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UNITED STATES NUCLEAR REGULATORY COMMISSION'S

ADVISORY COMMITTEE ON NUCLEAR WASTE

July 20, 2006

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON NUCLEAR WASTE

172[ND] MEETING

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THURSDAY,

JULY 20, 2006

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ROCKVILLE, MARYLAND

The meeting convened at the Nuclear Regulatory Commission, Two White Flint North, Room T-2B3, 11545 Rockville Pike, at 8:30 a.m., Michael T. Ryan, Chair, presiding.

COMMITTEE MEMBERS PRESENT:

MICHAEL T. RYAN	Chairman
ALLEN G. CROFF	Vice-Chair
JOHN T. LARKINS	Executive Director
JAMES H. CLARKE	Member
WILLIAM J. HINZE	Member
RUTH F. WEINER	Member

ACNW CONSULTANTS PRESENT:

HOWARD LARSON LARRY TAVLAREDES

RAY WYMER

ACNW STAFF PRESENT:

ANTONIO DIAS
LATIF S. HAMDAN
MICHAEL P. LEE
DEREK WIDMAYER

NRC STAFF PRESENT:

GORDON BJORKMAN	RES
ANNA BRADFORD	NMSS
DAVID ESH	NMSS/DWMEP
JOHN FLACK	ACRS
SCOTT FLANDERS	NMSS
ED HACKETT	SFPO
RONALDO JENKINS	NMSS
ASIMIOS MALLIAKOS	RES
JOCELYN MITCHELL	RES
JOHN MONNINGER	RES
CHRISTIANNE RIDGE	NMSS/DWMEP
ALAN RUBIN	RES

VIA TELEPHONE:

CHIP ROSENBURGER
DON WILLIAMS Oak Ridge

ALSO PRESENT:

ED ABBOT ABZ
KEN CANAVAN EPRI
JAMES LAIDLER ANL
MARTY MALSCH State of Nevada
KEMAL PASAMEHMETOGLU, INL
BUZZ SAVAGE DOE

C-O-N-T-E-N-T-S

P-R-O-C-E-E-D-I-N-G-S

7:59 a.m.

CHAIRMAN RYAN: Okay, ladies and gentlemen, we have a full day, so we'll come to order, please. This is the 4^{th} day of the 172^{nd} meeting of the Advisory Committee on Nuclear Waste. During today's meeting the Committee will consider the following; US Department of Energy Briefing on Advanced Fuel Cycle Initiative; Standard Review Plan for Activities Related to the US Department of Energy Waste Determinations; the Research/N/MSS Dry Cask Storage Probabilistic Risk Assessment Study and the Electric Power Research Institute Dry Cask Storage Probability Risk Assessment, Probabilistic Risk Assessment Study.

We'll also have a brief discussion of potential ACNW Letters at the end of the day. This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act. Is Antonio here. Derek Widmayer will be the designated Federal Official for today's -- oh, I'm sorry, John Flack will be the designated Federal Official for today's initial session, sorry, John.

MR. FLACK: No problem.

CHAIRMAN RYAN: We have received no

written comments or request for time to make oral statements from members of the public regarding today's sessions. Should anyone wish to address the Committee, please make your wishes known to one of the Committee staff. It is requested that speakers use one of the microphones, identify themselves and speak with sufficient clarity and volume so they can be readily heard. It's also requested that if you have cell phones or pagers, that you kindly turn them off.

Thank you very much and without further ado, I'll turn over today's opening session to Allen Croff, Vice-Chair. Allen?

VICE-CHAIRMAN CROFF: Okay, thank you, Mike. Our first session is on the Department of Energy's Advanced Fuel Cycle Initiative. I'm very pleased, we've got a number of representatives of DOE and in the National Laboratories here to talk to us about it and I'd like to introduce Buzz Savage, who is the Program Director of the Advanced Nuclear Fuel Cycle Initiative and also the Manager of Research and Development for the Global Nuclear Energy Partnership. And I'll let Buzz introduce his speakers and any introductory remarks. I think the only caution is that we are on the record, so in answering questions, you need to speak into the microphones and I'm not sure whether the microphone in that corner works or not.

So, Buzz, I'll turn it over to you.

Okay, thank you very much, Allen for the introduction. My name is Buzz Savage and I work at the Department of Energy Office of Nuclear Energy and my job for the last three years has been the Director of the Advanced Fuel Cycle Initiative which is the program from which the Global Nuclear Energy Partnership is now coming into the forefront as our premier vision for advanced fuel cycles of the future. It is a pleasure for me to be here today.

I have two speakers who are subject matter experts in the main facets of our advanced fuel cycle research and development in the area of spent fuel separations and treatment systems, Dr. Jim Laidler from Argon National Laboratory and in the area of fuel cycles and fuel development work, Kamal Pasamehmetoglu from Idaho National Laboratory. Also in the audience is James Bresee of our office in DOE. He is a subject matter expert in advanced fuel treatment technologies as well, so among us we hope to be able to answer any questions that you may have on the Advanced Fuel Cycle Initiative in the Global Energy Nuclear Partnership.

I want to point out that the Global Nuclear

Energy Partnership was introduced to the world only a few months ago and in the State of the Union Address by the President followed by the Department's budget roll-out in February of this year. The program is still under development. There are many aspects that are still not in the public domain as we work towards issuing various expressions of interest and request for proposals for contractual activities associated with the US activities in the partnership but we will be able to answer, as best we can, all of your questions.

So without further ado, I'd like to introduce Dr. James Laidler from Argon National Laboratory who will give you an overview of the GNEP vision, Global Nuclear Energy Partnership, and the specific technology presentation on the advanced spent fuel separations activity.

DR. LAIDLER: Thank you, good morning. As Buzz said, I'll give you just a few introductory slides on the Global Nuclear Energy Partnership and then talk about the development of advanced separations technologies that we propose to employ in this initiative. I'm the Director for -- the National Technical Director for the Development of Advanced Separations Technologies and let me begin.

The key elements of the GNEP program, Global Nuclear Energy Partnership, are to, as shown here, expand the use of nuclear power in the United States and in the world and in doing so, to minimize nuclear waste by demonstrating recycle technology so that it can be employed economically, to demonstrate advanced burner reactors in the transmutation of certain radio-toxic materials that are present in spent fuel, to establish reliable fuel services for our partners in GNEP, to demonstrate small exportable reactors that can be deployed worldwide and to also demonstrate enhanced nuclear safeguards technologies. Key to the GNEP is a reliable fuel services system. The intent, really the basic intent of GNEP is to permit the expansion of nuclear energy worldwide without spreading sensitive technologies, that is uranium enrichment and spent fuel reprocessing. The system under GNEP is organized into fuel cycle nations which would operate nuclear power plants and fuel cycle facilities both uranium enrichment and spent fuel reprocessing and reactor nations which would operate reactors under a condition in which they would lease the nuclear fuel and return the used fuel to the fuel cycle nations for processing.

And the system is schematically shown here where the fuel supplier nations or the fuel cycle nations would operate with a closed nuclear fuel cycle. The user nations would receive fresh fuel from the supplier nations and then return the used fuel to those fuel cycler or fuel supplier nations for reprocessing.

There are a number of projected benefits from GNEP. First, of course, these are motherhood statements, to reduce our dependence on fossil fuels for electrical generation, to provide that electric energy without generating greenhouses gasses. To recycle used fuel to minimize nuclear waste and also to curtail the proliferation concerns associated with the accumulation of an inventory of spent nuclear fuel in the so-called reactor nations. To safely and securely allow those nations to deploy nuclear power to meet their energy needs and raise their standards of living. To assure the maximum energy recovery from used nuclear fuel and, perhaps, most importantly to this Committee, to reduce the number of required geologic repositories to one for the remainder of this century.

And I'll show you how we're going to do that. If we were to continue with the once through direct disposal fuel cycle, without recycling, you can project significant growth in the accumulated commercial spent fuel inventory and in this graph, I've plotted the spent fuel inventory in metric tons as a function of time and I've extrapolated to the end of the century for two cases. The MIT study, which was published in 2003, was based on a growth rate of about 3.2 percent annually. They carried their projections only to 2050 at which point they had projected growth in this country to 300 gigawatts electric, about three times the present generating capacity.

CHAIRMAN RYAN: Jim, I hate to interrupt you but we need to make a phone connection that we thought was going to be made already. If you'd just stand by for a second, we'd appreciate it. Sorry to interrupt.

DR. LAIDLER: Sure.

MR. WILLIAMS: Good morning, this is the ACNW meeting making a phone connection for you.

MR. WILLIAMS: Thank you.

CHAIRMAN RYAN: Would you just tell us who you are and where you are and that way everybody in the room will know whose on the phone.

MR. WILLIAMS: This is Don Willams, with Oak Ridge National Laboratory.

CHAIRMAN RYAN: All right, Don, thanks for being with us.

MR. WILLIAMS: Thank you for having me.

CHAIRMAN RYAN: Okay. Jim, please proceed. Thanks.

DR. LAIDLER: The other projection is the EIA projection of 1.8 percent annual growth and these are assumed to take place in 2015 and beyond. And this is the projected accumulation at that growth rate. I've shown in red here two lines. The first is the well-known legislative capacity of the Yucca Mountain Repository, 63,000 tons of spent fuel, 7,000 tons of defense waste, and then the dotted line is adjustable, depending on who you talk to, but this is -- one value of the technical capacity of the repository based on limited exploration, it's about 130,000 tons.

And you see that we exceed those capacities early on in the game. By 2030 or so, we exceed the technical capacity of the repository and if you project at those rates, we would accumulate several hundred thousand tons of used nuclear fuel if we continue on the direct disposal path. To analyze the benefits of the GNEP system to the repository, we made certain design assumptions to do this evaluation. We really focused on two controlling design criteria that deal with the management of decay heating in the repository. The first criterion is that the rock temperature midway between drifts which are 81 -- or 81 meters center to center, should not exceed the local boiling point of water. At that elevation it's 96c and that second one is that the temperature of the wall of the drifts should not exceed 200c.

The first criterion has to do with the prevention of the formation of a vapor barrier over the repository which prevents the trickling down of surface water into the water table. The second has to do with the stability of the rock in the repository.

Using those criteria, we arrive at the reference loading for the repository drifts in terms of tonnage of spent fuel per meter of lights of the drifts and you see that at a loading of 1.17 metric tons per meter of lights, we reach the rock temperature, the midway point limit of 96c in this case of this loading system.

In GNEP, we're following two main paths for the development of advanced separations technologies. The first is the management of the spent fuel coming from the current generation of light water reactors and future advanced light water reactors; and secondly, to close the fuel cycle for advanced burner reactors. In the near term, we have the issue of the very large amount of spent fuel that's being generated by our commercial reactors which is now at a rate of about 2,000 metric tons per year and I showed you that accumulation will exceed the repository capacity greatly and previously I mentioned also our objective is to eliminate the need for a second repository in this century.

Longer term objectives deal with the closure of the advanced burner reactor fuel cycle to assure the economic sustainability of nuclear power in this country by providing assurance of a fuel supply at reasonable cost and to support the transmutation at high efficiencies of radio-toxic materials that are present in spent fuel. We're developing both aqueous and non-aqueous treatment processes for the near-term and treatment of commercial oxide fuel we're focusing on aqueous methods because they're highly mature. The longer term objective, the advanced burner reactor fuel treatment, because that fuel is possibly going to be a sodium-bonded metallic fuel, it may be more amenable to pyro-chemical and non-aqueous treatment methods. In both these cases, we're focusing an overriding concern on the economics of the fuel cycle and the protection of special nuclear materials.

We're using a solvent extraction process for the treatment of LWR spent fuel. It's highly mature. It's industrial practice in France, UK, Russia and Japan and it's most importantly capable of achieving very high decontamination factors from the separated products, and this is important because if we were to engage in thermal recycle, of the recovered materials, we have to eliminate the high cross section fission products. We we're requiring a decontamination factor, a DF, of greater than 10,000. Now, that may not make much sense to you but let me say that in the defense production of plutonium, decontamination factors for the plutonium product have historically been on the order of 10^7 to 10^8 so it's not an unreasonable target.

For the case of fast reactor recycle, we have to reduce the rare earth fission product content and achieve a decontamination factor of the lanthanides, the rare earth fission products, in excess of about 250. The special feature of aqueous solvent extraction processing is that it gets you a great deal of flexibility in the degree of partitioning of the constituents of spent fuel. And this is something that we may need to really capitalize on the future. We have been emphasizing a group extraction of the transuranic elements to control a degree of poor fission risk reduction to the process. What we're developing is a suite of processes for alternative applications.

Just quickly showing you something about the fuel that we're dealing with from the commercial reactors. This is probably old hat to all of you but typical PWR fuel assemblies are shown here. You see the makeup of those assemblies. They're significant. It's something that requires great attention when you come to processing these materials. Their length is shown here. It's about 13 or 14 feet long. It weighs about 1400 pounds and it's got a great deal of hardware associated with it; 154 kilograms, which is important because it becomes part of a significant waste stream. In fact it's probably the largest waste stream that we have.

I wanted to show you this. This is in response to one of the events, questions that we received. This is the important radio-nuclide content of spent fuel. Most of it is uranium. There is a significant quantity of uranium-236 in this spent fuel which is what impacts the potential for re-enrichment of the uranium. So if we were to re-enrich the recovered uranium, we'd have to compensate for that value 236 which has a rather high neutron absorption cross-section.

Krypton, one of the noble fission gasses, is present in a quantity of, as shown here, about 6.6 liters per ton of spent fuel if you bottle it at 10 atmospheres pressure. Xenon is much more significant. It's an incredibly large amount of material. At 10 atmospheres, it's 172 liters per ton and that's very important in how we deal with the noble fission gasses coming out of the spent fuel.

Radon, not much of an issue. Carbon-14, you see about .3 of a gram per ton; tritium maybe about .6 of a liter per ton at standard temperature and pressure. And then you see the transuranics. Plutonium is the dominant transuranics, about 85 percent or so at a burn-up of around 50 megawatt -- 50,000 megawatt days per ton. I wanted to emphasize these too, the technetium and iodine, the long-lived fission products. Technetium is a significant constituent of spent fuel, about one and a quarter kilograms per ton and iodine is maybe 424 grams per ton of spent fuel.

All of these are important because they dictate the choice and the details of the process that we intend to deploy. Technetium and iodine are important. This is an extract from the Yucca Mountain project EIS which shows, and it's probably outdated, but it shows the mean annual dose as a function of time. The purple line here, this curve is the technetium-99 contribution. The red is neptunium-237 which means that not only do we have to deal with the long-lift fission product, iodine technetium, but we also have to deal with the transuranics that contribute to the offsite dose as well as being a significant part of the radio-toxicity of the spent fuel.

So we have to not only deal with the neptunium but with its precursor americium-241. I mentioned that we're developing a suite of processes that we call UREX+. The variants UREX+1 and +1A are intended for fast reactor recycle of transuranics. Plus 1 leaves the lanthanide fission products with the transuranics for extended storage and UREX+1A produces a pure stream of transuranics. It separates the lanthanide fission products. UREX +2 and +3 are intended for thermal reactor recycle and we have chosen to separate in that case, plutonium together with neptunium. It provides some advantages in tracking the material if we include the neptunium with the plutonium. Plus 2 delays the removal of the lanthanides, +3 does the lanthanide separation as part of the process.

And this would be the standard thermal recycle process. UREX+4 is also a process intended for thermal reactor recycle, plutonium and neptunium, that goes one step further and separates americium from curium which enables us to do transmutation of americium in a thermal reactor. It does avoid fuel fabrication problems that are associated with the presence of curium but it also introduces the issue of having to store the curium, which is no small problem.

So here's the suite of UREX+ processes. I won't dwell on this except to say that each one of them follows the same path initially. We separate uranium as a pure uranium stream. We co-extract technetium with the uranium and then separate the technetium from the uranium. That's intended for immobilization in a highly durable waste form. We then separate cesium and strontium to eliminate the short-term decay heat load on the repository and then we go into the various separations of the transuranic elements.

When GNEP was first conceptualized, a very high level decision was made that we would process LWR spent fuel using a technology that did not involve the separation of plutonium, consistent with past US policy, that we would not engage in civil nuclear fuel cycle involving separated plutonium. And that led to a process that I showed you, the UREX+1A as our reference process in GNEP.

It separates pure uranium, highly purified, for future use, separates cesium, strontium, to take care of the short-term decay heat load and separates the transuranic elements as a group and this group of transuranics is intended for recycle in fast reactors. We have a number of performance targets that have been established for UREX+1A. We intend to recover at least 99.5 percent of the uranium at very high purity, at least 4/9. We've demonstrated 6/9 in laboratory tests and then that uranium would be converted to an oxide for storage or ultimate recycle. We want to recover 99 percent of the soluble technetium and convert it to a metallic form that would be incorporated in a metallic waste form. We want to clean the cladding hulls if possible to a non-TRU condition, less than 100 nanocuries of transuranics per gram of cladding for compaction and for disposal as a low-level waste. We'll take a portion of those cladding hulls and combine them with the sludge, the undissolved solids from the nitric acid dissolution step in the UREX process, and combine those with the metallic technetium to make that metallic waste. We want to recover 99 percent of the gaseous fission products, iodine and krypton.

We will recover the krypton and xenon together, isolate them, recover them by cryogenic means and then use cryogenic distillation to separate the krypton from the xenon and then vent the xenon, because xenon are all stable isotopes. We want to recover 95 percent of the tritium and carbon-14. We intend to recover 99.9 percent of the cesium and strontium. They'll come together with barium and rubidium and place those in a mineral waste form for sub-surface decay storage.

We want to recover 99.5 percent of plutonium, 99 percent of neptunium, 99.9 percent of the americium and 99.5 percent of the curium. And then overriding it all is we will produce no high level liquid waste that requires underground tank storage. Just to remind you of the reference case for the Yucca mountain loading with direct disposal of spent fuel. If we apply those same calculations to the same fuel with 99.9 percent of the transuranics removed, in this case 97 percent of the cesium and strontium removed, then we find that the limiting criterion is the drift wall temperature and that is reached at a loading of 202 metric tons per meter. Now that compares to the 1.17 tons per meter in the direct disposal case.

So it's a very significant increase in the effective capacity of the repository. And it's shown in another way here which may be a little more illustrative and in this case the z-axis is the relative increase in capacity of the repository as a function of the fraction of cesium and strontium remaining in the waste and the transuranics remaining in the waste. So if we had 3/9 recovery of the transuranics, and 3/9 recovery of the cesium and strontium, then we'd have a 225 factor increase in repository capacity.

This is a simplified schematic of the UREX+1A process where we separate pure uranium for storage, we separate the long-life fission products, technetium and iodine, separate cesium and strontium for decay storage, the transuranics for recycle and then the residual fission products, mainly the lanthanides and the transition metals for geologic disposal along with the fuel cladding that the other sub-assembly hardware. And this in its -- all its glory is the UREX+1A process. I'll just spend a little time going through this because you've basically seen the elements of it.

The light water reactor spent fuel is chopped and then dissolved in nitric acid. The solution from the dissolver is clarified to remove any particulate material and then it goes into the first solvent extraction process which is called UREX. And this is very much like the PUREX process but it doesn't remove plutonium so we took off the P. And it does that by addition of a complexant called acetohydroxamic acid and this suppresses the extraction of plutonium. The process simply uses tributyl phosphate, the same reagent or same solvent used in PUREX but with AHA present, it does not extract plutonium.

It also does a very efficient job of extracting the technetium along with the uranium. So then we strip out the technetium and send that to an alloying step where we will combine the cladding hulls, the sludge from the dissolver and produce a metallic waste form. Now the reason for doing that is if we can convert all the technetium to metallic state and put it in a large mass of zirconium, then it will remain in the metallic state rather than the oxide state. If it's present as an oxide, as you probably know, it's very soluble in groundwater and highly mobile in the Yucca Mountain geology. But if we can retain it as a metal, it will not dissolve. It will not become mobile and that large mass of zirconium that's present with it will prevent -- its basically a highly reducing atmosphere, so it will prevent the oxidation of the technetium.

The uranium extracted in UREX goes to a product conversion step, basically a calcining step where we convert it to oxide and store it. And this is very highly purified. It can be stored without any requirement for shielding. We expect to be able to store it in standard 55-gallon drums. The raffinate , the waste stream from the UREX process, and I should say the reason we call it UREX+ is that it's this process, UREX, plus all these other things.

So the next one in the step is to remove the cesium and strontium. We place that extraction step here. It could be at any point in the process but we do it here because having removed the uranium, the highly absorbing mass of uranium and removing the highly radioactive cesium and strontium, then it becomes easier to track the presence of the fissile materials. So we take out the cesium/strontium. We convert it by a steam reforming process into an aluminosilicate and put that into decay storage. Then the raffinate from that process goes into a process called TRUEX. TRUEX is a process that is well-developed. It's been around for a long time. It's actually in commercial application at Savannah River for tank waste treatment.

The TRUEX process is very highly specific to the transuranic elements. It also extracts lanthanides, the rare earth fission products. So the waste stream from the TRUEX process is the remaining fission products except for the lanthanides and that would go into high live waste from production. The raffinate from the TRUEX process then goes to the TALSPEAK process which is one that we can use to separate lanthanides from the fission products. And the lanthanides then go back into the high level waste form production. The transuranics go to a step in which we will blend a part of the uranyl nitrate solution from the UREX process with this aqueous stream from the TALSPEAK process and then send that to the fuel conversion process where we convert the liquid stream to oxides.

If the fuel that we're going to recycle is oxide, then that's it. If the fuel is going to be metallic, then we have to reduce the oxides to metals.

Then that goes into fuel fabrication. That fuel is sent to an advanced burner reactor, a fast spectrum reactor, and it operates its own closed fuel cycle so that the spent fuel from the advanced burner reactor then is processed. The recycled lanthanides go back to fuel fabrication, that closes the fuel cycle. The cladding hulls from the AVR spent fuel processing go into high level waste as well as the residual fission products and the cesium and strontium.

Now, we've very carefully looked at the amount of waste that we'd be generating in this process. It's a very important consideration. And I've normalized this to a scale of 100 metric tons per year. You can project to whatever size commercial or industrial plant you'd like. We kind of think about 2500 tons is about right for an industrial process. But for 100 tons of spent fuel per year, we generate about 13.3 cubic meters of uranium oxide which is classifiable as a low-level waste, a Class C waste.

The hulls, plus the technetium and the sludge would be in an iron zirconium allow. That's a high level waste stream about a cubic meter per year for 100 tons. Iodine, we're presently looking at potassium iodide but that's rather soluble in water, so we're looking at other waste forms but this, if it's KI it would be a high level waste, very small volume. Xenon and krypton, we would bottle up the krypton and have a very small volume of that. Tritium would be a high level waste. We are still looking at what that volume would be. Cesium, strontium as aluminosilicate, again, a Class C waste after decay. It's a significant volume, about 35 cubic meters per year. The residual fission products could be in a borosilicate glass or a different type of crystalline waste form such as a crystalline silicotitanate. That's a high level waste. If it's glass, it's around six cubic meters per year.

Carbon-14 we'd capture as a sodium carbonate also as a high level waste. Now if you add the high level waste volumes in this table, it comes out to around 10 or 12 cubic meters per year. For the same amount of light water reaction spent fuel in the direct disposal case, the unpackaged volume of that 100 tons is about 120 cubic meters. So we have about a factor of 10 reduction in waste volume. So we have both the benefits of reduced heat load repository and reduced waste volume. Now that's maybe a secondary effect, but it's going to result in fewer high expensive -- highly expensive waste containers.

Another way of looking at the UREX+1A

process is to consider the attractiveness levels of the various streams coming out of the process. And the main thing I wanted to show you here is that we're operating with very dilute streams, very dilute concentrations of transuranic elements in these process streams. If you're familiar with the DOE order on graded safeguards, these have attractiveness levels of either D or E and you see that it's D at this point, it's level D at this point, D at this point. It becomes a level C only when you've done the final product conversion of the oxide.

Now, that has to do -- and here's the table from that DOE order. At attractiveness level D, basically this says that we would not have to operate in a Category 1 security facility. Now, we will probably do that anyway, make it a Category 1, but the point I wanted to make is that the streams that are present in this process are really not a proliferation issue until you get to the final step where you convert it to the fuel form. Now, the status of the development of this process, we've demonstrated UREX+1A process at laboratory scale in 2005 and this year. We'll continue optimizing the process probably through 2009. We're planning a pilot scale demonstration of the process in the 2011, 2013 period at a scale of around 30 to 100 metric tons of LWR spent fuel per year at a location still to be determined.

We expect an industrial scale spent fuel recycling plant using that process to come on line and maybe 2025 to 2030 time period at a very large scale, 2500 metric tons per year, to match the expected output from our commercial fleet. It also helps to go to that very large size as far as the economies of operation because if you can capitalize on economy of scale with an aqueous process, you've gained significant reduction of cost.

Now on the fast reactor closed fuel cycle, we can either use the UREX+1A process if it's oxide fuel. If it's metal fuel, in the fast reactor system, then we use a pyrochemical process and that's illustrated schematically here. It's a process that involves molten salt electro-refining. In this case, we replace the chopped fuel pin segments into an electrolyte salt, apply a potential and deposit pure uranium on a cathode. Within -- of course, deposit salt along with that uranium deposit. We remove the salt by a process of distillation and cast uranium into an ink, that becomes our uranium product.

The cladding hulls, the noble metal fission products are left behind in the anode basket in that electro-refining process and that goes to metal waste form reduction. The remaining salt from the electro-refiner contains some of the uranium, all of the transuranics, and all the fission products except the noble metals. And that goes into an electrolysis step where we then recover the uranium transuranics together and that becomes a mixed uranium transuranic product with about 25 percent uranium and maybe five to seven percent lanthanides.

The salt that is remaining in this system is then sent to a polishing step where we remove the residual transuranic, send the salt to a cesium strontium extraction step and then that leads them to the formation of a ceramic waste form where we incorporate the other fission products. We've demonstrated a portion of the pyro processing flow sheet in the course of EBR-II spent fuel processing. We're not conditioning around 150 kilograms of spent EBR-II fuel per year. It's highly enriched uranium. The driver fuel is discharged at about 57 percent U-235. It's recovered and then down-blended to LEU.

The trues in this process are not recovered but are sent to waste. The GNEP program would complete the process by recovering the transuranics and recycling them and we envision that plants used in the ABR fuel cycle closure will be rather small, low throughput plants, co-located with a cluster of reactors, perhaps on the order of a gigawatt in the reactor part which means that the plant throughput can be something on the order of less than five tons per year at which point this process is very economical.

The final slide; we're looking at a number of advanced technologies for longer term applications including uranium crystallization, the user of super=-critical CO2, carbonate dissolution for the uranium step, decladding by means of voloxidation. We're even considering the recycle of zirconium. We believe that we can recover zirconium at sufficiently high purity that it can be sent to zirconium cladding fabrications for recycle. They've looked into it and at least one of them, Wachang (phonetic) has said that they'd be delighted to accept it if it's free.

We'd also like to have a single step extraction process for the transuranics to replace the combination of TRUEX and TALSPEAK. And these are, as I said, longer term application, probably for application in a second generation recycling plant. That completes my presentation. Thank you very much.

VICE-CHAIRMAN CROFF: Thank you, Jim. I

think we'd like to take a few questions right now. We're a little bit tight on time at this point, so I'm going to ask each person asking questions to limit themselves to one question at this point. If we have time at the end, we'll throw it open, but we'll see how the second talk goes, but, Professor Hinze.

MEMBER HINZE: A quick question, if I might; the hardware, is anything being done to look at the hardware to minimize the hardware as part of the waste stream?

DR. LAIDLER: It's something we're going to have to live with. If we can achieve the kind of decontamination that we hope, then it need not become a high level waste stream. The nice thing about the hardware is that it's not heat generated. So it really doesn't impact on the repository. It takes up some volume, of course, but you can compact it pretty well, even if it has to go into the repository.

MEMBER HINZE: Thank you.

CHAIRMAN RYAN: If we could just pull out that slide that was a table for a UREX+1A process projected waste generation.

DR. LAIDLER: Sure.

CHAIRMAN RYAN: There it is. Uranium, of course, on its own is Class A waste according to 61, so I guess what's making it Class C?

DR. LAIDLER: I guess I'm being a little conservative. It's pure enough that it would meet Class A. If we can achieve that level of purification in a large plant then it would be. Right now, we've only done it at lab scale. We down to -- we're up to 6/9th percent purity, which means just a few atoms of other materials in there.

CHAIRMAN RYAN: Well, I mean, to me that's an important difference and I guess the message I take away is all the decontamination factors really are going to drive what's in what category for waste.

DR. LAIDLER: Sure.

CHAIRMAN RYAN: That's interesting.

DR. LAIDLER: Now, you know, we're dealing, of course with a departure from current law. The Nuclear Waste Policy Act categorizes all this as high level waste.

CHAIRMAN RYAN: Right, and I think you just used in a radio-nuclide content which you know that has some merit as a risk-informed approach.

DR. LAIDLER: Sure.

CHAIRMAN RYAN: The other second part to the question is, you know, the European system, IAA and others there's an intermediate waste category. Do you see the current waste -- set of waste categories in the US as being -- as needing significant revision to address this new system?

DR. LAIDLER: I'd love to see that. That would give us an easy way to get rid of the hardware.

CHAIRMAN RYAN: One of the things that the Committee has commented on and thought about in other context is most of our definitions are origin based, where the waste came from or who generated it rather than what the radio-nuclide content is. And we've commented that, you know, to be risk informed, you'd take the approach of looking at the radio-nuclide content and perhaps not so much on what process generated it or where it came from. What do you think of that idea?

DR. LAIDLER: I'd love to see us evolve into that.

CHAIRMAN RYAN: Okay, thanks. I'm sure there will be other questions and again, let me apologize to our speakers. I do have a meeting at 10:00 o'clock with the Commission, so if you see me leave, it's not due to lack of interest, but I just have to make another meeting. Thanks.

MEMBER WEINER: Thanks very much for your presentation. It's fascinating. Has the reduction --

I can't read my question. Has the reduction and precipitation of technetium that you showed been tested in something other than laboratory scale? Can you do this on a large scale? Does it work?

DR. LAIDLER: We've not been able to do it on large scale. It's strictly at the laboratory scale. Now, our definition of laboratory scale is a kilogram of spent fuel.

MEMBER WEINER: Uh-huh.

DR. LAIDLER: And we're limited in that respect by two things, our budget and or facilities.

MEMBER WEINER: Do you anticipate any problems in scaling up that process?

DR. LAIDLER: We don't think so. We've done enough tests with recover of these materials. The only uncertainty is in the case of the dissolver sludge. We know that about 40 percent of the technetium will be in the sludge and we fully expect it to be metallic in that material. The key is to prevent it from oxidizing during the course of processing.

MEMBER WEINER: Thank you.

MEMBER CLARKE: Thanks, Jim. Just a quick question; you've given us a real nice analysis of the -- how the radio-nuclides follow through the process in waste streams that are generated, linking waste streams to different processes. I wonder, is there an ongoing effort to determine what the facility would look like at the end of its lifetime to identify decommissioning issues and seeing how they might be minimized as well?

DR. LAIDLER: We're presently in the midst of the conceptual design of the pilot scale facility that I mentioned which would operate at 100 tons per year. We are paying a lot of attention the how to decommission the facility. The present study that we're doing is looking at existing facilities because we're trying to do it on a fairly short time schedule. It's nice to be able to utilize existing concrete. So we have one facility existing that's contaminated already, one that is not, actually two that are not, and we're also looking at a Greenfield site for that pilot plant.

If we're in the contaminated facility, we're stuck with what's in there, but we're trying to conceptualize the facility equipment, the process equipment, so that it does make it easy to remove and decontaminate.

MEMBER CLARKE: It seems like a good time to be thinking about those things.

DR. LAIDLER: Absolutely.

MEMBER CLARKE: Thank you.

VICE-CHAIRMAN CROFF: I'll go next. A couple of slides before this one, you had -- you talked about process performance targets for your various recoveries. Where did you -- how did you come up with these, I guess, is the most straightforward way to ask it and is there a need for more regulatory guidance concerning the needed requirements or the process performance targets?

DR. LAIDLER: Absolutely. These are numbers that we've been wrestling with for about five years now. We even formed an OECD NEA working group to address performance criteria for advanced separations technologies. And every time I introduce a set of numbers to that group or even within our own program, I get the reaction, "Well, you're just being subjective". And I'm not entirely subjective. I'm looking at reductions in heat load and in radiotoxicity and in waste volume. And so that's where these numbers -- how these numbers are based but it would be nice to have some regulation which would give it some sort of an imprimatur .

VICE-CHAIRMAN CROFF: Okay, and now by way of a little explanation, the ACNW has initiated the development of a White Paper on fuel recycle to help us get smart is what this is for, and provide a basis for future recommendations to the Commission and it will address somewhat the history of recycle and to some extent the advance processes. And this is a good start, the talks today in providing information for that.

To prepare that paper, we've got three consultants on board and I'm going to give that a shot at the questioning here. The first is Ray Wymer.

MR. WYMER: Hi, Jim.

DR. LAIDLER: Hi.

MR. WYMER: I just have a small question. Tell me how you'll handle the tritium.

DR. LAIDLER: I wish I knew.

MR. WYMER: Okay, that's a good answer.

DR. LAIDLER: We are planning in the chopping step and in the dissolution step to carry out those operations in an enclosed cell where we would use an inert cover gas and then sweep that cover gas through scrubbers. And the intention is to pass that through a caustic scrubber and in that case get the CO2 in the form of a carbonate and hopefully the tritium in a titrated water, basically. The issue then is how we concentrate that stream and we're presently trying to design that.

Nobody, to my knowledge in the commercial world is worrying about it, but we're going to try. It's probably -- we're probably three years away on coming up with a process.

MR. WYMER: so that's a development activity.

DR. LAIDLER: Absolutely, yeah.

MR. WYMER: Thank you.

VICE-CHAIRMAN CROFF: Larry Tavlaredes?

MR. TAVLAREDES: Hi, I'm Larry Tavlaredes, Syracuse University. Thanks for you presentation, it's very illuminating. And I have one question, I'll ask this one first. You touched upon it and that is the DF's that you need to get the separations you are looking for to get in for high cross section fission products. You mentioned the DF of around 10,000 required. What do we know today about this and are there extractants that can achieve this that we know of? Are these developmental things?

DR. LAIDLER: Well, the DF of 10,000 for thermo-recycle is really a piece of cake. That's not a problem. In fact, we probably another couple orders of magnitude higher than that. That particular criterion is a number that was developed in concert with CEA and EDF, Electricity de France because they are doing thermo-recycle of MOX and that is their specification for thermal recycle pollute.

We think it's a pretty easy criteria to meet. The 250 for the fast reactor fuel is really a speculative number because we have very limited evidence that there is a fuel cladding interaction, pinnacle interaction between the lanthanides and the stainless steel cladding which could -- it's basically a liquid metal embrittlement process which could limit fuel lifetime. It's very limited basis for that criterion and there are those who think that we could get by with a lower DF but I'm trying to be very conservative at 250.

It's easy enough for us to do, certainly with the aqueous process. It's more of a challenge with pyro. The thing is that we need data, we need fuel performance data from fast reactor radiations of this fuel and we don't have any. We don't have a fast reactor.

MR. TAVLAREDES: Do we think we can get this data down the road in time for what we need? Are we planning to do this?

DR. LAIDLER: That's the next speaker's problem.

VICE-CHAIRMAN CROFF: Howard Larson?

MR. LARSON: Howard Larson, a consultant to the ACNW. In my private life, a long time ago in another world I was involved in commercial reprocessing and I understand why the pilot plant would be essentially a DOE activity because of the timing and other things but you're talking 10 or 15 years later for a 2500 metric ton a year plant. Is there any plans for industry participation in this program or development or building it or what? Or is it entirely a DOE effort all the way through as part of this program?

DR. LAIDLER: Well, I can give you my own opinion but maybe I should ask Buzz to give the official position.

DR. SAVAGE: The official DOE position is that we desire and intend to engage industry very actively from the beginning of the program, which is right now and we are working on our plans for doing so.

MR. LARSON: They do have them?

DR. SAVAGE: Yes.

VICE-CHAIRMAN CROFF: I think with that, we'd better get onto our second speaker. Buzz?

DR. SAVAGE: I'd like to introduce Kemal

Pasamehmetoglu from Idaho National Laboratory. He's our National Technical Director for Fuels Development for Advanced Fuel Cycles and his presentation will give you the perspective on the fuel development program which is a part of our advanced fuel cycle development.

DR. PASAMEHMETOGLU: Thank you both and thank you for the invitation. I am Kemal Pasamehmetoglu from Idaho National Laboratory. As Buzz indicated, I'm the National Technical Director for Fuel Development Activities for the Advanced Fuel Cycle Program originally, now merging into GNEP. So in my talk -- is this clear for you? Okay. I will talk about the fuel development activities basically taking over from where Jim leaves the transuranics and converging them into fuels and sending them to the reactors and then receiving those back, after Jim gets done with them, again, taking the transuranics and recycling back to the reactors. That's the part of the job that I'm doing.

As part of that development, there is also which was -- there is also a facility that we are planning on looking at a similar concept as Jim indicated, a pilot-scale fabrication facility for fuels supported by separations and other technology activities, so I'm going to give you a brief summary of where we are with respect to that and at the end of my talk, I have a number of view graphs looking at advanced safeguards concepts but that is really -- I put those at the end of my presentation. It's up to the Advisory Committee whether you are really interested in going through that or -- it is part of the package.

Let's start with the fuel development. Now, what is so different about the fuels that we are talking about under the fuel cycles programs as opposed to commercial fuels. As you know, all the commercial fuel today in the United States is really uranium oxide fuel and of course, in other parts of the world, it is also plutonium uranium oxide most fuel that's being commercially used. And it took awhile to develop that technology. Now we are talking about basically additional elements in our fuel. So it's no longer just uranium and plutonium but we are talking about adding neptunium, americium and curium to our fuel.

So we are dealing with multiple elements which complicates the problem from the get-go. And these transuranics, they do have varying thermodynamic properties. One of the important properties that is really challenging us is the vapor pressure of americium. Because it has a high vapor pressure at temperatures around 1400, 1500 degrees C, it is challenging some of the standard fabrication processes. Therefore, we need to develop processes that are lower temperature processes. We are no longer dealing with a very pure stream coming in, a pure stream of uranium. Now we have to deal with the impurities that get carried over from the separations process and as Jim mentioned, in many cases, I believe the purity that comes into the fuel is going to be more than adequate but depending upon the separation process that we use, we still have to obtain some data on the lanthanide carryover and how that effects the fuel's performance.

For thermo-recycle, the lanthanide carryover is really a big detriment, but if we're go to fast recycle, it is the criteria it is relaxed a little bit. On the other hand, we still need additional data to look at fuel clad interactions issue. Typically, when we are talking about closed fuel cycles, the economics and the fact that we don't want to lose too much material to the second -- to the waste streams, we want to achieve as high burn-outs as possible at 50 gigawatt days, the type of burnouts that are standard today are -- we don't believe are going to be economically feasible to go to a closed fuel cycle. We are talking about hundred gigawatt day per ton or higher type of burnouts.

The fuels that we are dealing with, especially those that contain the americium, they have a much higher helium generation compared to standard fuels, so we have to design our fuel to accommodate the high helium generation part of it, designing the fuel pellets to make sure that the helium gets out of the pellet, doesn't get retained in the pellet and part of it is designing the fuel pin so that the planning is adequate to accommodate that released helium.

And it's not really -- it's not merely the fission process. It is the capture and the decay process on americium that causes the additional helium generation. And finally, but probably one of the most important issues of that, when we introduce these elements, especially americium and curium and perhaps after one recycle, just americium along, the fabrication -- all the activities associated with the fuel fabrication and assembly needs to be done remotely. We can no longer relay on hands-on activities and fabrication itself -- by itself is not the issue, as you all know. All the fuel fabrication plants are automated, so everything gets done automatic, in an automated fashion anyway. It's just the maintenance and the quality control associated with that, that causes the problem.

And also just the nature of the problem, we are not dealing with a really specific fuel composition. We are dealing with a range of compositions that we need to be able to accommodate the fuel to. Obviously, our source material from the LWRs is variable. That depends on the burnoff that the initial fuel receives in the LWR in terms of the isotopic compositions, but it also depends on how long it's been cooled before it was separated and sent to the fuel refabrication plant. And as we transmit materials is fast reactors, in each step, there will be slight changes in the isotopic compositions and then again, every time they separate, there is -- and especially if we go from one separation process to another separation process during the recycling, say from aqueous for the first part and then the pyro for the second part, you have to deal with the impurities that are associated with those. So those are the things that really make the fuel issue a critical issue for this to be successful. I'm not going to dwell on this too much, but basically, this is where the current technology is in the US, that this technology we can say it is mature. It's been used in other parts of the world but as we start adding other materials to it, in terms of fabrication, we still have quite a bit of demonstration to do.

Now this is in a long view graph, it is an eye chart, I apologize for that but in one view graph I tried to show you the different steps of the fuel fabrication as well as the -- it's not the steps that are really important. I think, everybody does the fuel development and the fuel qualifications the same way. It is the facilities that we need and how many of them do we really currently have and how many of them we are going to have to rely on either foreign sources or start building them ourselves. Now, early on the concept development -- that is where we are with this transuranic fields, really. That's the step we are doing right now. We are doing a lot of small scale fabrication, doing a lot of out-of-pile characterization of those samples and some irradiations in facilities where we can get some irradiation time. Most of the time, even though these are fast reactors fields we are doing these in thermal reactors because that's what we have in our country.

We have advanced test reactors that's easily accessible to us and we are trying to do some fast reactor irradiation on collaboration with the French.

But as we go -- and these are facilities that we have and we are using. However, as we go to pin scale fabrication, with these kind of materials now we are really quickly talking about remote fabrication. When we are talking about those quantities of materials, we can no longer do those hands-on; therefore, we need to establish our hot cell capacities as quickly as possible to be able to fabricate those fuels and then we also need to go to more and more prototypic irradiation conditions. That means fast reactors, and eventually we will have to, as part of this phase, before we can define the process design, we really need to do a transient test as well to establish the power limits of our fuels.

So we have a facility in this country that's being shut down for awhile now and we are planning on restarting that or at least we are making proposals to DOE that we should restart that so that we can do the transient testing on those fuels. And now one step beyond that, now we are talking about assembly levels basically. We are talking about the engineering issues, the real engineering issues, associated with those fuels to lead onto the lead test, assembly testing. At that time, we need an engineering scale facility and that is the advanced fuel cycle facility that I'll talk about. And then we have assemblies of these fuels that we can test which is basically one step before we can say we have a qualified process for the fuel. At that time, we'll probably need a test reactor of our own as well.

Now, when we are talking about the test reactors anywhere in the world and obviously, the United States as well, we are -- and if we are talking about a test reactor that's aimed at qualifying the fuel, recycle fuel or the transuranic fuel, we are talking about two different types of fuels. First we need to be able to restock the reactor with a known fuel type which we refer to as the driver fuel and in our case that will probably be either a metal or an oxide driver, uranium plutonium driver, oxide driver. And then we should be able to introduce our transuranic fuels into that reactor in varying quantities with time, probably starting with pin level irradiations early on and working our way up to assembly irradiations, doing the lead test assemblies and qualifying the process and eventually being able to convert a fraction of the core to transuranic fuel and demonstrate that the reactor can run with transuranic fuels alone. So the fuel at that -- at this point, the fuel development program really divides into two.

There is an effort and granted that is not a development, that's just a fabrication and finding the fuel type of more of an engineering effort, to find the driver fuel and then to develop the transuranic fuel in parallel to that. Now, for our initial assessment, we've been doing the fuel development before GNEP, and it actually started under AFCI, all the way back to ATWP Program, Accelerated Transmutation of Waste Program and we've been looking at a number of different fuel forms and trying to find what is the best fuel form for transmutation and with GNEP coming along, we sat down and evaluated what we've learned, what we know so far. We've reviewed the data that's out there, not only in the United States as well as in other countries, who are looking at the transmutation technologies and basically our conclusion was that in an accelerated program the metal fuel and oxide fuel are the ones that are closest to implementation.

So we are going to proceed with development of the metal fuel. There are still some things that we need to solve even though we are fairly confident that the base data that we have so far shows these both fuel forms are feasible. We need to be able to demonstrate -- we have done fabrication at laboratory scales with very small loads of americium; however, those kind of techniques that we've been using in laboratory scale are not quite amenable for large scale production, so we have to be able to extrapolate that and we have a conceptual design for a production scale fabrication method and be able to demonstrate that and also the fuel clad interactions, especially for fuels that are containing large quantities of lanthanides from the get-go and we are talking on the order of four or five percent type of lanthanides in there and see what the fuel clad interactions was in there.

Now, there are some backup options, of course. If the americium, if we cannot do a fabrication directly with no loss of americium, then there are -- we also have backup designs where we try to recover the americium that are lost during the fabrication and introduce that as a target into the reactor to recover the americium. And then we are looking at the development of advanced clad materials especially if the lanthanides become an issue and we are looking at cladding of possibly liners, to he able to deal with larger amounts of lanthanides.

On the oxide side, when -- early on about five, six years ago when the partitioning and transportation program started in the United States, we have met with our international colleagues and at that time we had made a decision that US will focus on metal and nitrite fuels and Europeans and the Japanese, they were already doing a lot of work on the oxide fuels. So we were basically minimizing our investment on the oxide fuels with the full knowledge that we will be sharing our data as we go along and that's indeed, what we did and it turned out that the oxide fuels, the work that was done in Japan and France, so far showed that those fuel forms are, indeed, feasible as well for transmutation.

In other words, you can put the transuranics in a stable form, in an oxide pellet, and they do survive in a certain amount of irradiation and they behave fairly nicely without any gross failures after a certain amount of irradiation even in fast reactors. However, the issue really is that the process that we are using to fabricate that fuel is still a derivative of the MOX process. It is basically the same as the MOX process. It's a powder processing, pressing the powders, centering the powders and that is not a process that is very friendly to remote fabrication, not the remote fabrication, per se, but it is to remote maintenance of that facility.

So it does work and it will -- it is feasible to do it. The concern is, really, the economics associated with that. So in parallel to that is a backup option. We are also looking at the vibor-pac and the sphere-pac oxide fuels which simplify the fabrication quite a bit but again, it's a risk trade-off at that time, is those type of fuels do not have the same amount of data in terms of performance so we need to build that data base up fairly quickly to go down that direction. And the longer term technologies are the things that we have started looking. We are nowhere near basically being able to say, yeah, these fuels are indeed feasible, they can be deployed. Those are nitrite fuels and the dispersion fuels for second and third generation fuel forms. And the nitrite fuels have an advantage of the capability of high transuranic loading for transmutation purposes. They are nice for reprocessing purposes.

However, there is also the nitrogen-15 issue that we need to solve if we go with the nitrite fuel in the long -- that's the second generation. Dispersion fuels are good candidates for -- if we really want to go to really high burnoffs in the long run, those will be good fuel forms. But our research -- by the time GNEP came along, our research on dispersion fuels was in the really early stages so it is not a candidate for the first generation, perhaps not the second generation, but for the long run, they do offer some potentials.

Now, let me quickly summarize on what we have done so far with respect to the metal fuels in this country. As I have indicated, we have fabricated a number of metal fuel samples at the laboratory scale using a technique called arc casting, where we basically heat the materials really quickly and cast them really quickly so that there is not time for americium to be lost. And it worked really well, but this is basically one small batch at a time type of deal and there's no way we can do that on a really large scale. So we are looking at basically and extrapolation of that design which we call the induction casting where we would be flowing the materials but the materials will not be flowing in a molten state. They will be flowing as solid materials and powders and then they will be molten and casted very quickly into slugs so that there in no time for americium to vaporize. We have not done that process yet. However, we did -- as I've indicated, we did fabricate a number of samples. We have irradiated them in the advance test reactor in the United States. Those are thermo-irradiations. The French and the Japanese have done some irradiation of metal fuels in similar compositions in their fast reactors and we are sending basically two rod loads (phonetic) worth of fuels. Within two weeks it's going to be going to France to be irradiated in Phenix in the last two cycles of the Phenix, Phenix reactor.

And those have basically uranium, plutonium, americium, neptunium, just because we are limited so far on dealing with these fabrication with all these fuels. We have not dealt with curium at all. We don't -- we have not fabricated any curium bearing fuels. However, there are -- even though we believe the -- at least we have demonstrated the feasibility but there are some issues that needs to get resolved and I already talked about those in the previous view graph. And this picture here, this is the arc casting. This is how the metal fuel looks like, it's slugs after it's cast and then it's loaded into rods or pins and the metal field is always sodium bonded so it is sodium bonded.

This is the result of our very initial irradiation that we did in ATR around about eight percent burn-up levels. These are the PIE results after the fuel came out of the reactor, right around six to eight percent. That is really the swelling threshold for this fuel and we were able to achieve the swelling threshold. Some of the fuels did not swell and that has to do with the fission density as opposed to just a percent burn-up and some of the fuel was fully swollen that came out. But what we've seen in this fuel that contained americium and neptunium, the behavior was very similar to the uranium plutonium fuel that we had extensively tested in the past.

So that's why we feel fairly confident that this fuel form may be feasible for transuranic recycling. We have spent quite a bit of effort on nitrite fuels as well in this country, as I've indicated and I also wanted to summarize that for you. We were able to produce pellets under very carefully controlled conditions. We were able to produce pellets, irradiate the pellets in the advanced test reactor. We are also shipping a couple of rods for the irradiation campaign in Phenix. It's going along with the metal fuels but what we have observed with the nitrite fuel is that it is a very sensitive fuel form and especially with the addition of americium to the fuel, the centering temperatures, it's very difficult to control the centering temperatures. When we go to too low of a centering temperature, we cannot get the mechanical integrity in the fuel.

When we go to very high centering temperatures, then we start putting too much americium in the fuel. Americium nitrite is -- the vapor pressure is almost the same as americium metal, versus americium oxide vapor pressure is quite a bit lower than the americium nitrite. So with nitrite, we still have a long way to go in order to be able to do a large scale production with consistent results and we have also seen that there is an extreme sensitivity to pellets to oxygen, even small amounts of oxygen, whether it's in the -- it's introduced during shipping or whether it's introduced during characterization, small amounts of oxygen results in loss of mechanical integrity very quickly. And this is an example of that. This pellet was one of these.

It was a perfectly nice pellet. We put into a -- we were trying to measure the thermo-conductivity of that pellet and our thermo-conductivity was flowing around 100 ppm of oxygen in there. And after being exposed to 100 ppm of oxygen, that's what happened to the pellet.

Okay, this is a summary of the irradiation schedule, what you see in here. Everything in here is already done. I've shown you a few results of the PIE already, so those fuels are irradiated. The PIE is done. We are confident with the results of that and those are basically nitrites and metals and based on those results, we said metal is our primary candidate, nitrite is a backup option for longer term. We have a number of irradiations ongoing in the advanced test reactors, also these are for basically higher burn-ups. We are trying to achieve 20 percent or higher burn-up in these fuels in the advanced test reactor.

Starting next year we are going to have this campaign which we have been looking forward to, we have been getting ready for about three years in Phenix reactor. That's going to be really -- for our own fuels it's going to be the first time we're going expose them to prospect (phonetic) from irradiation. So it will go on for about two years and after that they are shutting down the reactor so it's really our last opportunity to do anything in Phenix in France.

And these campaigns that will start also next year are going to start dealing with the issue of lanthanide and start putting some lanthanides into the fuels and going to different amounts of lanthanides under different levels of burn-up trying to come up with a quantitative measure that we can pass onto Jim in terms of what the lanthanide clean-up factor needs to be on the fuel and then we are negotiating with the Japanese to be able to get into Joyo in late 2009, early 2010 and start doing some irradiation testing in Joyo as well for these kind of fuels.

Now, as part of that, at least as part of the long-term program, and if you have read the GNEP, overall GNEP objectives, one element of GNEP is a larger emphasis on modeling and simulation and being able to do more predictive work in the long run with respect to not only the fuels but the separations, the whole recycling technology. Now, as you know, even for the simple type of fuels that we have today, our predictive capabilities are really, really limited. It is a very difficult problem that we are dealing with. Everything is changing on us with time. There is really no steady state to speak of. Everything is a transient problem and everything is really an initial condition dependent problem depending upon how you fabricate the fuel. Two -- the exact same fuel is fabricated in two different places, typically behaving two different ways.

So but these are the type of predictions that we need to do in order to at least get a handle of -- from a pure fundamental understanding standpoint, to get a handle of what these fuels are really doing. And to us, that is important because as I have indicated in my early -- in my very first view graph, we are dealing with a variable range of compositions even though that's not a very wide range, but we are talking about perhaps the neptunium going from three percent to five percent and curium going from 500 ppm to up to 2,000 ppm. Even though it's a narrow range, it will be almost impossible to be able to hold qualification experiments for the whole series of compositions. Therefore, we need a tool that at least within a narrow range can guide us and do one set of experiments and then be able to extrapolate those experiments to at -- at least to different compositions.

So as part of that, we do have an effort where we are looking at an integrated fuel modeling. It is a multi-scale modeling, basically on the length scale going all the way from the nanometer scale to meter scale which is really where we see the engineering problems occur, but these are mostly the electronic structures, the molecular dynamics and in the time scale all the way from picosecond to seconds and hours to years to fuel performance. And this is one of our grand challenges, that we also communicated to our office of science partners in DOE to help us out with. There are two problems.

One problem is, do we really understand things at this level? Do we have a good understanding of it to be able to model it? And in many cases it turns out that yeah, we do have quite a bit of understanding to be able to model it. But computationally doing this kind of a computation over a decade's worth of scale, is also a challenge. And they are -- I believe they are really excited to help us with this problem and we are working with them closely on that. So that's part of our fuel development effort as well.

Now, I'm going to talk a little bit about the advanced fuel cycle facility, what it is. Basically, as I've indicated right now, we are trying to use our existing facilities, our plutonium facilities. There are not too many places in the United States where we can deal with transuranics, so we are using almost -- we are taking advantage of everything we can get our hands on to be able to do that work. And we are going to start converting some of the hot cells to help us out with that work, but eventually, those are really small facilities and we can deal with gram quantities of materials at the most, maybe tens of grams of transuranics. The advanced fuel cycle facility is to basically take everything that we've done in here, be able to bring that closer to an engineering reality and it does have -- it is targeting four technologies; advanced fuel fabrication, remote fuel fabrication for these different types of fuels, advanced processing, and primarily the processing of the fast reactor fuel as it gets recycled through the fast reactors, advanced safeguards concepts and advanced weight form associated with all these recycling operations, not only separations but the fuel fabrication.

And then it's supposed to be done at an engineering skill so that the data that we get out of it in terms of post-safety non-proliferation and environment can give us the input we need to make a decision whether we really -- those are technologies you want to commercialize or do we need to work on them more until we optimize some of this before we go commercial. It needs to be large enough. We don't want it to be too large. It's not a production facility. It's still a technology development facility but at the same time, it needs to be large enough so that the data that comes out of it is reliable for decisions on commercialization.

And as it's positioned currently, and this is at the very, very early stage of conceptual design, actually. It's pre-conceptual design I would say. It's envisioned that the size of it is going to be on the order of maybe 10 LTAs per lead test assemblies per year for fuel fabrication. These are fast reactor test assemblies, about one ton or per year of heavy metals, plus reactor fuels, the processing module and then it will be complimented by an R&D module where we will be doing small scale things before we carry them into the large scale engineering module.

We expect that it will have a pyro-process module and an aqueous process module tied to a remote fuel fabrication and that connection -- designing that interface is very important and I'm going to talk a little bit about that also. The idea is to -- for the materials never to leave the hot cells between separation and fuel fabrication. Basically, we separate the materials, ship them to fuel yards, into the next hot cell and do the fuel fabrication and then in a cartoonish sense, we expect from one end we'll get spent fuel coming in and from the other end of the hot cells fresh fuel will come out without the materials ever leaving the hot cells in between.

And also analytic laboratory obviously to support all these activities and an advanced control and monitoring center to not only around the plant but also to be able to test the advanced concepts on safeguards control and monitoring. So in that respect, we are trying to design it so that not only we do demonstrate something but also this becomes a facility for us to use for the next 50, 60 years so that we always maintain the state of the art. After we do the first demonstration of the fuel cycle, I'm sure we are going to learn second things and for the second generation we will want to improve certain things in terms of cost and performance and we will like this facility to be able to help us do that, too.

So it's not being designed just one single demonstration with a limited scope. I'm going to skip this but basically as I've indicated, we are in the early phases of the design yet, but we have a number of trade studies to complete in terms of exactly which way we are going to go with the AFCF, whether it will be a modular facility, how many modules it's going to have and how it's going to interact with the other facilities that are either under GNEP or that are -- that we are currently using.

Now, the rest of the view graphs really are related to the advance -- what we plan on doing in terms of the advanced safeguards research and how we plan on using AFCF to demonstrate advanced safeguards. I don't know whether this Committee is interested in listening to that or can we just leave it with the view graphs.

VICE-CHAIRMAN CROFF: I think with our time situation if you could -- I think we'd be interested in the safeguards thing, but if you could get through it relatively quickly, seven minutes or something like that, because I want to leave a lot of time for questions.

DR. PASAMEHMETOGLU: Okay, with respect to the NRC, this is one view graph that I want to talk about for a few minutes, that's now currently with separations and the fuel fabrication, really for advances safeguards, IAEA has certain goals that we would like to achieve in this kind of recycling plants and the IAEA goal, it's not a requirement, it's a goal, is to be able to detect a significant quantity and I guess I should go to the previous view graph, and that's a fixed amount. Basically, we should be able to detect any loss of eight kilograms of material within a year, that's the significant quantity and the uncertainty on that which they refer to as sigma inventory difference as 2.4 kilogram of plutonium.

So that's a fixed number. The regulations in the United States right now, at least the ones that are in there, granted that we haven't really operated -- built or operated a plant like this for a long time, are in terms of fractions of the inventory, a percentage of the inventory. And this is the NRC and this has really -- the basis for this has nothing to do with the separation plant. The basis for this is for a fuel fabrication plant and this is the DOE number. And the issue I want to point out which means that we really need to work closely with NRC as we go through this process in order to develop these kind of regulations, if you just convert the current numbers, existing numbers, to what it will take for us to operate AFCF, you are talking about basically an inventory difference uncertainty of 25 grams per year. That's impossible to detect.

Whereas IAEA would not -- regardless of the size, that would still be 2.4 kilogram for IAEA which you know, we are meeting. So these are the type of things that I think as we proceed in this technology in terms of the safeguard, these are the type of things we need to develop jointly. And the objective really is to be able to apply for what we currently know which is a robust safeguard technology for the PUREX plants that we are using all over the world, but be able to apply the same techniques to UREX and pyro and get the same robustness out of that, be able to achieve the IAEA goal of not losing any more than one significant quantity with a certain uncertainty with less intrusive means and by that I mean, we don't want to shut down the plants every other month to be able to take inventories and we don't want to take too many samples, and we don't want too many inspectors pushing too many buttons in there.

So reduce the requirements of continuous presence of inspectors and overall the objective is to reduce the risk of diversion from these facilities. And it is based on basically four different concepts. One is advanced instrumentation. We are working on a number of new instruments that are -- that may be more accurate, more robust, more reliable than what we had before to track down the materials as the materials flow in the plant and advanced control logic concept where we are basically looking at all the instrumentations that are in that plant, not just stuff that tracks down the nuclear materials, but the pressure sensors, temperature sensors, everything that we have in the plant and convert those into some sort of a safeguard envelope and every time something that shows that the plant is really not operating the way it should be operating, doesn't mean somebody is diverting something but there is something wrong with the plant operations, then you shut down the plant and do the inventory at that time. And these are mostly based on modeling and simulation and we are working on also basically an advanced virtual design of the plant fuel fabrication, plus separation plant jointly and do a lot of documentation and being able to embed safeguards into the plant design based on the virtual design, and then eventually demonstrate all those in the AFCF with a large enough scale so that you can really look at those materials.

And what I have indicated earlier the cartoonish concept as part of the safeguards by design is spent nuclear fuel comes from one end, the materials stay in the hot cells until it's converted to fresh fuel without leaving the hot cell so they are within a hot cell boundary until we have them in fresh fuel. There is no shipping in between; however I have to admit that on a cartoonish sense, it makes sense but designing that interface still requires a little bit of work to make sure that we can do an on-demand fabrication. And that's all really I want to say. I think that must -- and these are about -- I think that summarizes everything I want to say about safeguards.

VICE-CHAIRMAN CROFF: Okay, thank you very much. I think what I'm going to do is go around again and allow everybody one question to start and we'll see how much time we have left and I think at this point, I'll leave -- let the questioner direct a question to any of the folks up here as opposed to just Kemal, depending on, you know, where your interest lies. So with that, Jim?

MEMBER CLARKE: Thanks, Allen. Just a quick question for Kemal. You mentioned the americium and the high vapor pressure and you need to recover it given the current approach. It strikes me that if there were a way to keep it in the matrix and not compromise the quality of other operations that that would be preferable. Is there -- can you continue to look at that or is that --

DR. PASAMEHMETOGLU: Yeah, our baseline approach is basically keep it in the matrix and that's why we are looking at that induction furnace. If we just floated the solid materials, heat them very quickly, melt them very quickly, cast them very quickly, so that we don't lose any americium, that's our baseline approach. Recovering americium is a backup approach.

MEMBER CLARKE: Okay.

DR. PASAMEHMETOGLU: It's the first demonstration.

MEMBER CLARKE: Okay, I misunderstood. Thank you.

MEMBER WEINER: This is a kind of general question. And I address it to anyone who wants to answer it. When Dr. Laidler gave his presentation he was talking about nitrite fuel as if it were, you know, a done thing. And then I look at your slide and the nitrite fuel still has a great many problems. So my question is, generally, can one or all of you draw a line as to where you have actually tested something where you have some confidence that this is a going technology or where you're simply are -- I don't want to put it simply, but where you are still in a planning look at options stage? Is there some break point in here related to fuels, related to instrumentation? Can you give some idea because I'm a little confused as to how much of this is going to change -- have to change direction of necessity as we move toward the goal and how much of it is -- are you confident in.

DR. PASAMEHMETOGLU: Well, at this point for the fuels, I can speak for fuels, and then I'll ask Jim to comment on the nitrite, on the nitrite fuel, but for fuels, we are confident that we can make either metal or outside work. Therefore, those are our baselines and until we do some remote fabrication, in either one, it's very difficult to choose between the two because there are different issues and one is part of processing, the other one is this metal casting and we expect that after we do some hot cell remote fabrication, which will be within four to five years. At that time, we will be able to better make a decision on which one is our primary. So it will be metal or oxide.

Nitrites and dispersions have some nice futures to it, but as you have indicated, we have a long way to go; therefore, they will always remain the background research.

MEMBER WEINER: Thank you.

DR. PASAMEHMETOGLU: I don't believe, though, Jim is really basing his conclusions on nitrite fuels, but I'll let him speak to that.

MEMBER WEINER: I was simply using the

nitrite fuel as an example.

DR. PASAMEHMETOGLU: Yes.

DR. LAIDLER: Well, let me correct that. We're not developing processes for nitrite fuel, only for commercial oxide fuel and potentially for fast reactor metal or oxide. We know that we can handle nitrite fuel with a UREX process but we're not -- we're not including those tests in our repertoire, only oxide -- commercial oxide and fast reactor metal and oxide.

MEMBER WEINER: Could you extend the concept to the rest of the -- just in general to the rest of the processes or are you -- are you at a stage of confidence where these things can really go at an industrial level?

DR. LAIDLER: I'm very confident in the aqueous solvent extraction process because we have a lot of worldwide experience on that. The pyro-chemical process is at a very early stage of development and we just -- that's one of the reasons for having the AFCF. We can run that process on real spent fuel and do the real separations. But, again, that -- to give a time frame, it's probably two decades away.

MEMBER WEINER: Thank you.

DR. SAVAGE: I would like to make one general comment regarding the budgetary approach in the global nuclear energy partnership for the US program. The majority of our funding will be going into the demonstration projects to demonstrate the technologies that we feel are the most mature and have the least technical risk, but we retain an R&D component to the program which is a smaller amount to continue to investigate these higher risk processes to give us alternatives.

MEMBER HINZE: A brief general question with a few parts and this concerns the GNEP. What --where does the United States stand in terms of fuel developments compared with other nations and what's the level of cooperation and at what level is the cooperation being conducted among that nations and is there a -- any sense of an attempt to approach uniformity to our fuels on a global basis and is that important. And do others -- are others as concerned about non-proliferation in their development of these as we are? Is that a brief question?

(Laughter)

DR. PASAMEHMETOGLU: Yeah. The answer is not going to be very brief, though. No, actually there is quite a bit of collaboration among certain countries. Our collaboration with France in terms of the transuranic fuels, this transmutation fuels has been outstanding so far. There is the collaboration ongoing. Until GNEP came along, the sense of urgency was not there. So where we are with respect to transuranic or transmutation fuels in general is about equivalent of where Japan is, where France is. They are also doing similar thinks we are doing, small scale glove box fabrications at small scales and small scale irradiations and doing extensive characterization and trying to figure out what makes sense, what doesn't make sense.

Of course, with GNEP now, the program is going to get accelerated, hopefully quite a bit accelerated, and I'm hoping that those other countries will support that. It's really important to do this. That chart I showed you, the eye chart that I showed you, from the beginning to the end, it takes about 15 to 20 years to get there for one fuel type. Those experiments, they're not things that we do overnight and then look at it the next day and iterate again.

From a concept to qualified fuel, it takes 15 to 20 years and United States, regardless of how big of a budget we can throw at it, we can only do a few of those and it's very important that we do this internationally and share the data and make a decision on what really makes sense collectively.

With respect to proliferation, I think obviously other countries have different views of proliferation, because we don't do PUREX and they do PUREX and they don't see any problem with that. However, with respect to fast reactors, which GNEP is really looking at at the very end of the fuel cycle, I don't know any country that would disagree with the United States that if you're going to put this stuff into the fast reactors, this is the right way to put it in, in terms of group transuranics.

For thermo-reactors it is really difficult to put the group transuranics into thermo-reactors. That's why those other countries do PUREX and separate the plutonium. However, for what we are authorized to do on the GNEP, I think we will have full collaboration of other countries, regardless of what their view of proliferation issue is.

MEMBER HINZE: And the non-proliferation concerns in the development of the process, build in non-proliferation aspects of it, is that -- is that in accord across the nations?

DR. PASAMEHMETOGLU: The safeguards research that we are doing, we have received a lot of interest from the Japanese and the French to participate and work with us in terms of the safeguards by design approach as well as the advance instrumentation approach and they -- and I believe everybody realizes that if this is going to be a worldwide thing, we need to look at it.

MEMBER HINZE: Thank you.

DR. SAVAGE: I would also point out that there is another program in the Department's Nuclear Energy Office called the Generation for Advanced Nuclear Energy Initiative and there's a synergy between that program and this one. In fact, we feel that the nuclear power 2010 program to promote new reactor construction in the United States, the Generation For program, are all elements of the GNEP vision because without growth of nuclear in the US, the need for these technologies to deal with the waste management issue, the non-proliferation issues, our role in the world as a nuclear supplier state, are meaningless. So all of these programs work synergistically to achieve the ultimate goal, which would be a sustainable closed fuel cycle optimizing the use of the uranium resources and other fissile materials for energy production in a manner that is economic and promotes proliferation resistance.

MEMBER HINZE: And minimize waste.

DR. SAVAGE: And minimize waste, right.

VICE-CHAIRMAN CROFF: Howard?

MR. LARSON: Dr. Hinze has sort of the same question I did because when we looked at the safeguards segment, there's quite a difference between NRC, DOE and IAEA. I just wondered how the other countries feel with our goals being so much lower than the IAEA's. I know you said you wanted the plants to be able to meet the IAEA goals.

DR. PASAMEHMETOGLU: Yeah, but the funny part of it is, though, when I was looking at it with respect to the small pilot scale plants that we are trying to do before we go commercial, if we were to apply those numbers to a commercial plant, they'll all come out about the same and I think that's where the NRC's 0.1 percent number came from based on the JMOX plant in Japan. If we were to do it at the commercial scale, 0.1 percent would be roughly equal to what the IAEA is tracking.

But when you try to apply it to a small pilot scale plant, then all of a sudden it becomes impossible to apply. That's why I was making that comment.

VICE-CHAIRMAN CROFF: Okay, Larry?

MR. TAVLAREDES: I was curious about the scale-up issues that you mentioned a bit. And it seems to me it's going to be a challenge to go from the -- what you would say laboratory casting methods to a continuous process to make these and I have several aspects of this, questions related to that and that is first of all, it seems to me you have to go from a bench type continuous process to a larger scale and I think maybe the scaling up would not be linear. And so what problems do you see involved in going from the scale-up in the fuel fabrication and do you have any connections with the European community who may have facilities that may be helpful to you in doing this?

DR. PASAMEHMETOGLU: Let me answer -- I guess, let me answer the question in the reverse order. With respect to the European community, the only place where we can really do remote fuel fabrication in Europe right now, the only facility is -- at least the only facility that I'm aware of is in a place called the Transuranic Institute, TIU in Carlsrule (phonetic). However, they do not want to -- they do not want to contaminate their facility with powder processing so they are limited to a very few type of processes that they are willing to test in there. And they are not in the metal fuel business at all, so they don't have any equipment to doing metal fuel, therefore, that's not going to work.

In terms of the scaling, in Russia there is -- but they are mostly working no the vibro-pac and the sphere-pac technology for remove fuel fabrication issues. So if we can collaborate with the Russians, that will probably be a good thing in that respect. In terms of scaling the processes from the laboratory scale to large scale, on the pilot processing that's already done because if our scheme works, it's going to work just like the MOX fuel. For the metal, you're right, we still -- but the nice thing about it, it's not something that takes 10 years to develop and test. We can test the different concepts.

Once we have a hot cell facility up and operational, which we plan on having next year, after that it takes a few months to test a concept. If it works, great; if it doesn't work, you tweak a few things. So within a few years, I think we will find something is really the right scale for the scaling approach.

MR. TAVLAREDES: Thank you very much.

VICE-CHAIRMAN CROFF: Ray?

MR. WYMER: I had a couple of comments and then a question. Is that okay?

VICE-CHAIRMAN CROFF: All right, I'm soft.

(Laughter)

MR. WYMER: Okay, the comments, are, I was a little surprised that there was no mention of the fairly extensive Indian program on carbide fuels for fast reactors and the second comment was, I'm not sure you know both these things, that there's also over 40 years experience in fabrication and irradiation on a small scale of transuranic elements up through California at Oak Ridge and the RADC. And while the irradiate those in the thermo-flux reactor, still there's a lot of aspects of the performance that ought to be of some value and I'm sure you're aware of that and I mention it sort of as general information.

The question is, when you do the fast fuel reactor cycles, after awhile you build up a whole suite of higher actinides. You must have a bleed-off stream eventually because those become troublesome after awhile because they're parasitic. And I wonder what you plan to do with that bleed-off stream that becomes a waste stream.

DR. PASAMEHMETOGLU: Well, I guess that's more of a Jim question than my question because everything Jim gives me, I'll turn it into fuel.

MR. WYMER: Okay.

DR. LAIDLER: One of the beauties of the fast reactor is that you don't climb up the higher transuranics that quickly.

MR. WYMER: Not so quickly, right. So you can go around the loop a number of times.

DR. LAIDLER: Exactly, and there is -- in any of these schemes there has to be an exit strategy and we may exit from that cycle after 100 years or so at which point maybe we can apply accelerator transmutation to the residuals.

MR. WYMER: Okay, that's your fallback position.

DR. LAIDLER: Yeah, I'll be gone by then.

(Laughter)

VICE-CHAIRMAN CROFF: I'll let myself in on this. First, on your question, Ray, I've run recycle calculations for -- in fast reactor for a lot of cycles.

MR. WYMER: Yeah, I know you have.

VICE-CHAIRMAN CROFF: And it doesn't build up, period.

MR. WYMER: You don't get any in the higher stuff, the higher --

VICE-CHAIRMAN CROFF: Huh-uh, because

everything fissions before it gets there. Everything is fissile or fissionable in a fast reactor and --

MR. WYMER: I'm surprised that all of it does, Allen, but you're the authority, I recognize that.

VICE-CHAIRMAN CROFF: And actually, if you put LWR plutonium in a fast reactor and cycle it around a number of times, the quality improves.

MR. WYMER: Oh.

VICE-CHAIRMAN CROFF: Believe it or not, it ends up looking like very nice material.

MR. WYMER: Well, I'm talking to the father of the origin code that does all these calculations.

VICE-CHAIRMAN CROFF: But then my question, it will probably go to Buzz, I guess, I read through the -- I guess it was your report to Congress that you sent two or three months ago and I'm remembering, I think it was there, mention that you are at the beginning stages of preparing, I think it was a generic environmental impact statement. Can you talk a little bit about -- well, I'll call it the scope of it or what you're trying to decide through that process?

DR. SAVAGE: The initial scope that was announced for the Environmental Impact Statement was that it was strictly for out technology demonstration program which involved three demonstration projects; this larger scale, I call engineering scale demonstration of the UREX+1A technology separations, an advanced fast test reactor for testing transmutation fuels and the advanced fuel cycle facility. Those are the near-term projects in the GNEP vision for the US component of the program and the Environmental Impact Statement is evaluating the alternatives for those projects as far as technologies and site locations. And it will be a two-year process and we're -- we have a contractor on board to lead the effort and a draft of the EIS is due about a year from now.

VICE-CHAIRMAN CROFF: Okay, then let me go beyond it. I hadn't understood it was that narrow of a scope and preface it by noting that what is it, 30 years ago or so, the government, I guess the AEC actually, started it, the Generic Environmental Statement on Mixed Oxide, which, you know, basically appeared to be necessary for legal reasons I don't understand for the country to recycle plutonium, which they had wanted to do at the time. And that became a fairly contentious exercise that was not completed because of President Carter's policy decision.

And some regulations were put on the book, I guess sort of as a result of that, but what plans are there -- I'm presuming somebody will have to pick up that football again, at some point and complete it, you know, for the widespread deployment and finish that process. Is there any thinking about that?

DR. SAVAGE: There is. I'm not directly engaged in that activity. We are looking beyond the EIS for these initial demonstration projects to a programmatic level environment impact process afterwards. So there will be people evaluating that before we get into that programmatic but that will probably end up being in that programmatic EIS.

VICE-CHAIRMAN CROFF: Okay, and I'll note, I think something that flowed out of that at the time was an EPA, I guess it's a standard, 40 CFR 190, that is titled something like Releases from the Uranium Fuel Cycle, but it includes processing and fabrication. One part of that limits release of radioactive iodine and krypton, and if I work the numbers right, I think the DF for iodine, required DF was 300 and for krypton 100.

It's expressed in curies so you've got to do some gyrations to back it out. And in the Federal Register notice that promulgated that, the EPA indicated -- this is in the background information, of course, that they also wanted to look at let's see tritium and carbon-14. It's just they hadn't been able to assess the technologies to decide what a reasonable number was and at that point -- they never pursued it, of course, because again, President Carter's policy decision. But there is a little bit of information there and it seems to me that's probably going to come to the forefront in this Environmental Impact Statement. How much you put up the stack is the basic issue and that may have to be revisited.

I think with that, NRC staff, anybody have a question? Okay, we've still got a few minutes here. Anybody else, I'll throw it open. Anybody?

MR. FLACK: Allen, if I could just ask a question, with respect to the fuel, eventually that needs to be put into a reactor and I assume that reactor may be something like a liquid sodium reactor. Do we fully understand how the fuel will behave under the transient conditions that could evolve both for design basis accidents, and beyond the design like ATWS and that sort of thing, and how that would be addressed as you begin to evolve a model for the fuel, what the fuel should look like? Is that --

DR. PASAMEHMETOGLU: Well, that is the phase, the transient phase. You're right, we need to understand that. I cannot state at this point that we fully understand that based on the data that we have to date. We haven't done that. However, the data that we have obtained to date is showing that at least the metal field is behaving very much like uranium plutonium metal field, so we have expectations that the transient behavior will be very similar as well.

However, obviously, we have to test it and that's why we need to have that TRET (phonetic) facility, the transient reactor to do those transient tests and to put the -- before we can really say this is our fuel guide and what our power limits are and what our safety modules are. So that's -- it is part of the program.

MEMBER WEINER: This is really a question for Dr. Savage. If -- when these processes go commercial, when they become part of commercial fuel plants, of course, it will be regulated by the Nuclear Regulatory Commission. So I wonder to what extent, if any, you have been communicating with the NRC to design a regulatory framework for this.

DR. SAVAGE: We've already had several meetings with NRC and the problem has been recognized here at the Commission. A White Paper has been prepared for the Commission on what the regulatory issues are likely to be. Our current position with respect to our demonstration projects is that if they're built on DOE sites, they probably will not require NRC regulatory oversight. However, in our design efforts we want to bring NRC into the review of the designs as we develop them so that they can be licensable when do go commercial. So we will engage and keep NRC engaged throughout even the demonstration projects.

MEMBER WEINER: That's very forward thinking.

VICE-CHAIRMAN CROFF: I'll ask a question, probably for Jim. There was mention of very high burn-up LWR fuels up at the, you know, 100 gigawatt days per metric ton and maybe beyond. Are there any issues that arise concerning processing? Can these things be dissolved, for example? Are there any issues there that come up?

DR. LAIDLER: There are a lot of issues. The first issue is getting to 100,000.

VICE-CHAIRMAN CROFF: Well, I understand.

DR. LAIDLER: The second issue is any linings that are built into the fuel may complicate the processing. The third is that as you go to higher and higher burn-ups it becomes a little bit harder to get complete dissolution. So we may have to in those cases, resort to some either an advance dissolution process or, perish the thought, to the introduction of fluoride ion into the system.

VICE-CHAIRMAN CROFF: Understand.

DR. LAIDLER: I don't like to do that because of the complications of process equipment.

VICE-CHAIRMAN CROFF: I understand. Anybody else here?

DR. LAIDLER: Let me add one thing to that, Allen. The other point is that in some cases, these advanced fuels will require the introduction of reenable (phonetic) poisons and reenable poisons tend to be lanthanides.

VICE-CHAIRMAN CROFF: Okay.

DR. LAIDLER: And that just imposes a more severe restriction on the removal of lanthanides in our processes.

VICE-CHAIRMAN CROFF: I understand. Well, we're a whole three minutes ahead of schedule but I think that's not a problem. I'd like to -- I'd like to thank all of you for the presentations. They have been very helpful to us to get us into a common framework as to what's going on and I suggested in the number of questions that you did a really good job of that. I suspect in preparing this White Paper some of the fellows, you know, may have -- you know, may be on the telephone wanting a little bit more detail in some areas, but I hope that's not a problem. Yes, sir.

DR. SAVAGE: Can I make one final statement? DOE's office of Civilian Radioactive Waste Management still exists and Yucca Mountain licensing is their highest priority. That is one of the Secretary's highest priorities as well. So this program does not intend to do anything to divert attention on the path for Yucca Mountain.

VICE-CHAIRMAN CROFF: Okay, thanks. And my sincere thanks for coming by. It was really very helpful and I think an eye-opener a little bit on just how complicated some of this is going to be. There's a lot of boxes on those charts. So with that, I think we'll adjourn this session and we'll be back in session at 10:30.

(A brief recess was taken at 10:13 a.m.)

(Back on the record at 10:31 p.m.)

VICE-CHAIRMAN CROFF: Well, I'm still short a couple of Committee members but our schedule it tight and I think maybe yours is too, so let's go ahead and get going. I think for this session the designated official is going to be Latif Hamdan. And before we go, we've got somebody on the phone here. Would you introduce yourself, please?

MR. ROSENBURGER: Yes, this is Kent Roserburger with Washington Savannah River Company.

VICE-CHAIRMAN CROFF: Okay, thank you. Anybody else out there? No, okay. This session on Standard Review Plan for Activities Related to the US Department of Energy Waste Determinations. And staff has released a draft SRP and the ACNW proposes to comment on it and this is sort of a question and answer session on the draft SRP so you're going to, I guess, walk through some things and then we'll have the questions. Anna, do you want to take the lead?

MS. BRADFORD: Yes.

VICE-CHAIRMAN CROFF: Okay, take the lead.

MS. BRADFORD: Good morning. My name is Anna Bradford and I'm the Project Manager for Development of the Standard Review Plan for Activities related to Department of Energy Waste Determinations. And with me is Dr. Christianne Ridge and Dr. David Esh, the two other main staff contributors to the SRP.

And as you know, we were here a couple of months ago back in May and gave the Committee a presentation on the overall contents of the SRP but at that time the SRP had not been publicly released and the Committee had not had a chance to review the document, so that overview was at a pretty high level. But since then, the document was released for public review on May 31st. It's open for public comment until July 31st. Copies of the SRP were provided to the Committee and after you had a chance to look at it, your staff, as you mentioned, then transmitted to us some specific questions or comments from which you wanted to hear a little bit more specific information from us.

And that is the purpose of today's presentation is to really get to those specific areas. We're not going to go back over information you've heard before such as, you know, history of the NDAA and things like that. So Dave Esh and I will each cover several topics and hopefully, what you hear today will help you focus and clarify any recommendations you might have to give us for the final SRP.

And although I'm not going to go, like I said, to the background of the NDAA, I did want to talk for a minute about what we can and can't do under the law and these are the things we had to keep in mind when we were developing the draft Standard Review Plan. And the first is that DOE is only required to consult with the NRC. We do not have any regulatory authority over DOE and we do not have any authority over their activities with respect to this waste. Also, the NDAA does not apply to the cleanup of the entire site. It's not a site decommissioning law. All it does is provide specific criteria for determining whether certain waste required disposal in a geological repository or not. It really applies to only a small portion of all the clean-up activities that DOE might be performing at a site. And the SRP does not address all the other cleanup activities that might be going on at that same site

And it also particularly specifies the use of Sub-Part C of 10 CFR Part 61, not some other cleanup requirements. It specifically calls out Sub-Part C. Also that our monitoring role under the NDAA is limited to assessing whether or not DOE's disposal activities are in compliance with Sub-Part C. Again, we don't have any regulatory or enforcement authority over them in monitoring space. And we also don't have any authority or consultation role when it comes to other spills or leaks that may have already occurred at the site. And we'll talk more about monitoring a little bit later in this presentation.

And before we get into the technical details, I wanted to talk for a minute about the purpose of the SRP. And as you know, it's a document that provides guidance for the staff that may be conducting reviews of waste determinations. And it describes the types of information that may be assessed by the NRC staff during its reviews. For example, if we're looking at the performance assessment for closure of a high level waste tank, what types of things would we be looking for?

And having this documented in the SRP will help provide consistency across the different reviews we're doing and also because we'll be using different staff reviewers. I also wanted to point out that the SRP is deliberately written to be flexible and applicable to the wide variety of things that we might be analyzing in waste determinations. As you can imagine, it might be hard to be very prescriptive when we're looking at things such as closure of tanks in place, removal of waste which would then be treated and disposed of elsewhere in a low level waste disposal facility, maybe as grout, maybe as glass, looking at a piece of vitrification equipment such as a melter or looking at an evaporator for a tank farm. So we really needed to be broad.

If we had tried to be too prescriptive, this document would have been very large and probably still wouldn't have covered all the bases of all the things we might see in the future. Dave is now going to talk about some areas with respect to the PA. Following Dave, Christianne will talk about radio-nuclide removal and some cost benefit analysis. Then it will come back to me to talk on a few remaining issues such as existing guidance and monitoring.

Dave?

DR. ESH: Thank you, Anna. I guess now is the part of the presentation that we like to call Christmas in July because you get to hear me speak for 30 minutes. But I'm going to focus on performance assessment. It's a main part of what it's done in these reviews to demonstrate compliance. And this introductory slide is just providing a summary of the overlying elements and philosophy of the SRP with respect to performance assessment. We expect that performance assessment is going to be what's used, the analysis approach, to demonstrate compliance with 10 CFR 6141. The SRP provides guidance on general topics, such as data uncertainty and model support as well as the specific topics, such as say estimation of infiltration rates. And as Anna mentioned, SRP has to be written to consider site to site variability and also problem to problem variability.

Everybody tends to like to focus on tanks, but tanks are one incidental, one type of waste incidental to the processing review. There are other types of reviews too, that have different implications. So this review that we do, it's anticipated that they're performed with a risk-informed approach and that's necessary for a variety of reasons, mainly because there's a large amount of information and you have a limited amount of time and resources to perform the review, so you have to focus on those aspects that are most important to the -- most likely to influence the demonstration of compliance. Next slide, please.

In performance assessment review procedures, we have an allowance for deterministic or probabilistic approaches and the reason is that those different approaches can be used in different circumstances and they have their pros and cons. We had a separate section devoted to uncertainty and sensitivity analysis which we feel is an important part of the performance assessment process. We also have separate areas on evaluating the model results and defining the contributions of the barriers because if you can't evaluate your model results and define what's driving the calculations, then it's going to be very difficult to implement a risk informed approach to the review. Next slide, please.

The Committee had a number of questions about the performance assessment approach and the SRP and I wanted to reiterate here at the top that these reviews, we typically will measure or characterize the review not in say pages of documents but in inches of documents and the highest level documents may be hundreds and hundreds of pages and multiple documents and there might be hundreds of supporting references of various size, so if you're going to comb through that information and try to ask the right questions, you really need to focus on what are the areas that you think are driving the results.

The SRP does not prescribe a specific analysis technique to demonstrate compliance either deterministic or probabilistic but you can use different approaches and there's lots of reasons why you would use different approaches. And at the bottom here I say, "Compliance does not equal reality, compliance equals safety". I think this is one of the most important points that we're a regulator and our main goal is to insure that public health and safety is protected. And one way that you can do that is by being pessimistic or what people commonly say conservative in their analysis. That's a way to insure that you've protected public health and safety.

Ideally, the performance assessment would be a very close representation of reality. But when you have a lot of uncertainty, it's difficult to make a judgment as to whether you've not underestimated your impacts and therefore, that you're not protective of safety. So I think this is an area where maybe I'll spend a few minutes and talk about a little bit on my philosophy.

In the SRP we don't anticipate a particular approach. DOE can use whatever approach they want and justify. We certainly indicate a preference for probabilistic analysis. We think there's probably more advantages to disadvantages but a deterministic approach can be used. If a deterministic approach is used, we feel it has to be reasonably conservative because it's not explicitly representing the uncertainties. And it can be a very big challenge to represent that uncertainties in a deterministic calculation or to evaluate them, I should say, not represent them because they don't act in a linear manner and you can't look at them one at a time necessarily in these types of models.

The models respond in a non-linear way that if you look at one uncertainty or one sensitivity at a time, you usually don't get the full picture of what the sensitivity -- what the impact of the uncertainty is in that type of analysis approach. So we provide guidance on each approach in the SRP and we think that's appropriate and we indicate our preference for a probabilistic analysis but we can't prohibit the other analysis. All we can do is provide guidance as to what the shortcomings may be and the types of things a reviewer needs to look for if say a deterministic analysis approach is used.

We understand the problems with using a deterministic analysis. The Committee had some questions about well, shouldn't you be using a best estimate type of deterministic analysis with a pretty rigorous sensitivity analysis? And I would argue that the problem with that is if there's a lot of uncertainty and you're using a best estimate, the likelihood that you've under-estimated the impacts is much higher than if you've used a conservative analysis of some sort. So one of the issues is, well, if you use a conservative deterministic analysis and then you're trying to estimate the cost benefit of removal, which is related to the impacts that you've generated with your performance assessment, how is that valid because you have this conservative estimate of impact. And so when you're calculating the cost benefit, it's based on this number that's conservative.

Well, yeah, it is. What that would lead to, though, is you're going to make a decision to remove more waste than what you probably should which protects safety. It doesn't -- if you use a best estimate, you could maybe lead to the -- come to the decision that you don't need to remove more waste when you really should be removing more waste. So I understand that in an ideal world you would want to use your best estimate deterministic analysis but if you have a lot of uncertainty, there's a risk to doing that and I think that two approaches that we advocate either a probabilistic analysis or a conservative deterministic analysis are the two approaches you have to use if you have a lot of uncertainty and these problems have a lot of uncertainty.

Now, another complication if you use a deterministic analysis is how do you call -- what is conservative? How do you define conservative? It is a challenge because many things -- it's not obvious what the conservative answer is. And the example I would give is, say groundwater flow, is it conservative or over-estimate groundwater flow or underestimate it? It's actually dependent on the problem. If you increase groundwater flow, you're increasing the transport rate but you're also increasing dilution. So it depends on your specific radio-nuclides in your problem and your specific problem. Increasing the groundwater flow rate will increase the arrival time of the long-lived radio-nuclides but it will dilute the concentrations of the shorter-lived radio-nuclides or the more mobile radio-nuclides that may have been arriving at the compliance point already.

So there's a trade-off and the maximum might be in the middle or it might be at either end of the spectrum, but that's just one example. There's many examples in these types of calculations where it's not obvious what the conservative selection is, even though people will attempt to make conservative selections, what they call conservative. I have a problem with even using conservative because a conservative -- the terminology implies that you know what the answer is. And in these problems, the performance assessment, you're going your projection of what you think reality is.

We won't know what the real answer is but hopefully we can estimate an impact that will assure safety that we've over-estimated it. If you're designing a bridge, you'll put a safety factor in the design of the bridge. You will over-design the bridge. The performance assessments should be over-estimated. Even if it's a probabilistic analysis, you're probably over-estimating because there's some areas where you can't adequately represent the uncertainty or maybe you have variability that you don't want to handle, spend the effort to try to handle and so you'll try to make a conservative decision.

So, it's kind of a soapbox issue but I think it's important that we feel pretty strongly that the approaches in the SRP are the ways to go. A different approach, I think, could be problematic for us. Yeah, as a scientist, I want to know what the true answer is but as a regulator, I want to insure people are safe, and those are two different answers. And that's the point that I want you to take. Next slide, please.

In our performance assessments we strive, if we can, to perform our own independent analyses, given our resource considerations and schedule. These independent calculations may include a probabilistic performance assessment if we feel it's necessary. This review approach, we believe helps focus our review and strengthens the basis for the results of our review. As I indicated, it's a large amount of information and if you spend your time focusing equally on all areas, you're going to dilute your effort on areas that most influence the decision. Now -- or most influence your estimated risk.

The risk that we are estimating is a compliance risk, that's what I call it. We -- everybody talks risk and risk regulator. We're looking at the risk of exceeding a limit or a compliance type risk, which may be different than the actual or true risk. If you have a limited knowledge of your system, your compliance risk is probably going to be much larger, it's estimated to be much larger than what the true risk is if you really knew it.

As you collect more information, you can collapse those two closer and closer together. But if you don't have a lot of information and you have a lot of uncertainty, you almost -- by definition, your compliance risk is going to be quite a bit larger than what your true risks are, but that's the approach you have to use to protect health and safety. Just as if you were designing a bridge, you wouldn't design it at what you think the minimum strength is for that bridge to withstand the forces it's going to see; you over-design it so you're pretty sure it's not going to fall down and injure somebody or create a hazard.

We don't rely on these independent calculations as a basis for our decision though, only to inform the review process. Our decisions are based on the calculational results of DOE. Next slide, please.

Now there are a variety of questions on these higher level issues that were provided to the staff by the Committee. There's a whole list of them here. I really couldn't do slides on each one in the time that we have now, but we'll be happy to jump back and discuss them in detail. What I'll do now is just walk through them and say a few words about each one. The compliance period, 10,000 years, the Committee had a question of whether you could look at a shorter compliance period. Certainly if the risks were -- you can show that the risks occurred earlier, you could argue that you would evaluate a shorter period. The compliance period is kind of fixed, though, by the scope of the problem. The analysis period may be shorter, you can argue it needs to be shorter to demonstrate compliance.

The actual compliance period would be still our 10,000 years which we think is appropriate to look at the long-lived mobile contaminants and long-lived less mobile contaminants. Institutional controls, we are not attempting to do anything new in regulatory space here. We're following the Part 61 approach which specifies an institutional control period of 100 years. There were some questions about whether we could use an LTR approach, which the Committee says may be more risk informed. I would say it's different but it's not necessarily more risk informed. In the LTR approach you can analyze unrestricted release which means the people can access the site at year zero. Then you apply a 25 millirem per year dose limit or you can analyze restrictive release, where you evaluate that the institutional controls are in place as long as needed up to 1,000 years and the public receptors at the boundary of the site, the maintained area, but you also have to do a calculation that the controls fail at year zero and then you evaluated a dose limit of 100 or 500 millirem per year.

So it's a different calculation but it's not necessarily more risk informed and our calculations for the first 100 years there's no impacts assessed to a public receptor, it's -- the site's under control, the public receptors are only evaluated for ongoing operations at the site boundary, but during that time, there's no potential for an intruder to intrude into the system. So in many of these problems where you have a lot of cesium and strontium, on the order of 30 year type half-life, you're looking at an order of magnitude reduction in the risk over 100 years. So if you analyze the risk as year zero compared to year 100, you'll be looking at impacts 10 times larger than what we evaluate in this analysis.

And then when the controls fail, in the

analysis that we do for Part 61, that intruder is inside a buffer zone, which could be in the area where the waste is. The public receptor is outside the buffer zone. In the LTR analysis, the receptor is evaluated at the point of maximum exposure anywhere, so over top of the waste or wherever the point of maximum exposure is. So our approach is sticking with the Part 61 approach. Yeah, there's other things you could do but I don't see that the LTR approach is more risk informed, nor do I see that there would be a big benefit to extended the institutional control period for most problems, because we're looking at a situation where the technology is such that the intrusion occurs at 100 years. Where they have an intruder barrier that they may argue they can take credit for which will prevent intrusion for up to 500 years, but the risk from the long-lived contaminants, whether you start the release -- the processes that can lead to release 500 or whether you start them at 100, all it does is shift the arrival time of the peak by 400 years out some time in the future.

So maybe you're changing the arrival at 5,000 to 5400, it doesn't have a big impact for long-lived contaminants. So only if you went to the process where you allowed institutional controls for the whole analysis period and therefore, you could prevent -- you could argue for the prevention of contact with the waste or for a very large buffer zone effectively between the waste and the public. That's the only real benefit to allowing or arguing about what the institutional control period should be.

The use of water, I don't think we explicitly called it out in the SRP but this issue is that basically if the water is non-potable, would you allow the Act to evaluate the impacts from the water and that answer, of course, is no. If the water if not potable, we wouldn't assume that somebody is going to drink it. And my personal opinion is, that's one of the best ways to assure safety of a site is you put it some place where people aren't going to use the resources and the water is not either accessible at the yields or it has a state that people aren't going to use it. Over the long term, that's probably the best way to assure safety of one of these systems or sites.

Conceptual model uncertainty, there were some questions about how do we evaluate that. We don't evaluate conceptual model uncertainty different than any other uncertainty. We realize it's a little bit more of a challenge but when the staff performs one of these reviews, we basically have to ask yourselves, is there a different conceptual model that could be used that would result in a higher impact and -- or is the information sufficient to constrain it to the conceptual model that has been presented? So we evaluate the conceptual model uncertainty integrated with all the other types of uncertainties. It's not treated any differently.

Engineer barrier performance is a big part of these problems and it is a projection of performance into the future. We had quite a bit of guidance in the SRP about engineer barrier performance. We think that's needed and justified because these problems are going to rely on barriers. If you can't rely on barriers, the problems are done already which is, in most cases, they wouldn't meet compliance. You do need to rely on barriers to some extent. Estimating their performance is a challenge but I don't think we are constrained to saying that barriers can only last as long as the experience that we've had. There are a number of barriers out there that have lasted much longer than our recent experience. It may not be a barrier in a radioactive waste facility but there are natural analogues to many of these systems and processes that I think are reasonable representations of what we could expect to occur. And a couple of examples I'll give you is that for erosional stability, for instance, there's a native American burial mounds that have lasted for many, many hundreds to thousands of years in a variety of locations and environmental conditions. And yes, some of those have probably failed and some of them have partially failed, many of them have remained intact. That shows that they're basically -- they have a layered type system when they were designed and they're somewhat analogous to the layered type engineered caps you might see in these waste disposal systems.

They've lasted a long period of time and we're considering in decommissioning space doing some work to try to evaluate those more quantitatively and try to understand why they've lasted and develop guidance there. Certainly, if we did that, we would reflect that in our Standard Review Plan for incidental waste or if not in the plan at least in -- we would mention that guidance for the reviewers to consider.

Other examples are cementitious materials.

There are certainly examples of cementitious materials that have lasted for very long periods of time. The point I want to make about barrier performance is a lot of it comes down to functionality. There's a difference between saying a cementitious material can control the chemistry of a site for thousands of years compared to it will provide a hydraulic barrier for thousands of years. Cements and concretes, as you heard two days ago, there's been quite a bit of work, but there's still quite a bit of uncertainty. They're subject to discrete failure, cracking and it's hard to project when and to what extent they're going to crack.

That would limit the functionality of that barrier as a hydraulic barrier but the mass of concrete is still essentially there and if the pore fluids of the concrete are what's controlling release, you can estimate pretty easily how long that calcium hydroxide is going to be present and how long it's going to buffer the ph or the system which will limit the releases of the radio-nuclides. So performance, we really take a risk informed approach there. We don't view barriers as failed or unfailed. We view them as varying degrees of performance. And a barrier can start losing its performance but still be beneficial to the system, so people like to talk in failed and unfailed and I don't think that's really fair because all these things are a continual spectrum of results.

There was a question about the stability of tanks under variability saturated or saturated conditions. And this was a problem that was recognized in Part 61. If you look at the technical requirements, it basically says you cannot site a low level waste facility in an area -- in a zone of water table fluctuation basically. And that was because at the time, there was a lot of uncertainty about what that would mean for the release of contaminants.

There's still uncertainty with that, but in the SRP we don't take a prescriptive approach. We will consider that situation and consider how the risks were evaluated and if they were evaluated appropriately and if there's a basis for the release that's been considered but we don't say one condition is prohibited and one condition is favorable. We understand that there could be a variety of conditions that we'll see in our review and we'll evaluate them accordingly. Certainly, we'd probably focus more review effort on the situation that's more complicated. That should be understood.

Level of proof, we use reasonable assurance as our level of proof and we don't define it any differently here than in any other regulatory construct where the NRC uses reasonable assurance. So that's -- I guess I'll let it go at that. Climate change, we do consider climate change, natural -- climate change from natural processes. Climate change can influence a system but we don't consider human induced climate change and the reason for that, I think one argument I could make, in addition to how would you estimate it, which there's a lot of people arguing about climate change and they aren't arguing about what the 10,000 year value is. They're arguing about what's the impact of climate change in 50 years or 100 years.

But remember in these analyses, we do the intruder analysis where the intruder directly disrupts the waste, drills a well into it, puts a house above it, drills a well right beside it, something that puts them very close to the waste. Climate change, say human induced climate change is an indirect impact on the system from human actions. Intrusion is a direct impact on the system from human actions. I would imagine you could probably do the calculations to demonstrate that the -- in many and almost all circumstances, the direct intrusion is going to bound the impact from the indirect process. I can't say that definitively so, but that's my opinion.

Nearby contamination we heard about from Anna. We don't evaluate the impacts of a nearby contamination, although it can be very important and high from a risk perspective. We believe our language in the NDAA gives us an interpretation that we're supposed to focus on what's contained, not what the past releases are. The past releases are covered by other regulatory agencies and other processes. So if we were covering it, we're just duplicating that effort of how it's managed. What we do consider, though, is that the nearby contamination gives two pieces of information that we consider. It gives how is the -- how are the releases from our system likely to be transported in the environment, so that's an important piece of information.

And then what was that other one? Sorry, I lost my train of thought. I don't remember, I'm sorry, I'll think of it. The nearby contamination -- oh, I think it provides a decent analogue for how the system is going to behave. So a stakeholder might not like the fact that there's existing contamination but from a performance assessment perspective, it's good, you know. Yeah, if you look at the strontium-90 plume at West Valley, it's a big issue for the public and the management of it, et cetera. It gives you a great piece of information for how you expect the contaminants to move when they are eventually released from the high level waste tank. So in our analysis at West Valley, we made a GIS model and a 3D representation of the contamination. We're able to look at that and see, okay, whether our performance assessment model prediction for transport of these various contaminants are close at all to what's been observed in the system. I think those were some of the main topical areas you had questions on.

We didn't attempt to answer them in our slides but we figured it would be much more beneficial to have an open discussion on the topics with you that we could cover them more effectively. I'll pass onto Christianne now.

DR. RIDGE: Good morning. Is this microphone working? Okay. Well, we had the opportunity to come talk to you in May and you might remember in May Dave regretted that we had left the slowest speaker till last, and unfortunately we mixed that up a little today and Dave was second and unfortunately, that leaves the driest and stuffiest speaker third in the batting order.

But in addition, it leaves, perhaps a somewhat complicated topic for third which is radio-nuclide removal, which I think is something that we're perhaps a little less comfortable with because the tie-in to being risk informed isn't quite as clear and direct. With the performance objectives, I think it's very easy for a lot of us to understand that we want to do a risk informed review and meeting the performance objectives is our measure of risk and it's very easy and straightforward to see how that happens. Now, I'm going to talk for the next few minutes and the next few slides about radio-nuclide removal and why we're looking at radio-nuclide removal and what we're looking at and if you remember in May, unfortunately, there were a lot of protobations (phonetic) on this topic. I'm going to be looking at removal for waste determinations that were submitted after removal was completed, removal for waste determinations where the removal was submitted and they're looking at plans for what we will be removing. For instance in the saltstone review, we looked at salt waste processing facility which is not going to be completed for some time and yet we were looking at the waste determination before that removal action was complete.

So there's looking at the removal before and after waste determinations are completed, there's the difference in the language which I'm going to talk about on the next slide, between looking at the maximum extent practical and the maximum extent technologically and economically practical. So we've left, perhaps, the most protobatical section for last, the one with the more different little details we have to look at, but I'm going to try to do this simply, so if you bear with me.

First, in May we talked about radio-nuclide inventories, the selection of highly radioactive radio-nuclides, the selection of radio-nuclide removal technologies and the practicality of additional removal subdivided into a couple of topics. Now, the first two, I think we covered and were somewhat straightforward on most of the questions that we received from the Committee related to the selection of radio-nuclide removal technologies and the practicality of additional removal, so in the next few slides I'm just going to focus on those last two bullets.

Now, before I get to the last two bullets, I did want to talk briefly about why we're looking at radio-nuclide removal to the maximum extent practical. As I said, we appreciate that the performance objectives really give you a straight line towards assessing risk and doing a risk informed review and so one might ask why the SRP spends so much time and goes into so much detail talking about how to assess whether radio-nuclides were removed to maximum extent practical. The simple answer, of course, is that it's a guide for NRC reviewers and we're required to look at removal to the maximum extent practical by the language of various requirements including the NDAA.

The more philosophical question, perhaps, is why this requirement is included in the National Defense Authorization Act for 2005, the NDAA and also included in DOE's Order 435.1, which may apply to Hanford and the West Valley Policy Statement. Both include this type of requirement that radio-nuclides be removed to the maximum extent that's either technologically and economically practical or the maximum extent practical. There might be subtle differences between the two, which I'll address in a moment, and I'm an engineer not a philosopher but my interpretation of this is that all three bodies wanted to encode the preference that this waste that we're deciding is not high level waste that we try to minimize the amount of waste that is dealt with during this process. So maybe you could safely dispose of a little bit more of this waste in the ground, in near surface disposal, but it seems that all three bodies wanted to encode this preference that we reduce the amount of waste that goes through this type of waste determination for whatever reasons and I'm not going to speculate about what Congress was thinking or the philosophical positions of DOE or NRC, but my interpretation as a reviewer is that the reason we do this part of the review is that Congress and DOE and NRC have come to the same conclusion, that we want to minimize the amount of waste that goes through this process of being declared not high level waste or waste incidental to reprocessing as sort of an independent requirement in addition to meeting the performance objectives.

So the first step in this process that we outlined was selection of technologies and the NRC reviewer's evaluation of the technologies that DOE decided to use to remove radio-nuclides and the process that DOE used to select those technologies. And as a first cut, one of the things that we look for is the range of technologies that were evaluated and we expect those to include at the very minimum, technologies that have been used at other DOE sites. And one might think that that's a bit circular, where evaluating whether or not DOE is doing what it is that DOE does and they set their own bar and I appreciate that that is a bit circular.

Nonetheless, through experience we have found that that is a good starting point because the sites are different and the same technologies that perhaps could be adapted with some effort to be used under different circumstances with a slightly different type of waste or slightly different type of tank, we would like to see that those communications throughout the DOE complex are made. And one might assume that they are made, but we have found through experience that that's a good place to start, to say, well, you know, at Hanford they seem to be able to do this, they seem to be able to use this technetium from the waste, they seem to be able to use this type of technology. Could that he adapted for Idaho? Could that be adapted for Savannah River? Are there technologies that could be adapted for used under slightly different circumstances? So that's a first step.

As a second step, the SRP informs the reviewer that they would expect that the selection process that DOE would go through, might include some of the following topics; the expected effectiveness of the technology, the technological maturity of various technologies, schedule impacts that might occur from using different technologies, implementation costs, worker safety impact, systemwide effects of various technologies. Now, a couple of these terms might require a bit of additional explanation. One of them that, I think, cause some questions was technological maturity and the advice in the SRP perhaps isn't precise enough in saying exactly what level of technological maturity is required, but I think there's a reason for that, which in part, is due to the sort of complications that I alluded to earlier that a waste determination can be submitted after removal is considered complete by DOE, before removal is complete or even well before removal is considered to be complete, for instance, in the case of the salt waste processing facility at DOE which now is not expected to go online, my understanding is, until 2011.

So there is some time before some of these technologies will be implemented. And the degree of technological maturity, I'm not sure we could really draw a line that says if it's in development at a DOE site, then that's enough and you have to consider it or if it's actively being used at a different site, you have to consider it. Or if it's being actively used at your site, you have to consider it. I think that that comes down to a matter of judgment, in part because you would require a different level of maturity if the technology were going to be implemented within three months or if the technology were going to be implemented in 2011.

The degree of things you might consider depends, we feel, in part on what the other constraints are, when does this need to be used, when do you need to start building it, when do you need to start putting it in your budget? When do you need to put down the Erlenmeyer flask and the pipette and get out of the laboratory and into engineering, different levels of maturity might be applicable or reasonable in different situations. So I think that's in part the reason that the SRP left some flexibility in that region and maybe we do need to put a finer point on that in the SRP.

And then with respect to systemwide effect, I think some of these others are obvious, implementation costs, worker safety impact. With respect to systemwide effects, I think we were speaking there about effects that trickle down into downstream processes, so real physical chemical effects. For instance, you might not want to use oxalic acid, even though it cleans your tank out very well, if it causes downstream problems in another chemical system, if it means that the glass that you're eventually vitrifying does not turn out as well, so those kinds of downstream effects is really what we meant by systemwide effects.

The next topic we got several questions about was why we meant by looking at radio-nuclide criteria. Essentially, what we would be looking at is how DOE decided or will decide that they will stop removal activities. So I mean, once again, this is the real bug-a-boo of this kind of an analysis is that you're looking at either things that have taken place in the past or things that will take place in the future and the language is a little different, but essentially in meaning, the review criteria is the same. You want to know why did -- or will DOE stop removing radio-nuclides from a system. And so if you're looking at a system where you are yet to perform the removal activities, DOE may establish various radio-nuclide criteria for deciding when they're complete.

For instance, DOE might say, "We will stop when we reach this volumetric goal, when there are 200 gallons left in the tank, we're done". They might say, "When we've achieved a specified removal efficiency. So, for instance, if you have a chemical treatment process and you think it can achieve 80 percent removal of the cesium or technetium or whatever radio-nuclide in your system, DOE might say, "We're going to stop this chemical process when we have removed 80 percent because that is what we have decided is practical." And similarly, you might clean until you say, "We're going to pump on this pump until the pumping rate has declined to a gallon per minute, that's all we can do. Anything after that is not practical, we're not achieving much". And so any one of these types of criteria or different types of criteria, these were examples that we used, any one of these types of criteria might be a good reason for DOE to say, "When we get to this goal, we're done".

Now, in that case, we don't know if that has happened yet, but what the reviewer would look at, would be, "Well, they say they're going to stop when they've gotten out 80 percent of the cesium. Is that the best they can do? Are they doing better at other sites? Do we think there are other technologies that could do better?" Similarly, if you were going to say we are going to stop when we meet a volumetric goal, the NRC reviewer would look at, "Well, is that a fair goal, does that mean that they really did try to remove it and anything after that, yes, we agree getting down below 200 gallons, that would be impractical".

And so for waste removal activities that haven't stopped yet, that would be the type of thought process that a reviewer would go through. Now, those goals might not always be met. And they might be met. I should actually interject here, it's not as simple as a distinction between the top bullet is for future reviews and the bottom bullet is for waste removal activities that have taken place, because maybe you get a waste determination where the removal activities have taken place and the answer is we established this volumetric goal, we met this volumetric goal and we're done. So it's not quite as simple as a distinction between future and past but that's an easy way to think of it. But, of course, one reason you might have stopped is that you met your goals.

Now, you might stop for other reasons. You had a volumetric goal but then you worked at it and you were supposed to get down to 200 gallons. You got down to 300 gallons and your pump broke, and then you have to go through a process of deciding, well, is it worth taking out this pump and the worker dose that that would cause and the cost that that would cause and the delay that that would cause to remove that extra 100 gallons to get down to our goal? Well, maybe it is and maybe it isn't and you would need to evaluate that and the NRC reviewer would similarly want to understand DOE's thought process, DOE's evaluation to go through that decision and decide whether or not it's worth going on at that point.

So I may have over-emphasized this point too much but those are the types of decisions and essentially whether or not you call it the basis for a decision you have made or the criteria you're going to use to decide, it's the same thing. It's deciding -- it's evaluating the basis for the decision to stop removal.

Now, of course, another aspect of the same problem is that you look directly at would it be practical to perform additional removal. So you've stopped or you've decided when you will stop and then you also look at the flip side of that coin which is to decide is it practical to do more. So there are -- again, we list some reasons in the SRP that you might decide it's not practical to do more. There might be minimal expected benefits of doing more. The dose that you predict might be quite low and you can say, "Do you know what, it's not practical to do more because we just have nothing to gain". The economic cost in balance with those doses might be quite high. There might be programmatic and schedule impacts of additional removal. Again, there might be system impacts which I talked about a little earlier with respect to downstream processes.

Now, I think that the third bullet there, the programmatic impacts might require a little bit of additional clarification because that's a somewhat flexible and open-ended notion of what are these programmatic impacts. I think one example might be for instance, in the saltstone review that we did for Savannah River, one of the arguments that DOE made for why the schedule was so important was that any delays in treating salt waste would have an impact on the vitrification facility and would limit how much waste could be sent to the defense waste processing facility, the vitrification facility.

We don't know right now what all the programmatic impacts could be. That's one example but the reason that the SRP left flexibility in this area is that we recognize that we can't anticipate what all the mission impacts are going to be from DOE. We're not DOE. So we can't anticipate all those arguments but we did want to leave flexibility in that area, especially for analyses that are done under the NDAA. And I mentioned earlier that essentially we believe maximum extent practical and maximum extent technologically and economically practical to get to essentially the same point.

But if there is a subtle difference, it's that we might give more weight, perhaps, to these programmatic impacts under the NDAA because the language is more broad. It just says that we have to evaluate removal to the maximum extent practical, and practical encompasses a great many things. And so as one example that comes to mind is a mission impact such as limiting what can be vitrified in the vitrification facility. There could be others and that's part of the reason that the SRP left some flexibility in this area.

But now, again, since we are engineers and not philosophers, we did express the preference that to the extent possible, costs and benefits be quantified in terms of economic costs and expected doses because we understand those and their numbers. That's our preference but, again, there is flexibility left open for these other areas. So once you get into cost benefit analysis, the first question, of course that comes to mind is what is your metric? And we discussed this a great deal internally and whether or not we wanted to put into the SRP a number, this number of dollars for this dose that's averted. And we did not do that. Instead we recommended in the SRP that the costs and benefits be compared to costs and benefits of similar DOE activities, essentially recognizing that there are different -- there are reasons that activities performed by DOE are different than the type of activities that are performed, for instance, by our decommissioning licensees and we have guidance for ALARA analyses for licensees.

We recognize that for a variety of reasons, activities performed by DOE are different because they are part of the Federal Government, because they're a bigger organization than many of the licensees. There are various reasons, but essentially what we wanted to do going forward was to say, well, we assume that anything that DOE does someone at DOE believes to be practical. We are defining practical based on other DOE activities in the context of a site perhaps. In the context of similar environmental cleanup activities, what DOE guidelines does DOE use to say we are going to clean up this waste, we're not going to clean up this waste. And so the types of questions we're going to ask are the types of questions we've asked in the past, for instance, if you spent $600.00 -- and I'm, of course, making these numbers up, $600.00 a gallon to remove waste from Tank XYZ, why did you say it wasn't practical to remove the same number of gallons at $200.00 a gallon from Tank ABC? There might be good reasons for that but we would ask the question.

We would ask the question and expect that there would be a technical reason for the answer. And so that's the guidance that we settled on. We did discuss other NRC guidance, for instance, the guidance that's used in regulatory analyses or the guidance that's used for ALARA analyses for license termination under the LTR. And I don't need to go into it now, we discussed why we thought some of those might not be applicable to this particular situation. So that's how we addressed cost benefit analyses.

Now, of course, half of that equation is cost and half of that equation is the benefit and the Committee raised some very good questions about how do you assess the benefit when the analysis for the performance assessment might be quite conservative? And essentially, if DOE gives us a bounding analysis, and they say, "Well, this tank, do you know what, it's coming in, it couldn't possibly be greater than 15 millirem per year. We've met the performance objectives," if we agree that that's bounding, you come in at 15, you're done, it saves them time, you know, saves us time. You're done. That is problematic when you put that in the context of a cost benefit analysis because now you're chasing these 5 millirem that probably most people involved agree aren't there because maybe it's only a millirem, maybe it's a half millirem. We certainly appreciate that point.

The SRP emphasizes that uncertainties in the dose estimate will propagate into cost benefit analyses, so if you don't know if your dose is 10 or 50 or .1 millirem, the SRP does emphasize to the reviewer that those uncertainties are half of your cost benefit analysis and they're going to have an impact and the reviewer does need to be aware of that. And we do recognize this issue and it's a difficult one. And what I would say, the explanation I can give is that when we are evaluating a performance assessment, we certainly through independent analysis that Dr. Esh talked about and through just reviewing the analysis, try to assess the degree of conservatism of the performance assessment. So we do try to have some understanding, is this 15 the best estimate, is this 15 very conservative, and as Dave pointed out, that in itself is not simple but it is what we are trying to do.

And so we do recognize the issue and attempt to assess the degree of conservatism and indeed, DOE is free to and they certainly do point out to us any time they think an assumption that they're making is conservative. I think that those -- we can be confident that those areas will always be highlighted in the performance assessments we receive to make sure we understand and we investigate those and we decide if we agree, but certainly we do try to be aware of those areas.

We also received another question about

worker dose estimates and worker dose estimates are expected to be based on exposures from similar activities because they have been in the past in reviews that we've gotten. We don't require that and I think that the question probably was trying to get to the difference between a worker dose estimate, which probably is based on a best estimate based on similar activities that have taken place and DOE has experience taking pumps out of these tanks. They have a good idea of what the worker dose might be and so I think that probably what the question was getting at was this broader issue I just discussed of comparing a best estimate of a worker dose to a conservative estimate from a performance assessment perspective and I don't think I need to revisit that. I think I've probably went on about that a bit too long, but we are aware that one of those is a best estimate and one of those might be conservative and we do try to understand that in the comparison.

And so with that, I will turn things back over to Anna who will finish up a few last slides.

MS. BRADFORD: Right. I have just a few odds and ends types of things that came up in the questions and comments that we got from the Committee and one was on existing guidance. And I wanted to point out that the SRP uses existing guidance where applicable. We've looked a lot at NUREG 1573, which is performance assessment for low level waste disposal facilities, as well as NUREG 1757, which is the consolidated decommissioning guidance. But we didn't just cut and paste from these documents.

We really made sure we went and looked at the information we were using and tailored it to make sure it was applicable to waste determination reviews. And also because each of the sets of incidental waste criteria, be it NDAA, DOE, Order 435 or the West Valley Policy Statement, they all specifically cite 10 CFR 61, not the LTR or any other kind of requirement. And so, therefore, we thought using the guidance for 10 CFR 61 was the most appropriate approach in the SRP.

And for worker dose, 10 CFR 61 references for the most part 10 CR 20 and so the SRP lists those sections of CFR 20 that are applicable. We have ignored things like administrative things or enforcement because obviously, those aren't applicable to DOE but it lists the sections of Part 20 that should be considered and for the most part, DOE's own regulations in 10 CFR 835 are the same or in some cases a little bit more stringent than ours in Part 20 and so in their waste determinations, DOE typically provides a crosswalk between their requirements in 835 and our requirements in Part 20 to show that by meeting 835, they meet Part 20 and Part 61. And we don't plan, in the SRP to provide one of those generic type of crosswalks.

And then I wanted for a minute to just go over a few terms that there seemed to be some questions about that we used in the SRP. Reasonable assurance, Dave talked about that for a moment already. This is the same reasonable assurance that we use in all of NRC's or many of NRC's regulatory activities. It's the same here when we're looking t waste determinations. The comparable to, a few sets of the waste criteria will have a statement. For example, DOE Order 435 will say they should use 10 CFR 61 Subpart C or comparable safety requirements and the question was, what does comparable mean, and we would say that comparable means either the same or more stringent than the requirements of Part 61.

And as the SRP states, DOE has never in any of their waste determinations, tried to use some other set of criteria that are comparable to. They've always just gone ahead and used Part 61. The other phrase is "other characteristics", and this comes out of the first requirement of the NDAA, which is that it simply says the waste does not require disposal in geologic repository. And we feel that you show you meet this by meeting the other two criteria, which is you meet the performance objectives and you remove waste to the maximum extent practical. But we wanted to have some flexibility there. Maybe there's going to be some other characteristic of a waste stream that we haven't seen before that will come up in the future that would make you stop and think maybe this does require geologic disposal even though it meets these other requirements, for example, on non-proliferation concerns or some other -- something else. We just wanted to leave that flexibility there and not close the door on that. That's the reason for that phrase.

And then also the draft SRP was issued for interim use and comment. That interim use is just supposed to give the idea that we can go ahead and start using it immediately. Our reviewers can use the information in there on their waste determination reviews we have already ongoing and DOE can look at it to get an idea of what types of things they might want to include in future waste determinations that they plan to submit to us.

I wanted to talk again about monitoring for a minute. This is the last area addressed in the SRP is our monitoring of disposal actions under the NDAA and our monitoring will be risk-informed and performance based as the SRP says. We really plan to focus on the things that could effect the results. And we believe, as the SRP says, that non-compliance will be when there is no longer reasonable assurance that performance objectives can be met. And this might be the result of either a measured parameter or projected analyses such as a PA result.

And we intend to, as we do in our waste determination reviews, rely on DOE's PA as updated and revised. We would maybe look at how it's updated or revised or maybe perform our own confirmatory modeling to come to any conclusions about whether there's an non-compliance. And of course, we'd pay special attention to any parameters that are highly risk significant. And the scope of the monitoring plans may vary. We're really at the early stages of the monitoring. We haven't started monitoring anything yet in particular. So I think as we're going along, the scopes of those plans may change.

For example, right now, we're reviewing a waste determination for two tanks at Savannah River, and it would make sense to me if the first monitoring plan was for those first two tanks because that's what we've completed so far, but as we complete more reviews and as our monitoring activities are encompassing more tanks, it might make sense to consolidate a monitoring plan. Maybe eventually, it would be a plan for all of a tank farm but we're not there yet.

And I just want to repeat again that we do not have any authority with DOE with respect to monitoring. So we can't require them to monitor a particular aspect of their activities, but they do have their own internal requirements for monitoring and any documents and things like that are things we would expect to look at. I just wanted to, in conclusion, point out that the draft SRP is based on existing NRC guidance, like I mentioned, as well as staff experience. We've completed five incidental waste reviews and we certainly applied that experience when we were developing that SRP and I think we've found that it greatly informed what we thought should be in the SRP. Having had that experience of going through reviews, it really helped you understand what should be included in the SRP for future reviews.

Also the draft SRP is flexible and applicable to the many different types of waste determinations we may see in the future, while still providing the main purpose, which is the consistency for reviewers and for people to understand what it is that the NRC will be looking at. And with that, I hope what you heard today will help answer any questions you have and we look forward to receiving any comments you might have.

VICE-CHAIRMAN CROFF: Okay, thank you. Questions from the Committee, Jim?

MEMBER CLARKE: What I'd like to do Allen, is I'd like to make a comment, and then I'd like to ask Dr. Esh to comment on my comment. But I'd start out by saying I thought your comments concerning how the NRC will review the performance assessment especially with respect to the very difficult issues around long-term performance, I thought that they reflected a very thoughtful analysis and you don't have to comment on that, unless you disagree with it.

The observation seeing the barriers performance is limited to the experience is clearly overly conservative. What we've seen, if barriers are going to fail, they usually fail pretty quickly because they're not constructed properly or they were a bad design. However, saying that they will perform well into the future, and I don't know what that means, going back to experience, but to say that they will perform well, into the future, is probably overly optimistic unless we're prepared to intervene in a way that keeps them performing.

And the other thing is I think -- I can't recall how you did this but I think the way we define failure is important and I would define it as whether it's engineered barriers or institutional controls, is this loss of control. In other words, the barrier that failed to meet the design objectives or the institutional controls failed to perform, and I would add a caveat, with or without consequences, because I think if you try to wrap consequences into failure, just they are waste specific and site specific and many other factors reflect on that.

So I would come back to I think the importance of intervention in the long term if you really need a barrier to perform over a long term, in monitoring this, I think you have to be prepared to intervene. And so I would think that the way you propose to look at that or the way you propose to review how the applicant plans to deal with that would be important. That's my comment. I just throw it back to you.

DR. ESH: Well, I would agree with your wholeheartedly on your first part about experience base and going beyond experience based. I think certainly you can make arguments for going beyond experience based and of course, the $64,000.00 question is how far beyond that or maybe for some barriers it's a $64 million question, but I think it's -- the analysis approach has to consider a variety of things. It has to consider what you know now, the system that barrier is operating in, what's the processes mechanisms and how dynamic is that system and there are certainly some things that are going to be more controllable than others.

And the example I gave with respect to the burial mounds, the American Indian burial mounds is they've -- a number of them have survived for a long period of time from a stability standpoint. So the material is still where it was originally and it's still relatively intact. If that barrier was also trying to limit water flow through it, that functionality may have been lost much earlier than the stability functionality and also your type of design can be very important, too. So let's take the infiltration example.

And you have a source of something very short-lived, you may be able to put a geomembrane down which can be very impermeable if installed properly under the quality assurance procedures, very effective for a short period of time, essentially limit infiltration to nothing for 30 years, 40 years, 50 years, whatever the case may be. Of course, you wouldn't want to put a geomembrane down if you're worried about trying to limit infiltration 1,000 years out. Almost categorically, it's not going to last that long.

But another type of design, if your goal is to limit infiltration 1,000 years down the road, might be something like the water balance type covers that people have been investigating that try to mimic the natural system and I think those could potentially be very effective especially at the semi-arid sites. At the humid sites, there's just too much water. Plants can't use it all --

MEMBER CLARKE: We are totally on the same page here. I think --

DR. ESH: Yeah, so I think like in the SRP we tried to provide enough guidance that will allow somebody to make a reasoned judgment as to the validity or at least the reasonableness of the projection of the barrier performance. And we advocate multiple lines of evidence to support them and certainly if you're going beyond the experience base and you're going a lot beyond the experience base, then the amount of information you need to support that projection is much more comprehensive and stringent. You need a lot more support to justify that you're going to be able to achieve that objective.

Monitoring and maintenance definitely serves a role in barrier performance but also remember in our regulatory construct for disposal, we don't take the EPA approach. If you have monitoring and maintenance and it continues for a long period of time, great. But --

MEMBER CLARKE: I understand.

DR. ESH: But ultimately, you're trying to make a decision now and you're investing the cost to make a decision now, instead of continually deferring your decision and not making it based on new information. You may also add that in which will help insure that you don't have some problem down the line, but ultimately our process is trying to make a good decision now.

MEMBER CLARKE: I understand, David, but

all I'm pointing out is that if something happens, natural processes work against what we're trying to do, whether they be earthquakes or erosion or environment intrusion or whatever, I would submit that it would be important that the applicant has sort of that, they're telling them what they plan to do if that happens.

DR. ESH: Well, our analysis approach is you need to consider -- I mean, people like to look locally and I even fall victim to that. I'll give you an example. When I drive to work, I go over a railroad track that has no bars that come down, it just has lights. And I would just speed right over it. I think, you know, I've been driving this route for six years now. How many times have I encountered a train? What's my risk of needing to slow down at this railroad crossing?

MEMBER CLARKE: This does not come as a surprise to us, David.

DR. ESH: Well, anyway, so one day I'm driving and I'm approaching the railroad tracks and the lights are on and a train's gone through. And I'm like, you know, that's different. And the next day, I'm driving through and a train is going through again, at the same time. The same thing the next day. What happened is the Baltimore tunnel fire resulted in a rerouting of the train system that was sending more trains on the track that I crossed. It changed the system. It was a very complicated system and I was looking locally. But whenever you analyze these barriers or project performance, you have to think out of the box which engineers aren't usually good at and scientists are too good at. But you have to be somewhere in between, I think.

MEMBER CLARKE: Well, said, thank you.

MEMBER WEINER: First of all, I'd like to thank all three speakers for really clarifying this whole issue. I thought all three of you did a tremendous job. And Dave, I especially want to commend you for your discussion of deterministic versus probabilistic and conservative versus non-conservative. This is a very real problem because we tend to say, "Oh, my goodness, it's too conservative, it's not realistic, why are we doing this", but you have clarified the NRC take on this and that was really good.

I have questions for all of you. Your statement about potable water, David, does that apply across NRC regs? In other words, if you don't have potable water, you don't worry about anybody drinking it?

DR. ESH: Well, I can think of a decommissioning example. In Tennessee, I think it was maybe Kerr McGee (phonetic) where that was part of the argument for the dose assessment is that water was not likely to be potable. The states may have their own regulations and certainly EPA, they protect groundwater, I think, regardless of the potability. But then also in the recent EIS process for the uranium enrichment facility in New Mexico, I think, part of the argument for that is that the groundwater is likely not to be potable.

So --

MEMBER WEINER: Very likely not to be potable. It's very saline.

DR. ESH: Yeah, so I mean, it's not unique to our problem but -- and it's kind of a common sense thing. When we say risk informed, that applies across the board, so it applies to scenarios and parameters and models and all sorts of things, and this would be a scenario type thing.

MEMBER WEINER: Christianne, you talked a lot about doses and removal of radio-nuclides. To what extent do you use the concept of collective dose in making your regulatory decisions?

DR. RIDGE: Well, I think in the SRP what we outline is that we address the collective dose because it is what is used in ALARA analysis and basically the discussion in the SRP outlines some problems that would occur if that were to be used in a -- in this type of analysis. So to answer your question simply, so far we haven't. We do not expect to and the SRP discusses it basically in the context of reasons that it would not be applicable to this type of analysis.

MEMBER WEINER: That's very helpful. Do you -- in looking at these determinations, do you ever balance off work -- you must balance off worker dose against public dose or against dose to a potential intruder? Is that some kind of tradeoff that you do?

DR. RIDGE: Certainly worker dose is a very important consideration. And we fully expect and have in the past considered the impacts on worker dose. Now, in the SRP we do say that we think that a ratio of worker dose to public dose is very problematic and that worker dose is an accepted risk and public dose is not an accepted risk. And it makes us very uncomfortable with simply presenting a ratio; this much worker risk can be traded off against this much public risk. To our minds, they're very different things.

And so we certainly always consider worker dose and it's a very important consideration in the analysis but yet, we are uncomfortable and the SRP provides a bit of discussion on this topic. We are uncomfortable with the simple mathematical ratio of the two.

MEMBER WEINER: Well, I can understand that. Are you considering any discussion -- and I'm not -- I haven't read your guidance that well, I'll be perfectly frank about that, but are you considering some extended discussion of that dichotomy that you run into that you can decrease the public dose by increasing the worker dose or vice versa but worker dose is a -- the workers know what -- know that they're taking a risk. Is there a discussion of that?

DR. RIDGE: The discussion of the difference between the -- the discussion that I just provided basically, that one is an accepted risk and one isn't and that makes us uncomfortable with the simple mathematic ratio, that discussion is in the SRP.

MEMBER WEINER: Yeah.

DR. RIDGE: I don't think that we explicitly say that we would expect that worker dose would increase if public dose decreases. I'm not sure that that always would be true and so we don't say that in the SRP but we do discourage presentation of this simple tradeoff. There's a point at which this number of millirems to worker equals this number of millirems for public. We don't feel very comfortable with that.

DR. ESH: Remember the worker doses also have a much higher limit. So like of you look at the past experience for a worker dose, it's based on somebody trying to achieve that worker limit so the result is necessarily going to be probably much higher than what you're trying to achieve for the public dose and the things that you can do to control the worker dose in many cases are pretty straightforward. You put in more shielding or you put in more protective coverings and procedures, et cetera to minimize the worker doses. You could probably take the worker doses much lower than what they are, but why do you need to if you're meeting your limits.

So then if you take those numbers and try to compare them to the public numbers, it gets really sticky.

MEMBER WEINER: Yeah, I understand that. I just wanted to expand on the discussion. And I wanted to compliment you on your statement about reasonable assurance. That's always a problem and I really don't have any questions about it. So I just wanted to thank all three of you.

CHAIRMAN RYAN: I apologize for being late. I had a mission -- a meeting with Commissioner Yatsco (phonetic). He's the boss. I guess I compliment you on not using collective dose. In most examples it's silly, except for that relative evaluation for ALARA, do I do it by process A or B, and there is a metric that's very helpful in the work circumstance. I guess I'd challenge you to think about the fact that public dose in its broadest sense is accepted. People get medical exposure. We accept background. We accept radon up to certain levels and all of that so it is accepted.

It's not accepted, not by everybody, but I think it's a little risky to say you're comparing an accepted risk to an unaccepted risk. That's way too broad to be right over all schemes. So I would get you back to where you were a few minutes ago which is let's evaluate it in the context of the determination you're making whether it's a worker or a member of the public based on the system, the scheme and the process but I would be careful that language doesn't take you to that more philosophical place rather than the analytical place which is where you need to be.

DR. RIDGE: We always want to avoid the philosophical place.

CHAIRMAN RYAN: Yeah.

DR. RIDGE: And I understand your point, but I do need to comment that in the case of a medical exposure, there is some benefit that the public is expecting from receiving that dose and I --

CHAIRMAN RYAN: Radon, people accept radon all the time at much higher levels than they do from other things. I know it's voluntary, involuntary.

DR. RIDGE: Yeah, there's the whole voluntary/involuntary question and we probably don't need to get into that but it does need to be brought up.

CHAIRMAN RYAN: The comment is avoid it all. Stick to your knitting and I think you can avoid what would really be a complicated sorting out process. You might want to look at that language again and just touch on it.

And again, I apologize for coming in a

little bit late, so I missed some of the important conversation you had earlier on, so I'll just stop there and not continue, thanks.

MEMBER HINZE: Christianne, I'm a great believer in cost benefit analysis. That has great attributes. It also has problems and I'm sure you're well aware of them. And one of them is the problems that come from comparing apples and oranges and I'm wondering, you've also discussed or at least mentioned the uncertainty propagation that goes into the benefits, perhaps not the cost.

But I wonder if the important thing to emphasize here and maybe you have, is that once you compare technologies and removal limits, et cetera, within a site or within a problem rather than comparing that with other sites because as one compares the cost benefit from a site to another site, you're moving into another whole realm of uncertainty space and I think that the emphasis here should be on the comparison among the technologies, et cetera, within a site rather than between sites, if you will.

DR. RIDGE: I think that that -- actually, I think that we are already in agreement in that the SRP does indicate that we would expect that the best comparison would be to similar activities and one of the similarities we noted was activities at the same site. And so we did mention other environmental cleanup activities which conceivably could bridge sites, but we do actually mention in the SRP, I think in a couple of places, that when making this comparison, we want to look at similar activities and that one of those similarities that should be given weight is activities at the same site.

MEMBER HINZE: Yeah, I think your uncertainties are going to be common --

DR. RIDGE: Right.

MEMBER HINZE: -- within the site. Dave, in your presentation, I understand why we need or should provide flexibility in analysis procedures and deterministic versus probabilistic. I'm just wondering what kind of guidance that is in the document to make certain that people use the correct form of analysis. There are times when deterministic analysis is not a very good approach, as you are well aware and how are -- how is that guidance and assurance that we're really headed in the right direction both DOE and your own review?

DR. ESH: Yeah, I don't know if I can assure we're headed in the right direction but in the

SRP what we attempted to do was clearly indicate our preference and list the problems associated with certain approaches. The deterministic analysis can be very problematic in a situation where you have a complicated problem that you don't know much about and you have a lot of uncertainty because what ends up happening is you try to manage that uncertainty in each part of your calculation by being pessimistic or what people say is conservative and when you add that all up, the whole calculation can get pretty pessimistic.

MEMBER HINZE: Pretty mean.

DR. ESH: Yeah. If that approach, though, that very pessimistic calculation gives you a result which achieves that you're trying to achieve, shows compliance with your limits, then as a regulator, I don't have a problem with it. I can be pretty confident and argue that this is a correct decision action and that people are going to be safe. As a scientist, I don't like it at all because I'd like to know what the answer is, where is reality but in order to get to reality, you have to invest in the understanding which costs money.

People -- if there's a reason why people want to get to that understanding, they'll invest the money in it but usually the only reason they would want to know the truth is if it can save them a lot of money. So it's kind of a tradeoff. Our approach is generally, we start with a probabilistic analysis where we really liberally apply uncertainties and try to see exactly what can drive things in the problem and then we'll refine it and add in more complexity in the areas that we see driving it as needed and we might come to an understanding that well, the risks aren't as high as we thought. It was driven by our simplistic representation of process A.

But that process, I think, is iterative and also all we can do is indicate the disadvantages of certain approaches but we can't say you have to use a certain analysis technique. For all -- you know, somebody could -- they don't even have to use a performance assessment to do one of these things. They could do a hand calculation if they could demonstrate it. There's no impetus that they have to do something complicated but by the very nature, the activities associated with them and the projections, they are fairly complicated and that kind of drives towards the more complicated techniques, which I think you can get more out of.

Maybe we're kidding ourselves and you

aren't learning anything more by the complicated and probabilistic uncertainty analysis than you are with a deterministic but I tend to think we are because I think it really helps focus. When we're faced with a stack of documents this big, we want to know you know, I have 100 hours to look at it, can I put 90 of my hours on these two and 10 of them on the rest?

MEMBER HINZE: You also have the opportunity to go back to DOE and request additional information. Now, how binding is that or is that just a request but they need not comply with it? You need to have some of these iterative get-togethers.

DR. ESH: It certainly isn't binding. We can make the request and they can supply the information if they want to. Generally, they're very accommodating and if they have it, they'll supply the information. But there's no requirement that they have to. But then lacking the information, we have to make a decision. So if it's an important piece of information and we don't get it, then we're probably more likely to make an unfavorable decision because we don't have the information that we think is important to the decision.

MEMBER HINZE: You have to build in greater uncertainties.

DR. ESH: Yes, yeah.

MEMBER HINZE: Okay, thank you.

CHAIRMAN RYAN: I think the one thing that's really different for me and I think I heard, Christianne, you mention it a little bit, is that if you do the deterministic versus any kind of a either sensitivity study or probabilistic approach, you really end up missing what I think is your important point, is what's driving the system. One of the real key things that make the dose that I'm interested in go up or down. So, you know, I think that to me is one of the key elements is you really need to understand, do I need to spend more time on you know, sequestering radio-nuclides in a matrix, do I need to spend more time in water management? You know, where do I need to spend my time and my money?

So a little investment in studying the system might pay off and, you know, in what you actually have to do to manage the system. So to me that's a real focus and I believe that's reflected property in the guidance what you said today.

DR. ESH: Yeah, if I was on the other side of the fence and I was trying to solve or justify one of these problems, I would very much make a strong case that a small investment in understanding can probably pay off big in cost in terms of reducing the design or reducing the amount of waste you have to remove or all those things that are very expensive to do on these problems. So my opinion, though.

VICE-CHAIRMAN CROFF: I'll offer a few comments, I guess, and you know, whenever you want to respond, go ahead. First, concerning the use of water, it came to my attention, I think this is correct, is there is not necessarily one measure or whether water is potable. In other words, different agencies have different lists of you know, how much salt or whatever has to be in it to make it not drinkable water. And in some cases, I think some of these groundwaters can be close. And what I'm saying is, under one list it's potable, under another list, it's not.

And I think a suggestion there is be more specific on how potability is measured. In other words, if you have an official list or however it's done, I think that would be a good thing to do. I'm always sensitive to, you know, proposals, sort of trying to gain the system a little bit, if you will, and that's where I'm coming from. Nearby contamination with the LTN, I think we're sort of stuck with, you know, even if a tank has a residual

100 curies and there's 10,000 curies around it, well, the 100 still adds something whether -- by policy, whether we like it or not. So that's there.

Where I think nearby contamination is going to drive you nuts is in monitoring. If there's a -- whether it be leaks from tanks or other disposal sites nearby, if there's a comparable or a lot more radioactivity in it, you know, you're going to have a lot of trouble in monitoring, trying to figure out what is doing what, sort of unraveling the problem, if you will. And that's where I think it's really going to come to the forefront and be important.

DR. ESH: And that was my second point that resulted in the longest pause in ACNW briefing history, which was the impact of the nearby contamination on your ability to monitor. We would expect on the monitoring --

VICE-CHAIRMAN CROFF: I must have had a senior moment. Okay.

DR. ESH: Yes, I'm not that senior, but I guess it's maybe my young children that are causing this. In the monitoring, we would expect that they recognize that influence of their ability to see what's happening with their system from this nearby contamination. And we understand it could be a problem. On the other hand, we think that the monitoring should be much more focused on what Tim Nicholson from Research would tell you about are performance indicator type things rather than environmental monitoring.

The time that you're seeing the problem with the environmental monitoring, you've already created a significant problem that might be hard to remedy. If you use these performance indicator, such as the moisture content in the cap above the facility or something like that, you stand a higher likelihood of being able to take an action and a less costly action to remedy the situation. So that -- I agree with you, yes, it is an influence and we expect it to be considered in the monitoring.

VICE-CHAIRMAN CROFF: On the issue of conservatism, you correctly pointed out that you can use a conservative and deterministic analysis to show compliance has been done for years. I mean, there's no question about it. I begin to have concerns about it when it's used in the cost benefit situation. You know, your analogy with the bridge, I'm not sure that analogy flies with me, because safety factors in bridges, I think, you know may be factors of a few at most and some of these conservatisms as you've mentioned, you know, DOE, I think keeps -- in many cases, just keeps piling them on because they know they can still meet whatever the limit is. And the conservatism factors there, I would hazard in many cases can be orders of magnitude.

And when you start factoring that in, you know, doing this cost benefit kind of thing, I mean, you know granted, you know, it gives you a conservative answer there, too, but at some point, you know, you're driving the system to remove more and more waste when they really don't need to and those resources can be better used elsewhere. And that's part of the risk informed business and it gives me some concern there.

Then when you go to the monitoring thing and you've got this conservative performance assessment, and you get some kind of a monitoring result and the two are just apples and oranges --

DR. ESH: Yeah, but --

VICE-CHAIRMAN CROFF: So let me stop there and let you respond to any of that.

DR. ESH: Yeah, I share -- I understand your concerns. As I said earlier, from the regulator's perspective, we're trying to insure safety. As a taxpayer, I don't want somebody spending inordinate amounts of money on something that I don't think is an issue. And -- but as a regulator, we're trying to insure safety and these problems, if you have a bunch of things that are all linked together and there's data uncertainty and model uncertainty and all sorts of different types of uncertainty, if you have limited information, you don't have a good handle on the total impact of your uncertainty. So if you're using something like a best estimate deterministic analysis, the likelihood that you're underestimating the impacts is much higher than if you're using a conservative analysis to manage your uncertainties.

If you're using the best estimate, you're basically ignoring the impact of your uncertainties on the decision, which in these problems as you stated, the impact of the uncertainty can be large. You know, on something like plutonium solubility, it changes six orders of magnitude as you go from ph 12 to ph 9 or 8 or something like that, roughly speaking. That difference in six orders of magnitude can be the difference between flying way under your compliance limit and being way over your compliance limit. And that range -- the range of ph values I cited are what you get in a cementitious material as you go from a fresh cement to a very aged cement. It changes over that sort of range.

So if you don't have the information to say at what rate do we expect this ph to change and how is it going to change over our analysis period, if you just stick with your fresh value, you may be making a very bad and unsafe decision. You can invest the resources into defending how that's going to change and constraining it, and then your compliance risk is much -- is probably much closer to the true risk. But the down side -- I mean, this is like -- this is very analogous to I think our legal system. You don't want to put an innocent man in jail. You err on the side of letting guilty people out.

This is the same situation. You don't want to not protect people; you want to err on the side of over-protecting them. If it gets to the point of being ridiculous, I mean, that's what you worry about but I don't think that's what's happening in these problems. It's a matter of what you know and what you don't know. And I think we work in it much more closely. We understand how far from reality we, meaning the technical analysts, believe the results probably are from what the compliance calculation is and in many cases, I don't think they're inordinately out of line. They may be couched as conservative, but I think we tend to over-estimate what we know and if you just look at examples of -- in many of these cap systems, these RICRA type caps that they put in all over, where they've got around to analyzing them in detail, they find many times that the resistive layer, the hydraulic conductivity of the resistive layer, shortly after putting the system in place, is always a magnitude higher than what they thought it would be. And it's because they didn't plan for the complexity especially of like a dessication process that causes cracking of it in the near surface.

I mean, it's like that type of thing that can change things a lot. You have to factor into the analysis. If you can't analyze it, you have to be conservative to insure protecting people. So I mean I --

VICE-CHAIRMAN CROFF: Let me get back in here a little bit. I understand but again, where I'm coming from is let's postulate. You know, you received a conservative analysis. It shows that you comply with whatever the limit is. I don't know, the limit is 25 and the conservative analysis says 10.

Okay, you've complied. So you've already assured safety here. I mean, you've determined compliance with a conservative analysis. Now, the issue is how much further, if any, do you go.

DR. FLANDERS: Can I insert just for a moment? I think, Allen, I think I understand your question. I think one of the things we -- my name is Scott Flanders, NRC staff. I understand your question but I think one of the things you need to keep in mind is the cost benefit analysis is one piece of the information that we use to assess whether or not you remove radio-nuclides to the extent practical. And if you end up in a situation where you've demonstrated compliance, then it puts a pretty high threshold on the need to further remove radio-nuclides. And that's part of the reason why we don't necessarily establish a fixed dollar, $2,000.00 per -- is because it's a piece of the information that we take into consideration in terms of making a decision whether or not we believe they removed to the extent practical.

The word "to the extent practical", allows you the flexibility to consider other things like cost, and consider other things like dose and the fact that you've met the performance objectives. So

I caution that I don't want the thought to be that the staff looks at the cost benefit analysis and if it shows that even if you've already satisfied the performance objectives, that you know, you need to spend millions of dollars to reduce the -- you know, remove a few more millirem when there's so much uncertainty in removing a few more millirem. It's part of the information that we consider in terms of looking at removing to the extent practical.

And we recognize, I think, the point that I think Dave and Christianne are making, we recognize and we understand what you're doing in deterministic analysis and the uncertainty and the conservatism that goes into that analysis, how that influences what you see in terms of your dose estimate and that's factored into looking at your cost benefit analysis and factor that into your decision making on whether or not you remove to the extent practical.

So I mean, I'm not sure -- I think your question goes to the cost benefit analysis being -- you know a way looked at in isolation in terms of other considerations in terms of remove to the extent practical.

DR. ESH: I mean, I would look at it this way; if you do a conservative analysis and that over-estimates your impacts, you don't know that it's conservative first of all. It's your professional guesstimate that it's an over-estimate but besides that, you generate a result that is higher than what you expect realty to be. Then you decide, okay, based on that, I need to spend X amount of money to reduce it. Well, if you had the information to reduce your estimate, get constraining information that allows you to not be so conservative, that allows you to not spend the money to remove the source. You can either spend your resources on developing the basis and constraint of your analysis, or you can spend your resources on removing the source, but either one are tied to what you know and what you don't know.

If you are using a best estimate and there's a lot of uncertainty, you're running the risk that you're doing something that's not protective, and I think in that situation you have to err on the side of being protective. That's -- the whole -- I mean, I don't want to get into it, but the whole -- the way that we manage radiological risk in all of our systems is set up that way.

VICE-CHAIRMAN CROFF: I agree up to a point. You know and it's a matter of degree, you know, and go back to the bridge analogy. You know, maybe the bridge has a safety factor of two or three, but performance assessment has a safety factor of 100 or I think we're getting into a different part of space.

DR. ESH: But if the performance assessment results can range from 10,000 times unacceptable to 10,000 times acceptable, you have to look at it on a normalized scale. If you're 100 times over on an eight order of magnitude scale, that's not so bad.

VICE-CHAIRMAN CROFF: I agree with you and that's the kind of information I'd like to see it based on. You know, you've got the top, you've got the bottom and something in the middle. That's the idea. I think we may be headed in that direction anyway. We were talking a little bit yesterday, the recent Hanford Performance Assessment that I just sort of skimmed through is a best estimate deterministic. And we'll see what they use it for, but it's for the single shell tanks, so we've got to figure sooner or later we may be seeing it.

Let me try to move onto some other things. On radio-nuclide removal, I guess my -- you know, my thinking is to focus on whether it's worthwhile to remove the next gallon of waste and not so much whether removal is complete, whatever that means. I'm not sure focusing on the completeness leads you to anything very useful and for some of these, I'm not sure that they're even useful measures or meaningful measures. So it seems to me --

DR. RIDGE: It might be more helpful if you could be more specific about which other measures aren't meaningful.

VICE-CHAIRMAN CROFF: Efficiency, because, I mean, I's assuming by efficiency, you know, it would be a number like 99 percent.

DR. RIDGE: I think I can speak to that for -- I mean, not specific, I understand you're making a broader point, but I can speak to that specific point for a moment, about efficiency and I think that it might be clarified by giving a couple examples of how we have used it.

One is in the salt waste determination for Savannah River. One of the things we were looking at was the expected radio-nuclide removal of the various processes that we're using, one was the interim processes versus the final salt waste processing facility. So the final salt waste processing facility was going to get out five percent of the technetium. So I was thinking of that as -- you know, perhaps we should have defined it a little more specifically, but that's a treatment efficiency, five percent of the technetium.

Now, we would want to compare that to other technologies that maybe could remove 20 percent of the technetium that went through the chemical treatment process. And maybe there are, maybe there aren't, technetium can be a very difficult thing to remove. Are there other technologies that are being used at other sites that have removed a greater or lesser fraction of the technetium? That would be one way that we'd use a treatment efficiency. Now, I think if I understand your question correctly, you were envisioning efficiency more in terms of volume and --

VICE-CHAIRMAN CROFF: No, not necessarily.

DR. RIDGE: Okay.

VICE-CHAIRMAN CROFF: Let me go to first your example of --

DR. RIDGE: I do think that that efficiency was useful to us in that context. I'm not sure I understand why it would be not useful.

VICE-CHAIRMAN CROFF: I agree that the efficiency as defined as something like a percentage, can be useful in comparing processes. That's a very common use. But in determining when radio-nuclide removal is completed or is gone far enough, the difficulty you, you know, run into is if you say, you know, we can say it removed you know, 90 percent from the material from a tank, well, if they started with 10,000 gallons at the bottom of a million gallon tank, that's probably pretty good. If the tank was nearly completely full, it's probably not so good.

And the problem is, you know, your starting point is variable. And so the efficiency ceases to have meaning. You know, what's really meaningful is how many curies do you leave in the tank and how many curies are in the saltstone? That's the parameter that's really important and sort of how you get there and all these other measures isn't so important.

DR. RIDGE: I completely agree with you about the arbitrariness of -- the potential arbitrariness of the starting point and I think that that's one of the reasons that in the SRP we did ask the reviewer to look -- to make sure they understood if any percentages are presented by DOE, which in the past they have been. DOE has given us numbers that indicate we've removed 99.9 percent of the radioactivity due to this radio-nuclide, 90 percent of the radioactivity due to this other radio-nuclide and DOE has presented those types of numbers in the past. And I think that this arbitrariness of the starting point is exactly why in the SRP we encourage the reviewer to make sure they understand what the starting point was for that number, so that the understand was this 99.9 percent based on the all time high volume in the waste, was it based on treatments after bulk removal.

And there is a certain degree of arbitrariness. I think that it's important that the reviewer understand the starting point and I think you make a very good point that the matric might be more useful to compare processes. And maybe we need to put a finer point on that but certainly we haven't said once they remove 99 percent, they're done.

VICE-CHAIRMAN CROFF: I understand. And all I'm saying is I'd expend your resources on the -- you know, what's left and what's going to be disposed on site not what's removed and they're going to go into a glass log. Let me move on to programmatic and scheduling packs and sort of elaborate a concern there.

And that is on the programmatic impacts, and you've cited the Savannah River tank capacity example, which is, I would say a classic case here, what I discovered through hard experience is the -- at the DOE sites, the waste management systems are incredibly intricate, complex and huge. And it's very difficult to validate a claim that there's a programmatic impact. You know, the Savannah River tank capacity thing, if you try to track it all down and figure out, is there really a tank capacity crisis or is there not, and try to track down all the technical things of what they might be able to do to free up tank space and then whether they're really practical or not, you get -- I mean, it's an incredible amount of work and I say that from personal real experience, and you know, very often you can't get to a definitive answer to figure out is this claim really valid or not. And that leaves you in a very difficult position, I think using programmatic things and schedules sort of -- it's very easy, you know, for a milestone to be created here.

I mean, milestones can be created and uncreated at will and provisions in compliance agreements for that matter. So what I'm saying there is, I mean, you know, there can be practical implications there but on the other hand, it -- you know, there's ways that can be used and I think in th

SRP cautions need to be in there about sort of, you know, how much weight can you give to these, and validation of them? That's the thought process there.

On the cost benefit thing, in metrics there, you know, Mike talked a little bit about collective dose and the limitations in that. And, of course, this Committee is on record in saying collective dose isn't such a good thing to use as it was done traditionally for this kind of thing which is, you know, the integral overall space of micro-doses is what I'm referring to.

But then that leaves the question okay, what kind of measures and metrics do you use? In some of the waste determinations I've seen DOE seems to approach it more on a you know, "Gee, the pumping efficiency went down a lot, we're not getting very much out and it will cost a lot more", kind of a thing. And then in the most recent Savannah River waste determination, there were these metrics like dollars for 50 years of dose averted to the public receptor and a similar thing for workers.

And first, I've never seen a metric like that before so it was sort of novel, and I'm not sure whether it has any real conceptual validity or not.

And secondly, even if it has conceptual validity, you know, there were numbers like, I'm remembering numbers like the magnitude of like $10 million per millirem averted, on that order, and I'm sort of, you know, asking myself is that too high or too low? I mean, what am I measuring it against. And --

DR. RIDGE: I think the answer we would provide, the answer that we tried to provide in the SRP and that I've apparently unsuccessfully tried to provide in my slides was that we would try to compare that to other similar activities that DOE is performing.

VICE-CHAIRMAN CROFF: Give me a couple of for instances on the similar activity.

DR. RIDGE: For instance, removal of similar waste from tanks at the same site. If DOE wanted to move into this phase, I could imagine looking at dollars per public millirem averted for another environmental cleanup, maybe a spill at the same site. I think we wanted to keep it somehow similar and so we envisioned that maybe you would compare one weird determination to another but it's difficult. We don't --

VICE-CHAIRMAN CROFF: I realize this is a very tough issue and I'm not sure I have an answer to it, but the relative comparison, I don't think quite is going to make it because for a couple of reasons. First, if the next one was say, you know, they go ahead and they grout these tanks and it was 10 million per millirem. They go to the next one and its 50 million per millirem or something, well, maybe you should have done something to the first tank but you've already gone by it and secondly, these may all be too high or too low compared to other opportunities to use the researchers.

DR. RIDGE: I think something that gives us a benchmark as to whether or not we're out of the ball park is that they do have to meet the performance objectives. So whether or not -- I doubt they would all be much to low in the sense that really they should be spending 10 bucks per millirem because I think if they did that, they wouldn't be meeting the performance objectives. So in that sense, that does help to tie us into reality but I certainly appreciate that there is an unsatisfying aspect to only comparing it to other DOE activities. Unfortunately, we also didn't think it was reasonable to compare DOE activities to for instance the ALARA analysis we do for our licensees. That seemed to us to be a bit apples and oranges. So I certainly appreciate your point.

VICE-CHAIRMAN CROFF: Fundamentally, I think you have to assure that the conceptual validity of the measure they propose and I'm not -- you know, I mean, on one hand we say collective dose has a problem but it includes the population, but this measure doesn't include the number of individuals exposed. Mike wants to intervene.

CHAIRMAN RYAN: I guess I'm struggling with Allen's view of it a little bit. I mean, in one hand I agree and hear what he's saying, but I think to me it's better to get back close to what is important to risk. Are you effecting release rates or not? Are you effecting confinement or not? Does your system add containment or not? Those are the kind of relative measures where I think you have a much better handle of evaluating A versus B. Please stay away from collective dose as you say you're going to. It's a measure fraught with terrible uncertainty in and of itself. All those dose conversion factors are all conservative, sometimes by many orders of magnitude and that's ignored when we do dose calculations most of the time.

So you're compounding, if you use a dose metric, another set of conservatisms that you don't even account for in most cases. So my view of it would be get back to the things that you looked at that are risk significant and try and get your measure of relative value, you know, for doing something closer to those activities out to the receptor. You know, my version of it for students is, "Well, do you want to drive the bus sitting in the front seat looking out the front window or do you want to put it in reverse and sit on the steering wheel and try and steer it"?

You know, it's much better to be in the front seat, so get close to the work, get close to the radioactive material and you'll have a better way, I think, to make those kind of evaluations rather than the back end. And again, it's all in the context of what Christianne said, that if you are demonstrating compliance, that's done. Now let's see if we can optimize at the source or at -- you know, that kind of thing. So does that make sense to you? You folks, all three of you or --

DR. ESH: I think it does to me. I mean, the problem is, if you're operating in an overall construct that has some degree of silliness to it, how much do you refine some part within it?

CHAIRMAN RYAN: Yeah, exactly, well said.

DR. ESH: I mean, that's the problem you're dealing with. I mean --

CHAIRMAN RYAN: That made up for the pause, by the way.

VICE-CHAIRMAN CROFF: I think with this, we're at the closure time, so I'm going to shut up and turn it back to you.

CHAIRMAN RYAN: Well, no, I appreciate the discussion but it's always good to hear --

VICE-CHAIRMAN CROFF: Well, no, we're at 12:30. I mean, I could yak on forever but --

CHAIRMAN RYAN: That was clear. Again, I thank you all for your time this morning and for your insight. You've got a tough job that you've done really a very professional and well prepared document and, you know, our part now is to maybe offer some minor things that might help make it even a little bit better. You've all done a really wonderful job and thanks for letting us participate with you.

With that, hearing no other further business we'll adjourn for lunch and reconvene at 1:30. Thank you.

(Whereupon at 12:31 p.m. a luncheon recess was taken until 1:29 p.m.)

CHAIRMAN RYAN: Good afternoon, folks. If we could come to order, please.

We have two briefing schedules this afternoon on dry cask storage probabilistic risk assessments, first from RES and NMSS, and second from the Electric Power Research Institute. We'll have both briefings separated by a short break.

So without further ado, I will turn this over to our cognizant member for this session, Dr. Ruth Weiner. Dr. Weiner?

MEMBER WEINER: Thanks, Mr. Chairman. Our first presentation will be from Ronaldo Jenkins, who is Branch Chief for PRA Support Branch for the Division of Special Projects and PRA in the Office of Research. And he is joined by Gordon Bjorkman, who is Section Chief of Structural and Material Technical Review Group and SFPO.

So without further ado, gentlemen, it's all yours.

MR. HACKETT: Actually, Dr. Weiner, if I could chime in. This is Ed Hackett from the Spent Fuel Project Office. I had a few opening remarks, and then we'll turn it over to the staff.

MEMBER WEINER: Please.

MR. HACKETT: Dr. Weiner, Chairman, thank you.

Good afternoon. As I said, my name is Ed Hackett. I'm Deputy Director for Technical Review in the Spent Fuel Project Office. Just a few opening remarks relative to context and key messages that I'll go into here just very briefly.

But even before that, I'd like to express our thanks from the Spent Fuel Office to the Office of Research, many of whose representatives are arrayed around me here to the right. And it's been a long effort for them and for us working collaboratively, so we appreciate that.

We also appreciate prior communications here just recently from the committee with regard to some of your questions, so we have the benefit of those in advance. We appreciate that. The staff will endeavor to answer your questions during the course of the presentation, and, if not, I'm sure you'll let us know.

If I could have the next slide.

This effort was really initiated to help SFPO develop an initial look at risk-informing our regulatory approach for spent fuel storage. As you are aware, the framework in this area has historically been largely deterministic and prescriptive. As I just mentioned, the Office of

Research has had the lead for this effort, but we have worked very closely, sort of hand in hand, on this effort for quite some time.

The focus is an important thing to bring across here in the way of context and opening remarks. The focus has been on development of the methodology, and you'll see in here, and I've already reviewed, the limited pilot application, the limited scope pilot application that you see there.

Go to the next slide.

And the reason for the importance of that context, I think it's obvious that these PRA numbers are very low. I think that's in common between the study that the staff did and also from what I've seen of the EPRI study. However, that was not the focus of the study. The numbers come out small. I don't consider that myself to be a surprise.

I come from the reactor side of the house here, just recently to SFPO, and, of course, dry casks are decidedly not PWRs or BWRs, so you would expect a lower risk, and, in fact, a significantly lower risk. And that's, in fact, what we see.

The dry storage systems for spent nuclear fuel are also passive, obviously. They have significant margins on the structural integrity that have basically been designed in, and they are extensively analyzed and tested, so -- also, there are significant inspection and oversight efforts that we do here at the NRC that you're aware of that provide for continued maintenance of these margins.

So the bottom line there is that there are a lot of reasons these numbers would be low, but that's also not the focus. The focus was really kind of where you get into in the second bullet here is looking at, you know, where are we getting to in terms of what's risk-dominant or what are risk-dominant contributors to this study. And Gordon and Ronaldo will go through that in detail.

But one example you'll see is, again, not surprising that the risk is dominated by handling sequences. And there will be some discussion of that.

So that said, you know, we're here to present you with significant findings and conclusions and present an overall discussion, and try and answer your questions to the best of our ability.

With that, I'll turn it over to Ronaldo. Thank you.

MR. JENKINS: Good afternoon. My name is Ronaldo Jenkins, and I'm Chief of the PRA Support

Branch in the Office of Nuclear Regulatory Research. I'm joined by Dr. Gordon Bjorkman, Chief of the Structural and Materials Section of the Technical Review Directorate in the Spent Fuel Project Office within the Office of Nuclear Material Safety and Safeguards, NMSS.

I would also like to thank the committee for taking the time to hear this presentation.

Just to review the topics we will discuss today, I will cover the goals of the dry cask storage system PRA and an overview of the PRA methodology. Then, Dr. Bjorkman will provide a detailed discussion of the success criteria for this system. He will discuss the staff's analysis of the response of the multi-purpose canister or MPC to these stresses and fuel failure. Dr. Bjorkman and I will then conclude by summarizing the report findings and highlighting its conclusions.

When the Office of Research began this project, it was first intended to be a scoping study. As the staff examined the issues involved, the scope of the report changed and became more detailed to provide better understanding of the dry cask storage system operation and failure modes. The primary focus of the report was to provide guidance for future PRA studies such that we can encourage risk-informed activities in this area.

Just to review what we mean by "risk, risk equals frequency times consequences." Risk in this report is defined in terms of the probability of latent cancer fatalities per person per year.

The dry cask storage system operation is divided into three phases -- handling, transfer, and storage. As the equation on this line indicates, we examine and determine the risks associated with these three phases, and then add them together to obtain the total risk.

Just a brief discussion on the cask system itself. The Holtec Hi-STORM 100 dry cask storage system consists of a multi-purpose canister or MPC that confines the fuel, a transfer overpack which shields workers from radiation while the cask is being prepared for storage, and a storage overpack that shields people from radiation and protects the MPC during storage.

When the transfer overpack contains the MPC, the unit is referred to as a transfer cask. When the storage overpack contains the MPC, the unit is referred to as a storage cask.

The dry cask storage system operation, as

I said, is divided into those three phases. During the handling phase, the transfer cask is lowered to the bottom of the cask pit next to the spent fuel pool. Then, the spent fuel assemblies are loaded into the MPC. The MPC is then prepared for storage and lowered from the transfer cask to the storage cask.

The transfer phase begins when the storage cask with the MPC inside is moved through an airlock outside the secondary containment building. Then, the transfer phase ends when the storage cask is moved to its location on the storage pad of the independent storage -- independent spent fuel storage installation or ISFSI. Lastly, the storage cask begins its phase of storage for the balance of the 20-year licensing period.

In order to facilitate the risk analysis, the dry cask storage operation was divided in 34 distinct stages. These stages were developed in part due to the detailed analysis that the staff took to -- when they examined the overall process.

This composite sketch shows the movement of the transfer cask and storage cask through the secondary containment building, out the equipment hatch, to the ISFSI. A risk assessment will evaluate how the applicable initiating events affect MPC during each stage of operation.

Just so that we are clear on terms, in terms of this report, initiating events are those events that may lead to a release of radioactive material to the environment.

As we have discussed before, the initiating events were identified using NUREG-2300, PRA Procedures Guide, and from design operational data for the specific cask and the plant being studied. Information on the design of the cask system was obtained from licensing documents.

Analysts visit the plant to observe the operation and equipment used during the handling, transfer, and storage phase. Written descriptions of the procedures were obtained and studied, and additional details were provided through a discussion with plant personnel.

The total list of initiating events were reviewed by the NRC staff who had reviewed and licensed this particular dry cask storage system. This review drew upon the extensive knowledge and the diverse perspectives that the staff had on the system. Based on these reviews and the process used to develop these events, the staff constructed a complete list of all initiating events that would conceivably affect the cask system.

What you see on the slide is the final list of initiating events for the handling and transfer phase which were not screened out by other engineering analysis.

This line lists those initiating events relevant during the storage phase. Here we're concerned with external phenomena such as seismic events, strikes from aircraft, or thermally overloading the MPC due to vent blockage or fire. We are excluding tsunamis and volcanic activities as initiating events, because they are not applicable to the site.

Other events such as lightning, flooding, and shockwaves from pipelines, commercial trucks, and rail cars were screened out by engineering analysis.

Given that the applicable initiating events create mechanical and thermal challenges that could lead to failure, the PRA must now assess whether the barriers -- in this case, the fuel plan and the MPC cask system -- will be successful in performing its containment function.

In addition, for the subject plant, a release of radioactive material will actuate the containment isolation function. Therefore, the PRA must consider the reliability of those systems to isolate that release.

As shown in this event tree, we see that the applicable initiating event and the success criteria combine to determine whether or not you arrive at a particular end state, whether you have a release or no release. The evaluation of the release end state, or consequence analysis, provide us with the consequence portion of the risk equation.

In order to assess the radiological consequences, the staff used the MELCOR accident consequence code system. Release fractions were estimated, and the source terms were developed based on input from Sandia National Laboratory.

As shown, the model used input from radionuclide inventory, source term, meteorological data, population data, and emergency response to make these calculations. Estimated consequences in terms of latent cancer fatality probability for an individual was 3.6 times 10^{-4}.

Going back to our risk equation, we summarized the risk in each of the three phases -- handling, transfer, and storage -- to provide an estimate of the annual risk to an individual. We estimate 2.0 times 10^{-12} for the first year of operation, which includes the three phases. We estimate 1.9 times 10^{-13} per year for the remaining years of operation, which only involves the storage phase.

At this time, I'd like to turn the presentation over to Dr. Bjorkman, who will discuss specifically the staff's analysis of the mechanical and thermal loads on MPC and fuel.

DR. BJORKMAN: Well, thank you. Could I have the first slide?

Thank you. In terms of success criteria, what I'd like to talk about and highlight are basically the Hi-STORM 100 system. I'd like to summarize the events that could lead to containment or confinement boundary failure -- that is, MPC breach -- or fuel failure.

I'm going to concentrate on the high probability of failure events. I'm going to talk a little bit about the analysis models, failure criteria, failure modes. And when I'm finished with that I would also like to talk about the release fractions methodology that was developed.

Next.

Going to the Hi-STORM 100, as Ronaldo has already mentioned, there are three components -- the multi-purpose canister, which is the confinement boundary for the fuel; the transfer overpack shields the MPC and workers during transfer operations; and the storage overpack, which shields the MPC during storage.

Next, please. Thank you.

Just to give you an idea of what these look like, the transfer overpack -- these are pretty much to scale. The interior volume is occupied by the MPC, and those are approximately the same. The transfer overpack consists of an exterior one-inch thick plate, an interior three-quarter inch steel plate, and four and a half inches of lead shielding. And it's surrounded by a water jacket for a neutron shield.

The storage overpack is -- has a steel shell about three-quarters of an inch thick, an interior shell of approximately one and a quarter inches thick, and a concrete -- filled in with concrete that is about two feet thick. It also contains a concrete shield lid, as well as two two-inch thick plates that cover the top of the storage overpack.

Next, please.

The multi-purpose canister -- the multi-purpose canister is basically made up of three components. There is the shield lid, the structural shield lid, which is a nine-inch thick stainless steel lid; an inch and a half -- or, excuse me, a half-inch thick steel shell; and a two and a half inch thick baseplate.

With respect to the seals that occur at the junction of the lid and the shell -- of course, we have to have a double seal there, and that is formed by the exterior shell. And the lid -- there's a structural weld at this location. The welds that prevent leakage through the event and drain ports are here.

These two welds, in this group of welds, provides the first seal. The second seal is provided by an annular plate, which is then welded to the shell and welded to the lid. And that provides the second confinement boundary seal. So it's a double containment or double confinement as required.

The lower region there is a full penetration weld that connects the shell to the baseplate. That is right down here at this location. This will be a very, very important -- of interest. This will be a -- really, a region of focus down here in terms of MPC potential breach and failure.

Next slide, please.

Release of the radionuclides -- well, radionuclides are released from the environment if -- first, we have cladding failure or CRUD spallation, and the MPC confinement boundary breaches.

Okay. Next.

Now, the Table 19 in the report summarizes the various stages. We have summarized them right here. We have 34 stages. We talk about initiating events or frequencies, and these range in these orders of magnitude for all of the 34 events.

We then have the conditional probability release from the MPC or from a fuel rod, and these range typically from zero all the way up to about 28 percent conditional probability failure.

We then have the probability of secondary containment failure, the consequence, and risk numbers, and these are the ranges. What I am going to talk about specifically is this column. Virtually my entire presentation will be dealing with this column -- conditional probability of release from the multi-purpose canister or from fuel rods.

MEMBER WEINER: Excuse me?

DR. BJORKMAN: Yes.

MEMBER WEINER: Gordon, can we go back to that slide a moment? What are the units of consequence that you have?

DR. BJORKMAN: Cancer fatalities per year, I believe?

MEMBER WEINER: Consequence?

DR. BJORKMAN: No. I'm not sure.

MR. JENKINS: It's the probability of latent cancer fatalities.

MEMBER WEINER: I thought that was the units of risk.

MR. JENKINS: It's frequency times the consequence.

MEMBER WEINER: Oh, okay. Thank you. So the consequence there are latent cancer fatalities, is that correct?

MR. JENKINS: Right, probability.

MEMBER WEINER: Probability. Thank you. Okay. Sorry.

DR. BJORKMAN: No, that's fine.

MEMBER WEINER: Please continue.

DR. BJORKMAN: Okay. So what I will be talking about is that second column -- conditional probability of release from the MPC or fuel rods. Okay. Event categories -- there are two event categories that could produce fuel failure or MPC breach -- thermal events and mechanical load events.

Under thermal events, to evaluate the thermal events, a computational fluid dynamics model of the MPC and the storage overpack were developed to do the thermal evaluations. This is the storage overpack. A detailed thermal analysis model was constructed, a computational fluid dynamics model using fluid.

Okay. And this model was used to evaluate two particular thermal events -- that is, aircraft fuel fire, so the entire fuel load from the Gulfstream IV aircraft, which is the largest aircraft that could land near the -- this particular site. The entire fuel load was then discharged and burned for three-hour duration.

We know that this is quite a conservative duration. We know that in aircraft failures or aircraft crashes that we have a large fireball much of the fuel is burned up in the first few seconds or few minutes. All of this -- all of this fuel was also pooled around the storage overpack. We know that that's a very unlikely event as well. So it's quite a conservative analysis that was done here.

MR. HACKETT: Gordon, could I interrupt for just a second?

DR. BJORKMAN: Yes.

MR. HACKETT: This is Ed Hackett again. I should have mentioned at the beginning as a caveat to this, and it's maybe obvious to a lot of folks, but what Gordon is talking about here from the aircraft perspective is an accidental crash. This study specifically excluded accident, sabotage, and terrorism related to those factors.

MEMBER WEINER: Thank you. Your report makes that very clear.

DR. BJORKMAN: Okay. Very good point. Thank you, Ed.

And, again, these are from accidental crashes of aircraft.

Blocked vents was another event that could take place. Blocked vent -- duration for the blocked vents, the vents cool -- convection cooling of the MPC shell is done through air circulation if these vents are blocked. The temperature of the MPC could go up, and the temperature of the fuel could go up as well.

A 20-year duration for this was assumed, although steady-state temperature are actually reached in less than 30 days. Also, it would be very difficult for this to occur, because inspections are done -- several inspections are done yearly to particularly look at whether or not the vents are actually blocked.

But the 20-year duration was assumed, because as I'm going to talk about one of the other failure criteria, which is a structural failure criteria, is creep rupture, and we try to prolong the duration of this fire, so we can get as much duration to see if we could get creep rupture.

Okay. Next slide, please.

Now, results of the thermal events with respect to fuel cladding failure. These are the two events -- the Gulfstream IV fuel fire and the blocked vent. The maximum cladding temperatures in degrees Celsius are shown here, and the accident limit or the accident temperature limits are shown here, 570 degrees. And, obviously, from this we see that there are no cladding -- fuel cladding failures.

I should mention as an asterisk on this that cladding failure is actually not expected until we get to temperatures well above this, temperatures in the vicinity of 750 degrees Celsius. So this was quite a conservative failure criteria, and we never reached those temperatures.

Next slide, please.

Now, thermal events and MPC failure, thermal events and the multi-canister failure. We're looking at a loading in the MPC and internal pressure due to the filled gas. The MPC canister is filled with helium. The helium is there to cool through convection, to cool the fuel. It's at approximately five atmospheres, about 82 psi, and there are two failure modes that could be generated from this internal pressure loading.

One is a limit load failure, and in that case what happens is you get a -- we use a flow stress model, and what we want to do is -- what are the stresses causing continuous plastic flow? Could I get continuous plastic flow and breach? And what we wanted to make sure is the actual stresses in the shell, in the MPC, are actually less than the flow stress.

Now, the flow stress itself, though, is a function of the yield stress of the material, the ultimate strength of the material. In turn, the yield and ultimate strength are functions of temperature. So what was done is probability distributions were developed from the literature for all of these quantities, Monte Carlo simulations were performed, and no failures were predicted at all.

For creep rupture, creep rupture being under sustained stress, long-time -- long term-stress. Is there a sustained straining such that a strain limit is reached and rupture occurs? And that's what we'd like to determine here.

So it's a time to failure data, or as much time to failure data on the stress and temperature for stainless steel weld and base metal was obtained. The Argonne National Laboratory creep model was used to predict creep damage for any time-temperature-stress condition, and in this model the stresses were magnified to account for weld flaws as well.

And using all of this data and running it through a Monte Carlo simulation, again, no creep rupture failures were predicted. None whatsoever.

Next slide.

So we see that from thermal events we have no failures, either for the fuel rod cladding or for the MPC confinement boundaries.

Now, mechanical load events. What was considered? What were the results? Explosions -- a gasoline tanker traveling on the nearest highway. Well, the explosion of that tanker of course is an overpressure at the location of the storage overpack of about one pound per square inch, significantly less than the design external pressure of 10 psi. Again, pipeline failure from the nearest pipeline and explosion overpressure one psi, much less than 10 psi design.

Strikes by heavy objects -- could they tip the storage cask over? Could they penetrate it? Well, we looked at vehicle impact. We took a 10,000-pound vehicle traveling at 150 miles an hour. You could not tip over the cask. If the cask does not tip over, there is really nothing that really stresses the cask whatsoever, unless it tips over.

Tornado missiles -- again, the mass and velocity of these missiles were insufficient to cause storage overpack perforation or tip over.

Again, strikes by heavy objects continued -- aircraft. The Gulfstream IV aircraft is the largest aircraft that can be handled at the local airfields. This is a twin-engine jet. The two jets are mounted at the rear of the fuselage. The plane weighs approximately 74,000 pounds.

We're looking at the possibility of crashes on landing and takeoff as well as crashes due to overflying aircraft that don't land at the airfield. Landing and takeoff, it's the -- Gulfstream IV is the largest aircraft. We want to look at the hard components that are in the Gulfstream IV.

This would be the landing gear or the engine shaft, and the engine shaft is where the -- the hardest, smallest diameter piece that could hit the storage overpack. And that does not penetrate the storage overpack, let alone even get to the MPC.

The mass and velocity also of this aircraft are insufficient to tip the cask over as well.

Okay. Now, that's for takeoff and landing. What about overflights? Well, we assume that all over-flying aircraft are larger than a Gulfstream IV and traveling at high velocity. We, therefore, assume that all impacts cause cladding failure and MPC breach. We made that assumption.

Rather than trying to do an analysis for all of these aircrafts, okay, we just said let's just see what happens to the risk numbers if we made the assumption that all overflights -- that these are large aircraft traveling at high velocity, and they could potentially breach the MPC and cause fuel cladding failure.

Based on that, the conditional probability of a release is then the probability or frequency of overflight crashes divided by the sum of the frequency or probability of overflight crashes and takeoff and landing crashes. And the number that is reported here and is in the PRA is .14.

Well, I want to tell you that this number is wrong. Okay? In reviewing this section last night, I discovered that the calculation for overflight pressures, you have to have -- you have to know the size of the target area that the aircraft will hit. Well, in that calculation, on page 32, second from the bottom paragraph, they had a calculation in there which the aircraft engines of the Gulfstream II were 100 meters apart.

Well, we know that that's not true. They are actually a lot closer than 100 meters, and that number is going to be reduced by a factor of more than 10. This number will then go down to .01, will be one percent, and will change the risk number accordingly by an order of magnitude. And this will be corrected in the PRA.

Next slide.

Other mechanical load events -- seismic. An ABAQUS soil structure interaction mode, ABAQUS is a finite element package that is used for non-linear analysis as well as elastic analysis and explicit dynamics.

A soil structure interaction model that included the storage overpack, the ISFSI concrete pad, and the soil was modeled, and the coefficient of friction between the cask and the pad -- that is, the frictional coefficient that resists sliding or tipover, particularly sliding of the cask, was varied between .25 and .53.

Earthquake magnitudes were increased from their site design basis value by 9 to 11 times. The site design basis value was taken at half of the seismic margins earthquake value, which is .3g, and we use .15g peak ground acceleration. Again, these are increased by 9 to 11 times, the design basis earthquake, no cask tipover whatsoever under those conditions.

Okay. Thank you.

Mechanical load events continued. Cask drop events. Okay. There are two categories of cask drop events. One is when the MPC is unsealed, open, the lid has not been welded yet. Okay? Those obviously, in terms of the calculation of whether the MPC breaches or not, don't really matter. We must consider that the MPC is breached for all of those evaluations.

Now, when the MPC is sealed, there are

really four conditions and four general categories. One is when the transfer cask is moved over the refueling floor. The maximum drop height at that point is about three feet. The other case is when the transfer cask is lowered through the equipment hatch we have a maximum drop of 100 feet.

And the other is when the MPC, the multi-purpose canister, is lowered into the storage overpack from the transfer cask. That's a 19-foot drop, and that storage overpack moved to the ISFSI pad and the maximum drop is only one foot.

Now, in evaluating the MPC drops there were two significant drops. One is the 100-foot drop through the equipment hatch. We have the refueling floor, we have approximately a 100-foot drop. If the storage overpack, if the cask hits the storage overpack, that ends up being a soft impact, because the storage overpack acts as an impact limiter, absorbing much of the energy in that impact.

If the storage overpack is either not here or the transfer cask misses the storage overpack on its descent, it will hit the concrete floor. That is also a soft impact. This transfer cask, as I described earlier, is a fairly robust, very heavy cask. It goes about 10 inches into the concrete floor, and that 10 inches of deformation and crushing absorbs a significant amount of energy. So that is relatively soft impact.

On the other hand, the 19-foot drop of the storage overpack -- of the MPC into the storage overpack -- and I should explain what happens here -- it's lowered through the equipment hatch down to and rests upon -- on the top of the storage -- on the top of the storage overpack, and then independently the MPC is then lowered after the door is slid sideways, opened, the MPC is lowered into the storage overpack.

There is a possibility in this particular transfer that it could drop 19 feet. This is a hard impact. There is very little energy absorption here. The MPC hits the bottom of this plate. This plate is spread over a large area. Very little deformation takes place. It probably only sees -- well, it sees on the order of probably only a fraction of an inch. We're talking about maybe an inch deformation here, very small amounts of deformation. That's a very hard impact.

And as we will see, just to give you -- you know, let you see what's going to come here, this is the dominant contributor to risk, this drop right here, not that one. And that comes out of this study.

Yes?

MR. DIAS: One quick question here. How wide is the shaft? You know, is there any chance of some rotating momentum to be applied to the canister, or as the transfer canister -- as it's coming down that would cause it to hit some of the floors in between? I'm thinking out loud here.

DR. BJORKMAN: I really depends upon what actually happens, what the event is that causes --

MR. DIAS: Yes. But if it's wide enough, we know, then, that could be a little less probable.

DR. BJORKMAN: I couldn't tell you exactly what the width of this is.

MR. DIAS: Okay.

DR. BJORKMAN: My estimate is that it is probably 30 feet or, you know, more. I'm --

MR. DIAS: Okay.

DR. BJORKMAN: I'm just guessing, but I don't know for sure.

MR. DIAS: Okay.

DR. BJORKMAN: I mean, I have looked over equipment hatches before and looked down and --

MR. DIAS: I haven't.

DR. BJORKMAN: I don't -- I don't recall

what the exact --

MR. DIAS: Okay.

DR. BJORKMAN: But, no, you know, if it is brought over and the event -- the drop takes place as it's coming over and certainly hits something and tips it could then -- and it would go down, that would -- that would probably be a less damaging event for the MPC than the direct impact all the way down.

The likelihood of breach under those conditions is probably less. That's just a guess at this point.

Yes?

MR. MALLIAKOS: This is Asimios Malliakos from the staff. Actually, this failure is being drawn to scale. So I have engineer here --

DR. BJORKMAN: This is 20 feet. Then, this is on the order -- this could be almost 30 feet.

MR. MALLIAKOS: Yes.

DR. BJORKMAN: So it could be close.

MR. MALLIAKOS: Yes.

DR. BJORKMAN: Okay. So this is the event that will dominate right here. It's not intuitive at all, not intuitive at all. But this is what comes out when you do this kind of a detailed evaluation to determine what the dominant event is.

Okay. Next, please.

To do this analysis, a detailed LS-DYNA finite element model was developed to perform the drop impact analysis. This is a continuum mechanics model. This is the geometry. It's a quarter scale, taking advantage of two planes of symmetry. It's a quarter scale model. This shows the concrete floor and the wall under the concrete floor that this cask would impact.

Next slide, please.

We zoom in at the bottom there. We zoom in at the bottom corner, and, you know, this is hard to, you know -- in a 10-second glimpse it's hard to see what's going on here, but you can begin to see some of the detail.

This is the baseplate of the MPC. This is the baseplate. Here we have the shell -- the shell, the half-inch thick shell. And there were a lot of elements through the thickness, and you see that going up this way.

This yellow here is a basket support, and I will talk about that in a minute. That's a basket support that is welded to the MPC shell. You see that in a very coarse model the actual basket in green is modeled. The actual fuel rods are actually modeled, and they are modeled so that the mass -- the mass of the system is actually modeled correctly. So they're in there just to make sure that the mass and the dynamics work properly.

Next slide, please.

If we look at the MPC -- and, again, I talked about that weld in the corner between the shell and the baseplate. If we look at a location away from the basket support -- the basket support that I'm going to be looking at in this case is a bar that may be an inch and a half thick and maybe two inches wide. The basket supports are welded fairly -- at anywhere from 15 to 20 degrees around the interior of the MPC shell. They're there to prevent any movement of the basket inside the cask. That's their function.

If we look at the deformation -- and this is for the 19-foot drop at the same time at five milliseconds into the event, if we look at a location away from the basket support we see a nice gradual curvature taking place, a very nice deformation.

If we look directly at the basket support, we see that what is happening here is we get high constraint. Virtually much of the deformation -- all of the deformation takes place just in this lower section right down here. So the basket support is constraining the deformation into this localized region.

Next slide.

And if we look at the stresses, or in this case the strains, the effective plastic strains in here -- and this is exactly the same picture as I showed you before, and now we're going to look at it more closely. This is a closeup of that same section, and I'm going to show you the maximum value of strain that comes out of here, which is .459 or about 46 percent strain. You'll remember that number.

What I also want to show you is another thing that's very important for the PRA to recognize how this analysis was before performed. Notice this maximum occurs at a single element -- right here -- a single element through the thickness. There are six elements through the thickness.

So when we discuss the failure probability of the MPC or the possible breach of the MPC we're really talking about the failure of that one element through the thickness. And we're making the assumption that this crack or this initiation of failure would propagate through. That is not always the case, however.

So this is a conservative analysis in that case. It will take additional -- additionally more rigorous analysis to actually go through and fail it all the way through and do the multiple simulations that would have to be done. So I want you to keep that in mind. We're talking about a single element here.

Okay. Thank you.

What is the failure criteria? I showed you how we calculated the stresses, or in this case the strains. I showed you how we calculated the strains.

What's the failure criteria? The most highly stressed region of the MPC is at the circumferential weld joining the shell to the baseplate, and you saw that. The material, the weld material, is Type 308 stainless steel. We have a strain-based failure criteria based on test data of Type 308 stainless steel weldments taken from nuclear powerplant piping, nuclear powerplant piping that was in service. These coupons were cut up from those welds, and tests were done on those two failures to determine strain at failure.

From this data, the mean and standard deviation of the true strain at failure was calculated, and the true strain at failure is really what we want, because this is consistent with the output in LS-DYNA. The data that we used to compare with our analytical model should be the same and consistent. In this case they are.

The data have to be adjusted, however, for strain rate and temperature. The data is for room temperature at static loading. We have to adjust it for high strain rates, high impact loads at elevated temperature. A factor of .88 was applied to the mean failure strain.

Okay. And based on that, the actual data now -- I can show you, this is in Table B2 in the PRA -- we now have the standard deviation from the mean. The mean value for the strain at failure is about .73 or 73 percent strain. Seventy-three percent strain, for those of you who aren't familiar with strain, this would be a 73 percent -- in general, a 73 percent increase in the length of the material prior to failure.

Okay. So a one-inch bar would fail when it got to 1.73 inches approximately. That's not exactly the definition of "true strain," but it's the definition of engineering strain.

Anyway, so .73 or 73 percent strain, and that is really what we were calculating -- that is at the 50 percent probability effect. That is, we have a 50 percent chance that the actual failure strain is less than the calculated value. Okay? So this is incorrect. This should be switched around. It's correct in this table in the PRA report, however.

So this is the probability. This is the probability that the actual failure strain is less than the value that was calculated in the LS-DYNA program. Okay. And these are the values for several standard deviations.

Next slide.

We also have to adjust it for the state of stress. We adjusted it for strain rate and temperature. Now we have to adjust it for state of stress.

Okay. The strain at failure is based on uniaxial tension -- that is, pointing it in one direction, stretching it this way, failed. Okay. In the actual LS-DYNA calculation, we have a complex three-dimensional state of stress going on. Okay? So we need to -- and this triaxial state of stress, this three-dimensional state of stress, may constrain plastic flow and lower the strain at failure, particularly if it's tension. It'll constrain the plastic flow and lower the strain at failure.

So what is calculated as a triaxiality factor for each element -- so for each element in the analysis a triaxiality factor was calculated, and the failure strain was modified.

And this is the final data -- MPC failure probability. For various drop heights -- 19-foot drop, 100, and five-foot drops. The maximum strain in LS-DYNA -- I'll just go through the 19-foot drop, the maximum strain in LS-DYNA, approximately 46 percent strain. Notice the 100-foot drop is considerably less.

Okay. Now, adjusted for the effects of triaxiality, what we did was we took the triaxiality factor and bumped up the LS-DYNA value -- rather than lowering the failure value, we bumped up the LS-DYNA value by the triaxiality factor to get this strain, before comparing it to the table I just showed you before, to compute the failure probability. And this is, again, the probability of weld failure.

So we end up with approximately a 28 percent conditional probability failure, okay, given that the event has occurred. And, again, asterisks -- this is the probability that one of the six elements through the thickness has failed.

Next slide. Thank you.

Okay. So we've talked about MPC failure. Now we also have to talk about cladding failure, the drop events, mechanical drop events. We have end drop impact. The most likely drop scenario is that of an end drop impact. These are high impact loads on the fuel rods.

If we were to go and use what we call static buckling formula for a fuel rod, and use static buckling formulas where you just -- you know, we all take the yardstick and put some load on it and it bows out, and that -- that is buckling.

Well, if we did and used those formulas to predict the failure of the fuel rod for the g loads that are -- it is subjected to, we would have the fact that a one-foot drop predicts buckling and fuel cladding failure. And this, of course, is not physically correct.

What happens is that magnitude and the duration of the loading are important. We have high loads but very short duration. And this is a dynamic problem and must be treated as a dynamic problem.

What we did is we developed a fuel rod model, a single-pin model, and this is -- the artist has taken a great deal of liberty here in creating --

this is a straight pin. It has a slight blow in it. That bow is only one one-hundredth of an inch, but it's highly exaggerated here, just for the point of illustration.

These lateral springs are the grid spacers between -- okay, the grid spaces in the assembly. These distances are typically 20 inches, 20 inches each. Okay? And there's a small amount of bow.

And the rod can displace laterally through some gap, and that gap is determined by distance between adjacent rods, how much gap there is between the fuel assembly and the fuel basket itself, and the maximum gap was assumed.

Now, if we use the single rod model -- and that was dictated by computational efficiency. In a 10 by 10 fuel assembly, we have 100 rods. All of a sudden we have 100 rods buckling, interacting with one another. This is a very complex problem. It's only recently that this problem has begun to be tackled computationally.

This single pin rod by itself has 20,000 elements and 10,000 nodes. Okay. We use a cask to ground spring. I will just -- you know, we have a rod and there's the cask mass and the MPC mass are all in here, and we have a cask to ground spring.

They'll say, "Well, how do you choose that cask to ground spring?"

Well, what is the fuel rod field? The fuel rod fields -- what it's resting against. It's resting against the MPC baseplate. Well, how does the MPC baseplate move? Well, what we do is we determine the stiffness of this spring such that it has exactly the correct displacement characteristics, and we go through an iterative process until we get it right, so that it displaces and the fuel rod thinks it's resting against the MPC baseplate.

The mechanical properties of high burnup fuel were used, and a cladding failure strain limit of one percent was used. And this is near the lower end of the strain failure data. Other values could certainly be used. We used one percent in this particular study.

Okay. I want to show you one of the results, and then this is -- again, this is not intuitive. Fuel rod response -- these are basically impacts from the same height. There is a 20-foot drop onto the concrete floor, and this is the MPC 19-foot drop of the -- from the transfer cask into the storage overpack. I talked about that before.

Look at the behavior of this. This is a

fairly soft impact. Okay. The 20-foot drop, the transfer cask, onto the concrete floor. We get deformation. The transfer cask is very heavy. It penetrates an inch or two into the floor for a 20-foot drop, and we get this very classic buckling mode, very classic.

This is one grid spacer. This is the next grid spacer. This is about 20 inches.

Now, MPC hard drop. This is a hard drop. Same drop height. Totally different buckling characteristics. This buckling characteristic, this is the exact buckle shape you would get if you took a rod -- free rod -- a fuel rod, dropped it 19 feet onto a rock hard surface, steel plate or something, freely, without any support or anything, you just drop it, bang. This is the buckle shape you get. It's a classic textbook. You can open a textbook. That's exactly what you get.

Well, isn't this nice? The model predicts it, so the model works. It's not biased by our own -- how we constructed the model or anything like that. It is giving us exactly what it wanted to do. In this process, the strains are very, very high, as we'll see on the next slide. If we look at what goes on here, and we say, well, drop height -- the maximum principal strain with drop height onto the concrete floor -- and what we see is for about 20 feet we're less than the one percent strain limit.

At 40 feet we've exceeded the one percent strain limit, so we could say, well, we -- by our criteria, we're getting failure of somewhere between 20 and 40 feet.

Look at the 19-foot drop. Nineteen feet -- we are way up there. Way up there. Okay. We're probably at -- for the same drop height we're more than 10 times higher in the strain value. So it is a much more severe impact again.

Go ahead.

Okay. That ends the discussion of the success criteria that basically lead to MPC, breach, or cladding failure. Now I'd like to talk about release fractions methodology, and this methodology was developed from a number of references. Dr. Bob Einzinger put this together, did a great job.

The release fractions methodology -- what's the governing equation? It's actually pretty simple in its most fundamental form. The release fraction -- that is, the amount of radionuclides that get out into the atmosphere is based upon what?

Well, if I have a three by three fuel

assembly, certainly based on the number of rods that fail -- let's say the red ones fail, so four out of nine rods fail. This is four over nine. That's the release fraction.

Now, I've got to look at it and say, "Okay. Those rods failed." Now, of those rods, how much of what is in that rod gets into the MPC canister, into the cask? How much gets into the cask environment? So that's this quantity -- F sub from rod to cask. Then, if there's a breach, you have to say, "Well, of all the stuff that's in here, how much actually gets out into the environment?" So that's the third component.

And I'll go through very, very briefly and discuss how we went about or how Bob went about calculating each of those quantities.

Okay. Source terms -- the source terms. The source term for the I^{th} radionuclide -- we have quite a few radionuclides. What is the source term for each radionuclide? We have F sub K. This is the release fraction.

And the source term -- the amount of stuff, the amount of radioactivity that is going to get out is, what is the fraction of the total inventory that gets out summed over the various -- summed over the various types of radionuclides that we can have?

And we have basically three larger classes of radionuclides. We have noble gases and volatile gases. Okay. And as I'll explain later, we're not going to be talking about volatile gases, just noble gases. And this will be krypton-85.

Fuel particles, fuel particulates, and we're also -- and we're going to be talking about not only the body of the fuel pellet but also the rim of the fuel pellet as well. And we'll also talk a little bit about the CRUD.

Okay. What are the model limitations? It's only applicable for impact events. The effect of fire on volatility of fission products and change in material properties are not considered because the MPC failures -- because no -- no MPC failures occurred due to thermal events.

And, therefore, thermal events which would produce volatile fusion projects -- if the temperatures got high enough -- are not considered. The temperatures are not high enough to release these volatile fusion products -- fission products.

Next.

Fuel properties. BWR, slight modifications would have to be made for PWR, but it's BWR fuel, 60 gigawatt days per metric ton burnup, and the rim effect in the fuel pellet is considered. And the reason it's considered is that the actinide inventory -- actinide inventory in the rim is higher than in the body of the fuel. That's number one.

And the particulate size is small. And what I mean by "small," I'm talking about sub-micron size, .1 to .3 microns. And, therefore, the rim and body are considered two distinct regions in this methodology.

Next.

Okay. Release from the rods, F sub RC. Release from the rods into the cask. How is that done? Well, as I just mentioned, the particulate release from the rim and the body regions were analyzed separately.

Now, the fracture of the fuel into fines is based on modifications of the equations from the DOE Handbook that relate the fraction of the fuel fragments, the fraction of the fuel fragments that are generated, that are of respirable size, versus the specific energy or the impact energy.

If we know the impact energy, we can go up and using the DOE methodology we can calculate the percentage of particles less than 10 microns. Okay.

I should say that the PRA adjusted this curve downward to be more consistent with the data, and that is explained in the PRA.

Okay. F sub RC. F sub RC, release from the rod to the cask is dependent upon what? The number of fracture sites in the rod, and anywhere from one to seven sites were considered. Five is the default value.

Entrainment of the fines in the gas stream during depressurization of the rod. Rod breaks, the gases want to stream toward the opening, the gases as they're moving at some velocity want to pick up the particles. How much of those particles are picked up by the gas and get out of the rod? That's the entrainment.

Now, the extent to which the rim region actually fractures -- how much of the rim region actually does fracture? Okay. Well, uncertainty is considered in both of these parameters -- number of fracture sites, entrainment, and the amount of rim material that is actually fractured. And with those ranges you end up getting release fractions for this particular quantity, from rod to the cask, that vary from 7 times 10^{-5} all the way up to 1.2 times 10^{-2}. So variability in these is significant.

Now, the next quantity is the cask to environment release. So now we've got the particles in the cask. They've come out of the rod; they're in the cask. Okay. Now what happens? Well, the particles not settling out by gravity or plating out onto surfaces is assumed to be 10 percent, so 90 percent are assumed to settle out or plate onto surfaces.

And, again, this -- in this environment we have the internal five atmospheres or the original 82 psi, plus the fill gas pressure that is now relieved. So the internal pressure in the cask is greater than five atmospheres. It also depends upon the particles exiting the depressurized cask.

How many exit the depressurized cask? Of those that it suspended, how much exits the cask? It is assumed here that it's 100 percent, because we're going from five plus atmospheres down to one atmosphere, and in this process we're going to get -- depending upon how much the fill gas contributes, we're going to get up to the high 90s in terms of percentages of actual material that will go out when the cask actually ruptures. So we were assuming 100 percent here for that.

CRUD -- what is the basis for CRUD

inventory? CRUD -- Chalk River Unidentified Deposits -- it bounds -- the value that is used of .72 curies per rod bounds 90 percent of the rod data of the data for assemblies that's out there. The inventory was decreased, or the radionuclides were decreased by decay of cobalt-60. It's assumed that CRUD is made up of cobalt-60. The decay of cobalt-60 was assumed over 10 years, so that's also contributing and went into the value.

Reduce the CRUD values -- reduce by a factor of two for axial variation on the rod, because the data is based on peak values. So it was smeared across the rod. It was scaled up for burnup. Okay? Scaled up for burnup because the data is really for low burnup fuel, but it does not include the influence of water chemistry.

PARTICIPANT: (Inaudible comment from an umiked location.)

DR. BJORKMAN: Right. Ten years is the age of the fuel since it has come out of the reactor. Correct, right.

And this is basically a summary of the release fractions. These are for the three basic groups -- noble gas particulates and CRUD. The inventory came from the ORIGEN program here. This was basically developed for the CRUD inventory curies per rod. The fraction of rods that fail -- 100 percent of the rods when they failed -- 100 percent of the rods were assumed to fail in this analysis, or the fraction from the rod to the cask -- again, for noble gases, 12 percent.

This was the range of values. You saw these numbers before when I talked about the uncertainty. These are the range of values, and this range of values pertains to the amount of rim fracture which can be almost zero to one, and the entrainment. How much of it actually gets entrained in the gas as it flows out of the crack? Anywhere from zero to one, and that gives you this range.

How much actually gets out of -- okay. Well, for the CRUD we've got 0.05. And how much actually gets out from the cask to the environment? For the noble gas it's all of it. For the particulates it's 10 percent. And for the CRUD it's also 10 percent. And that gives you the -- basically, the release fractions for each of these three groups.

And now I'd like to turn it back over to Ronaldo to talk about issues that are out of scope.

MR. JENKINS: Now that we've discussed

basically what went into the report, we should also talk about what didn't go into the report or was not explicitly addressed.

As the slide indicates, terrorism, sabotage, or military accidents were not addressed by this PRA. Fabrication errors or design changes were not considered in this study. But we did consider the weld failure evaluation of the MPC, as Gordon talked about, to reflect normal flaws that might exist in well deposits of stainless steel.

Plant damage -- the casks would travel along a designated load path that was selected to ensure that should the cask be dropped on the floor the floor would be able to hold the cask. The cask -- excuse me, the train carrying the transfer cask along the load path is also designed at this plant to be single failure proof.

The frequency of misloading, while not estimated, deterministic calculations were performed to investigate the effects of misloading on thermal loads, and the failure probability of the MPC and the possibility for criticality. With respect to human reliability issues, the operational data was used in order to derive the frequency of the handling initiating events to occur. Therefore, human performance is implicitly implied, so we did not do a human reliability analysis. But the data does reflect human performance.

Similar to nuclear powerplant PRAs, worker risk was not addressed. And except for possible cask and fuel corrosion, aging effects was beyond the scope of this PRA.

Lastly, we considered individual initiating events and not multiple events. Individual factors were investigated one at a time using sensitivity studies.

Including the issues outside the scope of this report -- unloading, offsite, transport, and repository storage was not addressed in the report. On the subject of uncertainty analysis, we do recognize today that we would formerly perform a quantification of the model uncertainties, but the decision at the time was to forego that step.

Now, as to conclusions, the PRA report determined that there was no prompt fatalities, and the risk in terms of latent cancer fatalities was very low. The risk was dominated by accident sequences in the handling phase where the significant contributors were the drops of the MPC and transfer casks.

This comprehensive evaluation of the initiating event success criteria and accident consequences sets the stage for future PRA studies in this area.

At this time, we'll entertain any questions you might have.

MEMBER WEINER: I'm sure that we have -- I certainly have a great many, but I will defer first to my colleagues on the committee. Dr. Hinze.

MEMBER HINZE: If I may ask, these out of scope issues that you've just talked about -- did sensitivity studies indicate that these could be considered outside the scope?

MR. JENKINS: I'm sorry. The --

MEMBER HINZE: Sensitivity studies.

MR. JENKINS: -- sensitivity studies --

MEMBER HINZE: Considering the range of uncertainties?

MR. JENKINS: The sensitivity studies were conducted on selected parameters. You know, Dr. Bjorkman talked about those kinds of sensitivity studies. When we talk about uncertainty analysis, we're talking about how probability distributions may vary depending on how they're propagated through the analysis.

So sensitivity studies are typically where you'd take one particular parameter and you would bury that and determine how sensitive your results are, your bottom line results are to --

MEMBER HINZE: I'm familiar with what --

MR. JENKINS: Okay.

MEMBER HINZE: I guess I'm a bit confused. This is a PRA, but in many places, as I understand it, you selected conservative conditions and used those in a -- as a single value.

MR. JENKINS: We selected the best --

MEMBER HINZE: And so is this really a probabilistic risk assessment?

MR. JENKINS: Well, we tried to select best estimate values.

MEMBER HINZE: Well, I heard "conservative" quite often. Perhaps I misheard. I don't know when they are conservative and when they aren't, but, you know, it's a brief presentation.

Let me ask -- this was for a particular site?

MR. JENKINS: Yes.

MEMBER HINZE: What were the criteria that were used to select the site for this analysis? Why was this one chosen?

MR. JENKINS: I believe it was due to the -- having information readily available to start the work.

MEMBER HINZE: I would think that you would have this kind of information available at every dry cask storage site. Were there particular attributes of this site that made it more desirable from a failure standpoint?

MR. JENKINS: No. I don't think there was any bias one way or the other regarding --

MEMBER HINZE: I was trying to -- is this where you had data? Well --

MR. JENKINS: First, you had to have a cask at that particular --

MEMBER HINZE: Yes, okay.

MR. JENKINS: -- facility. Okay?

MEMBER HINZE: Sure, I understand.

MR. JENKINS: And I think it was more driven by the fact that we had design data from the dry cask storage manufacturer. So once you picked that particular design, then you say, "Well, where is it? Where is the facility?" And then, we made arrangements to contact the licensee to allow us to go and, you know, walk down the system.

MEMBER HINZE: One of the things that I was

-- I was surprised to see out of scope issue was this aging effects of fuel during storage. That has a lot to do with CRUD. It has a lot to do with thermal aspects. How sensitive are your results to the age of -- the storage age of the waste?

MR. JENKINS: The report talked about looking at a cask -- I forget the name -- a Victor 21.

MR. MONNINGER: There were -- yes. This is John Monninger from the Office of Research. For the past several years, the NRC has had a research program ongoing up at Idaho National Laboratory, wherein they have taken fuel and opened up casks to look at the evaluation of the fuel.

And the fuel has actually been in very good shape. I don't have the exact reference to the research reports, but this issue on the aging effects of the fuel, aging effects on the dry cask, or dry storage cask systems, was also considered in the staff's license renewal assessment, for example, for the Surry site, etcetera. So the staff has looked at aging effects, but it just wasn't explicitly included within this PRA study.

MR. JENKINS: The particular system I think you're talking about, John, is there's a canister

V/24, and it was like 14 years of storage. And so they pulled it out and examined it, and there was no indication of degradation. So I believe that kind of lends credence. We can't rule it out, but it's -- it wasn't explicitly addressed.

MR. HACKETT: I think if I -- this is Ed Hackett. I think if I could back up our questioning, I think just to try and paraphrase where you're at with the questioning, it's really going to criterion for what was in scope and what was out of scope. And I don't think -- or I think it is fair to say that was not addressed in a systematic way. I think a lot of these were out of scope based on the magnitude of the resources or the level of effort that would be required in certain areas.

One I could speak to, for instance, from my own technical background, when you look at -- the slides not up there, but fabrication and future cask design changes. But just to stick with fabrication, you could probably have spent several years worth of effort going into weld flaw distributions and how they, in turn, might initiate cracks.

There are certain stress events, like Gordon was referring to, and where that might go. It would be a very large effort. And I wasn't involved at the time, but I would have assumed that one of the reasons for excluding that probably were twofold -- one, because of the magnitude of probably -- one, because of the magnitude of the effort; and then, also, when you look at the complexities involved in trying to do this on a pilot sense and getting the methodology down, that piece was excluded. I don't know if that's helpful, but I see where you're going. You're trying to get to a criterion.

MEMBER HINZE: Sure. Sure. One of the things that was going through my mind as Gordon was talking was the effect of corrosion. Both the effect of strain on accelerating corrosion and the effect of corrosion on the strength characteristics, and I gather that's excluded because it's a multiple initiating event. Did you consider corrosion?

DR. BJORKMAN: No, corrosion was not considered -- was not considered in this at all. Typically, when one designs a nuclear powerplant, piping and things like that, a corrosion allowance is included at the beginning. But in these analyses, no reduction in thicknesses of materials was assumed due to corrosion that might occur over time, particularly given that this was -- these were stainless steel casks.

MEMBER HINZE: I'm taking time away from my colleagues. I'll just ask one more question. This earthquake magnitude confused me, 9 to 11 times the design basis earthquake. Are we really talking about earthquake magnitude here? Or are we -- you know, the log of the energy? Or are we talking about 9 to 11 times the acceleration?

DR. BJORKMAN: Nine to 11 times the acceleration.

MEMBER HINZE: Okay. I really think you ought to be very concerned about using earthquake magnitude.

DR. BJORKMAN: Correct.

MEMBER HINZE: That has a very specific meaning. I was quite sure you didn't mean that.

DR. BJORKMAN: No. I mean -- it has nothing to do with moment magnitude.

MEMBER HINZE: Right.

DR. BJORKMAN: Exactly.

MEMBER HINZE: It couldn't.

DR. BJORKMAN: No, it couldn't.

MEMBER HINZE: But you -- that's something you should try to not use, please.

DR. BJORKMAN: All right. Thank you.

CHAIRMAN RYAN: Page 18 and 19. Just

clarification questions. I want to make sure I understand. If you wouldn't mind, just for everybody's benefit, putting it up on the screen.

There we go. The 3.6 times 10^{-4} is a fairly standard reference for cancers per rem of radiation exposure. Is that -- am I understanding that right? What's the 3.6 times 10^{-4}? I'm at Slide 18, right down at the bottom.

MR. JENKINS: I'm sorry. Your question was?

CHAIRMAN RYAN: The question is: what is 3.6 times 10^{-4}. That's the probability of latent cancer --

MR. JENKINS: That's the probability of latent cancer fatality.

CHAIRMAN RYAN: Fatal cancer for an individual.

MR. JENKINS: For individuals.

CHAIRMAN RYAN: Per what? Integrated over an accident or --

MR. JENKINS: Well, for this particular release -- high burnup fuel, fuel and thee release height of 50 meters. I believe there is a certain area that's specified on the table.

CHAIRMAN RYAN: Okay. I'm just trying to

-- and I realize in the interest of time you just summarized that, but I'm trying to figure out, are you calculating doses to one individual? Are you integrating over a population and a sector? How is it done? Is it rem? Is it something else? Can you help me out a little? Thank you.

MS. MITCHELL: Jocelyn Mitchell from the Office of Research. The Max code takes the inventory, the specific inventory released, multiples it times the release fractions, which you heard discussed, takes the population and the meteorology for the specific site, and then transports the plant -- or the plume away from the site.

For that particular number, we looked solely between zero and 10 miles, 16 kilometers, from the site, and then calculated an individual risk from that distance only. The reason that that was chosen was to try to compare with the reactor safety goal.

CHAIRMAN RYAN: Yes, I understand.

MS. MITCHELL: Okay. So it is not a total integrated latent cancers for this accident. If I were doing it again, I would probably choose to quote that number, because it's a lot easier to explain.

CHAIRMAN RYAN: No, I'm with you. And I -- that really helps me understand it. I also just have a little bit of trouble from a fundamentals point of view of taking very small doses, multiplying, and then adding them up, and trying to relate that to cancer. Just -- it's wrong. In spite of the fact we use it a lot, it really is a gross overestimate of cancer risk I think.

MS. MITCHELL: Well, that surely is a subject of discussion, and I know that the ACNW is having a very large meeting, which I wouldn't miss for the world --

CHAIRMAN RYAN: Okay. Great.

MS. MITCHELL: -- later this fall. I think whatever it is, November or something, I will be there --

CHAIRMAN RYAN: My simple-minded analogy is --

MS. MITCHELL: -- to hear the discussion.

CHAIRMAN RYAN: -- I'd rather be hit in the face by a one mile an hour wind for 200 hours than a 200 mile an hour wind for one hour.

(Laughter.)

So low dose or no dose rates really -- and, again, from a relative standpoint -- I'm now on page 19, it sort of washes out. I mean, you can compare different scenarios or different accident scenarios for the absolute values of those numbers relative to one another.

One is 10 times higher or lower, but I just -- I just wanted to make sure I understood that we we're on the page where there is some uncertainty and how that's -- what it really means in terms of absolute values. Thanks.

Ruth?

MEMBER WEINER: Jim?

MEMBER CLARKE: I just had a quick question following up on Dr. Hinze on the out of scope issues. Based on what you learn from this, is there any interest in going back and looking at any of those? I was particularly interested in the last one. Are there any plans to -- uncertainty distribution and propagation?

MR. JENKINS: At this time, I don't believe there is -- we're not going to revisit that particular issue. However, in the future work we'll consider that. The focus of this report was to provide the staff with, you know, sort of a road map on how to do these PRAs. And once having done it, you know, future applications will become easier.

Ed, did you have anything?

MR. HACKETT: Yes. This is Ed Hackett.

Very good question, and I think the answer is, yes, there is definitely interest. The caveat is: are there resources? And are we going to be able to pursue that relative to some of our other priorities?

For right now, as Ronaldo indicated, what we're looking at doing, as far as the user office, the Spent Fuel Project Office, is looking at how this can inform our regulatory approach in a number of areas as you've seen in the report, with an easy example being the inspection effort. So we're focusing on that right now, but there is absolutely interest in that. It's just going to be a question of where we can go with resource limitations for the future.

MEMBER CLARKE: Understood. Thank you.

MEMBER WEINER: You've called this a pilot program. Just to follow up on that, so your intent from here is to go where? Revisit some of these issues, simply use it to inform the regulatory approach as you just said? Where are you going -- what is this a pilot for?

MR. HACKETT: Again, a good question. And the original view was that there would probably be several phases to this effort, I think it's fair to say, wherein this was the first phase and it was a pilot. I think there was envisioning that we would go beyond to address these other items that are out of scope. And as I just said, we may or may not be able to do that, subject to resources.

So our next steps, so to speak, are to go down the path of looking at, what does this mean for us in dry cask storage space from the standpoint of risk-informing the inspection process, the oversight process, licensing, possibly even the regulations themselves, was basically an initiation and a first look for us at being able to do that with what has largely been historically a deterministic approach.

CHAIRMAN RYAN: Why did you use latent cancer fatalities and not dose? Because surely you have to calculate dose before you get to latent cancer fatalities.

Jocelyn? Jocelyn, why don't you stay up here?

(Laughter.)

MS. MITCHELL: As I mentioned, the desire was originally to compare with the reactor safety goals, and they are both expressed in terms of impact, early fatalities, which can calculated zero, and latent cancer fatalities.

CHAIRMAN RYAN: But the basis wasn't the

same. You didn't do it for a whole integrated population, so how do you compare it? I'm sorry. The basis wasn't the same. You didn't do it over the same integrated population, if I understood you right.

MS. MITCHELL: The safety goals are --

CHAIRMAN RYAN: Oh, no. This case versus the reactor case.

MS. MITCHELL: The reactor safety goal, when you compare with the safety goal, you -- the qualitative statement is that the latent cancer fatality risk to the population should be a small fraction of the naturally-occurring, and they define the small fraction as .1 percent, and they define only the first 10 miles, because if you -- for exactly what you said, you have so many cancers naturally-occurring in the huge population that the amount that you would get from this accident would be small. So they look only between zero and 10 miles.

CHAIRMAN RYAN: Oh, and you did the exact same thing.

MS. MITCHELL: Yes.

CHAIRMAN RYAN: And integrated over the whole population.

MS. MITCHELL: No, only between zero and 10 miles.

CHAIRMAN RYAN: In that -- the whole population in that 10-mile annulus.

MS. MITCHELL: Yes.

MEMBER WEINER: I see. Thank you. I would encourage you in all of these to at least go back to dose, because you're just introducing another uncertainty. But that's just a parenthetical comment.

MS. MITCHELL: The problem with dose is that not all radionuclides are the same. So if you talk about some sort of a dose, you have a hard time putting short-lived and long-lived activities on the same, and inhaled versus not inhaled.

CHAIRMAN RYAN: Figure that out to apply the risk.

MS. MITCHELL: Yes, that's correct. Which dose --

CHAIRMAN RYAN: You have to calculate it anyway.

MS. MITCHELL: -- which dose would you --

CHAIRMAN RYAN: Fifty --

MS. MITCHELL: We go on an organ-by-organ basis. Well, for -- for organs we look at the lung and the breast and -- on an organ-by-organ basis for early fatalities. For instance, we look at the red marrow in the lung, and the GI tract to determine -- in this case it happened to be zero. Okay?

But that's the dose we look at. For latent cancer fatalities it's the thyroid gland. What dose went to the thyroid gland? What number of cancers would you get, and what fraction would be fatal? So we add up all those cancers on an organ-by-organ basis.

CHAIRMAN RYAN: Is this methodology outlined in the report, or is it --

MS. MITCHELL: No. You can get the Max reports.

MEMBER WEINER: It is outlined in the Max reports. This is not to say that there aren't -- there isn't controversy over it.

I'm confused as to why you selected certain parameters. Why a 20-year fire, for example? I'm just -- you know, why not, if you're going to do 20 years, why not 10 or 100 or what?

DR. BJORKMAN: The actual selection of the -- the 20 years has to do with a block event. The actual fire duration was from the aircraft fuel, which was a three-hour fire.

MEMBER WEINER: So that was based on the

aircraft fuel.

DR. BJORKMAN: Right. The aircraft fuel was the basis for the fire, and even that was longer than it probably should have been. But, again, it was more extreme than it had to be, but it showed that there were no possible breaches of either the multi-purpose canister or the fuel for a rather severe fire.

MEMBER WEINER: And I'm curious as to, since there was a degree of uncertainty in your input parameters, sometimes more, sometimes less, as to why you didn't use distributions and sample on them. I mean, it seems to me you could have said the value of parameter X is between A and B, and I will assume a certain kind of distribution, or my data looks like a certain kind of distribution. Why so many point values? Why not use distributions?

DR. BJORKMAN: I think that, for example, the -- you know, the example of the fire, I didn't -- I didn't do the analysis, but I know that computationally, if you're going to start to use distributions around -- you know, you're going to have to use distributions around the material properties, you know, obviously, the inputs, the fire, the duration. You would have to use changes in the meshing scheme for the model. That's a variable that has to do with our knowledge as opposed to a random variable. So there would be so many things to vary.

So here, rather, point estimates were made, and one then looks at the result and one says, "If I had begun to chose -- or choose distributions based on all of these parameters, how different a result could I get? And what would be the probability that I could even achieve that result of, say, cladding failures or MPC breach?"

And based upon these point estimate analysis, what it looks like is that even with accounting for distributions for all of these parameters, we couldn't get to the point where even the worst combinations could get us to a failure. And that's really what these point estimate problems begin to show us.

MEMBER WEINER: I can understand that when you don't get to a failure. But you do have a case where you do get to a failure. And you don't have to distribute everything. In fact, you could have simply given the range and reported this as an error bar. And I'm a little bit concerned -- I'm concerned about reading a report like this where there is a single number -- this many latent cancer fatalities per year.

I mean, it seems to me at the very least with all of the uncertainties in the parameters you used there should be a range reported.

MS. MITCHELL: We did look at a sensitivity. If you look at the appendix, I'm not sure that it was actually carried forward into the executive summary or the main body of the report, but the appendix we did consider the value of the source term. So there was what we called the higher source term, which is the number that goes into the two times 10^{-12}, and then used the lower value -- a lower value of the source term for the particulates in CRUD.

MEMBER WEINER: I see. I'm going to --

MR. RUBIN: I'd like to give a little perspective to answer your question. My name is Alan Rubin with the staff. I had been involved with the study early on when this got started. There was a lot of different analysis going over time on this report. The initial scope was to do sort of a scoping study, preliminary pilot study, and then look to see where you're getting some dominant contributors and do a more refined detailed analysis of those dominant contributors.

We did that, and you see the results. The risks are extremely low. To expend staff resources on doing more refined detailed analysis for very low risk was something we had to weigh based on other priorities. And that was kind of a -- sort of an overall decision, where we were going to spend the resources.

We also, in light of earlier studies, had picked some parameters that were much more conservative and came up with some results earlier. We had much longer duration fires, for example, that were assumed in earlier draft studies. And even in those cases, with our sensitivity study, the risk was still extremely low. We have refined the analysis. We had shorter duration fires that were more realistic but still somewhat a little conservative maybe, and each time we did that we got lower risks.

So to spend more resources, detailed sensitivity studies -- you might change the order of magnitude a little bit, but you're still so low beyond other risks that we see normally in reactor studies that it was felt that it was not the most prudent thing to do. So --

MEMBER WEINER: Thank you for that. Staff?

Antonio or --

MR. DIAS: I've got a very quick question. I understand this is site-specific, but what really caught my eyes was the fact that, you know, the whole transfer process has to follow a very specific path. Is this really something that utilities will, you know, follow without ever, ever making any change? I would always expect there is always something on the way and all of a sudden, you know, they have to move it to one side or the other.

And how would that affect your calculation? Your calculation always assumes that it's either a beam or a concrete wall underneath the path that the transfer cask is following. If that was not the case --

MR. JENKINS: Well, my understanding is that this process, this moving the cask, is a very deliberate, very slow --

MR. DIAS: Yes.

MR. JENKINS: -- paint drying kind of process to observe. And the licensee is very deliberate in following every step of the process. Okay? So --

MR. DIAS: This is not something that is in any tech specs. I mean, it's just -- it's there --

DR. BJORKMAN: Actually, what it is is -- and this all --

MEMBER WEINER: Please talk into the microphone.

DR. BJORKMAN: Oh, I'm sorry.

MEMBER WEINER: Okay.

DR. BJORKMAN: This is really something that evolved out of the NRC's document, NUREG-0612, on the control of heavy loads back in the early '80s. And what plants have done because of that is they have basically had to do several things.

Number one, they have to evaluate the consequences of a drop, if they do not use a single failure-proof crane. If they have a single failure-proof crane, they're not required to evaluate the consequences of a drop as far as plant operations are concerned and safe shutdown of the plant, etcetera.

When they do not have a single failure-proof crane, the rigor with which they have to prescribe a load path is very constrained. In other words, they have actual markings on the floor. They get to a certain point, they have certain checks, they have to be no more than six inches above the floor at this point when they start to transport. The rate at which they can move across the floor is determined, so there are basic procedures that they must follow for the control of their heavy loads.

And, you know, I've been away from this for a long, long time, and got involved in the original analyses for drops into the reactor and other kinds of things. But I have not, in fact, written one of these procedures myself, but I know that they are required to have these procedures, yes.

MR. DIAS: Okay. Thank you.

MEMBER WEINER: Are there any other questions? Anyone? Hearing none, we are at the time for a break, and we will come back at quarter past 3:00.

(Whereupon, the proceedings in the foregoing matter went off the record at 3:01 p.m., and went back on the record at 3:15 p.m.)

CHAIRMAN RYAN: If we could come back to order, please. Please take your seats.

MEMBER WEINER: Our next presentation will be from EPRI, Probabilistic Risk Assessment of a Bolted Dry Spent Fuel Storage Cask Revisited. And the presenter is Ken Canavan. Have I pronounced it correctly?

MR. CANAVAN: That's correct.

MEMBER WEINER: It's all yours.

MR. CANAVAN: Thank you very much. Welcome to the last --

MEMBER WEINER: While Mr. Canavan is getting wired up, he is the Senior Project Manager for EPRI, and his main area of technical expertise is risk technology. His experience includes unique applications of risk technology including nuclear power and the aerospace industry.

MR. CANAVAN: Well, welcome to the last presentation of the last day of the ACNW meeting. I guess I will be challenged to both inform and entertain you. I'll try and keep it brief.

Prior to joining EPRI -- a little pertinent background for you, prior to joining EPRI I was employed by Data Systems and Solutions as Manager of Risk Technology there as well, and we were contracted by EPRI to perform the first and second version of this report. So I can't really disclaim much of what is in between those pages in that first I was the principal investigator, and then I joined EPRI and became the project manager.

So it's a little bit hard, but I will mention that we're going to talk about both versions of the report. We're going to focus on the revised version; hence, the title "Revisited." The first version was done in 2002 and completed in 2003.

And as a result of review and comments received on that report, another version of that report was generated to address some of the conservatisms in the study, and that was published in December of 2004. So a little bit of this was me looking back at some of the older materials and preparing for this presentation.

Our outline was to first go through some of our goals. We'll have some slides on methodology overview. There aren't too many, and they aren't that detailed. We'll talk a little bit about the Phase 1 study, the Phase 2 study, show you a little bit about the results, and talk about some of the conclusions and what the industry and EPRI sees as the future uses of cask PRA type technology.

Well, our goals in developing the spent fuel cask PRA were to develop a bolted cask PRA based on transnuclear cask. We knew at the time that the NRC was embarking on doing a welded cask study, so we thought we would look at another vendor, to collaborate with the NRC in some of their work, better understand the risk and consequences of onsite dry cask storage, and to develop some risk insights regarding the dominant contributors and potential cost reductions of cask handling and dry fuel storage.

And the last part, which is in bold, it's the more important part of what we were looking at as an industry, which was to develop the tools required to support a risk-informed framework in the area of onsite spent fuel cask handling, it says transportation. That's probably more appropriately transfer and storage.

As you saw earlier, we're dealing here with the same basic risk equation. Risk is frequency times consequence. We're answering our three basic risk questions. What can go wrong? How likely is it? And what are the consequences of what goes wrong?

For the dry spent fuel storage, the risk problem is, again, divided into three phases. Now, the reason why we divide it into three phases is because some of these questions differ among phases. What can go wrong? might be different in the case of loading or transfer than it is in storage. How likely is it? is certainly different. And certainly, the consequences can vary as well. So the reason for the three phrases is slightly different answers to the same type of questions.

In the area of dry fuel storage, risk is calculated very similar to standard probabilistic risk assessment. And it's using commonly used terms and procedures that are used in the operating nuclear plants. That makes sense since most of the people who work on these studies are taken from that area of expertise and simply work on the cask part.

So our elements tend to be the same. We go through an initiating event analysis, a data analysis, a human action analysis. We look at some success criteria, as you heard of before. It's a little bit different when we talk about casks.

Our success criteria is structural analysis and thermal hydraulic analysis, which isn't really typical in an operating plant, although the thermal hydraulics is, the accident sequence analysis, and then some work on consequences.

Our scope -- some of the items that are not in scope -- acts of sabotage and terrorism. Those are actually covered by other programs. The RAM cap process is a process that's applied to both operating facilities and spent fuel storage, so that's a risk-based approach to looking at dry fuel storage.

We don't look at damage to the nuclear

facility. Again, in most cases, this is handled by another analysis, which is one of the major reasons why it doesn't appear here. For example, it might be handled in the -- either the PRA or other analysis such as the fuel handling and fuel load drop analysis and accidents work that's done at the nuclear facility.

We don't look at worker risk. I'm not sure why we don't look at worker risk, but it's pretty typical. As a former worker, I'm a little concerned about that, but --

(Laughter.)

-- worker risk is typically not included within the scope of risk analysis. We're really looking at public risk, and it's because our metrics are the safety goals, which is public risk.

And, last, we don't look at transportation to the final repository. Again, there is quite a bit of analysis in this area that's being done and being performed as we speak. So this is covered under another type analysis.

Events that are in scope. Okay. We look at the design basis accidents, and we look at the beyond design basis accidents. We look at events resulting from the handling, which would be onsite transfer and the storage, and we look at all types of external events, including seismic fires, high winds, floors, nearby facility accidents, pipelines, aircrafts, and others. And the list includes such things as even meteorites, so it's pretty -- it's a pretty big list.

Okay. In the case of the bolted cask design, we were very careful to make sure that we were performing a realistic estimate of the frequency of occurrence as well as the consequences. And as such, most of the work represents what I would call average cask risk. It's average enrichment, average burnup, and average fuel age.

To give you an example, just one example of the many as you go through the study, a burnup of zero to 25 megawatt days per kilogram of uranium is probably about an eight percent strain. If you look at 25 to about 50, you're looking at a failure at about four percent strain. If you look at items that are greater than maybe 55 megawatt days per kilogram of uranium, you're looking at failures in the area of the strains of one percent.

So when we look at the fuel failing within the bolted cask, we're looking at failures around four percent, because that's an average for the current fuel inventories. Recognizing that reactors are running longer and higher burnups, in the future casks may be loaded with higher burnup fuel. But for now a good average is the average burnup in the range of 25 to 45 megawatt days per kilogram of uranium.

There are several more examples where we strictly look at average risk. They are noted throughout the report.

I included some selected highlights and the methodologies employed, because I thought it might be interesting, even to non-PRA type people. That was our initiating events.

We looked at a combination of generic lists to get to our generic list of initiating events, but we went a little bit beyond that and did a master logic diagram approach, which is a fault tree type -- tree type structure where you go through and you look at what different things can happen to fail different barriers of consideration -- so, for example, fail the fuel and fail the cask boundaries.

The frequency of cask drops was calculated from a fault tree of a typical nuclear power operating nuclear facility refueling building crane. So we took the crane, we divided it down into its pieceparts, assessed failure modes and effects and analysis, and developed a fault tree style approach to assessing that drop. Then, we used that fault tree to assess the various kinds of drops that we could have in our analysis.

We did look at the potential for misloading fuel, so there is some human action type analysis that was performed. Some more selected highlights of our methods employed in the case, the structural analysis for our success criteria. We use a fragility approach.

That approach is significantly different from the finite element analysis that was employed by the staff. In the fragility analysis approach, we were lucky enough to get a hold of some of the design basis calculations for use in this report.

In each design basis calculation we removed the margins of safety that are typically added in those type of design basis calculations, including margins of safety on materials, margins of safety on any of the structural parameters, and created basically a new structural capacity for the cask based on a median set of properties.

Then, we looked at acceleration dependent on target hardness. So there was some previous work done on how hard or soft a target is, and what the acceleration is. And they tell me I should continue to use acceleration, although I always feel it's deceleration when you're dropped. But the acceleration that -- the fuel experience is very dependent on whether the target is hard or soft.

So if you're looking at an asphalt roadway, or you're looking at a compacted gravel roadway, versus something that is 10 feet of steel reinforced concrete, there's a significant difference in the energy that the fuel will see.

So using a combined of these two we can calculate -- we can use the fragility approach, develop a fragility curve, and calculate a probability of the cask value for the different surfaces it won't land on.

Again, for thermal hydraulic analysis, we assume average fuel, average burnup, average decay heat, average storage times.

Accident sequence and consequence analysis -- in our case, we assume there are two fuel pins now for all acceleration events. There is a nice writeup in the report that talks about where that information was derived from. It was derived from previously done work by Sandia where they did a crash into a non-yielding surface, where the fuel experienced about 100g.

We took that and on the basis of how many fuel pins failed we recalculated those numbers back to what we thought the fuel would see for the work that we did, given average burnups.

Initially, in Phase 1 of the study, which was the initial study, we didn't model building -- buildings mitigating release. So we didn't model -- we took it as the refueling building didn't exist. There was a really good reason for that when we did that, but we decided in the future phases to include the HVAC systems that are designed to mitigate releases in the refueling building in the analysis.

Initially, we had assumed a ground-level release. In the first study, we removed that as well and assumed elevated releases where appropriate. And, last, we looked at some source terms -- conservative source term treatment. That was in Phase 1, and we looked at removing that in Phase 2.

We'll talk a little bit about -- more about that later. But before we move too far along, a couple of more interesting highlights that haven't -- well, let's see if they appear on the next slide. Yes. I will say that both Phase 1 and Phase 2 studies rely significantly on literature that was available and published to the team.

So aside from myself there were different people involved at different times in the study, approximately four to five engineers, all with a specific background and a specific item. Some had human action analysis experience. We had a structural gentleman involved with structural analysis and a gentleman who did the thermal hydraulics work, myself as the accident sequence lead and principal investigator.

But each of us brought to bear a lot of the previous work that was done by Sandia, and others, to support some of the work that was done here. But we did study -- in Phase 1 we looked at a bolted cask design. It was performed at a representative BWR. That's a really nice way of saying this is a generic study, non-site specific.

The NRC was a specific study done on a specific plant, and we're generic in that no particular sites modeled, although you'll see significant reflections of both the P and a BWR layout in it. And they might look a little bit like Prairie Island and Peach Bottom. That's where the team went and observed a cask movement, but yet still no particular sites modeled.

Where required, you assume location is the Eastern United States. When I say that, what I mean is when you look at wind hazard or you look at seismic hazard, it's very nice to be able to have a site so you can go get a fragility curve, so -- or go get a wind speed -- information wind speed. So where it was required to get these items they are either extrapolated to an Eastern U.S. site or they are actually from that Eastern European -- Eastern U.S. site.

Some hazards had to be assumed -- natural gas pipeline explosion. The plants that we visited did not have a natural gas pipeline located nearby, but we chose to include a natural gas pipeline in our generic study.

You might ask why. The reason why we did that is because we were trying to make the study generic enough that if someone wanted to take the generic study and make a plant-specific study out of it, that they could see how all of the hazards were handled within the study, and they could decide, "Well, I don't have a natural gas pipeline." It's much easier to remove it than it is to -- for them to go figure out how to include it. So we showed them how to include it, and if they need to remove it they can.

And I already mentioned that the general layout is based on Prairie Island and Peach Bottom. There are quite a few other little things that come in now and then based on a generic site. For example, we don't really know how the site is laid out with respect to nearby airports. So our aircraft crash is based on flyover only.

If you have a specific site, you might look around and find out that three sides of the ISFSI can't be approach by plane. We didn't have a specific site, so you can approach it from all four, which would probably be pretty rare for most nuclear powerplants.

As with all PRAs, we need to perform some simplifying assumptions in order to make the analysis tractable, to be able to perform it. One of those is that word "generic study." Cask loading was assumed to be a two-step process. I won't go into too much detail on cask loading, but with bolted casks it's a little bit different in that the lid is put on before the cask is physically removed from the fuel pool.

So it's submerged, the lid is put on, the cask is lifted as it breaks the surface of the water. Somebody climbs on top and screws down four of the bolts hand-tight. Then the water is pumped out via the drain as the cask is lifted. You don't want to lift it out of the water. You drop below tech specs and the fuel pool water level. So as someone mentioned earlier, the ink-drying thing, that's actually exciting compared to the campaign I saw.

(Laughter.)

So they basically move it two inches, two to six inches out of the water, pump some water out, move it another two to six inches, pump some water out. They're concerned about fuel pool level.

When that's all done, they decon and then move it. While it's still suspended, they decon it and move it over to a preparation area where it's deconned further, it's fully evacuated out, dried, fill gas is put in, the remainder of the bolts are tightened, and then it's ready to go outside.

In that interim, let's assume that they have put it down. They need to pick it back up. Putting down and picking up makes a difference to our fault tree and our calculated probabilities. So we're assuming two steps.

Acceleration-related events -- drops -- are always assumed to fail two fuel pins, not all the fuel pins. That's the subject of some debate because of the stress and strains calculated.

Horizontal drops within the refueling building, and actually even outside, were assigned -- were a high epistemic uncertainty, and, therefore, a higher probability of cask value. Okay. Nice big word -- epistemic uncertainty. All the PRA guys can shake their hands.

Epistemic uncertainty is the sequence of events. Uncertainty of the sequence of events. For example, you drop the cask sideways, what will it hit? What will it land on? When we were looking at horizontal drops within the refueling building, we had assumed that intervening wall underneath the cask, and that intervening wall would create stiffness. That stiffness on a horizontal drop could be problematic in that it was on a small area and focused all of the energy, for example, worst case midline of the cask.

So we assigned a pretty high epistemic uncertainty in this part of the analysis to that probability that we don't know exactly what's -- we're dealing with a generic study. We don't know exactly what's underneath when we drop it. We don't know what they've left in the movement path of the cask. So we were a little concerned of what it might hit.

And as a result of using a higher uncertainty that broadens our 5ths and 95ths percentiles of the curve, and makes the mean move higher. So if you have less uncertainty, with the same parameters you would have a lower mean value.

Building mitigation and potential doses was not modeled. This was because it was not initially modeled in Phase 1. This was because we knew of one utility that did some handling outside. And, therefore, we assumed immediately that, well, we shouldn't model building mitigation. We'll talk a little bit more about that when I get to Phase 2.

Ground level doses were also assumed. Again, if you're not going to model building mitigation, you're probably close to the ground. Limiting weather conditions were assumed.

And I -- for reference I provided the EPRI report number that was completed in 2003. Let's see if you have a nicer laser pointer than me. Okay. You do.

Okay. So Phase 1 was completed in December of 2003, approximately a year after it was started. Phase 2 was begun shortly after that, and it had a slightly different set of goals and objectives. The first one was to reduce some of the conservatives in the Phase 1 study. Lower, more realistic assessment of spent fuel cask risk was desirable, and we wanted to make sure that we had a better comparison with the NRC PRA when it was completed, a more flexible tool for risk-informing regulations and informing the public, and a reduced potential for misinterpretation of the results.

In other words, we didn't want to come out with something and then be saying, "Well, that's actually a little bit higher than it should be." So we went and did the update, which was completed in November of 2004. The update was to revise the cask drop probabilities from NUREG-0612 to incorporate the lessons learned and items in NUREG-1774, to reevaluate some of the uncertainties, specifically the one concerned with the horizontal epistemic uncertainty of the cask.

We wanted to evaluate additional source terms. We initially ISG-5, which was not intended for use in PRAs. We subsequently changed that. We revised assumptions associated with mitigation of releases and aerosol deposition and building HVAC. So we went and said, "If you're handling a building, here's a fault tree of a typical HVAC system. What's its availability, and how much mitigation would it provide?"

We considered elevated pathways for releases from the buildings. We investigated the impact of alternative, more realistic weather conditions. Our initial analysis has pretty much just the right wind speed that if someone were standing in the plume that they got the maximum amount of dose that they could receive. They stood there an awful long time, too.

So we investigated alternative, more realistic weather conditions. We investigated -- we wanted to do a couple of other things, which was investigate intact versus damaged fuel rods. You know, we have tight cracks and pinholes which are generally classified as non-damaged currently and larger defects. And we assumed initially that the fuel that was put into the cask was non-damaged, and that, therefore, took completely intact which is not always the case.

And last was to assess the conservatisms in the storage phase, and look at, you know, 20-year duration, knowing that someone might simply take the year -- if you give them a yearly risk, someone might just take it and simply multiple by 20. Since we were a little conservative, because the number was low, but you start multiplying the conservatisms by 20 and they start adding up.

Unfortunately, Items 7 and 8 were not evaluated in Phase 2.

I should have mentioned earlier, but it was mentioned in the last presentation, that our results are in terms of latent -- both prompt and latent cancer fatalities per cask per year. And in the area of prompt fatalities we have 0.0. The reason why these metrics are chosen is -- again, is because they are very typical of online risk.

And if you start looking at a site and saying, "Well, I want to know what the risk of operation is, the risk of shutdown, the risk of spent fuel storage," you need common metrics. This is a pretty typical metric. So we wanted to stay true to the metrics at least that are typically used.

And you'll notice these are the Phase 1 results and these are the Phase 2 results. The biggest thing to note is that we have a factor of 62-1/2 reduction from Phase 1 to Phase 2. But even Phase 1 had a very low value -- $3.5E^{-11}$ per cask per year is a substantially low number. Most of that came from the loading phase.

If you look, here is the loading phase with a significant fraction, basically 80 percent of the risk. Then, if you look at the storage phase, we had about 12 percent of the risk with this absolute value. And then, the transfer phase made up the remaining eight percent.

When we took a look at some of those conservative assumptions that we had, Phase 2 came out and said, okay, well, we're still at zero prompt fatalities, but the total cancer fatalities go from $3.5E^{-11}$ per year to $5.6E^{-13}$ per year. And if you'll notice, one interesting thing happens.

This is now the loading phase, as opposed to that. So there's a -- most of the reduction takes place in the cask loading phase. and if you think about it most of our conservatisms were related to the cask loading phase, right? They were building mitigation ground-level releases and the horizontal epistemic uncertainty. So that gave us a very different picture of the risk and said, "Hey, you know, cask loading is still a significant fraction, though. I don't want to throw it away." It's still 11 percent, but it dropped significantly.

Storage came up and transportation -- the transfer also becomes a larger fraction, although all of the absolute values are a little bit lower.

Okay. Let's talk about some sequences. In Phase 1, on the left-hand side of this graph, is the Phase 1 of the project results, and on the right-hand side it's Phase 2. And if you look, initially Phase 1, number one accident sequence -- if this is hard to read, it should be decent to read in your handouts hopefully -- that's the on-edge or horizontal drop. And it says -- easy to read on my screen. It says during loading. That's what in the brackets. That's the loading phase.

Then, we have the refueling building failure, another horizontal drop, but this is during transfer. These two are a function of the larger uncertainty that we've spoken about. The next one is heavy loads exceed the structural limit. This is a first year only. It's a function of the assumed frequency of the high winds. So dependent on location.

And again, this one, which is the high temperature, is assumed a function of the distance from some of the fixed hazards. So a gas line -- you know, we assumed a gas line. There are several others that contribute, but they're all the result of assumptions of this generic site. And the last one is the high temperature fire during transfer.

Okay. In the second one, the top sequence is the high temperature fire during transfer. So this one right down here is now here. And then, heavy loads exceeding structural limit, the high temperature -- temperature and forces during storage, that's the assumed hazards.

The on-edge drop during transfer, the refueling building failure, which is both random and seismically induced, and then the last, cask impacted by missiles. And I can give you some details on each one of those initiating events. I wrote it down, so I'd get them right.

In this case, this high temperature fire during transfer is a transporter fire. We all know that occasionally vehicles catch fire. In this case, one of the transporters we were looking at had very large wheels. They were rubber. Rubber burns nice and hot and for a long time.

Some of the other transporters we knew were tracked, but in this particular case we noticed this one. We did note it in the combustible loading, that this was a function of the type and size of a vehicle. If you look at a tracked vehicle, this number might be significantly different.

Heavy loads exceeding the structural limit -- this is floods, tsunamis, wind, seismic. This high temperature force during storage is the fixed and non-fixed transient sources. The on-edge drop during transfer is the horizontal drop. The refueling building failure we spoke about is the seismic and the random failures. And the last one is actually missiles, which are wind, flood, and a meteorite is I believe included in that list.

Let's talk about some conclusions. The Phase 1 project conclusions was that there's a pretty low risk for the bolted design dry fuel storage systems. We felt that in general it might apply to all design systems. It's driven by a relatively small number of key assumptions as well as site-specific hazards. So if you should happen to be sitting next to a liquid natural gas plant, you might have a different set of site-specific hazards, but in general it's a very low number.

The use of a risk-informed approach could achieve both cost and safety benefits. So we came to the conclusion that a risk-informed approach could be beneficial in this area.

So then we did Phase 2, and we confirmed the low risk for the bolted design and even found some areas that could be improved upon. We showed that the risk is, again, still driven by a small number of assumptions in plant specifics, although we think that plant specifics are more related to seismicity and weather than they are to near site facilities.

We thought additional analysis was only warranted if the cost benefit could be justified through a burden reduction. At this point, the risk is so low when compared to the operating risks, if you consider the site as a whole, putting money into doing additional analysis or making this generic analysis plant-specific is not really warranted unless you can justify it on a beneficial basis.

The use of the risk-informed approach to dry fuel storage, though, could achieve, if used correctly, both cost and safety benefits.

So what are some of the future uses of the cask technology? Well, to improve public perception of spent fuel storage options. Cask storage is a very low risk activity. There were some other things. Going through the literature, maybe you look at performing a risk tradeoff of analysis between repairing versus just leaving it as found.

If something, for example, is slightly

above the design thermal loading of the cask, you might look and say, "Well, you know, it's really not worth lifting it up, transporting it back inside, taking out some fuel assemblies, putting in some fuel assemblies," and retransporting it outside, because the risk of leaving it as it is versus moving it is -- it's a better situation to leave it outside.

Enforcement discretion for discovered deficiencies, identify areas for reduced margins in future cask designs, it is interesting that drop dominates some of these -- some of the areas of transport. Dropping is close -- is a function or at least partially a function of weight. If you can reduce weight you might reduce situations where drop is a problem.

Identifying reduced burdens associated with regulatory and environmental requirements -- so you might be able to increase allowed boundary doses or reduce inspections, something that was mentioned earlier. And then, lastly, review regulations to assist in licensing of new storage or expansion of existing facilities. Again, it's a low risk activity, and some of the effort that goes into the licensing of it might be better served if it was applied somewhere else.

MEMBER WEINER: Thank you very much. We'll start at the other end with questions. Dr. Clarke?

MEMBER CLARKE: I guess just a couple of things to clarify. The metrics are the same in both studies, is that correct, or --

MR. CANAVAN: That's correct.

MEMBER CLARKE: If I recall correctly, the prior study incorporated human factors indirectly through the data. Do you get into that at all, or --

MR. CANAVAN: We have a separate -- we incorporated human actions directly as a function of human action analysis. So there was actually human action analysis performance tests. For example, we did look at corrosion, and as part of that we looked at the introduction of the wrong gas, introduction of liquids.

We looked at the handling procedures that they use around the cask for those types of items. And there was actually human performance analysis done by looking at the procedures and the steps in those procedures and determining whether or not mistakes could be made at various steps. And so there was the specific handling of human actions.

MEMBER CLARKE: And both of you came up with very low risks.

MR. CANAVAN: I meant to point that out. I had another presentation where I stuck in a little bit of slides the similarities and the differences. There is a factor of 3.6 difference between the first year calculated by the NRC and the EPRI report. And at this level of resolution, those are identical numbers.

Matter of fact, I am amazed that the numbers are as close as they are, given the different designs, given the different approaches that were taken in several areas. While the overall methodology remains similar, there's a lot of things that go on in the details that can easily affect a number. And 3.6 is spot on. I don't think we could do it if we tried, and it did happen relatively independently.

And I'd also note that storage is exactly the same -- $1.9E^{-13}$. That is the same number.

MEMBER CLARKE: Thank you.

MEMBER WEINER: Dr. Ryan?

CHAIRMAN RYAN: No additional comments. Thanks.

MEMBER WEINER: Dr. Hinze?

MEMBER HINZE: Is your work, especially on the storage, transferable to the aging pad at Yucca

Mountain with the proper seismic and meteorological conditions?

MR. CANAVAN: You're not the first to ask that question. I believe it is substantially applicable to Yucca Mountain.

MEMBER HINZE: When you considered some of the potential far-out factors, did you -- would you consider volcanic ash that has come from a remote volcano as a factor in analysis of the cask?

MR. CANAVAN: The TN bolted design does not rely on that, so we did think about it and dismissed it based on it would have to remain totally covered for a substantial period of time.

MEMBER HINZE: Totally covered.

MR. CANAVAN: Totally covered.

MEMBER HINZE: Okay. Very good. I gather that from NRC's work and EPRI's work that there is no difference between a bolted and a welded covered cask?

MR. CANAVAN: Each design has some advantages and has some disadvantages. Since I have never been in the operational aspects of welding a top on versus bolting a top on, I will say from the risk perspective the tradeoffs seem about even.

MEMBER HINZE: Thank you very much.

MEMBER WEINER: Why two fuel pins? Why not five? Why not all of them?

MR. CANAVAN: Actually, on page H4, so you can see I prepared for this --

(Laughter.)

On page H4, Sandia did an analysis where they took a cask with I think PWR fuel and accelerated the fuel and had it hit a non-yielding surface. The fuel inside experienced about 100g. They had a certain amount of fuel failures that occurred in that test.

What we did is we took that test, and we took the forces that the fuel experienced, and we translated that to our fuel, which was four percent -- approximately an average of four percent strain. And then we looked at how many fuel pins do we think would -- based on the stresses that they would see would exceed that strain. And we came up with a very small fraction, something like $2.7E^{-4}$. We took that and we multiplied it by the number of pins and came up with about two.

MEMBER WEINER: You certainly did prepare for that question.

(Laughter.)

That was very good.

What went into your particular choice of -- let me ask the question the other way, another -- a more general question. Did you correspond or communicate at all with NRC to have some comparison between the two analyses?

MR. CANAVAN: Well, let's see. Yes. But the communication was intended to be more frequent, but what ended up happening is we had some early communication where I did the site drop-in up here. We shared some -- shared some early information. After that, the EPRI schedule was quite aggressive, and I was a paid contractor at the time, paid to meet schedule milestones. And our work quickly got ahead of the NRC. So at that particular time we didn't share much more, so I do think the efforts are relatively independent.

MEMBER WEINER: Does anybody from NRC want to comment?

MR. RUBIN: Yes, let me comment. My name is Alan Rubin. I was involved at the beginning of the study where there's initial interactions with EPRI, basically the methodology of identifying initiating events, and I think there are many similarities in that. We had an early start.

We had initiating events identified. I

think EPRI had meetings with us, and there was an intent to share more information. Because of the unavailability of the NRC's report to be publicly available, that was not -- we couldn't do that. We limited the meetings to what we could discuss. And until a public meeting such as this, when we could share documents and review and compare, the interactive discussions were more limited.

MEMBER WEINER: Thank you. Does anybody on the staff have questions? Antonio?

MR. DIAS: It's very interesting the numbers come so close, because you have a boundary that's about 300 meters, isn't it? Between 100 and 300 meters. That's the boundary for the public that you assume.

MR. CANAVAN: Yes, that's correct.

MR. DIAS: And I didn't see in your presentation -- do you go into a very elaborate model for release fractions or not? How did you address release fraction?

MR. CANAVAN: Yes. There's a pretty elaborate --

MR. DIAS: Okay.

MR. CANAVAN: -- model for release fractions. We don't use the Max code substantially, so we're not looking at what is the population around the site, because we couldn't. So we put our member of the public at the site boundary and made him stay there --

MR. DIAS: Okay.

MR. CANAVAN: -- until the release passed him.

MR. DIAS: Okay. Thank you.

MEMBER WEINER: So you basically calculated the reasonable and maximally exposed individual, or just the site --

MR. CANAVAN: At the site boundary.

MEMBER WEINER: Yes.

MR. CANAVAN: Yes.

MEMBER WEINER: At the site boundary.

Anyone else have any comments, questions? Come up and identify yourself, please.

MR. MALSCH: Yes. Marty Malsch. I'm with a law firm that represents the State of Nevada. I just had two clarifying questions. One is, did your PRA include consideration of errors in the fabrication of the cask or canister?

MR. CANAVAN: A commonly-asked question. Yes, I would say that it does, because when you use the fragility approach to assessing, for example, cask drops you assess an average strength of materials. So you're looking at an average. And then, uncertainty is applied to that average in terms of both epistemic uncertainties and randomness uncertainties.

In the case of randomness uncertainties, they incorporate things like strength of materials and other properties that could be random throughout. Could there be a flaw? Could there be a partial flaw? Could there be a manufacturing problem? All those come together to produce the mean value of the cask. So the short answer to the question is I believe they're in there.

MR. MALSCH: Okay. My second question is: in looking at aircraft crash risks, what kind of aircraft did you assume, and what did you assume was the aircraft crash probability? I'm trying to guess because your slides say you associated the study with a typical site in the Eastern U.S., and I was guessing what you might have assumed by way of aircraft and crash probability, but I wasn't sure.

MR. CANAVAN: I want to be careful and not misspeak and give you a probability that I am -- that I don't know off the top of my head. But I will say it looked at the random -- the statistics from the

FAA on random failures per -- the typical random failures per hundred square miles and looked at ratio in that area and to the approximate area of what an ISFSI normally consists of. It might have even been a little conservative on that, because I think if you actually do that number it's a really small one.

And it was a larger -- for the purposes of doing cask impacts, it was an extremely large plane. I believe -- and it is cited in the report, I'm going to say a 757. It's a big plane, but it -- and the engine sizes are all there, and the fact that the hardest parts of the plane are the engine shaft and the wheels. They're all -- that's all accounted for as well as the fire, a resulting fire. And conservative bounding analysis is done in a lot of that case.

MR. MALSCH: Just to point out, you mentioned earlier that you thought your study was applicable to Yucca Mountain. Just to point out that on initial analysis DOE has concluded that the probability of an aircraft crash at the site -- I'm not sure what the footprint was, but at the same from military aircraft associated with a nearby test and training range, flunked the NRC criterion of 10^{-4} per year.

So the aircraft crash probability for Yucca Mountain is likely to be considerably higher than the typical aircraft crash probability associated with overflights in Eastern U.S.

MR. CANAVAN: Yes, that could be true.

MR. MALSCH: You should be careful about whether this aspect of your study is directly applicable to Yucca Mountain.

MR. CANAVAN: Yes. When I said it was directly applicable to Yucca Mountain, I would never assume that the site-specific values were directly applicable. I will say that the study did look at large military aircraft, by the way. It looked at air taxis, large aircraft, and small aircraft. So it does -- it did look at the range of our aircraft.

But I wasn't insinuating that all of the values -- for example, the study looks at a natural gas line being located next to this particular ISFSI. I assume there aren't a lot of natural gas line at Yucca. So we'd have to look at some of the items that are in the study and decide whether or not that they need to be considered for that risk or not.

MR. MALSCH: Okay. Thank you.

MEMBER WEINER: Is there anyone else? Yes?

MR. ABBOTT: Hi. My name is Ed Abbott with

ABZ. If you were talking to a member of the public about this, would you consider these events credible from a public health and safety perspective?

MR. CANAVAN: That's a good question. Ed doesn't remember me, but I worked for GPU many, many years ago, and we met several times. I would say that some of the -- we took an approach of trying not to screen. There is the word "screen" used very rarely in this report. My intent, since it was generic, was not to screen when we did the analysis. My intent was to be additive.

So when you look at missiles, we looked at anywhere from wind-produced missiles all the way to a meteorite. I was actually surprised how non-rare a decent-sized meteorite is, but it's still probably not -- it might be on the verge of non-credible. The idea would be to add up those hazards, use them as the initiating event, that being sort of a bounding value, but not conservative because it's calculated on the individual pieces.

Then, we didn't throw anything out. So if somebody suddenly feels that they have a reason for changing the wind speed or there -- you know, there's a meteor shower coming by and it's going to affect that. They could adjust the values in the study and take the generic to specific.

So the short answer to the question is individual initiators might be non-credible. But if they are, they shouldn't have impacted the total that we looked at very significantly, because the more credible hazards should dominate.

Did I answer your question, or was that too much tap dancing?

MR. ABBOTT: That's okay.

MR. CANAVAN: Okay.

MEMBER WEINER: Any further questions? Anyone? Hearing none, I'll turn the meeting back over to the Chairman.

CHAIRMAN RYAN: Thanks very much, and I'd like to thank all our participants and speakers for this afternoon session on two very informative presentations on work done in separate places by separate people and showing similar results. It's always interesting to see that.

With that, I believe we are at the end of our agenda for presentations. I think we've got a brief bit of business for the committee to discuss, potential letters for the rest of the day, whether we will or won't write them. Beyond that, we're finished.

I want to suggest for folks that do want to participate in the last part that you do that. But other folks that may want to leave, we'll just take a short five-minute break and the reconvene.

(Whereupon, the proceedings in the foregoing matter went off the record at 4:05 p.m. and went back on the record at 4:16 p.m.)

CHAIRMAN RYAN: Okay. We're ready to go, so we'll go on the record.

I think we just need to cover one bit of business for the end of today's activities, and the question is: will we have letters on today's activities, which would include, first, the advanced fuel information that we heard in two briefings this morning.

VICE CHAIRMAN CROFF: Not yet.

CHAIRMAN RYAN: Not yet.

VICE CHAIRMAN CROFF: We want to wait for the White Paper.

CHAIRMAN RYAN: And I think with the White Paper under construction by Ray and colleagues that it's best to integrate that into that White Paper. So, and the information we heard, while very informative, is generic and early on.

VICE CHAIRMAN CROFF: Right.

CHAIRMAN RYAN: And that's a good place for it. Okay. That's fine.

The standard review plan for waste determinations -- I think from yesterday we agreed we want to modify the current draft that we read out late yesterday. Right, Allen?

VICE CHAIRMAN CROFF: Right.

CHAIRMAN RYAN: And then, the two briefings this afternoon on the dry cask storage -- first, the RES presentation, second the EPRI presentation.

MEMBER WEINER: What I would like people to do --

CHAIRMAN RYAN: Well, before we ask people to do stuff, I'm curious what the letter would focus on and what we would be reporting on the information.

MEMBER WEINER: Well, I think we need to report that we -- on these two studies and the differences, the similarities, a number of the questions that we had about -- particularly about the NRC study, number of the suggestions that were made as to how it could be improved, and I -- if no one has any comments, then we could just write a very general letter. But my guess is, just from the comments that I heard, that everyone has some comment to make on the letter.

And out of that I would guess we could get some recommendations. One recommendation is that this was a pilot study. I'd like to see a final -- a study that is not a pilot study, that is more generic.

CHAIRMAN RYAN: Jim?

MEMBER CLARKE: I think she's asking us to send her what we would put into a letter if we write a letter. Now, can we take that approach, or do we have to decide to do --

CHAIRMAN RYAN: Well, I guess I'm reaching -- now that it's fresh in our minds -- and, again, I'll hold my views until the end, but what would be the main conclusion or the main recommendation, or where are you leaning? I mean, we had I think a productive dialogue and understanding what's in the reports.

But here -- and I'm just offering a comment. We have two reports, two different approaches on slightly different but similar casks and similar purposes and endpoints. And in spite of my stumbling through how the risk calculations are done, just not having as much familiarity as I perhaps should, we end up with what by all reckoning relative to anything are extremely low probabilities.

So I wonder what it is we're going to say. And I guess, frankly, I take up the point that was made by one of our presenters that, does it make sense that we spend the time, money, and effort on such low probabilities and refining and fine-tuning? So I'm challenging us to think about, does this rise to the point where we have something terribly substantive to add?

Now, I think we did have good dialogue on perhaps things that could be better clarified, better stated, clearer, crisper definitions, and things that, like I said, I stumbled through. I just wonder what it is we're going to report.

MEMBER WEINER: I think one of the things worth reporting is that there were two quite different, uncoordinated approaches, and they come up with very similar risks.

CHAIRMAN RYAN: And very low risks.

MEMBER WEINER: And very low risks. And within -- well within an order of magnitude of each other, and that I believe is significant, because this is an area that the public does look at.

CHAIRMAN RYAN: And I think if that's the main conclusion, and then the observation is there are a number of points discussed, and, you know, these are listed in the appendix for the benefit of the authors to consider as they finalize and review documents, and so forth, that's about as far as it goes.

I just want to leave with a little bit better structure of what we were talking about here if we're going to write a letter.

MEMBER WEINER: Fine.

MEMBER HINZE: I think, if I might --

CHAIRMAN RYAN: Bill, please. Yes.

MEMBER HINZE: I think Ruth said the magic words there. There's a lot of public interest in this. And I think it's very important. I'm very impressed that they came up with similar values with two different types of canisters, and they are low values. I think this is going to be of interest to everyone.

CHAIRMAN RYAN: You know, and one point that struck me is after I sorted out that all of the probabilities that I was asking about were conditional, it turns out the real driver is the frequency of the accident. That's the driver.

MEMBER WEINER: And that's --

MEMBER HINZE: The seismic activity.

MEMBER WEINER: Yes.

CHAIRMAN RYAN: Right. So there's a couple of things we could observe for the benefit of trying to translate it into, you know, a different kind of a summary for our own purposes. But that's where I think the letter ought to go. It's not to say things ought to be thrown out, or it's not good, or it's just, you know, here are some interesting observations from the two sessions, and the one conclusion is the probability of impacts are pretty low. So --

MEMBER HINZE: Put a positive spin on it.

MEMBER WEINER: Yes.

CHAIRMAN RYAN: Well, I don't think we spin it either way. I think we simply say what we reported.

Allen, any thoughts?

VICE CHAIRMAN CROFF: I think we should give it a try. The point on the public is public interest is well taken, and I think there is pretty clearly an interest on the part of one Commissioner, since he took the time to come down and listen to it himself. And I think he -- I think it's worth trying to put our views down.

CHAIRMAN RYAN: Okay. All right, good.

I'm just -- I'm glad we focused it up a little bit to help Ruth --

MEMBER WEINER: Thank you.

CHAIRMAN RYAN: -- shape it up a little bit more.

MEMBER WEINER: May I say one more thing? I'd like to have a draft that we can -- that would be final by the August meeting. I think that was your intent, wasn't it?

CHAIRMAN RYAN: That's up to you.

MEMBER WEINER: So if you're going to send me comments, please send them in a timely fashion.

CHAIRMAN RYAN: Okay.

MEMBER WEINER: Otherwise, I'll ignore them.

CHAIRMAN RYAN: Okay. That concludes our review of what letter-writing we had not discussed. Are there any other items? Hearing none, the meeting is adjourned.

(Whereupon, at 4:23 p.m., the proceedings in the foregoing matter were adjourned.)

CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of:

Name of Proceeding: Advisory Committee on Nuclear Waste 172ND Meeting

Docket Number: n/a

Location: Rockville, MD

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and, thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

Lindsey Barnes
Official Reporter
Neal R. Gross & Co., Inc.

The Global Nuclear Energy Partnership

Global Nuclear Energy Partnership

Greater Energy Security in a Safer, Cleaner World

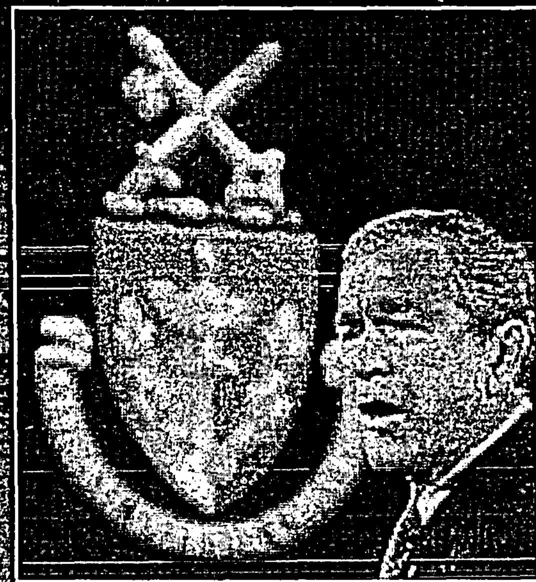
Advanced Separations Technology Development

James J. Laidler
Argonne National Laboratory
National Technical Director
Separations

ACNW July 20, 2006

Key GNEP Program Elements

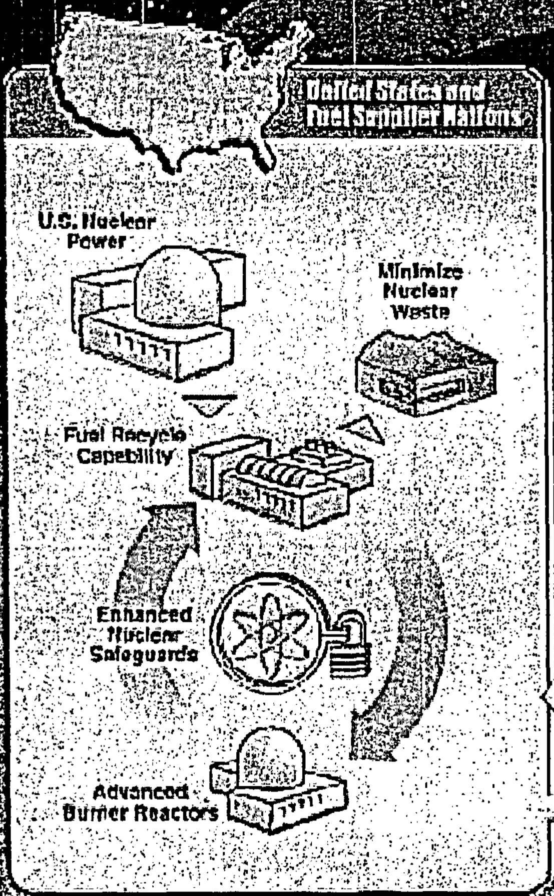
- Expand use of nuclear power
- Minimize nuclear waste
- Demonstrate recycle technology
- Demonstrate Advanced Burner Reactors
- Establish reliable fuel services
- Demonstrate small, exportable reactors
- Enhanced nuclear safeguards technology



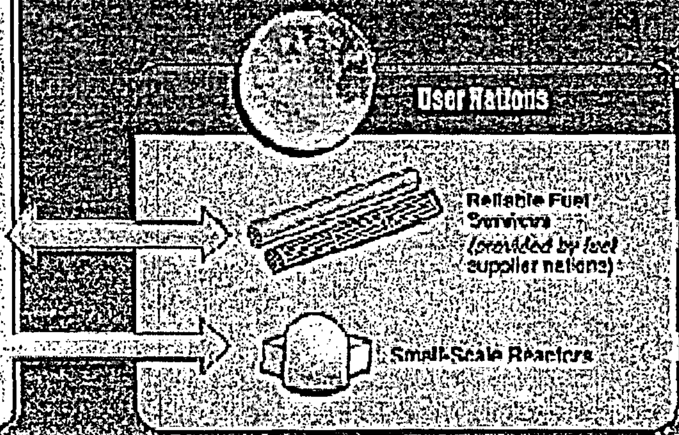
"To build a secure energy future for America, we need to expand production of safe, clean nuclear power"

President Bush, 06/2004

Reliable Fuel Service Model



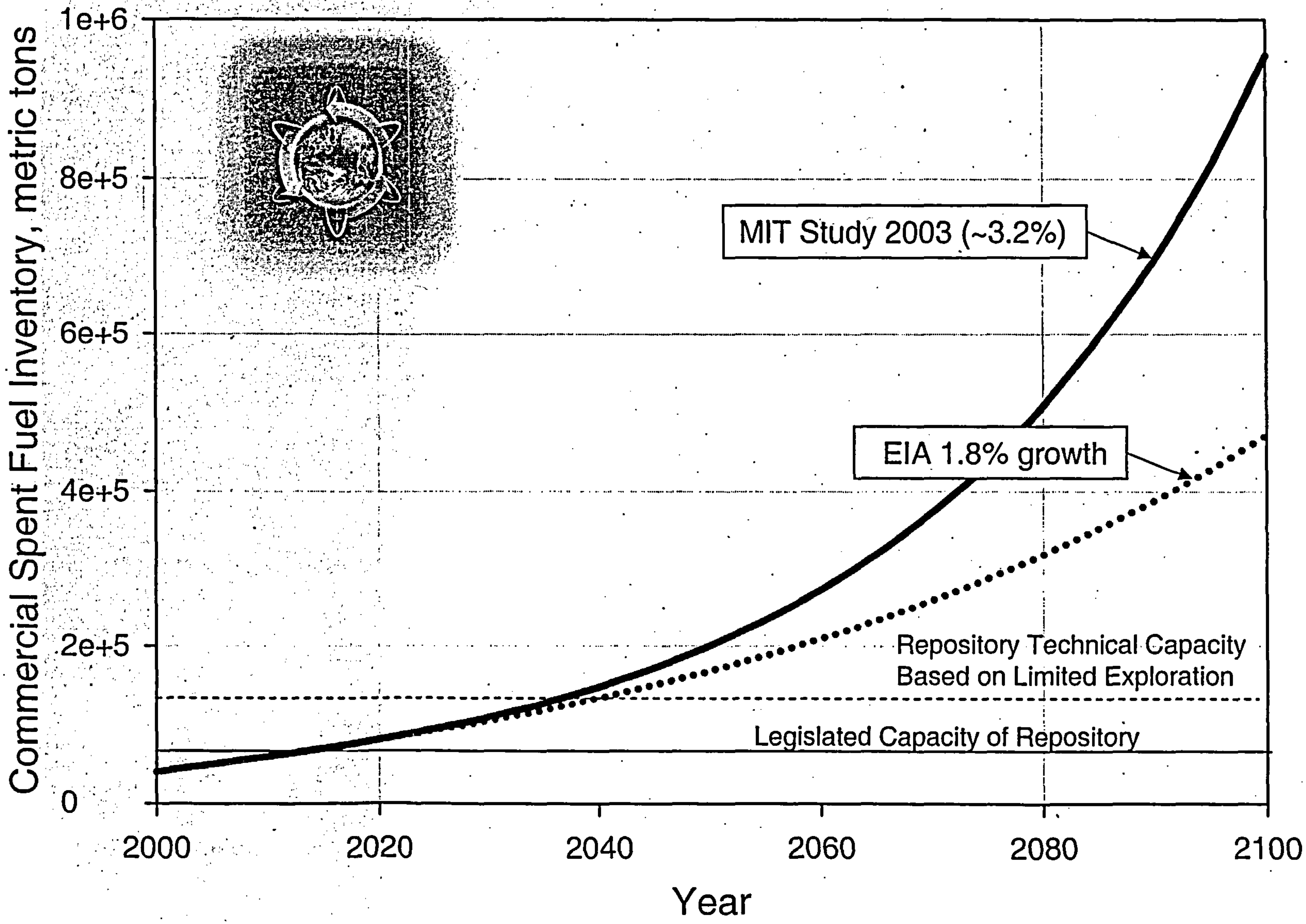
- Expand nuclear energy while preventing spread of sensitive fuel cycle technology
- Fuel Cycle Nations – Operate both nuclear power plants and fuel cycle facilities
- Reactor Nations – Operate only reactors, lease and return fuel



GNEP Benefits

- Reduce America's dependence on fossil fuels
- Provide abundant energy without generating carbon emissions or greenhouse gases
- Recycle used nuclear fuel to minimize waste and curtail proliferation concerns
- Safely and securely allow developing nations to deploy nuclear power to meet energy needs
- Assure maximum energy recovery from still-valuable used nuclear fuel
- Reduce the number of required U.S. geologic waste repositories to one for the remainder of this century

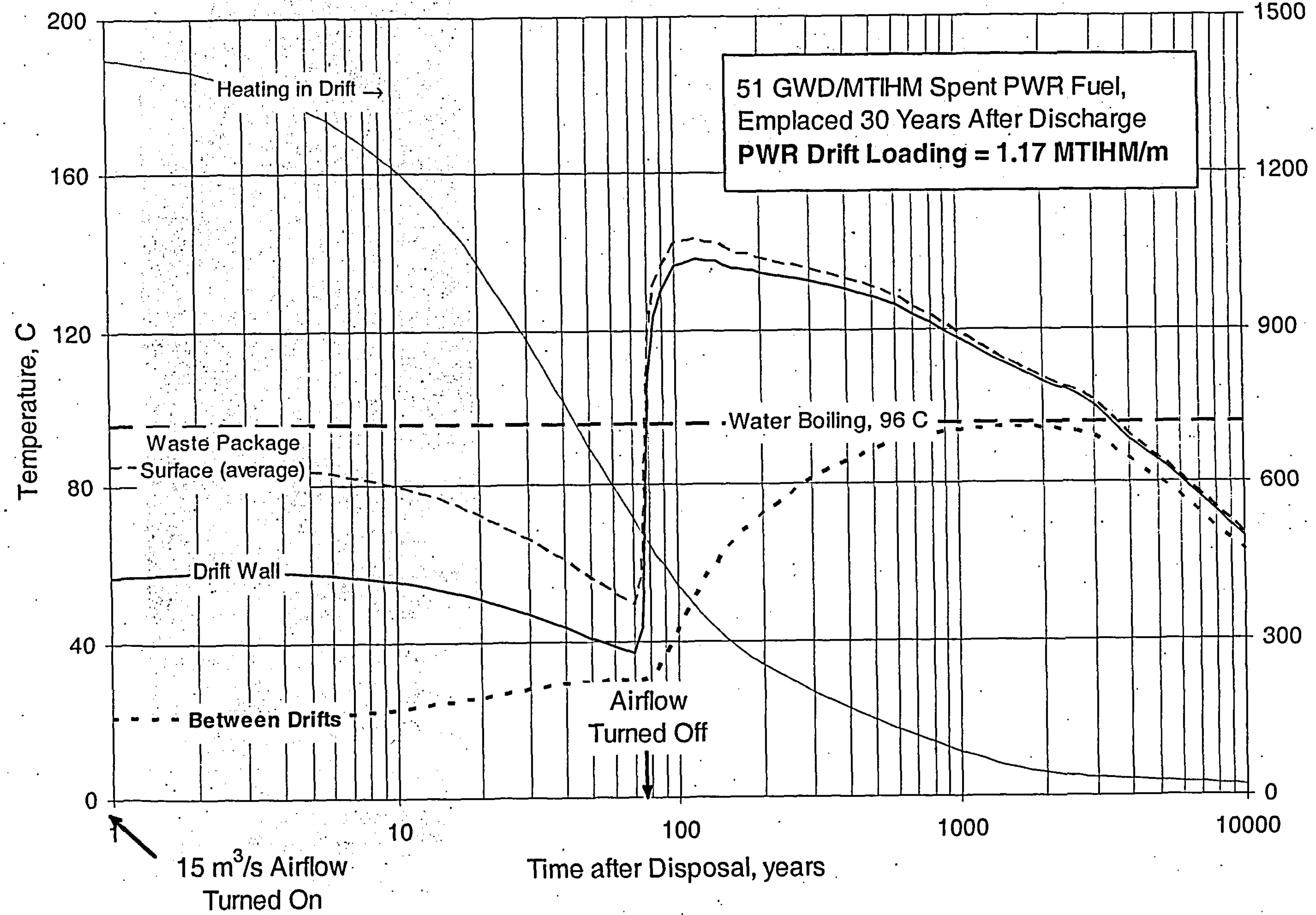
Projected Spent Fuel Accumulation without Recycling



Yucca Mountain Design Assumptions

- Certain design assumptions were made, in consultation with YMP staff, in order to assess the repository benefits of spent fuel processing
- The two controlling criteria deal with the management of radioactive decay heating
  - The rock temperature midway between adjacent waste emplacement drifts (81 meters center-to-center) must not exceed the local boiling point of water (96°C)
  - The temperature at the wall of the 5.5-m diameter drifts shall not exceed 200°C

Yucca Mountain Reference Case



GNEP Separations Technology Development

- Two main program paths
  - Management of the spent fuel generated by current generation LWRs and future ALWRs
  - Closure of the fuel cycle for advanced burner reactors
- Near term objective is dealing with the large amount of spent fuel generated by the current fleet of commercial power reactors
  - Spent fuel is currently generated at a rate of about 2,000 metric tons (heavy metal) per year
  - Projected accumulation of commercial spent fuel will greatly exceed the legislated capacity of the Yucca Mountain repository by 2050
  - Goal is to preclude or significantly delay the need for a second geologic repository

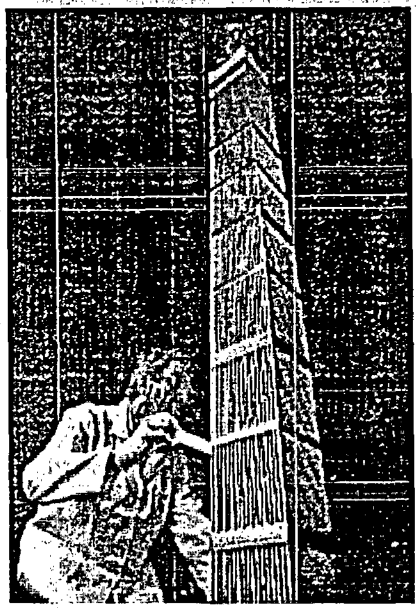
GNEP Separations Technology Development

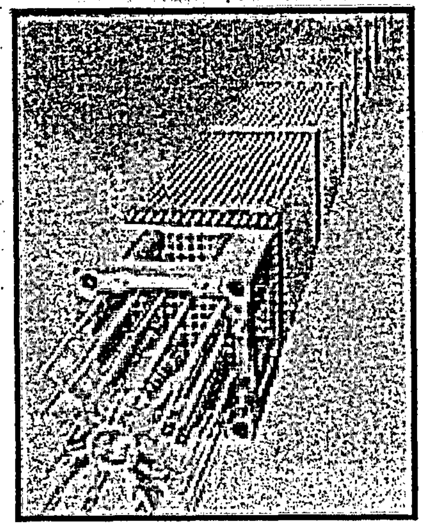
- Longer-term objective is closure of advanced burner reactor system fuel cycle
  - Economic sustainability
  - Assured fuel supply
  - Support high efficiency of transmutation
- Both aqueous and non-aqueous spent fuel treatment processes are being developed
  - Because it represents a near-term issue, the more technologically mature aqueous methods are being developed for the treatment of commercial oxide fuel
  - Advanced burner reactor fuel is likely to be a sodium-bonded metallic fuel and may be more amenable to treatment with non-aqueous methods
  - Overriding concerns are with fuel cycle economics and protection of nuclear materials

Advanced Aqueous Processing of LWR Spent Fuel

- Aqueous solvent extraction process is the reference method being developed for treatment of LWR spent fuel
  - High degree of technological maturity; industrial practice in France, the United Kingdom, Russia, and Japan
  - Capable of achieving high decontamination factors for separated products
    - For thermal recycle, must eliminate high cross section fission products nearly completely (DF>10,000 required)
    - For fast reactor recycle, must reduce the rare earth content (DF>250 for separation of lanthanides from transuranics)
  - Provides flexibility in degree of partitioning of radionuclides present in spent fuel
    - Emphasis on group extraction of transuranics to confer a degree of proliferation resistance to the process
    - Suite of process variants under development

Pressurized Water Reactor Fuel Assemblies





Typical PWR Fuel Assembly:
Length: 13.5 feet
Weight: 1400 lbs. (636 kilograms)
No. of fuel rods: 264
Active fuel length: 12.7 feet
Heavy metal content: 936 lbs. (425 kilograms)
Fuel composition: UO_2 (482 kilograms)
Hardware: 154 kilograms

Composition after Irradiation:
Uranium: 398 kilograms (93.6%)
Plutonium: 3.9 kilograms (0.9%)
Neptunium: 0.5 kilogram (0.1%)
Americium: 0.5 kilogram (0.1%)
Curium: 0.07 kilogram (0.016%)
Fission Products: 22.5 kilograms (5.3%)
[50,000 MWd/t burnup]
Hardware: 154 kilograms

Radionuclide Content of LWR Spent Fuel Important to Process Design

- Uranium
  - U total = 9.22E+05 g/MTIHM
  - U-236 = 5.27E+03 g/MTIHM (~0.6% of total U)
  - U-234 = 1.86E+02 g/MTIHM
  - U-232 and U-233 = 1.07E-02 g/MTIHM
- Krypton
  - Kr-84 (stable) = 205 g/MTIHM
  - Kr-85 = 19.2 g/MTIHM
  - Total krypton = 6.6 L/MTIHM (10 atm)
- Xenon
  - In 10-year cooled spent fuel, all xenon isotopes are stable
  - Total xenon = 172L/MTIHM (10 atm)
- Radon
  - Total radon isotopes (10-year cooling) = 1.2E-11 L/MTIHM (STP)
- Carbon-14
  - C-14 = 0.28 g/MTIHM
- Tritium
  - Tritium = 0.62 L/MTIHM (STP)
- Transuranics
  - Plutonium = 1.43E+04 g/MTIHM (85.8% of TRU)
  - Americium = 1.06E+03 g/MTIHM (6.4%)
  - Neptunium = 1.12E+03 g/MTIHM (6.7%)
  - Curium = 1.83E+02 g/MTIHM (1.1%)

- Technetium
  - 1.25E+03 g/MTIHM
- Iodine
  - 4.24E+02 g/MTIHM

Radionuclides Contributing to Offsite Dose

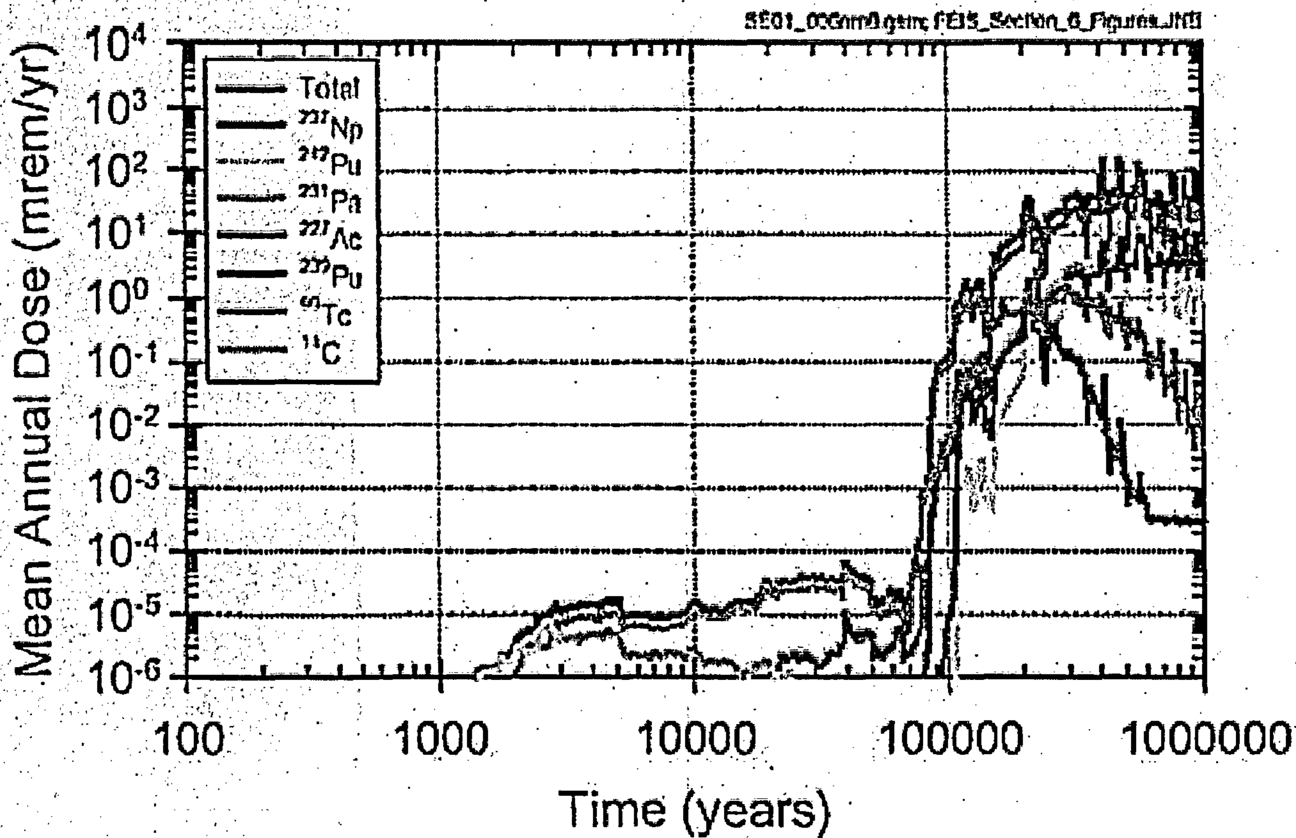


Figure 6-4 Total Mean Annual Dose and Mean Annual Dose for Radionuclides Contributing to Total Mean Annual Dose from the 300 Probabilistic Simulations for the HTOM for the 70,000-MTHM Inventory at the RMEI Location Downgradient from a Yucca Mountain Repository

Suite of UREX+ Processes

- UREX+1, UREX+1a are intended for fast reactor recycle of the transuranic elements
  - UREX+1 leaves the lanthanide fission products with the transuranics, for extended storage until a fuel fabrication capability is ready
  - UREX+1a produces a pure stream of transuranics for recycle
- UREX+2 and UREX+3 are intended for thermal reactor recycle of plutonium together with neptunium, as a mixed oxide recycle fuel
  - UREX+2 delays the removal of the lanthanides until fuel fabrication is possible
  - UREX+3 would be the standard thermal recycle process in a close-coupled separation/fabrication process

Suite of UREX+ Processes (continued)

- UREX+4 is intended for thermal recycle of plutonium and neptunium together, with the further separation of americium from curium
  - This permits the transmutation of americium in a thermal reactor, either in specially designed fuel elements or as transmutation targets
  - It avoids the fuel fabrication problems associated with the presence of curium, a strong neutron emitter

Suite of UREX+ Processes

Process	*Prod #1*	*Prod #2*	*Prod #3*	*Prod #4*	*Prod #5*	*Prod #6*	*Prod #7*
UREX+1	U	Tc	Cs/Sr	TRU+Ln	FP		
UREX+1a	U	Tc	Cs/Sr	TRU	All FP		
UREX+2	U	Tc	Cs/Sr	Pu+Np	Am+Cm+Ln	FP	
UREX+3	U	Tc	Cs/Sr	Pu+Np	Am+Cm	All FP	
UREX+4	U	Tc	Cs/Sr	Pu+Np	Am	Cm	All FP

Notes: (1) in all cases, iodine is removed as an off-gas from the dissolution process.
(2) processes are designed for the generation of no liquid high-level wastes

U: uranium (removed in order to reduce the mass and volume of high-level waste)
Tc: technetium (long-lived fission product, prime contributor to long-term dose at Yucca Mountain)
Cs/Sr: cesium and strontium (primary short-term heat generators; repository impact)
TRU: transuranic elements (Pu: plutonium, Np: neptunium, Am: americium, Cm: curium)
Ln: lanthanide (rare earth) fission products
FP: fission products other than cesium, strontium, technetium, iodine, and the lanthanides

Technology Choices

- At the inception of GNEP planning, a top-level decision was made to process LWR spent fuel using a technology that did not result in the production of a separate stream of plutonium
  - Consistent with past U.S. policy to eschew the use of separated plutonium in the civil nuclear fuel cycle
- Process selection led to a process (UREX+1a) that
  - Separates pure uranium for future use
  - Separates cesium and strontium to alleviate the short-term decay heat load on the repository
  - Separates the transuranic elements as a group for recycle as fuel for fast reactors (advanced burner reactors)

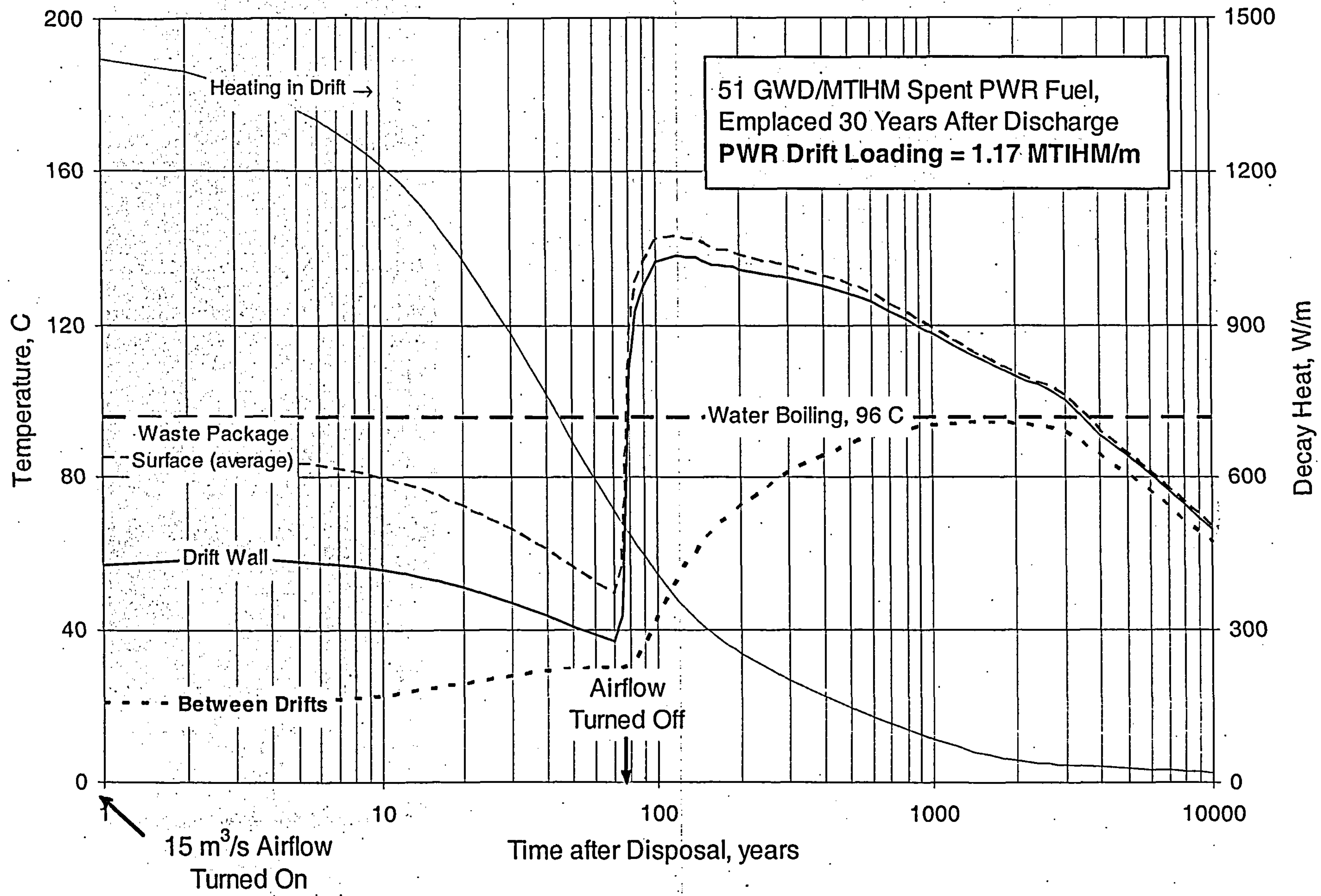
UREX+1a Process Performance Targets

- Separate uranium with 99.5% recovery efficiency and 99.99+% purity; convert to oxide for storage/recycle
- Recover 99% of the soluble technetium and convert to metallic form for incorporation in a metallic waste form
- Clean cladding hulls to a non-TRU condition (if possible) for compaction and disposal as low-level waste
- Combine a fraction of the cladding hulls with dissolver sludge and metallic technetium to produce a metallic waste form
- Recover 99% of the gaseous fission products iodine and krypton for immobilization; vent xenon after separation of krypton by cryogenic distillation; recover 95% of tritium and ^{14}C

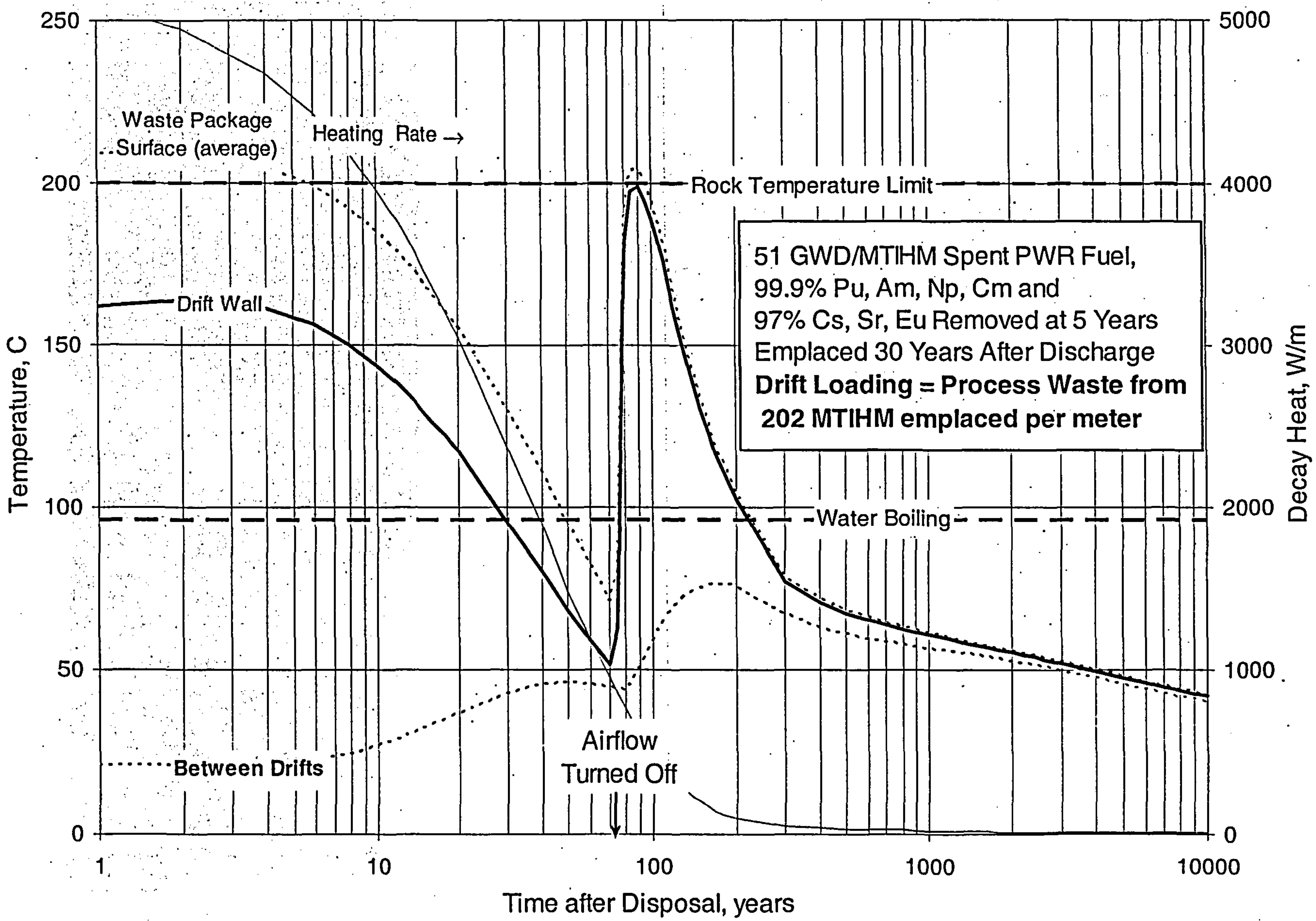
UREX+1a Performance Targets (continued)

- Recover 99.9% of the cesium and strontium (together with barium and rubidium) for incorporation in a mineral waste form for decay storage
- Recover 99.5% of the plutonium, 99% of the neptunium, 99.9% of the americium, and 99.5% of the curium present in the original spent fuel and convert to a oxide product blended with UO_2
- Produce no high-level liquid waste requiring underground tank storage

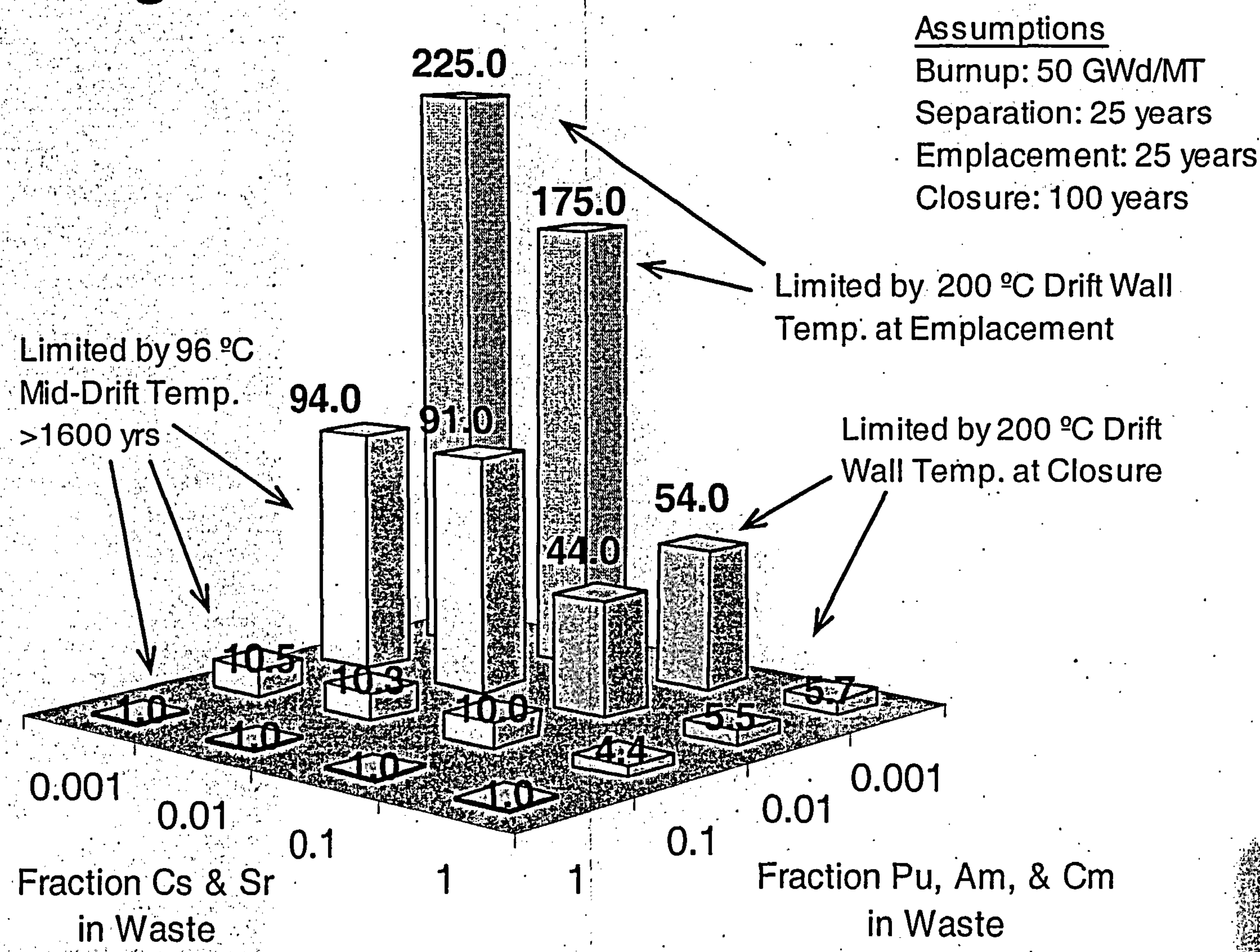
Yucca Mountain Reference Case



Enhanced Repository Capacity Case



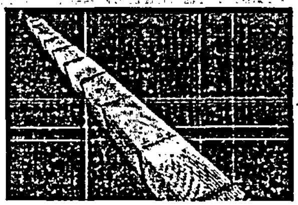
Relative Increase in Repository Capacity by Recycling



Reference: R. A. Wigeland et al., Nuclear Technology, 154 (April 2006), pp 95-106.

GNEP Commercial Fuel Reprocessing

Spent Nuclear Fuel



Uranium – 94%
Plutonium - 0.9%
Neptunium – 0.1%
Americium – 0.1%
Curium – 0.015%
Fission Products – 4.9%

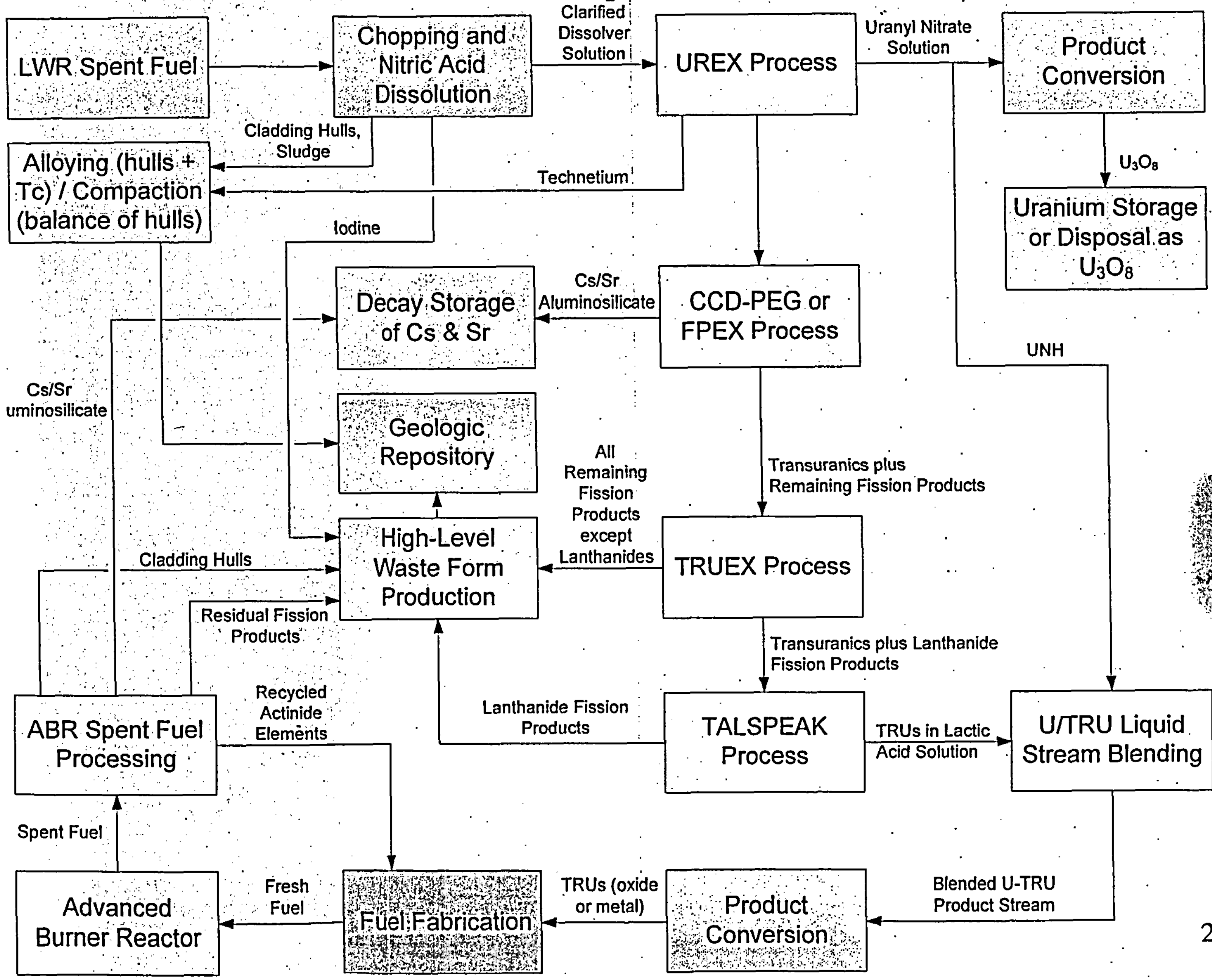
- High Radiation Level
- Highly Radiotoxic
- High Level of Decay Heating

UREX+1a Process

- Pure Uranium → *Storage for future recycle*
- Long-Lived Fission Products → *Technetium and iodine in durable waste form for geologic disposal*
- Cesium and Strontium → *Near-surface decay storage*
- Transuranic Elements → *Recycle fuel*
- Residual Fission Products → *Geologic disposal (with fuel cladding and assembly hardware)*

- Highly efficient recovery of transuranic elements for recycle and destruction; elimination of radiotoxic materials
- Greatly reduced volume of high-level waste requiring geologic disposal; fewer expensive waste containers needed
- Greatly reduced heat generation rate of high-level waste allows increase in repository capacity; one repository will be sufficient to support an expanded nuclear program for at least the rest of the 21st century

UREX+1a Process: Group Extraction of TRUs



UREX+1a Process: Projected Waste Generation

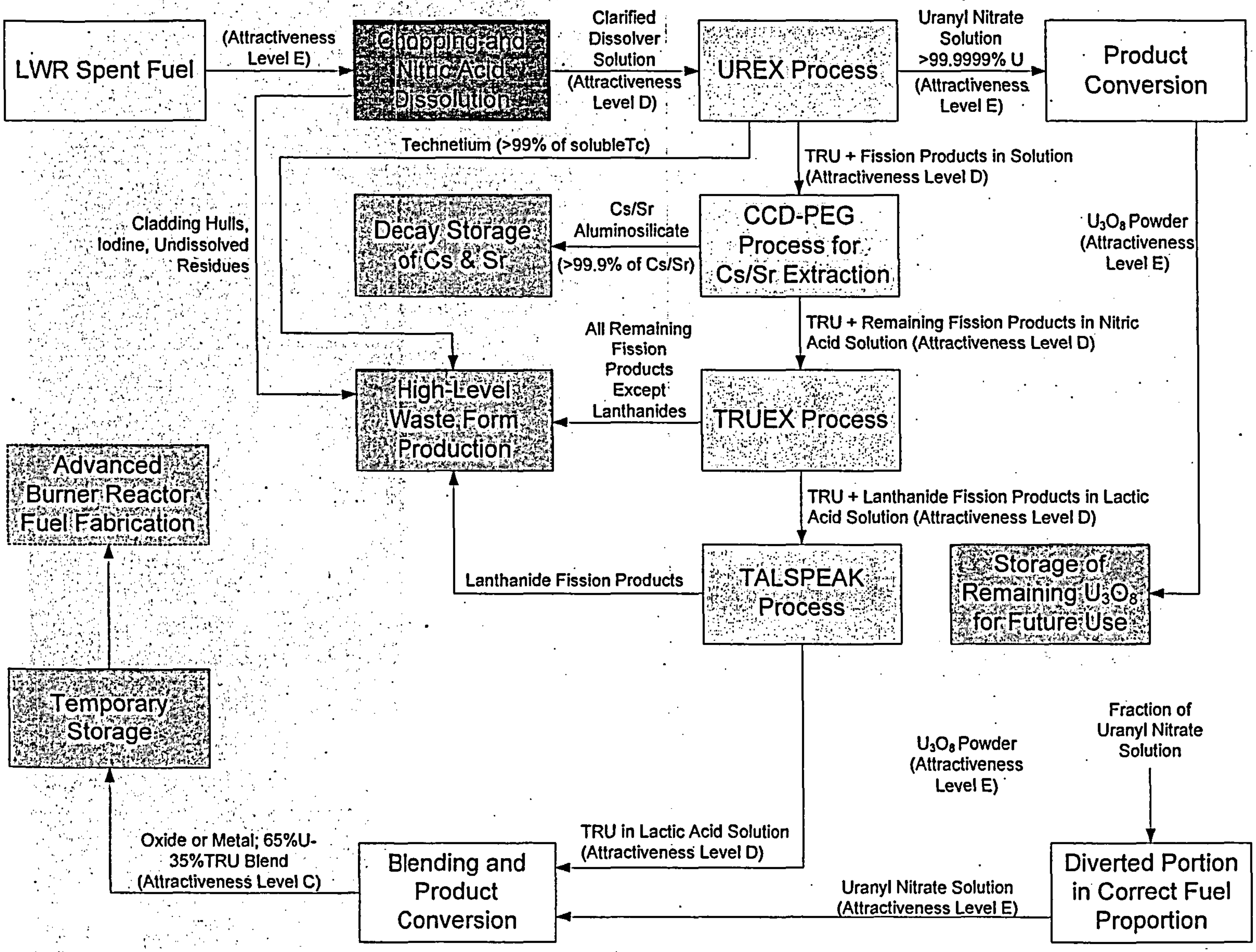
- At a scale of 100 metric tons of LWR spent fuel per year, the UREX+1a process is estimated to generate the following wastes:

Waste Stream	Waste Composition	Category	Volume, m^3/y
Uranium	U_3O_8 powder	Class C	13.3
Hulls + Tc, sludge	Fe-Zr base alloy	HLW	1.0
Compacted hulls	Non-TRU Zr	HLW	4.2
Iodine	KI	HLW	0.018
Xenon/Krypton	Compressed gas	HLW	0.056*
Tritium	Tritiated water	HLW	TBD
Cesium/strontium	Cs/Sr aluminosilicate	Class C	35.1
Fission products	Borosilicate glass	HLW	5.7
Carbon-14	Sodium carbonate	HLW	0.034

* - Kr only; Xe can be vented to the atmosphere

- For comparison, 100 metric tons of LWR spent fuel has an unpackaged HLW volume of approximately 120 m^3

UREX+1a Process



DOE Order 470.4 - Graded Safeguards

	Attractiveness Level	Pu/U-233 Category (kg)				Contained U-235/Separated Np-237/Separated Am-241 and -243 Category (kg)				All E Materials Category IV
		I	II	III	IV[1]	I	II	III	IV[1]	
WEAPONS Assembled weapons and test devices	A	All	N/A	N/A	N/A	All	N/A	N/A	N/A	N/A
PURE PRODUCTS Pits, major components, button ingots, recastable metal, directly convertible materials	B	≥2	≥0.4<2	≥0.2<0.4	<0.2	≥5	≥1<5	≥0.4<1	<0.4	N/A
HIGH-GRADE MATERIALS Carbides, oxides, nitrates, solutions (≥25 g/L) etc.; fuel elements and assemblies; alloys and mixtures; UF_4 or UF_6 (≥ 50% enriched)	C	≥6	≥2<6	≥0.4<2	<0.4	≥20	≥6<20	≥2<6	<2	N/A
LOW-GRADE MATERIALS Solutions (1 to 25 g/L), process residues requiring extensive reprocessing; moderately irradiated material; Pu-238 (except waste); UF_4 or UF_6 (≥ 20% < 50% enriched)	D	N/A	≥16	≥3<16	<3	N/A	≥50	≥8<50	<8	N/A
ALL OTHER MATERIALS Highly irradiated forms, solutions (<1 g/L), uranium containing <20% U-235 or <10% U-233[2] (any form, any quantity)	E	N/A	N/A	N/A	Reportable Quantities	N/A	N/A	N/A	Reportable Quantities	Reportable Quantities

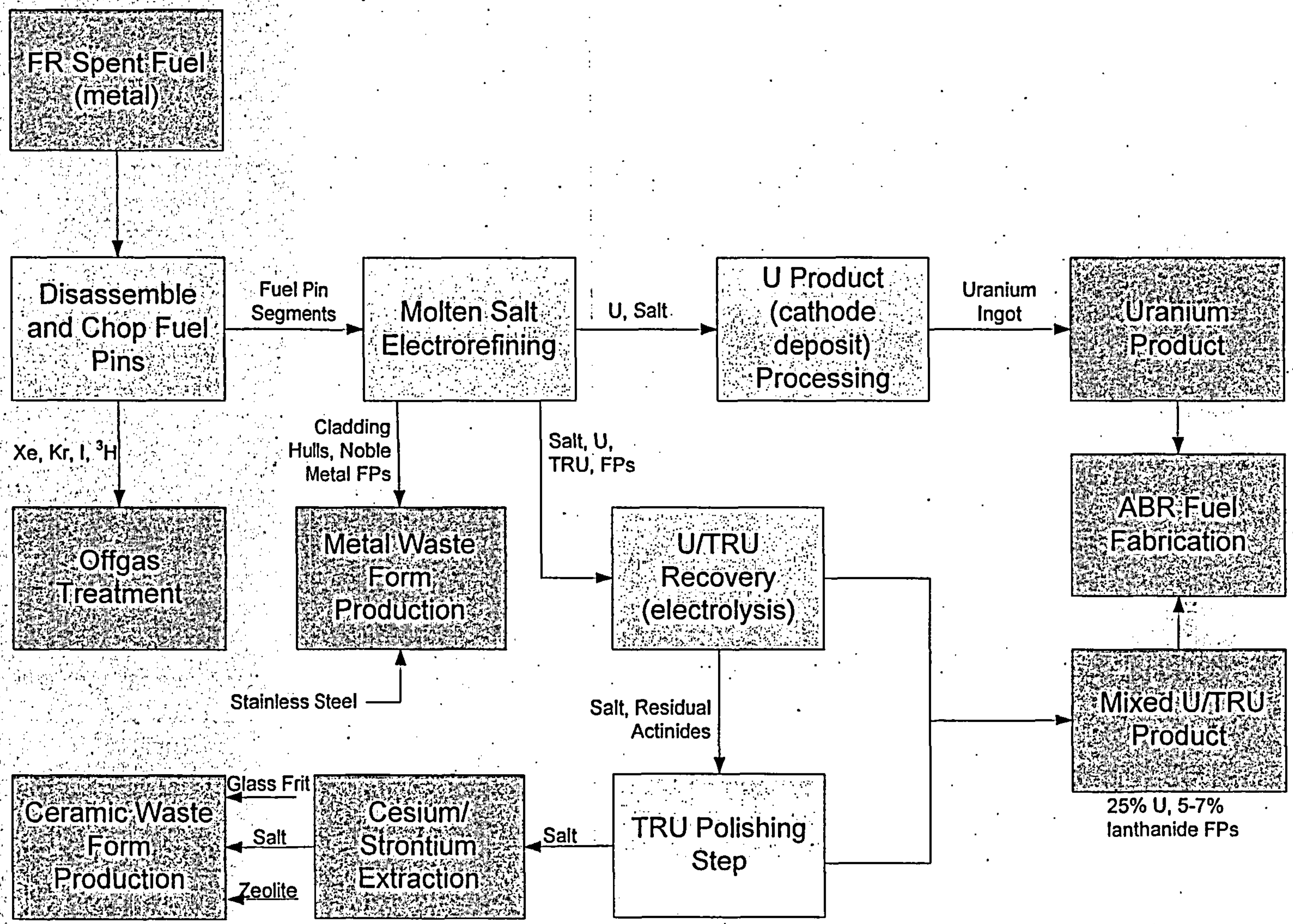
[1] The lower limit for Category IV is equal to reportable quantities in this Manual.
[2] The total quantity of U-233 = [Contained U-233 + Contained U-235]. The category is determined by using the Pu/U-233 side of this table.

- Although it may prove appropriate to incorporate Category I safeguards in a future industrial-scale recycling facility, the materials are seen to be Attractiveness Level D or less until the fuel fabrication step

Status of Technology Development

- The UREX+1a process has been demonstrated with LWR spent fuel at laboratory scale in 2005 and 2006
- Further optimization of the process will continue through 2009
- A pilot-scale demonstration of the process is planned to commence in 2011-2013
  - Scale: 30-100 metric tons LWR spent fuel per year
  - Location: TBD
- Industrial-scale LWR spent fuel recycling plant based on the UREX+1a process is projected to come on line in 2025-2030
  - Scale: 2,500 metric tons LWR spent fuel per year

Pyrochemical Processing of FR Spent Fuel



Advanced Pyroprocess Development

- A portion of the pyroprocess flowsheet is being demonstrated in the course of EBR-II spent fuel conditioning
  - Processing about 150 kg spent fuel per year
  - Highly-enriched uranium is recovered and down-blended
  - TRUs are not recovered, but sent to waste
- GNEP program planning includes activities aimed at developing the complete pyroprocess
  - Process is directed toward treatment of Advanced Burner Reactor spent fuel in low-throughput plants collocated with reactor parks on the order of 1 GWe capacity

Advanced Aqueous Process Technologies

- A number of innovative technologies are being considered for longer-term applications
  - Uranium crystallization
  - Supercritical CO_2 carrier for TBP and HNO_3
  - Carbonate dissolution
  - Decladding by voloxidation
  - Zirconium recycle
  - Single-step extraction process for transuranics
- These technologies might find application in a second-generation recycling plant (post-2040) if proven feasible

FUEL DEVELOPMENT AND ADVANCED FUEL CYCLE FACILITY

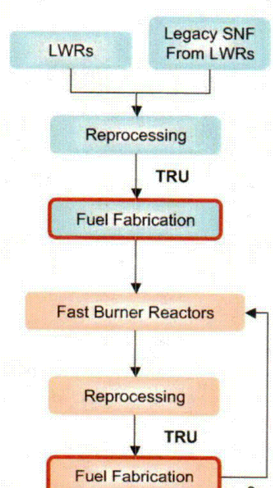
Kemal O. Pasamehmetoglu, Ph.D.
AFCI Fuels Development National Technical Director

Presented at
ACNW Meeting
July 20, 2006

Development of a transmutation fuel is a critical element for the implementation of GNEP

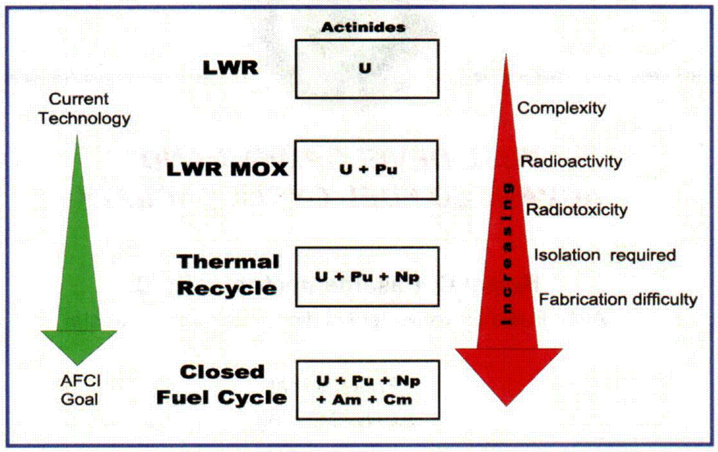
- **Multiple elements in the fuel**
  U, Pu, Np, Am, Cm
- **Varying thermodynamic properties**
  e.g. High vapor pressure of Am
- **Impurities from separation process**
  e.g. High lanthanide carryover
- **High burnup requirements**
- **High helium production during irradiation**
- **Remote fabrication & quality control**

- **Fuel must be qualified for a variable range of composition:**
  - Age and burnup of LWR SNF
  - Changes through multiple passes in FR
  - Impurities from separation process (lanthanides)



C01

The required fuels need remote fabrication, maintenance, and QA



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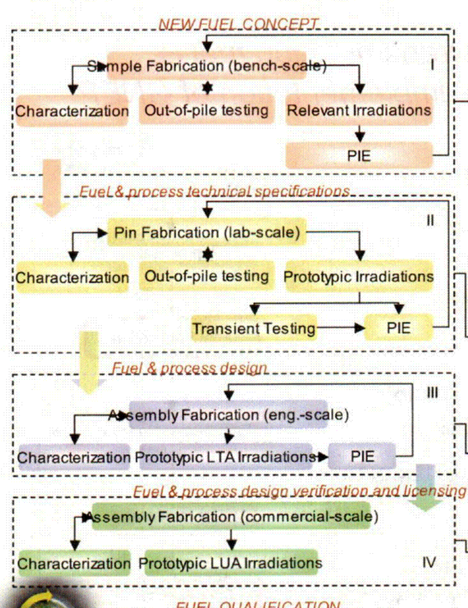
The development and qualification process requires multiple facilities



Sample fabrication in Glove Boxes	ATR, MTS Joyo
Laboratory-scale Remote Fuel Fabrication Facility **(Hot cells)**	ATR, Joyo ABTR ACRR, TREAT
Engineering-Scale Remote Fuel Fabrication Facility **(AFCF)**	ABTR
Commercial Fuel Fabrication Facility	ABTR

4

CO2

An accelerated program requires a fast test reactor and work on two types of fuels

- Driver fuel:
  - Minimal additional testing for qualification
  - Metal: (HEU, Zr) or (U,Pu,Zr)
  - MOX
- Transmutation fuel with group TRU:
  - Considerable additional development before qualification
  - Metal and oxide are ahead of other forms in terms of feasibility assessment
  - Qualification process using the test reactor after the fuel and process design is completed

5

Metal and oxide TRU fuels are candidates for the first generation transmutation fuel.

Metal Fuel

- Successful small-scale fabrication and irradiation on limited amount of samples
- Large-scale fabrication without loss of Am must be demonstrated
- Fuel-clad interactions at high burnup must be investigated

→

- Am recovery and use in moderated targets
- Development of advanced clad materials (possibility of using liners)

Oxide Fuels (powder processing)

- Successful small-scale fabrication and irradiation on limited amount of samples (France, Japan)
- Effect of group TRU on fabrication process unknown
- Large-scale fabrication amenable to hot-cell *operations must be developed*
- Limitations on linear power

→

- Sphere-pac or vibro-pac fuel technology

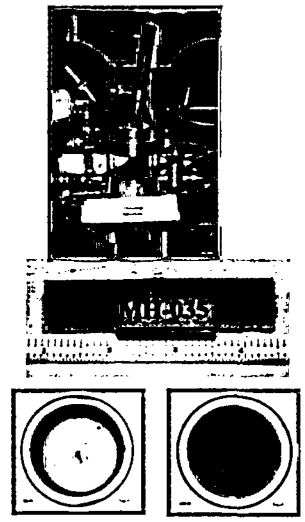
  Risk trade-off: fabrication versus performance

Long-Term Fuel Technologies (> 20 years)

- **Nitride - candidate for 2nd or 3rd generation**
  - High TRU loading potential
  - Fabrication process requires further work
  - N-15 enrichment
- **Dispersion - candidate for 2nd or 3rd generation**
  - High burnup potential
  - Fabrication process requires further work
  - Separations process must be developed

Metal Fuel Development is progressing successfully

- AFC-1 metal fuel fabricated and characterized (U, Pu, Am, Np, Zr)
  - High-burnup ATR irradiation started in Aug'05
- FUTURIX-FTA R&D report and fuel fabrication report completed.
- FUTURIX-FTA rodlets (including sodium bonding) are completed
- Metal fuel thermal characterization successful
- Initial PIE of low-burnup (~8%) ATR irradiated samples completed with positive results
- Remaining major feasibility issue:
  - large scale remote fabrication without Am loss
  - Fuel-clad chemical interactions
  - Fast-flux irradiation performance/high-burnup
  - Fabrication with co-precipitated materials.

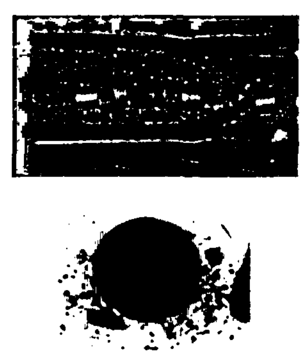


Pu-12Am-40Zr U-29Pu-4Am-2Np-30Zr

AFCI

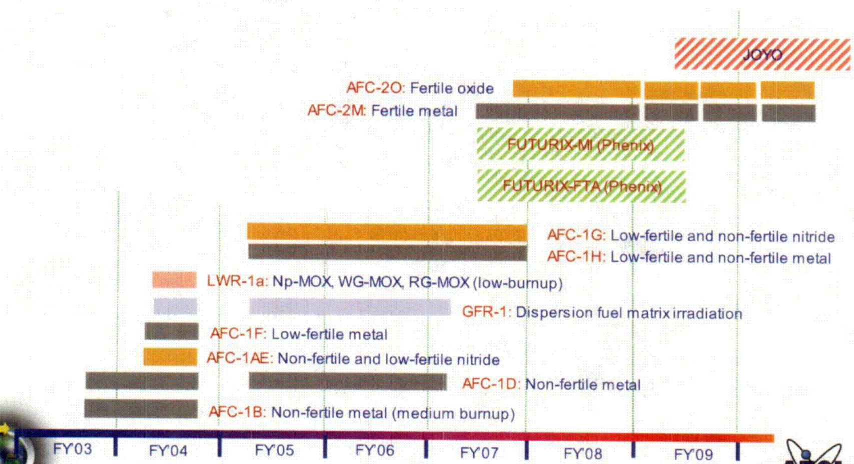
Nitride fuels require more work to demonstrate feasibility.

- AFC-1G pellets fabricated and characterized (6 rodlets)
  - Only one is being irradiated because of unexpected failure after 2 months
  - Sodium bonding process considerably improved
- FUTURIX FTA pellet fabrication completed
  - Moderate success/improved quality
  - Sodium bonding is challenging with nitrides
- The thermal properties measurements will be repeated
  - Failure due to oxygen exposure
- PIE on low burnup samples irradiated in ATR show no gross fuel failure observed but considerable pellet fragmentation
- TRU nitrides are showing extreme sensitivity to small amounts of oxygen exposure
  - Loss of mechanical integrity
  - Uranium addition helps
- onsiderable Am, and some Np losses during brication.



AFCI

Irradiation schedule reflects emphasis on fast-reactor transmutation fuels



Fuel performance prediction requires integral understanding of multiple phenomena

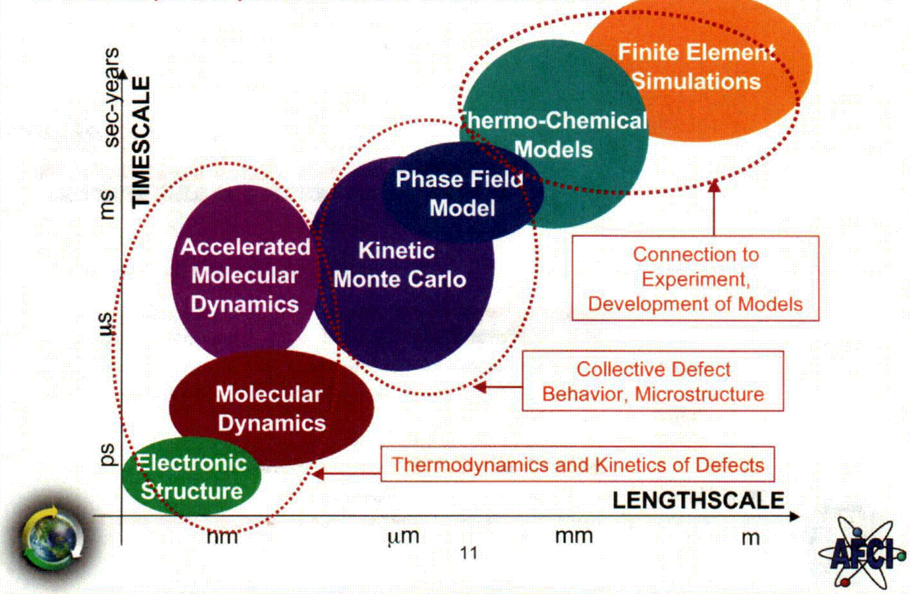
- Thermal conductivity
- Fission gas formation, behavior and release
- Materials dimensional stability
  - Restructuring, densification, growth, creep and swelling
- Defect formation & migrations
- Diffusion of species
  - Pu and Am redistribution
- Radial power distribution
- Fuel-clad gap conductance
- Fuel-clad chemical interactions
- Mechanical properties
- Thermal expansion
- Specific heat
- Phase diagrams

Dynamic properties:
Changes with radiation effects, temperature, and time

Nonlinear effects
Initial condition dependence (fabrication route)

C03

Multi-scale modeling approach is being used to develop fuel performance suite of codes



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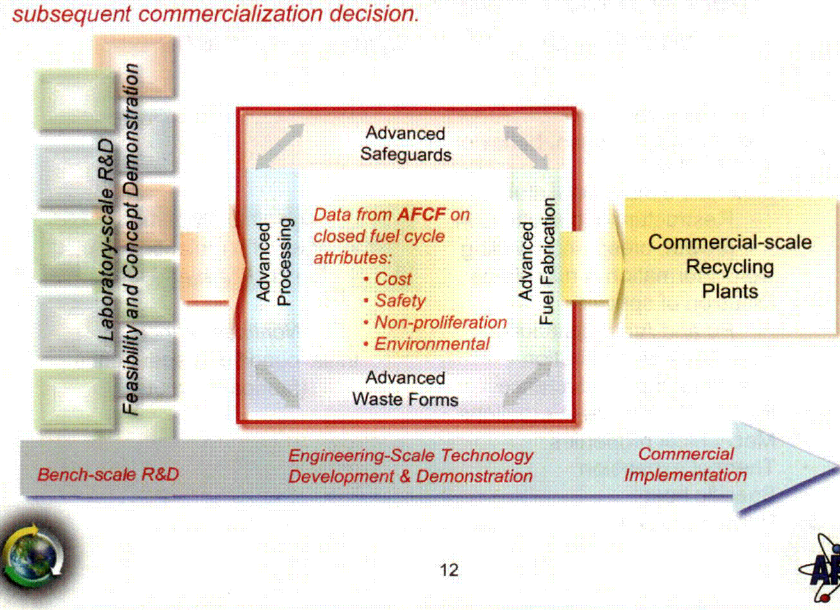
The over-arching objective of the AFCF is to perform fuel recycling technology development and demonstration of at a scale sufficient to provide input for subsequent commercialization decision.



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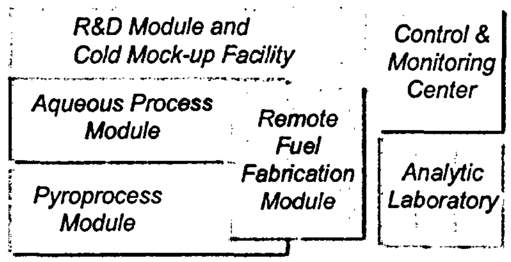
Pre-conceptual design considerations is being reviewed under GNEP structure

- Remote fuel fabrication module ~ 10 LTA/yr
- Fast reactor fuel reprocessing modules ~ 1 ton/yr
- R&D module: small-hot cells and shielded glove-boxes for bench-scale research.



AFCF is being designed to REGAIN and RETAIN US leadership

A facility for near-term demonstration of baseline technologies (<20 yrs)

Flexibility for technology enhancements (> 20 yrs)

13

A number of strategic trade studies will be conducted in the near-term

Trade Studies

- Single facility with modules
- Modular construction phased over time
- Minimum size and capability for each module
- Basis for bench-scale support module with flexible design of engineering-scale modules

Design Features

- Safeguards & security (design basis vs technology demonstration)
- Technology - risk ranking & success criteria
  - Technology choice for initial demonstration
  - Integration with ESD
- Support facilities

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ENHANCED SAFEGUARDS TECHNOLOGY DEVELOPMENT AND DEMONSTRATION

There are multiple types of Safeguards Measures

- Nuclear Material Accounting
  - Establishing the quantities of nuclear material present within defined areas and the changes in those quantities within defined periods.
- Containment and Surveillance
  - Ensures previously measured material in containers remains in place and unaltered or moves along declared and authorized paths
  - Examples: video and seals
- Process Monitoring
  - The use of process data (flow rate, temperature, pressure, density, etc.) to draw frequent conclusions on plant configurations, operations and material flows/inventories
  - Necessary for In-process, interim inventories

Sigma ID Requirements Based on the Current, Most Stringent, Category I Requirements for NRC and DOE

Agency	Goal/ Requirement Terms	Sigma ID	Frequency of Long-Term, Shutdown Inventory	Frequency of Interim Inventory, Shutdown Not Required
IAEA	Material Unaccounted For (MUF): 8kgs Pu abrupt in one month and 8kgs protracted in one year	Sigma ID <= 2.42 kg of Pu	Annual	Monthly
NRC	Standard Error of Inventory Difference (SEID)	Sigma ID <= 0.1% of active inventory	Semi-Annual	Monthly
DOE	Limit of Error (LOE)	Sigma ID <= 1% of active inventory of material	At least Annually	Bi-monthly

Rough Estimate of Sigma ID for AFCF shows the challenge!

- Capacity Per Year
  - 1 MTHM/yr
- Active Inventory
  - Assumed to be the amount of Pu processed in one month from FR spent fuel
  - Crude estimate because this is a technology development facility that will not have continuous operations
- The estimated NRC Sigma ID requirement of 25g is a challenge !

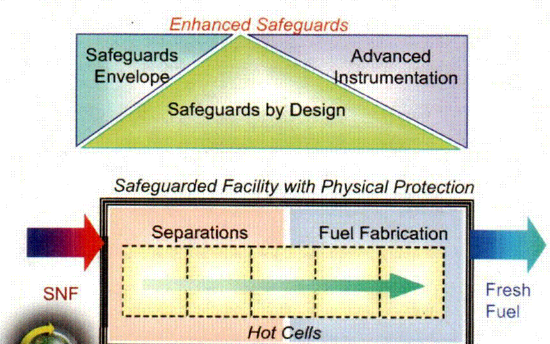
FR Spent Fuel Processed per year	**1 MTHM**
Pu processed/yr (30% of spent fuel)	**300 kg**
Pu processed per month (assumed to be active inventory at time of shut down inventory)	**25 kg**
NRC's Sigma ID is 0.1% of active inventory	**0.025 kg**
DOE's Sigma ID is 1% of active inventory	**0.25 kg**
IAEA goal for Sigma ID	**2.42 kg Pu**

Why do we need safeguards technology development ?

- Application of PUREX techniques to group TRU separations
  - UREX+
  - Pyro
- Achieve IAEA goals by less intrusive means
  - Reduce the shutdown inventory requirements
- Reduce the requirement on continuous presence of inspectors
- Overall, reduce the risk of diversion/theft

Focusing on fuel recycling facility, the proposed approach focuses on the following key elements

- Advanced instrumentation
- Advanced control logic ("safeguards envelope") (SESAME)
- Advanced modeling and simulation to guide and optimize the "safeguards by design" (SESAME)
- Engineering-scale research facility to demonstrate "safeguards by design"



The safeguards by design approach includes demonstrating an integrated separations & fuel fabrication plant.

C05

Advanced instrumentation and multi-attribute control logic are essential elements of enhanced safeguards.

Standard Instrumentation
- Density
- Weight
- Flow
- Temperature
- Pressure
- Conductivity
- Neutron Monitors
- Plutonium a Monitors
- Gamma Spectroscopy
- Radiation Monitors
- Chromatography
- Mass Spectrometry

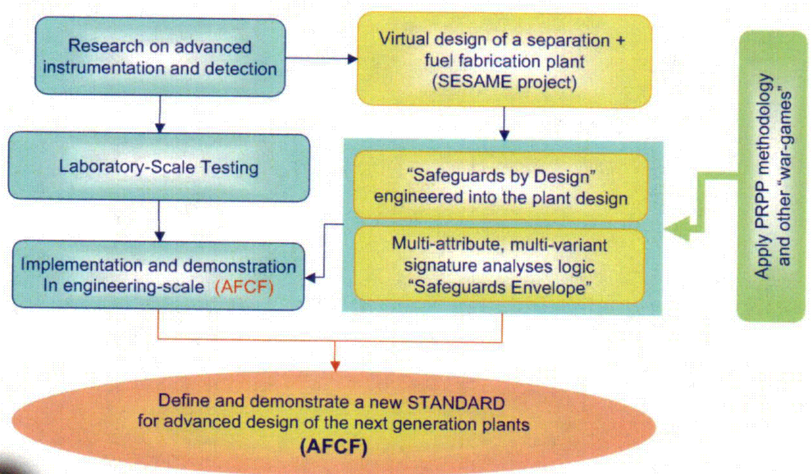
Precision
Sampling rate
Reliability
Durability
Cost

Automated, optimized sampling.

Active Interrogation:
- Accelerator stimulated x-ray fluorescence
- Neutron integral cross section spectroscopy
- Delay neutrons;
- Delay neutron lifetime measurements
- Delayed Gammas.

Multi-measurement, cross-correlated "safeguards envelope"

By implementing advanced monitoring and safeguards technologies, separation and fuel fabrication plants can be shown to be of minimum risk for proliferation.



C06

Standard Review Plan for Activities Related to Waste Determinations

July 20, 2006

Anna Bradford, David Esh, Christianne Ridge

Presented to the 172nd Meeting of the Advisory Committee on Nuclear Waste

1

United States Nuclear Regulatory Commission

Background

- NMSS staff gave a presentation to ACNW on May 25, 2006
- The Standard Review Plan (SRP) was publicly released on May 31, 2006
- The SRP is open for public comment until July 31, 2006
- The purpose of today's briefing is to address ACNW questions on specific areas

2

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The NDAA Do's and Don'ts

- DOE does consult with the NRC; NRC does not regulate DOE
- The NDAA does not apply to entire site cleanup
- The NDAA does specify which criteria should be used to determine if waste needs to be disposed of in a geologic repository
- NRC does monitor to assess compliance but has no enforcement role

3

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Purpose of SRP

- Provide internal guidance for use by NRC staff
- Describe the types of information that may be assessed by NRC staff during its activities
- Meant to be flexible and applicable to a wide variety of situations, rather than prescriptive

4

United States Nuclear Regulatory Commission

Performance Assessment

- Performance assessment (PA) is expected to be the analysis approach used to demonstrate compliance with 10 CFR 61.41.
- The SRP provides guidance on general topics, such as data uncertainty and model support, as well as specific topics such as estimation of infiltration rates.
- The review is anticipated to be performed with a risk-informed approach. That is, the reviewer has flexibility to focus on those elements more likely to influence the demonstration of compliance.

5

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Performance Assessment Review Procedures

- Review procedures for deterministic or probabilistic approaches are provided.
- A separate section is devoted to uncertainty and sensitivity analysis.
- Evaluating model results and defining contributions of barriers and the natural system are important to implementing a risk-informed approach to the review.

6

United States Nuclear Regulatory Commission

Performance Assessment – Staff Review

- Review of a performance assessment typical entails the evaluation of a large number of documents, some of which can be quite large.
- The SRP does not prescribe specific analyses to demonstrate compliance (e.g., deterministic or probabilistic).
- Different approaches to performance assessment may be used depending on the circumstances.
- Compliance ≠ Reality, Compliance = Safety.

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Performance Assessment – Staff Review

- If necessary (and possible given resource considerations), staff perform independent calculations.
- Independent calculations may include a probabilistic performance assessment.
- This review approach helps focus the review and strengthen the basis for the results.
- NRC does not rely on the independent calculations as a basis for a decision, only to inform the review process.

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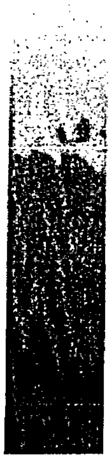
Performance Assessment – Staff Review



- A variety of questions on higher-level issues were provided to the staff by the Committee, including:
  - Compliance period (10,000 years)
  - Institutional control
  - Use of water
  - Conceptual model uncertainty
  - Engineered barrier performance
  - Stability of tanks under variably saturated or saturated conditions
  - Level of proof
  - Climate change
  - Nearby contamination

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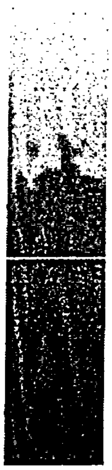
Radionuclide Removal



- Radionuclide inventories
- Selection of highly radioactive radionuclides
- Selection of radionuclide removal technologies
- Practicality of additional removal
  - Basis for stopping removal activities
  - Costs and benefits of additional removal

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Radionuclide Removal

- Applicable Requirement for Hanford and West Valley

  The waste has been processed (or will be further processed) to remove key radionuclides to the maximum extent that is technically and economically practical

- Applicable Requirement for Idaho National Laboratory and the Savannah River Site

  Waste . . . has had highly radioactive radionuclides removed to the maximum extent practical

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Technology Selection

- Range of technologies evaluated is expected to include technologies being used or developed across the DOE complex
- Factors affecting the choice of removal technologies may include
  - Expected effectiveness
  - Technological maturity
  - Schedule impacts
  - Implementation costs
  - Worker safety impacts
  - System wide effects

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Radionuclide Removal Criteria

- Criteria for concluding waste removal is complete may include
  - Meeting a volumetric goal
  - Achieving a specified removal efficiency
  - Reaching a specified decline in removal effectiveness
- Other reasons for stopping removal may include
  - Failure of key pieces of equipment that would necessitate unacceptable costs or worker doses to replace
  - Unexpected waste characteristics that prevented reaching a volumetric goal but made additional removal impractical

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Practicality of Additional Removal

- A decision that additional removal is not practical may be based on
  - Minimal expected dose reductions
  - Economic costs of additional removal
  - Programmatic and schedule impacts
  - System impacts (e.g., effects of chemical treatments on downstream processes)
- If possible, costs and benefits should be quantified in terms of economic costs and expected dose to workers and members of the public

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Cost-Benefit Analysis

- The SRP recommends costs and benefits be compared to costs and benefits of similar DOE activities
- Potential difficulties with other metrics used by the NRC (e.g., ALARA analyses for license termination) are discussed in the SRP

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Dose Estimates for Cost-Benefit Analysis

- The SRP emphasizes that uncertainties in dose estimates will propagate into cost-benefit analyses
- Reviewers generally attempt to assess the degree of conservatism of performance assessments
- Worker dose estimates expected (but not required) to be based on exposures from similar activities

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Existing Guidance

- The SRP uses existing guidance where applicable (e.g., NUREG-1573, NUREG-1757).
- The incidental waste criteria specifically cite 10 CFR 61; therefore, guidance for 10 CFR 61 was appropriate.
- For worker dose, DOE can generally use its own 10 CFR 835 to satisfy 10 CFR 61.

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Terms Used in the SRP

- "Reasonable assurance"
- "Comparable to"
- "Other characteristics"
- "Interim use"

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Monitoring

- Monitoring will be risk-informed and performance-based.
- Non-compliance is when there is no longer reasonable assurance that the performance objectives can be met.
- Scope of monitoring plans may vary.

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Conclusions

- The draft SRP is based on existing NRC guidance and staff experience with waste determination reviews.
- The draft SRP is flexible and applicable to many different reviews, while providing a consistent review basis.
- The staff looks forward to receiving any recommendations the ACNW may have for the final SRP.

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A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System – Key Messages

Ed Hackett
Deputy Director, Spent Fuel Project Office

Meeting with Advisory Committee on Nuclear Waste
Rockville, MD
July 20, 2006

Background and Focus

- SFPO requests RES support in development of a PRA for Dry Cask Storage
- Objective – NRC Regulations for Storage of Spent Fuel are Largely Deterministic. This effort provides an initial look at risk-informing the regulatory approach in this area.
- Focus – Development of the Methodology and a Limited Scope Pilot Application

Context and Key Messages

- Although the PRA numbers are very low overall – this was NOT the focus of the study
- Provided Identification of Risk-Dominant contributors – useful tool to provide insights for licensing and inspection/oversight activities
- Example: Risk dominated by handling sequences
- We are here to provide the Committee with an overview of the effort and to discuss findings and conclusions

Information Briefing
A Pilot Probabilistic Risk Assessment (PRA) of a Dry Cask Storage System at a Nuclear Power Plant

Presented to

Advisory Committee on Nuclear Waste

Presented by

Ronaldo Jenkins and Dr. Gordon Bjorkman

U.S. Nuclear Regulatory Commission

Agenda

- Goals of the Dry Cask Storage System PRA
- PRA Methodology
  - Initiating Events
  - Success Criteria
  - Event Tree Analysis
  - Consequence Analysis
  - Risk Calculation
- Detailed Discussion of Success Criteria
- Conclusions

Goals of Dry Cask Storage System PRA

- Develop PRA study of a dry cask storage system
- Identify dominant contributors to risk
- Provide guidance for the development of future PRA studies in this area
- Improve understanding of dry cask storage system operation and failure modes
- Encourage the development of risk informed regulatory activities

PRA Methodology

RISK = FREQUENCY * CONSEQUENCES

- What can go wrong?
- Likelihood of initiating events
- What are the consequences

PRA Methodology (Cont'd)

For the system, risk can be divided into:

R^H- risk during handling phase

R^T- risk during transfer phase

R^S- risk during storage phase

$$R = R^H + R^T + R^S$$

PRA Methodology (Cont'd)

- The cask system analyzed is the Holtec International HI-STORM 100 which consists of three major components:

  - a multipurpose canister (MPC) that confines the fuel,

  - a transfer overpack (or transfer cask) which surrounds the MPC and shields workers from radiation while the cask is being prepared for storage, and

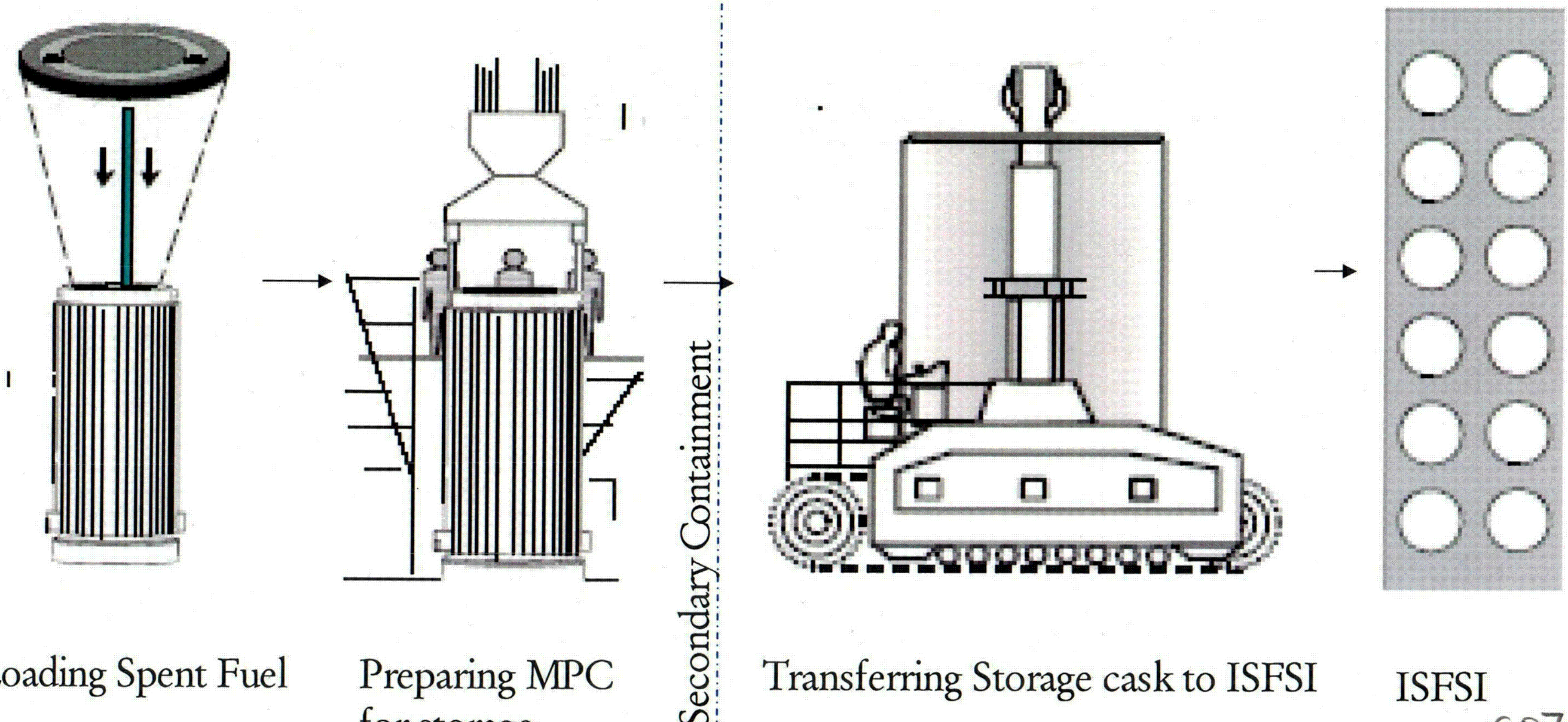
  - a storage overpack (or storage cask) which surrounds the MPC and shields people from radiation and mechanically protects the MPC during storage

PRA Methodology (Cont'd)

Dry Cask Storage System Operation

- HANDLING PHASE
  - Spent fuel assemblies are placed into the MPC which is inside the transfer cask
  - MPC is prepared for storage
  - MPC is removed from the transfer cask and placed in the storage cask
- TRANSFER PHASE
  - Storage cask is moved out of the secondary containment to the Independent Spent Fuel Storage Installation (ISFSI)
- STORAGE PHASE (Years 1 -20)

PRA Methodology (Cont'd)



Loading Spent Fuel

Preparing MPC for storage

Transferring Storage cask to ISFSI

ISFSI

C07

PRA Methodology (Cont'd)

An initiating event is a disturbance in the operation of the dry cask storage system which may lead to a release of radioactive material to the environment.

PRA Methodology (Cont'd)

Initiating Events During the Handling Phase

- ☐ Drop of the Fuel Assembly,
- ☐ Drop of the Transfer Cask (Into the cask pit, onto storage cask, concrete floor),
- ☐ Cask tip-over, and
- ☐ Drop of the MPC into the storage cask

Initiating Events During the Transfer Phase

- ☐ Drop of the Storage Cask, and
- ☐ Tip-over of the Storage Cask

PRA Methodology (Cont'd)

Initiating Events During the Storage Phase

- **Mechanical Events**
  - Seismic events
  - Tornados
  - Strikes from Aircraft, Meteorite

- **Thermal Events**
  - Vent Blockage from flood water, snow, ice, and hail
  - Fire from aircraft fuel

Excluded Initiating Events (not applicable to subject plant site)

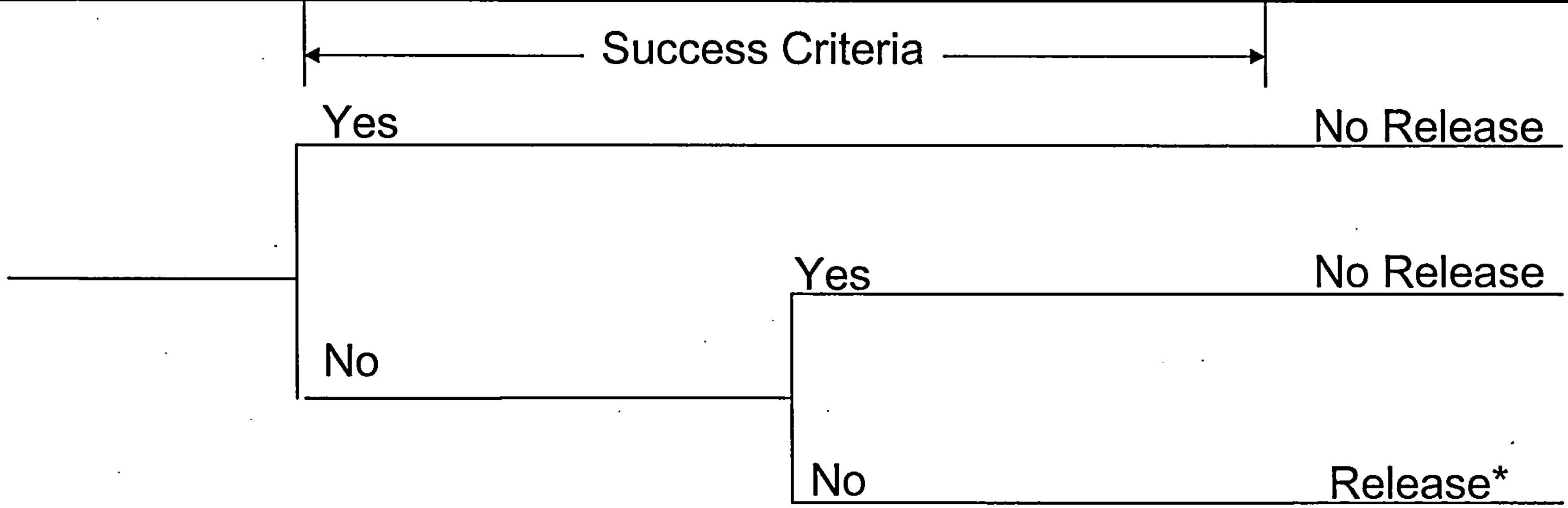
- **Tsunamis**
- **Volcanic activity**

PRA Methodology (Cont'd)

- Success Criteria
  - Response of the fuel and the MPC to the mechanical and thermal loads associated with the initiating events were determined
  - An engineering analysis estimated the reliability of the secondary containment isolation system

PRA Methodology (Cont'd)

Initiating Event	Fuel Cladding Integrity	Cask/MPC Mechanical Integrity	End State



*Release of radionuclides other than noble gases is dependent upon the reliability of Secondary Containment Isolation System

PRA Methodology (Cont'd)

- Consequence assessment was performed using the MELCOR Accident Consequence Code System (MACCS2).

- Specific data needed to model the HI-STORM 100 Dry Storage System in the MACCS2 consequence calculation were collected and used for this report.

- In order to examine the radiological consequences, release fractions were developed. Source term was derived from input from Sandia National Laboratories.

PRA Methodology (Cont'd)

- **Major Input for Consequence Calculations**
  - **Radionuclides Inventory**
  - **Source Term**
  - **Meteorological Data**
  - **Population Data**
  - **Emergency Response**

- Estimated consequences (probability of a latent cancer fatality for an individual) from a release of the inventory of the MPC for high fuel burnup and a release height of 50 meters, include essentially negligible contribution from noble gases and the contribution from other radionuclides is $3.6x10^{-4}$.

PRA Methodology (Cont'd)

$R = R^H + R^T + R^S$

OR

The estimated risk to an individual (probability of a latent cancer fatality per year) is:

- 2.0×10^{-12} for the first year of operation. This includes handling, transfer, and storage phase, and
- 1.9×10^{-13} during subsequent years of storage.

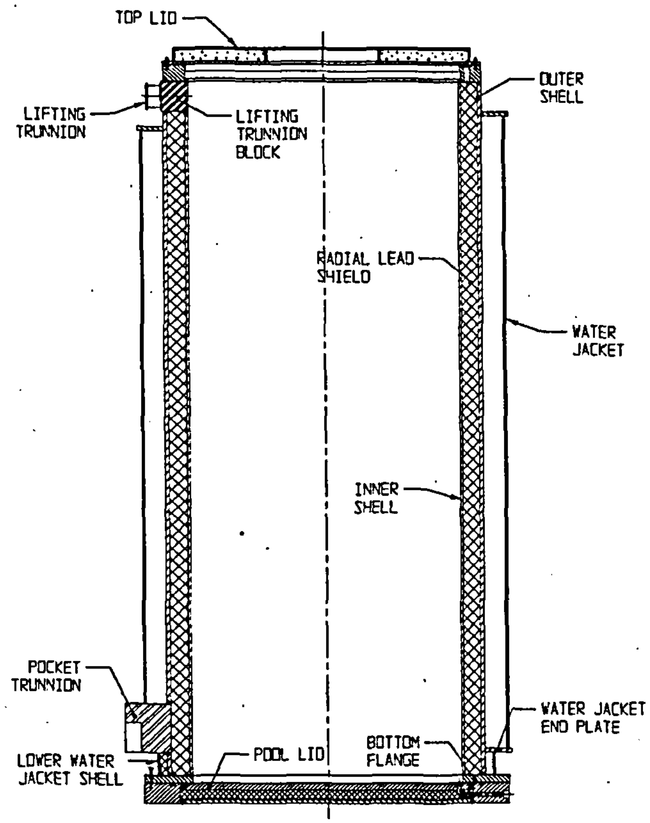
Detailed Discussion of Success Criteria

- HI-STORM 100 System
- Summarize Events that could lead to Confinement Boundary or Fuel Failure
- Concentrate on Higher Probability of Failure Events
  - Analysis Models
  - Failure Criteria
  - Failure Modes
- Release Fraction Methodology

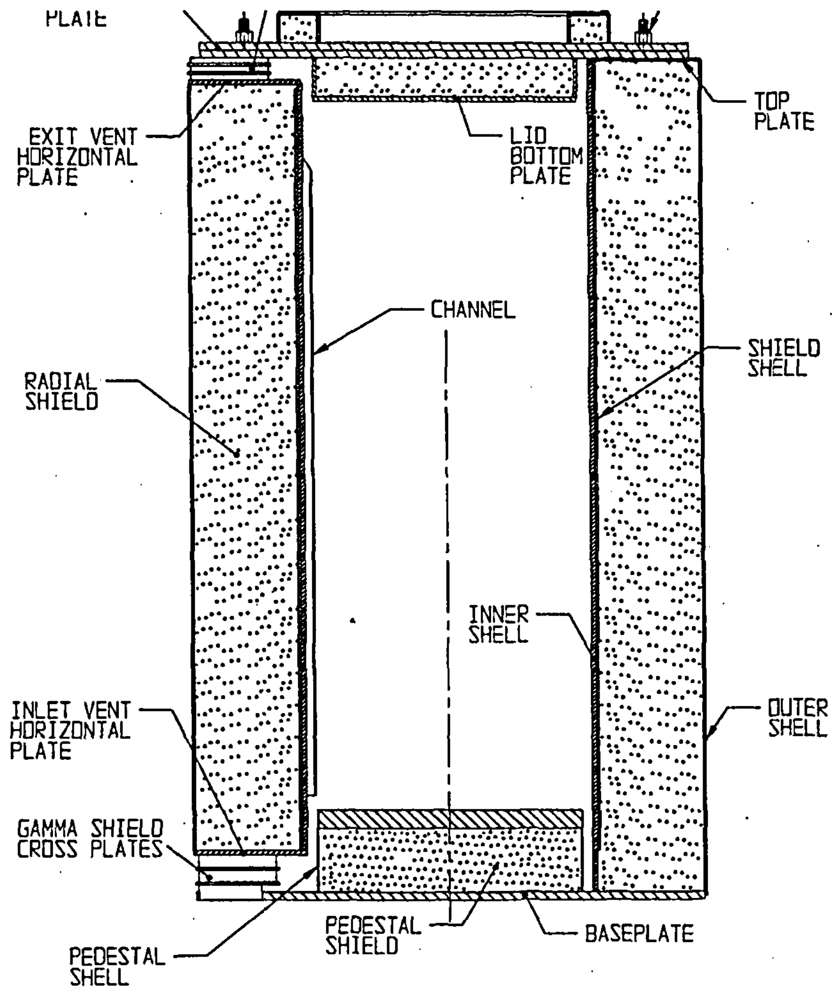
Components of the HI-STORM 100 System

- Multi-Purpose Canister (MPC)
  - Confinement Boundary for the Fuel
- Transfer Overpack (TO)
  - Shields MPC during Transfer Operations
- Storage Overpack (SO)
  - Shields MPC during Storage

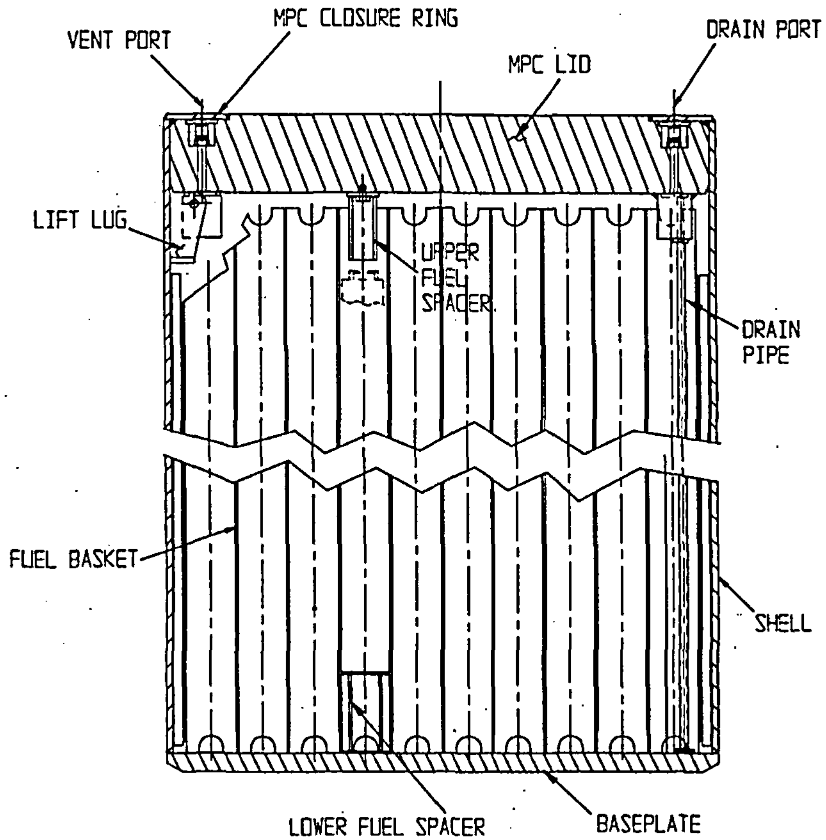
Transfer Overpack

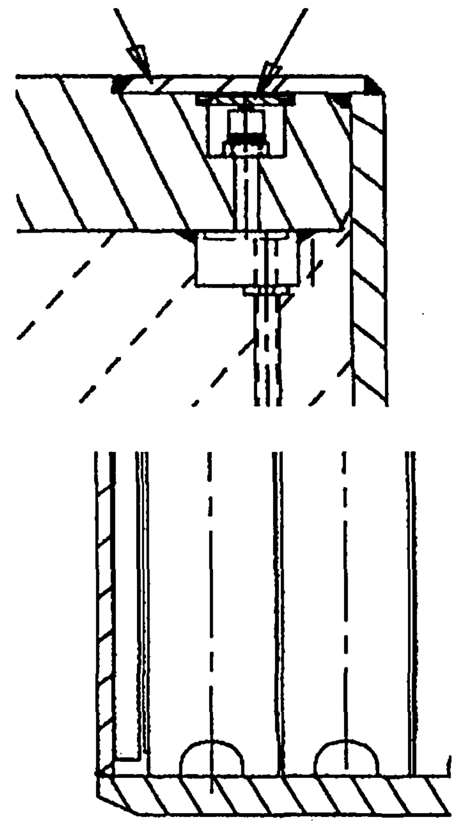


Storage Overpack



Multi-Purpose Canister (MPC) Details





Release of Radionuclides

Radionuclides can be Released to the Environment if

- ☐ Fuel Cladding Failure or CRUD Spallation

And

- ☐ MPC Confinement Boundary Breaches

Conditional Probability of Release

Stages	Initiating Event Frequency (A)	**Conditional Probability of Release from MPC & Rod (B)**	Probability of Secondary Containment Release	Consequence	Risk
1 to 34	1×10^{-3} to 1×10^{-14}	**P(B given A)** **0 to 0.28**	1 to 1×10^{-4}	1×10^{-4} to 1×10^{-12}	1×10^{-12} to 1×10^{-17}

Event Categories

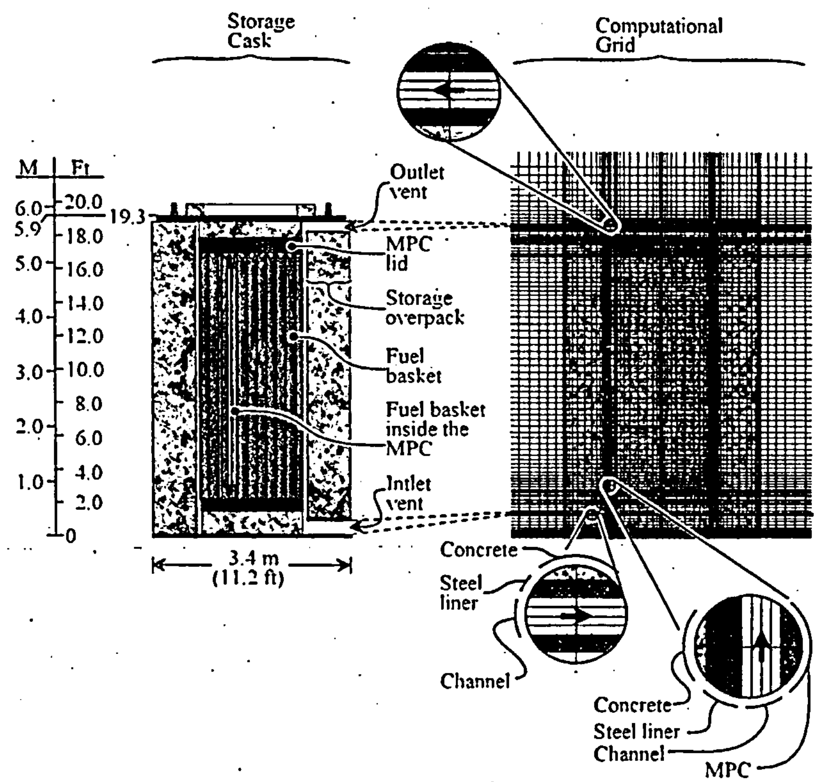
Two Event Categories that could cause Fuel Cladding Failure or MPC Breach

- ☐ Thermal Events
- ☐ Mechanical Load Events

Thermal Events

Computational Fluid Dynamics Model of MPC & SO Developed to Evaluate

- Aircraft Fuel Fire
  - Fuel load of Gulfstream IV (3 hour duration)
- Blocked Vents
  - 20 year duration (Steady State 29 days, Inspections)



Thermal Events and Fuel Cladding Failure

Event	Max. Cladding Temp.	Accident Temp. Limit
GIV Fuel Fire	298 C	570 C*
Blocked Vents	461 C	570 C*

NO Fuel Cladding Failures

* Cladding failure not expected until temperatures of 750 C are reached.

Thermal Events and MPC Failure
(Loading: MPC Fill Gas Pressure = 82 psi)
Two Failure Modes

Limit Load

- Flow Stress Model (Stress causing continuous plastic flow)
- Actual Stress < Flow Stress
  - Flow Stress ~ f(yield stress, ultimate stress)
  - Yield & Ultimate Stress ~ f(temp)
- Monte Carlo Simulations Performed
- **NO Failures Predicted**

Creep Rupture

- Time to Failure data as a function of stress and temperature for Stainless Steel weld and base metal
- ANL Creep Model used to predict creep damage for any time/temperature/stress condition (stress magnified to account for weld flaws)
- Monte Carlos Simulations Performed
- **NO Failures Predicted**

Mechanical Load Events

- Explosions
  - Gasoline Tanker (overpressure 1 psi < design external pressure of 10 psi)
  - Gas Pipeline (overpressure = 1 psi < 10 psi design)
- Strikes by Heavy Objects
  - Vehicle Impact (10,000 lbs at 150 mph, No Tip-over)
  - Tornado Missiles (mass and velocity insufficient to cause SO perforation or tip-over)

Strikes by Heavy Objects (continued)

- Aircraft Impact
  - Gulfstream IV is largest aircraft that can be handled at local airfields (74,000 lbs)
  - Landing and Takeoff
    - Engine Shaft does not perforate Storage Overpack
    - Mass and Velocity insufficient to cause tip-over
  - Over-flying Aircraft
    - Assume all over-flying aircraft are larger than Gulfstream IV and traveling at high velocity
    - Assume all impacts cause cladding failure and MPC breach
  - Conditional Probability of Release =

$$\frac{[P(\text{over-flight crash})]}{[P(\text{over-flight crash}) + P(\text{take-off and landing crash})]} = 0.14$$

Mechanical Load Events (continued)

- Seismic Events
  - ABAQUS Soil-Structure Interaction Model included Storage Cask, ISFSI Concrete Pad and Soil
  - Coefficient of Friction between Cask and Pad varied between 0.25 and 0.53
  - Earthquake Magnitude 9 to 11 times site Design Basis Earthquake (0.15g PGA)
  - No Cask Tip-over Predicted

Mechanical Load Events (continued)

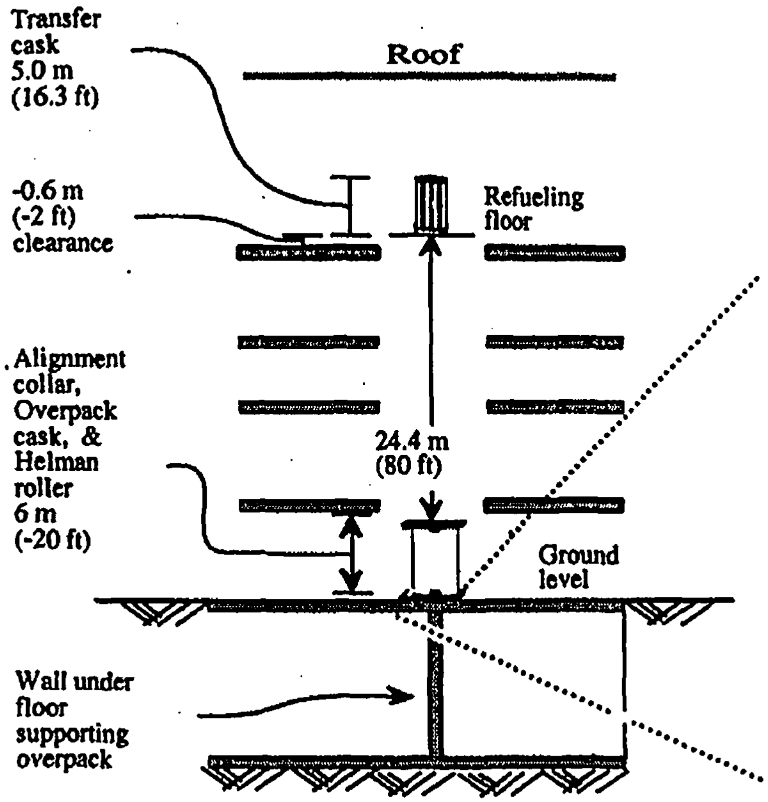
- Cask Drop Events
  - MPC Unsealed
    - Transfer Cask (TC) drop into Cask Pit
    - TC drop onto Refueling Floor
  - MPC Sealed
    - TC moved over Refueling Floor (Max Drop = 3 ft)
    - TC lowered through Equipment Hatch (Max Drop = 100 ft)
    - MPC lowered into SO (Max Drop = 19 ft)
    - SO moved to ISFSI pad (Max Drop = 1 ft)

Evaluating the MPC for Drop Events

Two Significant Drops

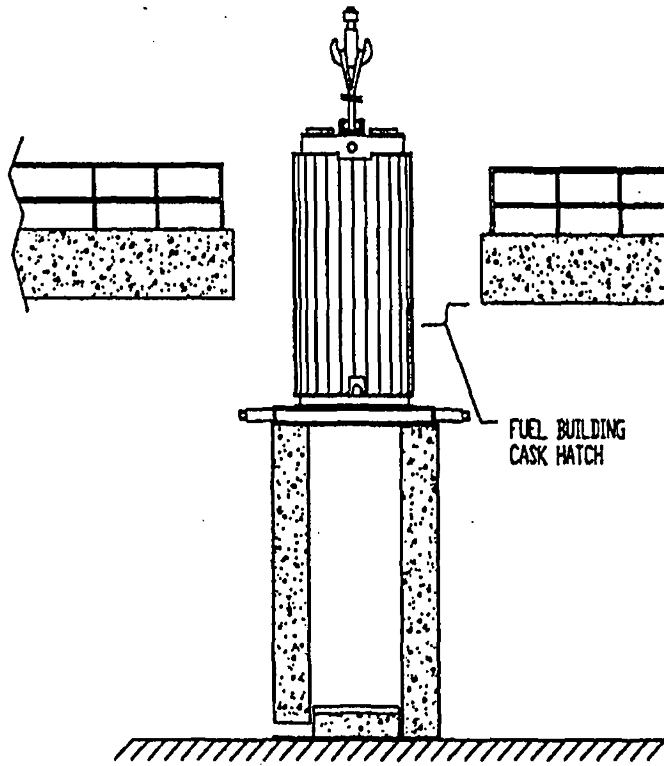
<u>100 ft Drop thru Hatch</u>

("Soft" Impact)



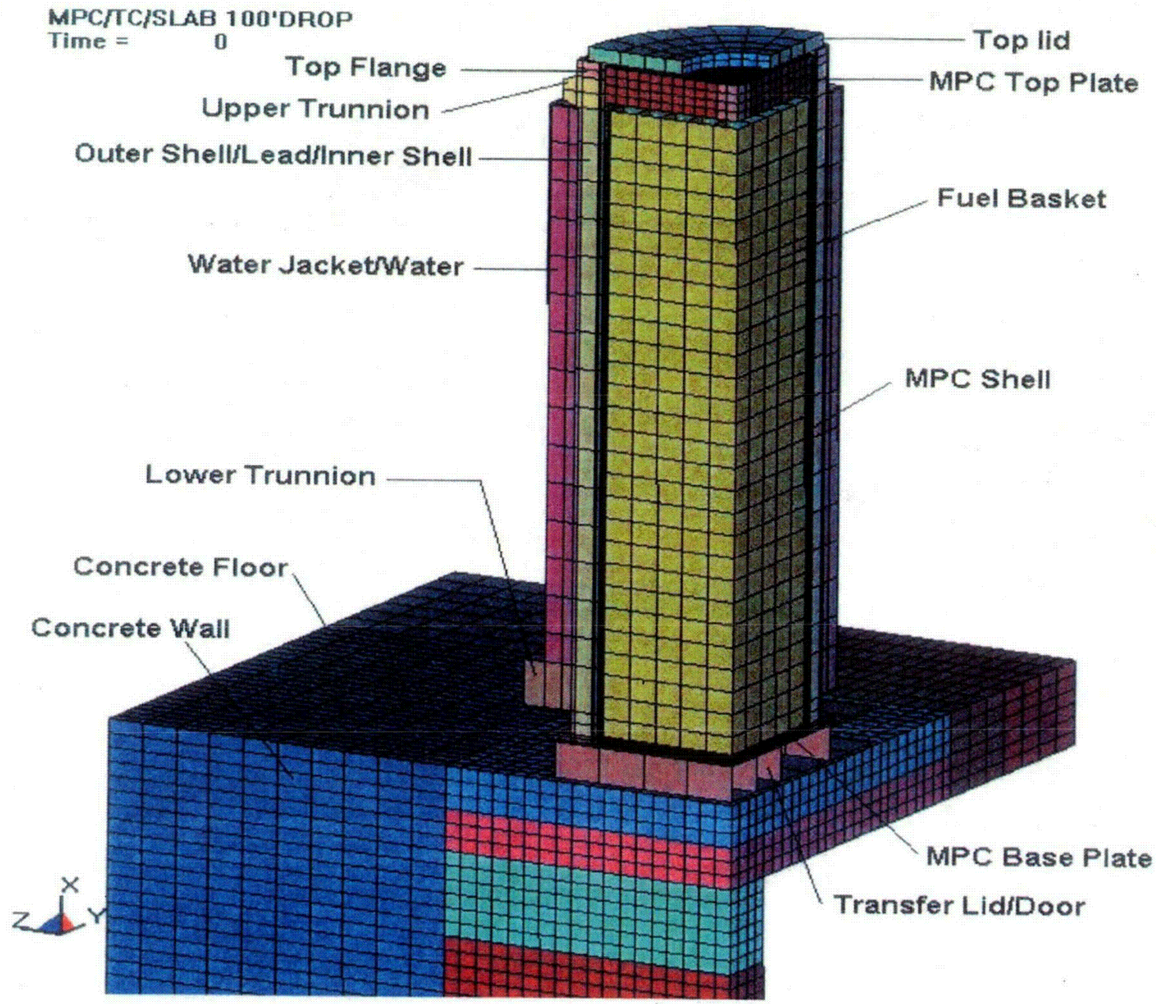
<u>19 ft Drop into SO</u>

("Hard" Impact)



LS-DYNA Finite Element Models Developed to Perform Drop Impact Analyses

- Geometry →
- Continuum approach



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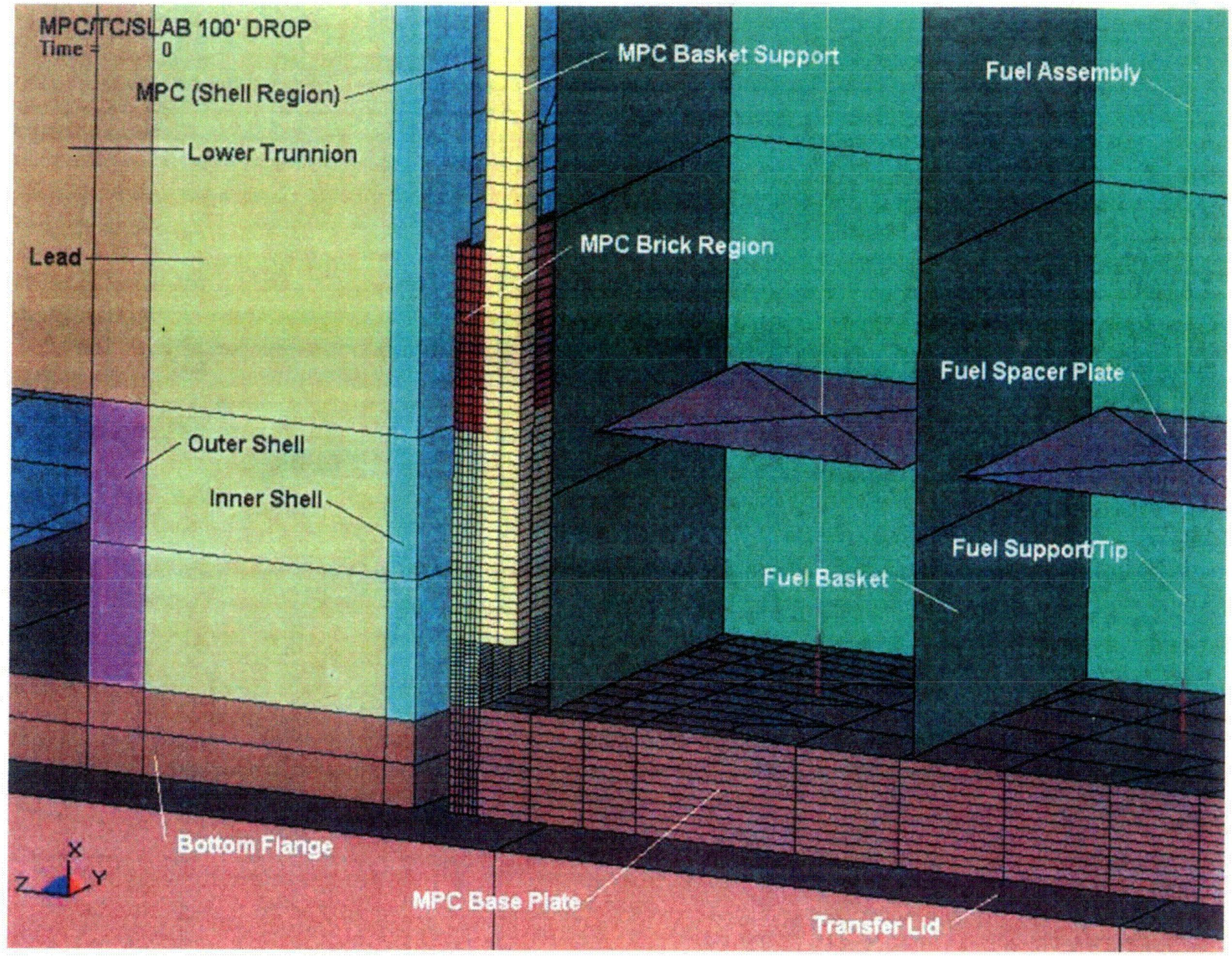
LS-DYNA Finite Element Models Developed to Perform Drop Impact Analyses

- Geometry



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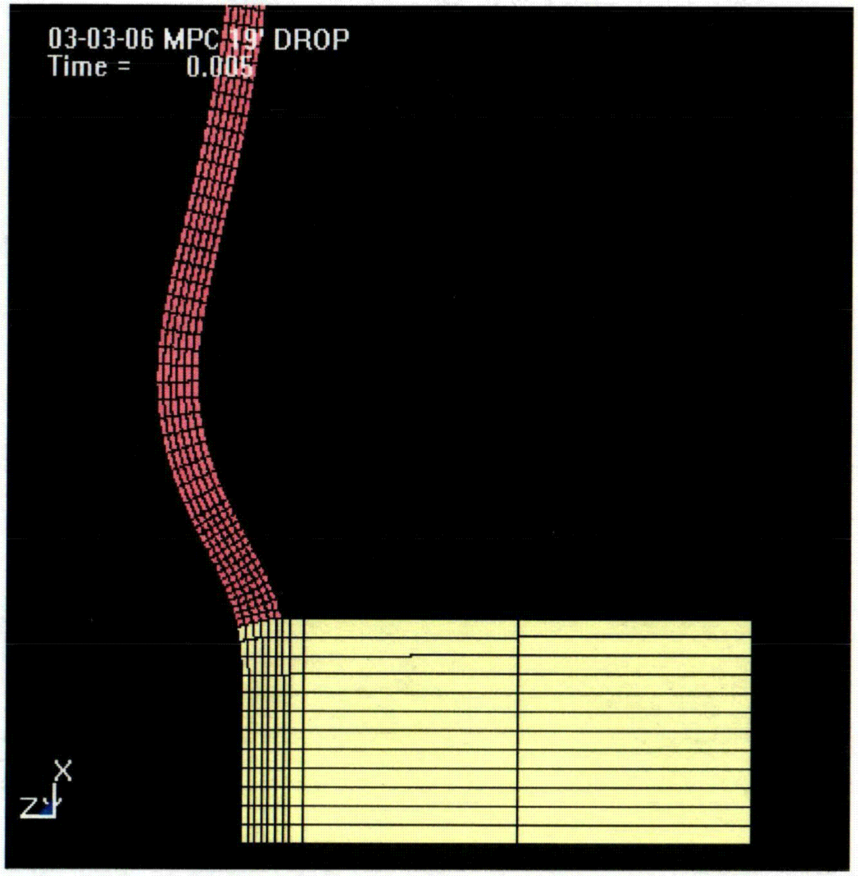
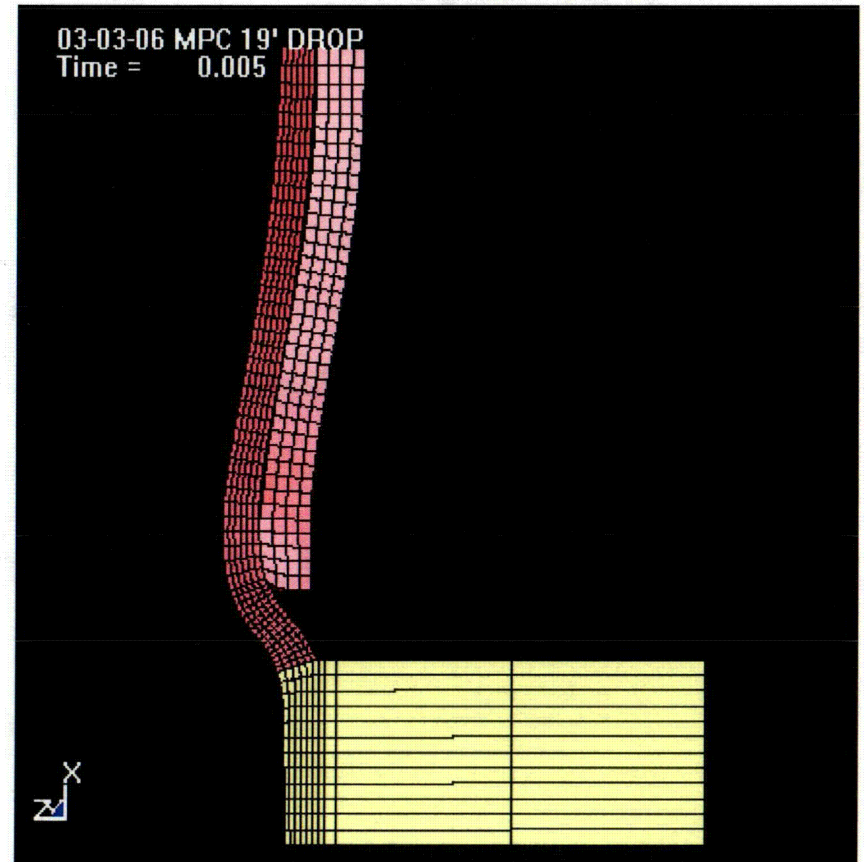
LS-DYNA Drop Impact Analysis

Deformed Shape at MPC Base

Away from Basket Support



At Basket Support



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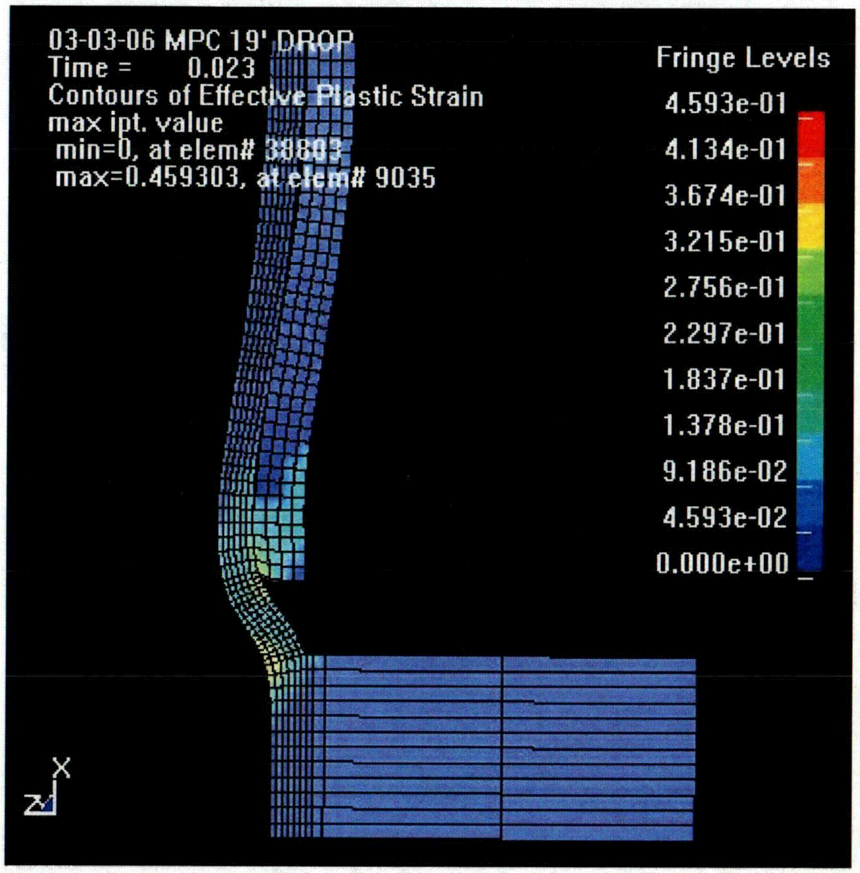
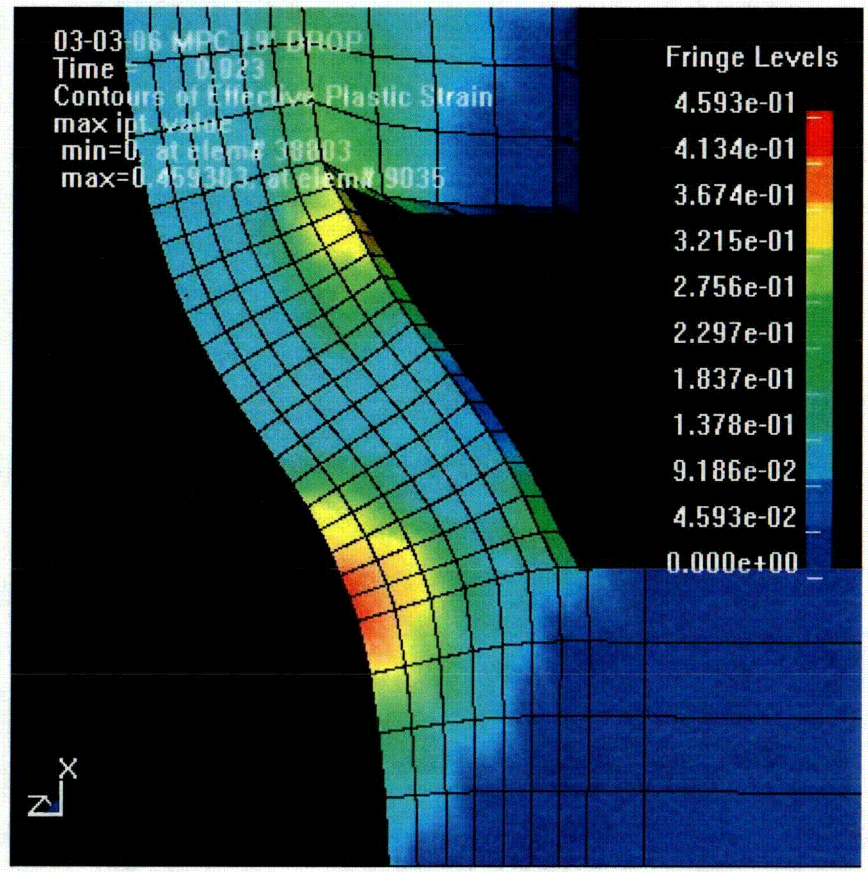
LS-DYNA Drop Impact Analysis

Strains in MPC Shell at Base for 19 foot Drop





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MPC Failure Criteria

- Most highly stressed region of the MPC is at the circumferential weld joining the shell to the base plate.
- Stainless Steel Weld Metal – Type 308
- Strain-based Failure Criteria based on test data of Type 308 SS weldments taken from NPP piping.
- From this data the Mean and Standard Deviation of the True Strain at Failure was calculated.
- Consistent with LS-DYNA

MPC Failure Criteria (continued)

- Failure Strain data adjusted for temperature and strain rate
- Factor of 0.88 applied to Mean Failure Strain

Table B.2

Standard Deviation from the Mean	Strain at Failure (in/in)	Probability Calculated Value < Failure Strain
0.0 (Mean)	0.73	0.5000
1.0	0.55	0.1587
2.0	0.40	0.0228
3.0	0.26	0.0013

MPC Failure Criteria (continued)

Adjust Failure Strain for State of Stress

- Strain at Failure based on Uniaxial (1-D) Tension Test Data
- Strain in LS-DYNA calculated from a complex Triaxial (3-D) State of Stress
- Triaxial State of Stress may constrain plastic flow and lower the Strain at Failure
- Triaxiality Factor calculated for each element from the 3 Principle Stresses in the element

MPC Failure Probability

Drop Height (feet)	Max. Strain LS-DYNA	Max. Strain Adjusted for Triaxiality	Probability of Weld Failure *
19	0.456	0.623	0.282
100	0.256	0.385	0.0196
5	0.024	0.048	< 0.000001

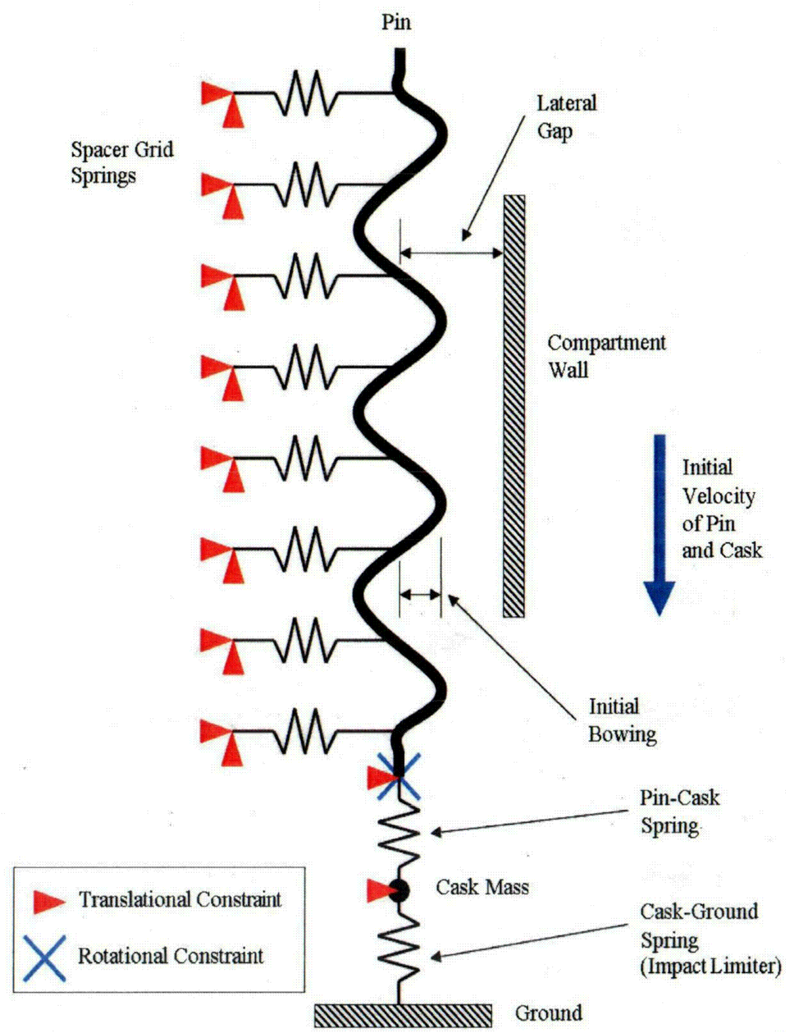
* This is the probability of one of the six elements thru the thickness of the MPC shell failing.

Evaluation of Fuel Cladding for Drop Events

- End Drop Impact
- High Impact Loads on Fuel Rods
- Static Buckling Formulas produce very conservative results (e.g., 1 ft drop predicts buckling and cladding failure)
- Magnitude and Duration of Loading Important
- High Loads, but very Short Duration
- Dynamic Problem

LS-DYNA Fuel Rod Model Developed

- Single Rod Model dictated by computational efficiency
- 20,000 Elements & 10,000 Nodes
- Cask-Ground Spring Stiffness developed thru iterative process to duplicate MPC Base Plate displacement time-history
- Cladding Mechanical Properties based on High Burn-up Fuel
- Cladding Failure Strain Limit for High Burn-up Fuel set at 1% Strain. (near the low end of the failure strain data)



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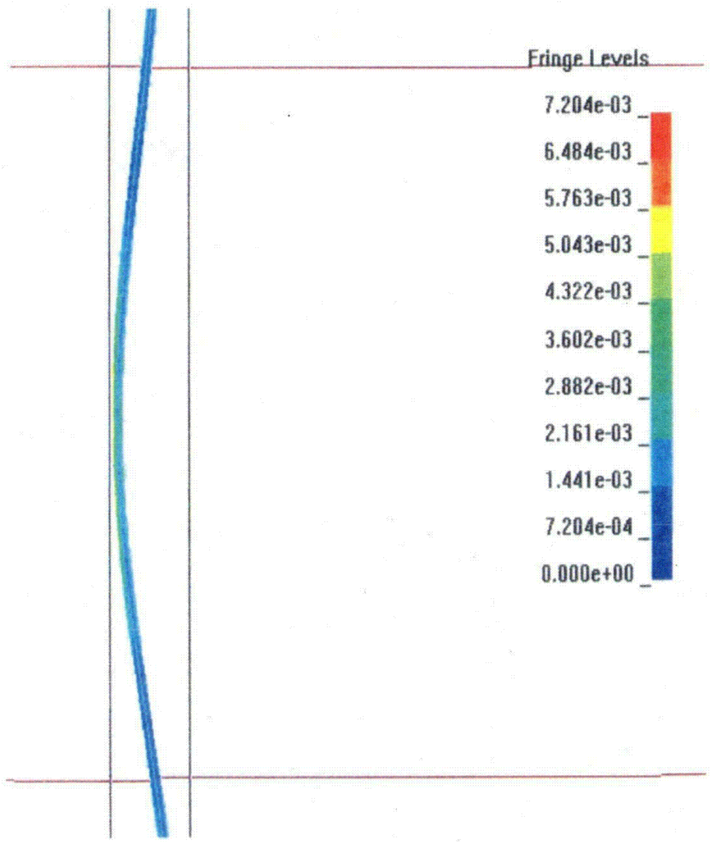
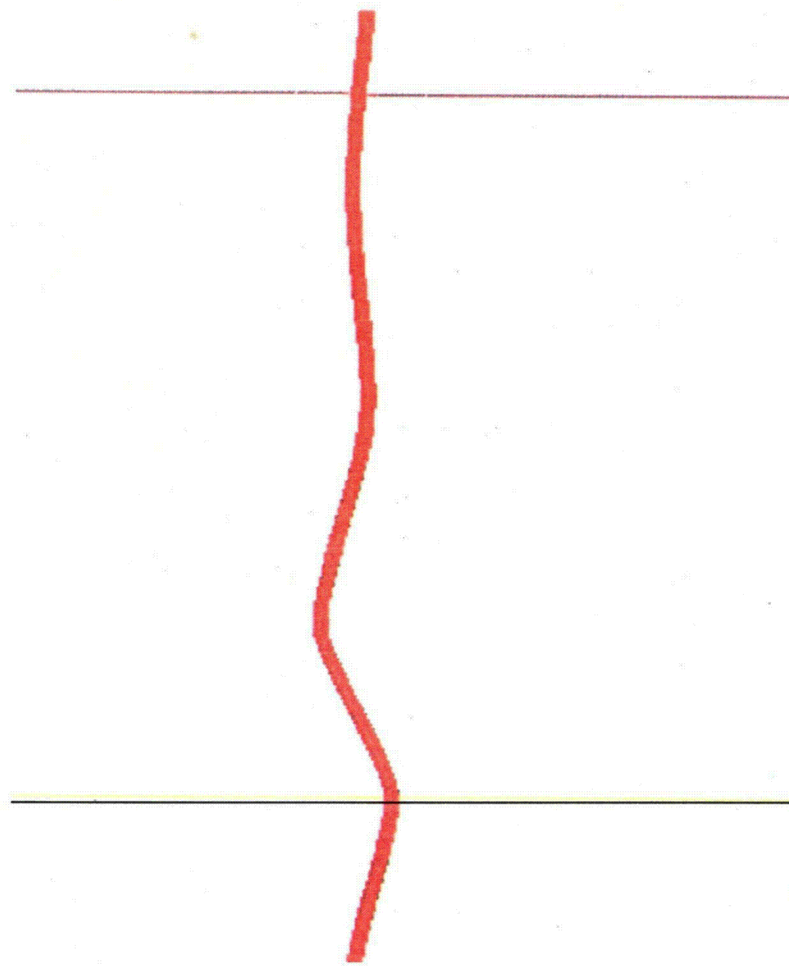
Fuel Rod Response to Impact

TC 20' Drop on Concrete Floor ("Soft" Impact) MPC 19' Drop into SO ("Hard" Impact)





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Fuel Cladding Failure Results

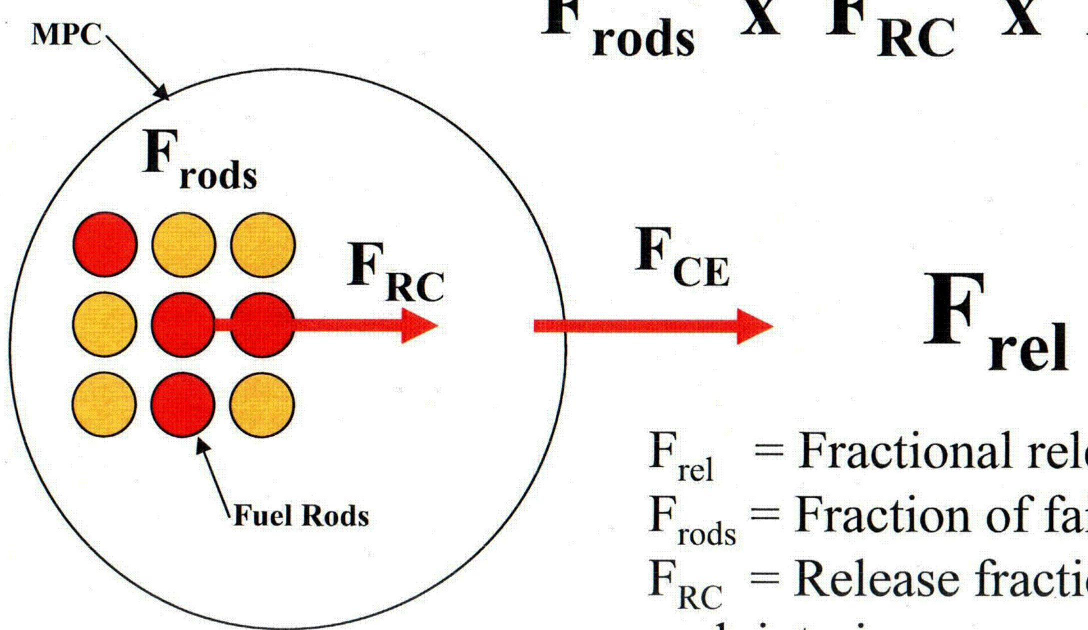
Table C.2

Event Scenario	Impact Surface (Target)	Drop Height (feet)	Maximum Principal Strain (in/in)	
Transfer Cask Vertical Drop	Concrete Floor	1	0.0043	
		5	0.0062	
		20	0.0072	< 0.01 in/in
		40	0.011	> 0.01 in/in
		55	0.025	
		70	0.037	
		100	0.052	
MPC Drop into Storage Overpack	Storage Overpack Pedestal	19	0.090	>> 0.01

Release Fractions Methodology

Governing Equation

$$\mathbf{F_{rods} \ x \ F_{RC} \ x \ F_{CE} = F_{rel}}$$



F_{rel} = Fractional release from the cask

F_{rods} = Fraction of failed fuel rods

F_{RC} = Release fraction from the rod to the cask interior

F_{CE} = Release fraction from the cask to the environment.

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Source Terms

$$ST_i = I_i \sum_k F_{rel,k}$$

ST_i = Source Term for i th Radionuclide

$F_{rel,k}$ = Release Fraction for k th Group

k = 1- Noble gases and Volatiles

2- Fuel particulate (rim and body)

3- CRUD

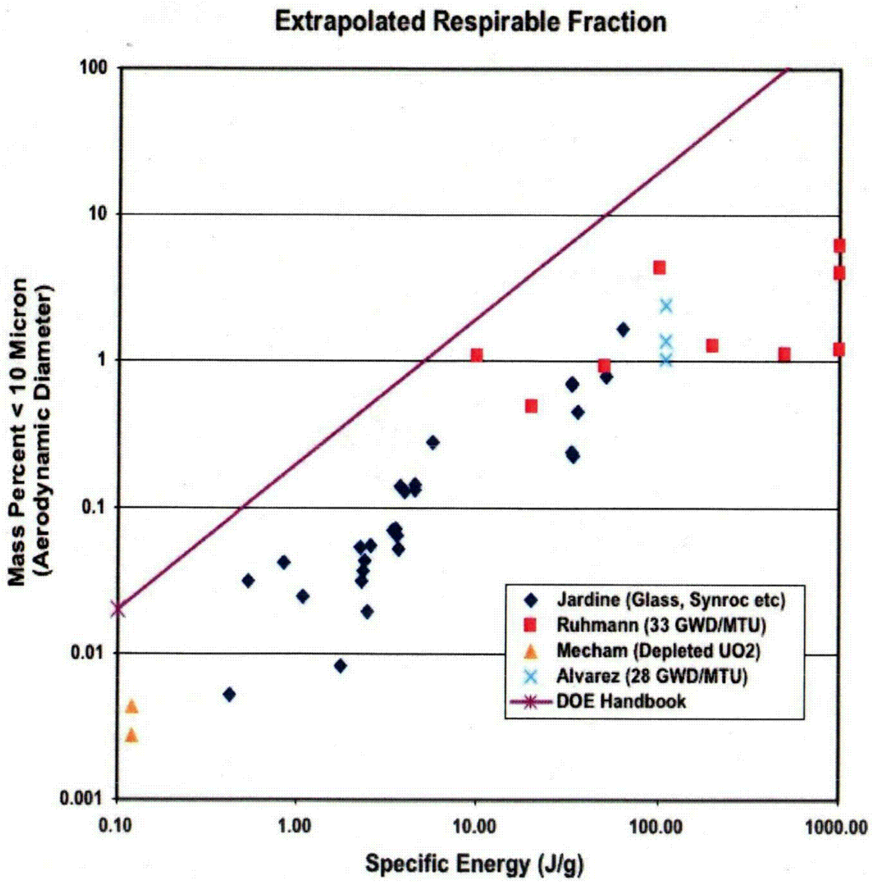
Model Limitations

- Only applicable to Impact Events
- Effects of fire on volatility of fission products and change in materials properties are not considered because no MPC failures occur due to thermal events.

Fuel Properties

- BWR (slight modifications for PWR)
- 60 GWd/MTU burn-up
- Rim effect in the fuel pellet considered
  (Actinide inventory in rim is higher than in body of the fuel and particle size is smaller, therefore rim and body considered as two distinct regions)

Release from Rods to Cask , F_{RC}



- Particulate release from rim and body regions analyzed separately.
- Fracture of the fuel into fines based on modification of equations in the DOE Handbook that relate the fraction of fuel that fragments into a respirable size to the impact energy that the fuel experiences.
- F_{RC} is dependent on
  - number of fracture sites in a rod,
  - entrainment of the fines in the gas stream during the depressurization of the rod, and
  - extent to which the rim region actually fractures
- Uncertainty in parameters leads to range of 7×10^{-5} to 1.2×10^{-2}.

C15

Cask to Environment Release, F_{CE}

	Fuel Fraction
☐ Particulates NOT Settling or Plating-out of the Pressurized Cask (5 atm + fill gas)	10 %
☐ Particles Exiting the Depressurizing Cask (Cask pressure drops from 5^+ atm to 1 atm, Ideal gas law)	100 %

Basis for CRUD Inventory

- Bounds 90% of Rods (0.72Ci/rod)
- Decreased for decay of Co-60 (10 years)
- Reduced by 2 for axial variation on Rods (data refers to peak)
- Scaled for burn-up but does not account for influence of water chemistry. (data is for low burn-up fuel)

Release Fractions

Chemical Element Group (k)	Inventory	F_{rod}	$F_{RC,k}$	$F_{CE,k}$	$F_{rel,k} = F_{rods} F_{RC,k} F_{CE,k}$ respirable
Noble Gases	ORIGEN	1	0.12	1	0.12
Particles	ORIGEN	1	7×10^{-5} to 1.2×10^{-2} *	0.1	1.2×10^{-3} (based on 1.2 x 10-2)
CRUD	0.72Ci/rod	1	0.015	0.1	1.5×10^{-3}

* Rim Fracture Fraction varied from ~0 to 1 and Entrainment varied from 0 to 1.

Out of Scope Issues

- Terrorism, sabotage or military actions
- Plant damage/accidents
- Fabrication/Future Cask design changes
- Misloading of spent nuclear fuel
- Explicit Consideration of Human Reliability Issues
- Worker Risk
- Aging effects of fuel during storage
- Multiple initiating events/failures

Out of Scope Issues (Cont'd)

- Unloading, offsite transportation, and repository storage
- Uncertainty Analysis of frequencies, probability distributions and consequences

Conclusions

- Risk is solely from latent cancer fatalities and no prompt fatalities are expected.

- The risk is dominated by accident sequences during the handling phase which involve drops of the MPC and transfer cask. The aggregated risk values are quite low.

- Many of the scenarios have zero risk because either their initiating events cannot occur at the subject plant or no radiological release will result.

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Probabilistic Risk Assessment of Bolted Dry Spent Fuel Storage Casks: Revisited

Ken Canavan
Senior Project Manager
Risk Technology
July 2006

Presentation Outline

- Spent Fuel Cask PRA Goals
- Methodology Overview
- Phase I – Overview
- Phase II – Overview
- Phase I & II Results
- Project Conclusions
- Future Uses

Goals of Spent Fuel Cask PRA

- Develop bolted cask PRA – based on Transnuclear cask
- Collaborate with U.S. NRC
- Better understand risks and consequences of on-site dry cask storage
- Develop risk insights regarding dominant risk contributors and potential cost reductions
- **Develop the tools required to support a risk-informed framework in the area of on-site spent fuel cask handling, transportation and storage.**

Methodology Overview (1 of 5)

- Basic Risk Equation:
  - Risk = Frequency * Consequence
- Answers three basic questions:
  - What can go wrong?
  - How likely is it?
  - What are the consequences?
- For dry spent fuel storage the risk problem is divided into three life-cycle phases:
  - Loading
  - Transfer
  - Storage

Methodology Overview (2 of 5)

- Dry spent fuel storage risk calculated similar to standard probabilistic risk assessment (PRA) techniques commonly used in operating nuclear facilities.
- PRA Elements:
  - Initiating events
  - Data Analysis
  - Human Action Analysis
  - Success Criteria
    - Structural Analysis
    - Thermal-Hydraulic Analysis
  - Accident Sequence Analysis
  - Consequence Analysis

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Methodology Overview (3 of 5)

- Scope of the Spent Fuel Cask PRA
  - Events not in scope
    - Acts of sabotage / terrorism
    - Damage to nuclear facility
    - Worker risk
    - Transportation to final repository
  - Events in scope
    - Design basis and beyond design basis events
    - Events resulting from handling, transfer and storage
    - External events including seismic, fires, high winds, floods nearby facility, pipelines, aircraft, and others

Methodology Overview (4 of 5)

- The PRA analysis is a ***realistic estimate*** of the frequency of occurrence as well as the consequences
- Represents ***average cask risk*** (e.g., average enrichment, average burnup, and average fuel age)
- Selected highlights of methods employed:
  - Initiating events
    - Final initiators result from a combination of generic lists and master logic diagram (MLD) approach
    - Frequency of cask drops calculated from fault tree of typical nuclear power refueling building crane
    - Potential for spent fuel mis-loading considered

Methodology Overview (5 of 5)

- Selected highlights of methods employed:
  - Success Criteria – Structural Analysis
    - Uses fragility approach
    - Acceleration dependent on target hardness (as opposed to unyielding surface)
  - Success Criteria – Thermal-Hydraulic Analysis
    - Assumes average fuel
  - Accident Sequence & Consequence Analysis
    - Assumes two fuel pins fail for all acceleration events (i.e., cask drop events)
    - Initially no building mitigation of releases (phase I)
    - Ground level release assumed (phase I)
    - Conservative source term (phase I)

Phase I – Project Overview (1 of 2)

- Studied bolted cask design (NRC work is on welded cask)
- Performed representative PWR study (NRC study is BWR)
- Study is “generic” (NRC is plant specific) in that:
  - No particular site is modeled
  - Where required, the assumed location is Eastern US
  - Some hazards assumed (e.g., natural gas pipeline explosion)
  - A general layout and handling is based on Prairie Island and Peach Bottom
  - Others

Phase I – Project Overview (2 of 2)

- As with all PRAs, simplifying assumptions employed. Some of the more significant assumptions include:
  - Generic study
  - Cask loading assumed a two step process with two lifts
  - Acceleration related events (drop) assumed to fail two fuel pins
  - Horizontal drops within refueling building have high uncertainty and therefore higher probability of cask failure
  - Building mitigation of potential dose not modeled
  - Ground level doses assumed
  - Limiting weather conditions assumed
- "Probabilistic Risk Assessment (PRA) of Bolted Storage Casks: Quantification and Analysis Report", EPRI 1002877, Topical Report, December 2003.

Phase II – Goals and Objectives (1 of 2)

- Goal: To reduce conservatisms in the phase I study
- Objectives:
  - Lower, more realistic assessment of spent fuel cask risk
  - Better comparison with NRC Cask PRA (when completed)
  - More flexible tool for risk-informing regulations and informing the public
  - Reduced potential for mis-interpretation of results
- "Probabilistic Risk Assessment (PRA) of Bolted Storage Cask (Update)"; 1009691 (November 2004).

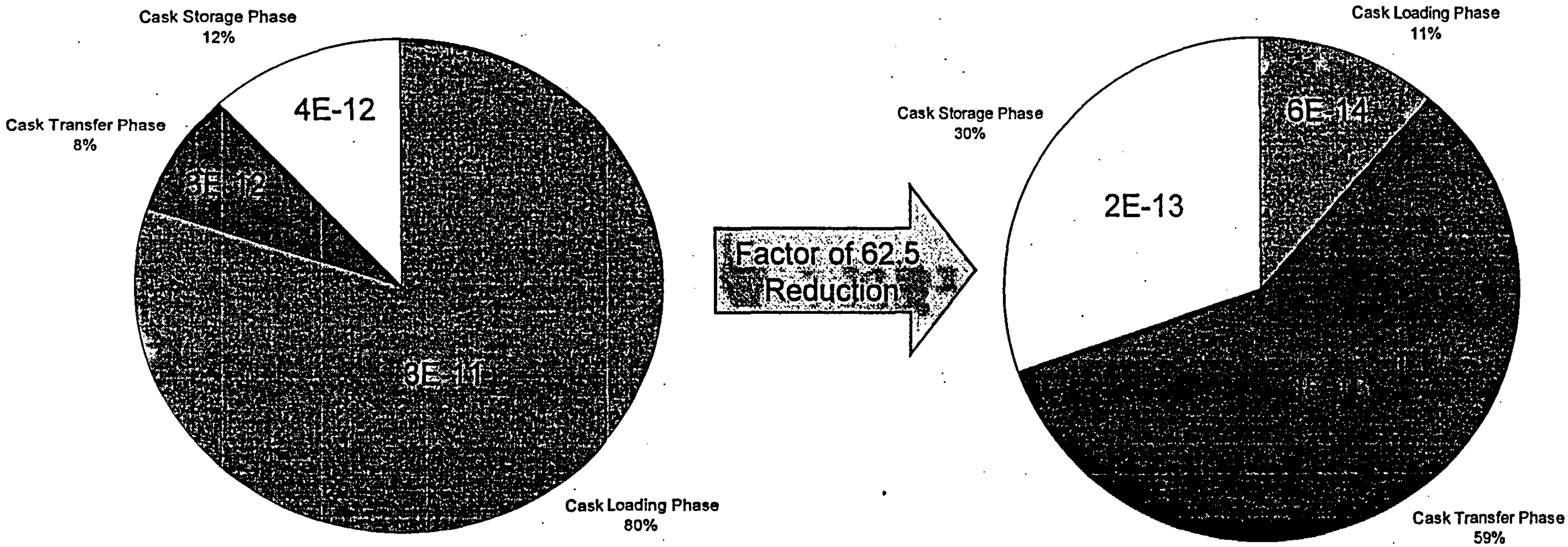
Phase II – Activities (2 of 2)

1. Revise cask drop frequency (NUREG-1774 vs. NUREG-0612)
2. Re-evaluate uncertainty
3. Evaluate alternate source terms. ISG-5 not intended for use in PRAs
4. Revise the assumption associated with the mitigation of releases. Aerosol deposition and building HVAC mitigation
5. Consider elevated pathways for release from buildings
6. Investigate the impact of alternative, more realistic, weather conditions
7. Investigate "Intact" versus "Damaged" fuel rods. Two levels of damage:
    a. Tight cracks and pinholes (classified as "non-damaged" at present)
    b. Larger defects
8. Assess the conservatisms in the storage phase, and potential additional accident sequences for a 20-year duration of the storage phase.

Items 7 & 8 Were Not Performed in Phase II

Phase I & II – Project Results (1 of 2)

- Phase I Results
  - Total prompt fatalities = 0.0
  - Total cancer fatalities = 3.5E-11 per cask per year
- Phase II Results
  - Total prompt fatalities = 0.0
  - Total cancer fatalities = 5.6E-13 per cask per year



Welded Design Results: 0 prompt fatalities, 2.0E-12 first year, 1.9E-13 per year subsequent years

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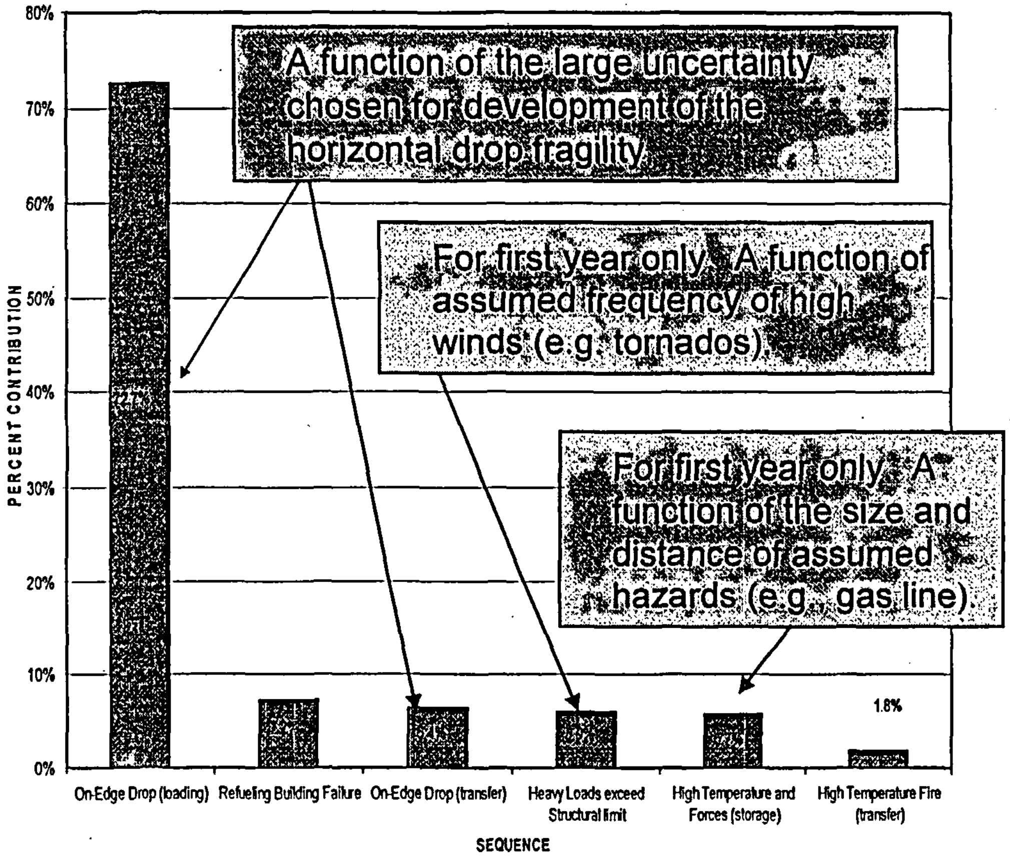
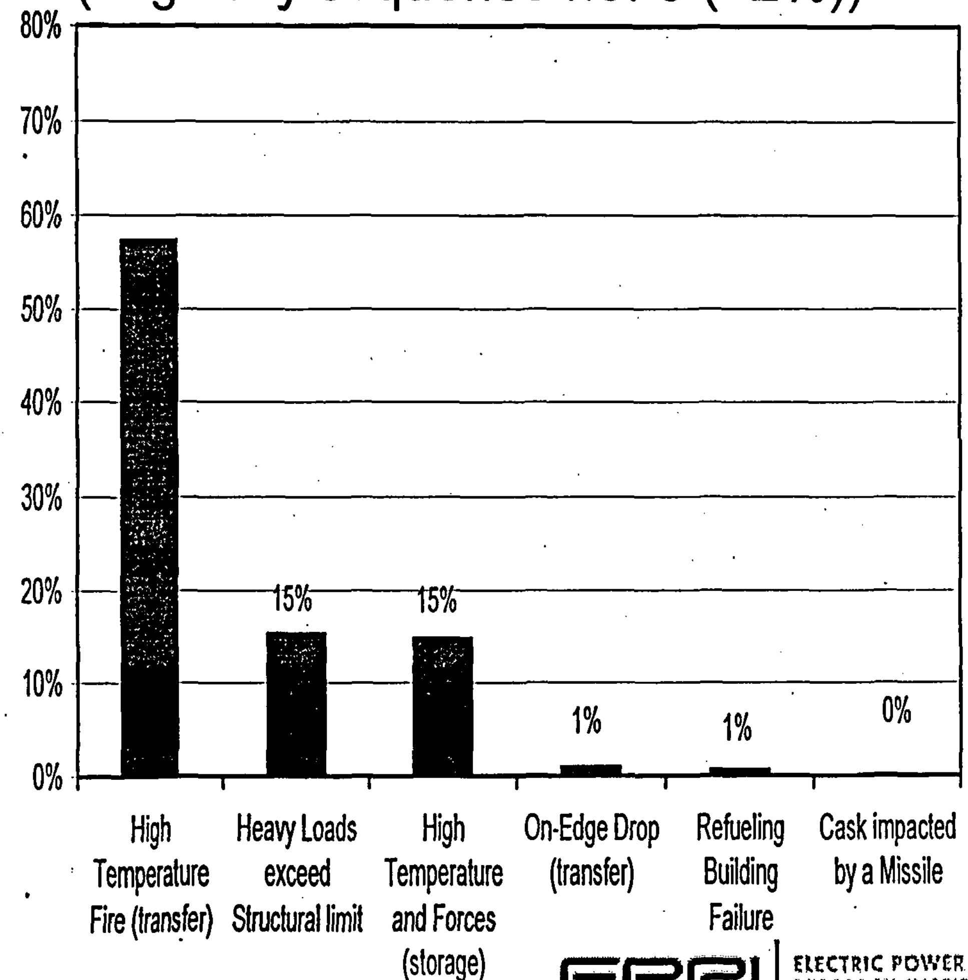
Phase I & II – Project Results (2 of 2)

- Top sequence (73%) is a horizontal drop of the cask in the refueling building



- Top sequence (58%) is a high temperature fire during transfer (originally sequence no. 6 (~2%))



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Project Conclusions

- Phase I Project Conclusions
  - Low risk for a bolted design dry fuel storage systems
  - Driven by a relatively small number of key assumptions as well as the site specific hazards.
  - The use of a risk-informed approach could achieve both cost and safety benefits
- Phase II Project Conclusions
  - Confirmed low risk for a bolted design dry fuel storage systems
  - Showed that risk is significantly driven by small number of assumptions and plant specifics
  - Additional analysis is warranted if cost-benefit can be justified through burden reduction
- The use of a risk informed approach to dry fuel storage could achieve both cost and safety benefits.

Potential Future Uses of Cask PRA Technology

- Improve public perception of SF Cask storage options
- Perform risk trade off analysis between repair versus as-found
- Enforcement discretion for discovered deficiencies
- Identify areas for reduced margins in future cask designs
- Identify and reduce burden associated with regulatory environment and requirements
  - Increase allowed boundary dose
  - Reduce inspections
- Review regulations to assist in licensing of new storage or expansion of existing facilities

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