures immediately beneath the waste package which is the area of zone volume where they would be the most concentrated. Now, there are other possibilities for accumulating concentrations in reducing zones, organic materials, and so forth which were treated differently and which were not found to be critical either. But, in this instance that Dave is alluding to, if we look at what is predicted to precipitate or absorb in the fractures, it is much too small to have anywhere near criticality. If we look at the maximum volume we could stuff into fractures that are there in the manner of some other analysis that's been done, then 15 cubic meters in that footprint would be close to critical. It would not be .95; it would still be well under that. But, in order to do a consequence analysis, we have to have a critical mass. And, so what we did, we artificially compressed that into one cubic meter which made the K effective up to one and so we could do a consequence analysis which led to the increased radionuclide that Dave quoted. But, we have a conservatism. We have an extreme conservatism on top of another extreme conservatism in order to get to a point where we can do a consequence analysis. So, it's only for illustrative purposes. It is not to be considered in any way a criteria for accepting the waste.

HAUGHT: I'd like to make sure that I can clarify what you've said, Peter, in terms that might answer Carl's question. I believe from what Peter just said that the answer of how we believe that the five kilograms deposited in a 15 cubic meter area or volume below the waste package is that we--in running the EQ 3/6 codes in the near field area, that those were the results that we got. Is that--did I characterize what you say correctly, Peter?

GOTTLIEB: No.

HAUGHT: No. I'm glad I asked then.

GOTTLIEB: When we ran the EQ 3/6 codes--you see, what we find when we tried to--some people take the fractures and say you can stuff the fractures with plutonium and uranium and get criticality. The whole purpose of running the EQ 3/6 code is to say, all right, what else is going to happen? You don't have pure water with uranium and plutonium in it. You have lots of other stuff and that's going to be competing for space in those fractures. And, when we do that analysis, it comes out far below anything approaching criticality. So, in order to get close to criticality, we have to go a factor of 10 or so on top of that in order to get anything close to criticality. That still isn't critical over the 15 cubic meters which is sort of where it would all be coming out. So, if we compress into one cubic meter, then it is K effective equal to one and we can then conduct an analysis of the evolution or the consequences of criticality.

Now, is that your question, Dave?

DI BELLA: It was my question and this Carl. Thank you very much. I only wanted to know not about your criticality calculations, but how you know that five kilograms, not 5.1 is the most--or 10 or whatever number it's going to be, is the most plutonium-239 that is going to come to rest in a 15 cubic meter space under the waste packages. Actually, I would think it would be much less than that and you could come forth with a plausible explanation.

GOTTLIEB: Well, the five kilograms is approximately what we get out of the EQ 3/6 calculation. That's straight from calculation. That's an approximate figure. It isn't that sharp between five and 5.1.

BULLEN: Another question, Carl?

DI BELLA: Yeah. Then, I have a question about the criticality calculations you did where you said you had six kilograms of Pu 239 and one cubic meter volume would be critical and chug along at 500 watts for 4,000 years. What are the basic parameters that go into that calculation? It seems to me it's going to be dependent on seepage flocks. I'm looking for that number particularly.

GOTTLIEB: Can I do this again, huh? The inflow of dripping water into the waste package for that case would be approximately 5mm/yr. That's the flow rate and then that's over the area of the waste package. So, you can multiply that to get the cubic meters of about less than half a cubic meter.

DI BELLA: What if it were 10, as someone put on a slide today, maybe you in this morning's presentation? Or what if it were 20 or 60 even, as some of the people from the expert elicitation have offered as a possibility particularly with climate change? What do you think the consequences might be in that circumstance?

GOTTLIEB: Well, with a higher infiltration rate, it's possible that you would have a higher--with higher drip rate, it's possible you would have a higher power level approximately linearly proportional because one of the limiting factors in the evolution of criticality is the heat dissipation and the heat will be removing water. And, if you dissipate too much heat, you remove too much water and you go subcritical. So, presumably, if you could replenish the water at a faster rate, you could sustain a higher power level.

BULLEN: I just have one quick question about the stack of 20 hockey pucks with the poisons in them. I'm assuming that that's a subcritical assembly if I immersed it in water. Is that not correct?

HAUGHT: Yes.

BULLEN: Okay. Now, you mentioned radiation effects and you're looking at fundamental thermodynamics of hafnium in the materials. What fraction of the hafnium do I have to take out before I have to worry or is there a very large margin and it sits in never mind?

HAUGHT: There is also gadolinium.

BULLEN: Gadolinium and hafnium. So, we have two BPs or two burnable poisons in there?

HAUGHT: Yeah.

BULLEN: And so, what fraction of those do I have to remove? What kind of margin do I have if thermodynamics isn't necessarily my friend in the radiation damage environment?

HAUGHT: I don't know what the fraction of the gadolinium and hafnium have to stay in. I can answer the question a little differently. I believe we're considering that the hafnium is going to stay and that the gadolinium excepting a couple of low pH scenarios will migrate with the plutonium. In fact, the low pH scenario that would allow that to happen, we haven't quite convinced ourselves that it actually can. That would be the forming of chromic acid due to the corrosion of the canister and perhaps the can itself. So, Peter, do we have any real hard numbers on that or is that the best answer we can give?

GOTTLIEB: I'm not prepared to give a specific number, but it's, at least, 90 percent. We could lose, at least, 90 percent and still be subcritical.

BULLEN: Thank you.

Any other questions from the Panel?

DI BELLA: A quick one, I hope. The weapons-grade plutonium, some of it anyway, has a small percentage of gallium in it. That, I believe, causes the MOX people some problems, an extra process step or two. Does it cause any problems in this ceramic process, particularly; the gallium?

GOULD: The answer at the levels of the gallium that would be coming in with the weapon-grade plutonium, the answer is no.

BULLEN: Any other questions from the Panel?

(No response.)

BULLEN: Questions from the audience?

(No response.)

BULLEN: Thank you very much, Dave.

HAUGHT: Thanks.

BULLEN: I notice by the agenda that we have until 5:00 o'clock. Let me make a couple of comments first. There was no one who signed up for the public comment period, and I would like to make one last call prior to closing remarks for anyone who would like to make public comment. If so, please, step to the microphone at this time and be recognized?

(No response.)

BULLEN: Seeing none, we move on to the last item on the agenda which is closing comments or closing remarks by Dan Bullen. I see that we're not supposed to be out of there until 5:00. So, that means I have 45 minutes to speak. Is that not correct?

(No response.)

BULLEN: No, I'm sure my classes would argue that I could take a two minute talk and make it 45 minutes, but today I would just like to express the appreciation of the Board and specifically the Repository Panel to all the speakers and to our DOE and Savannah River Site representatives who have gone to great lengths to organize both today and tomorrow's tour.

With that, I would like to call these proceedings closed. Thank you very much.

(Whereupon, at 4:15 p.m., the meeting was adjourned.)