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NUCLEAR REGULATORY COMMISSION

Title:	Toshiba's Request for Pre-application Review of the Super-Safe, Small, and Simple (4s) Reactor Design: Public Meeting
Docket Number:	Project No. 0760
Location:	Rockville, Maryland
Date:	Tuesday, October 23, 2007

Work Order No.: NRC-1841 Pages 1-119

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U.S. NUCLEAR REGULATORY COMMISSION

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TOSHIBA'S REQUEST FOR PRE-APPLICATION REVIEW OF THE SUPER-SAFE, SMALL, AND SIMPLE (4S) REACTOR DESIGN

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PUBLIC MEETING

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TUESDAY,

OCTOBER 23, 2007

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The public meeting convened at 1:00 p.m. in the NRC Auditorium, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, Donald E. Carlson, Office of Nuclear Regulatory Research, presiding.

NRC STAFF PRESENT:

DONALD E. CARLSON, New and Advanced Reactors Branch, Office of Nuclear Regulatory Research

JOHN R. JOLICOEUR, Chief, New and Advanced Reactors Branch, Office of Nuclear Regulatory Research

TOSHIBA PARTICIPANTS:

NORIHIKO HANDA

OSAMU MAEKAWA

AKIRA OZAKI

YOSHIAKI SAKASHITA

CRIEPI PARTICIPANT:

NOBUYUKI UEDA

WESTINGHOUSE PARTICIPANTS:

REGIS MATZIE

BRADLEY MAURER

RICHARD WRIGHT

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

1:02 p.m.

MR. CARLSON: Welcome, everyone. My name is Don Carlson. I am from the NRC's Office of Nuclear Regulatory Research, the New and Advanced Reactor's Branch and I will be leading today's meeting.

In addition to the people here in the room, we also have people who have phoned in on the bridge line as public observers. We will let them introduce themselves in a moment and there will be as shown on the agenda an opportunity for public comment at the close of the meeting.

The purpose of this meeting is to discuss items related to Toshiba's request for a preapplication review of its 4S reactor design and to allow Toshiba the opportunity to present details of its design and safety and security characteristics.

This meeting was, in fact, requested by Toshiba and granted by the staff pursuant to the Commission's Advanced Reactor Policy Statement which encourages the developers of advanced reactor designs to engage the staff in early interactions in order to provide all interest parties including the public with a timely and independent assessment of the safety and security characteristics of advanced reactor designs.

Okay. The next slide please. Okay. There we go. Thank you.

First, I'll start off with some formalities and then I'll be talking about some high-level questions that both the preapplicant and the NRC need to think about, go over the preapplication objectives, get a few examples of possible preapplication topics and then briefly review the regulations for radioactive material transportation as being a potential area that this design would have unique attributes and then finally, I'll provide a summary of some acronyms.

First of all, this is a Category 1 public meeting. The protocol for which is that members of the public are welcome to observe both in person and by the bridge line phone and other than the introductions in a moment, their comments and questions should be reserved for the final session of the meeting which is scheduled to start at 4:45, the public comments session.

Please be careful to speak into the microphone. We do have a court reporter and so, there will be full transcripts issued of this meeting by next week sometime.

And so, now, we will go around the room

starting first with the people at the table from Toshiba.

Regis, you want to start?

MR. MATZIE: Yes. My name is Regis Matzie. I'm the Senior Vice President and Chief Technology Officer for Westinghouse Electric Company.

MR. UEDA: I am Nobuyuki Ueda, Senior Research Scientist and Sector Leader over Advanced Reactor Systems in Nuclear Technology Research Laboratory, Central Research Institution of Electric Power Industry, CRIEPI. Thank you.

MR. MAURER: I am Brad Maurer with Westinghouse. I'm a Principal Engineer with Westinghouse.

MR. WRIGHT: I'm Rick Wright and I'm from Westinghouse and I work in the New Plant Projects Area.

MR. MAEKAWA: Osamu Maekawa, Technology Executive of Toshiba Nuclear Energy Systems.

MR. HANDA: I'm Norihiko Handa. I am a Senior Fellow of Toshiba.

MR. SAKASHITA: Yoshiaki Sakashita, Manager from Toshiba Corporation.

MR. OZAKI: Akira Ozaki, Senior Manager of Toshiba Corporation.

MR. JOLICOEUR: And I'm John Jolicoeur. I'm Chief of the New and Advanced Reactor's Branch in Office of Research.

MR. CARLSON: I would like to remind everyone here in person that to use the sign-up sheets. If you haven't signed up already, please do so during the breaks.

Also, out in the lobby or the atrium of this auditorium, you will see that our Toshiba friends have brought with them a very nice scale model of their reactor design. So, you'll want to look at that during the breaks as well.

Now, I'm going to give the people who have dialed in, and I really don't know how many have at this point, a chance to introduce themselves and how many people do we have? Does anybody have an idea?

MR. SHAHROKHI: This is Farshid Shahrokhi with Areva NP in Lynchburg, Virginia.

MR. CARLSON: Okay. Thank you. Next.

MS. RUNNER: This is March Runner, Louden Tribal Council in Galena, Alaska.

MR. CARLSON: Louden Tribal Council in Galena, Alaska, could you say your name again?

MS. RUNNER: March like the month. Runner like around a track.

MR. CARLSON: Thank you. Next.

MR. DEMOSKI: Peter Demoski from Nulato, a small village 50 miles down river from Galena, the proposed site of this nuclear facility.

MR. CARLSON: Did we get that name? Could you repeat it please?

MR. DEMOSKI: Peter Demoski D-E-M-O-S-K-I.

MR. CARLSON: Thank you. Peter Demoski. Next please.

MR. BURKE: Richard Burke, ISO New England.

MR. CARLSON: Thank you. Next please. Is anyone else on the line?

MR. ROSENFELD: Yes, Robert Rosenfeld R-O-S-E-N-F-E-L-D. I work for the Yukon River Inter-Tribal Watershed Council which is a treaty organization made up of 66 indigenous government from one end of the Yukon River to the other. I'd like to be recognized tack on some comments towards the end and invite Toshiba to call me at their earliest 907-388-2683. Thank you very much.

MR. CARLSON: Thank you, Robert. Next. Do we have anyone else on the line? Bodony. Yes, I have your name. Thank you. Is there anyone else? Last call. Okay. Thank you.

Okay. Yes, you heard that we have some members of the First Nations and Tribal Nations in the area who are interested and we do have a Tribal Consultation Working Group in the Agency to work with them. That, however, is not the purpose of this meeting and for the purposes of this meeting, they are public observers and they'll be welcomed to have comments at the end.

The purpose of this meeting again is to address Toshiba's request for a preapplication review for approval of the design of their 4S reactor. This is one of a number of processes that the Agency will have leading up to design approval or design certification and site approval or early site permits and all of these have various opportunities for public participation at various levels.

So, thank you for your participation.

I would also note that we have public comment feedback forms posted outside on the table and members of the public are invited to use those. Welcome your feedback.

Next slide then.

First of all, let's look at what some of the high-level questions that the preapplication needs to consider.

First of all, what level of review is desired for the preapplication? What will be the submittals? Informal? White paper submittals? Formal? Topical reports? Safety analysis reports? How open items will be treated in going forward after the preapplication review to the review of an eventual application which would be read -- which would be lead, excuse me, by the Office of New Reactors.

Also, what is your schedule for submittal of the application and finally, what will the design be formally aligned with a customer? This, of course, would effect how the Agency views the priority of allocating resources to these review activities.

Next slide.

High-level questions for the NRC, what priority do we, the NRC, give these efforts relative to ongoing efforts on advanced light-water reactor combined license applications and also, the support for DOE's next generation nuclear plant in global nuclear energy partnership initiatives. The pebble-bed modular reactor preapplication review which has been ongoing and potential preapplication reviews for other proposed reactors such as the IRIS, the High Temperature Teaching and Test Reactor, Hyperion and so on.

A key point in all of this in determining what priority and level of effort we allocate to this is the availability of NRC resources in the coming years as well as technical review staff with appropriate expertise for non-LWR efforts. This are very important considerations.

Next slide.

The preapplication objectives are to, in our view, familiarize the NRC with the design. In this case, an LMR design, liquid metal cooled reactor, building on insights from our past liquid metal cooled reactor reviews many years ago for the Clinch River Breeder Reactor Project and in the early '90s for the PRISM Reactor.

Also, we have ongoing interactions with DOE on liquid metal reactor technology in the context of the Global Nuclear Energy Partnership Project and finally, we have plans for LMR, liquid metal reactor, infrastructure research in the Office of Research.

We would like to provide feedback to the Applicant on key safety, design and licensing issues and their technology development programs in support of licensing.

Furthermore, these activities should help us plan and inform our infrastructure development work for research in the area of liquid metal cooled reactors and provide for early identification of any policy issues that may come up.

Just listing some examples of preapplication focus topics that we have seen in past reviews and we might anticipate in upcoming preapplication reviews:

Analysis codes and their validation for analysis of reactor safety is a frequent area of consideration;

The approach to defense-in-depth, testing and monitoring both for the reactor R&D program for design and for the operational reactor once it's built;

Safety classifications and treatments of systems, structures and components;

The selection of licensing basis events, transportation of radioactive materials, fuel cycle and waste issues unique to the fuel cycle for a new design, source term generation that refers to the radio nuclides available for release in severe accidents and long-term fuel performance might be of interest in this particular review;

Probabilistic risk assessment and how it is used;

Nuclear analysis meaning in some reactors there's an interest in reactivity feedback coefficients for example;

Physical protection and security especially in the aftermath of the events of September 11th, 2001, and;

Emergency planning.

There is no clear script that the NRC staff follows for preapplication reviews. For licensing reviews, there are standard review plans. There are specific regulations that those correspond to.

In the case of preapplication review, it really is the preapplicant's responsibility and prerogative to identify the topics that it wants to staff to cover in the preapplication review, of course, in consultation with the staff if desired.

Finally, I'll give a little overview of the transportation regulations for radioactive material. The reason I felt this might be of interest is following the evolution of the 4S design, there have been statements about transportability and so, those seem to be of interest. So, reminding what the regulations are in that domain, they are 10 CFR; that is Title 10 of the Code of Federal Regulations Part 71 for domestic shipments. For international shipments of radioactive materials, there are corresponding and very similar regulatory requirements in IAEA Safety Standard Series Number TS-R-1.

These regulations apply to transporting nuclear fuel from the fabricator to the proposed reactor site, radioactive waste from the proposed reactor site to a centralized low-level waste facility and finally, spent nuclear fuel from the proposed reactor site to a centralized geologic repository or interim storage facility or reprocessing facility.

And you can quickly go through the acronyms yourselves if you're interested.

And with that, I thank you and would like to turn it over now to our guests from Toshiba.

MR. MAEKAWA: Good afternoon, ladies and gentlemen. My name is Osamu Maekawa, Technology Executive of Toshiba Nuclear Energy Systems.

On behalf of Toshiba, I'd like to thank the Nuclear Regulatory Commission for this opportunity.

It's our great pleasure to be here to address the NRC for 4S which stands for Super-Safe, Small and Simple Reactor preapplication review.

4S has the collaboration of Toshiba and

the Central Research Institute of Electric Power Industry of Japan since 1988. 4S is a small sodium-cooled fast reactor with long-life core.

We believe 4S is the best fit to the requirement of energy for remote or developing areas.

Nuclear is an affordable emission free energy source and one with the most promising choices. Again, this global warming by carbon dioxide emissions.

We wish to contribute to the benefit of nuclear energy to a wide area of the world through our 4S reactors.

Let me introduce our company, Toshiba, briefly. We entered nuclear power business in 1966. Since then, we have been continuing to construct nuclear power plants. We have constructed 22 boiling water reactors over the past 40 years. Seventeen of them were as a main contractor.

In addition to those BWR construction projects, we played a major role in constructing advanced reactors such as fast breeder reactors, JOYO and MANJU.

We cover the full spectrum of nuclear technologies including research and development, detail design, production and procurement, construction through the operation and maintenance.

In the first reactor development in Japan, we played a major role as the management company in the experimental reactor JOYO and provided coring tunnels, control rod drive, the containment vessel and other equipment.

For the prototype reactor MANJU, we supplied the reactor closure head, the intermediate heat transport system, the traveling generator system and other equipment.

In addition to those, we have been continuing design and research and development activities for demonstration and commercial first reactors.

Now, I'd like to introduce the 4S team. Along with Toshiba, 4S Development Team consists of Central Research Institute of Electric Power Industry of Japan, Argonne National Laboratory and Westinghouse Electric Company.

Our team hopes today's meeting will give you a good understanding of our plan and technology. We look forward to working with you to achieve the certified design of the 4S.

Thank you all for coming to this meeting and thank you for your attention.

At this time, I would like to introduce Dr. Ueda of CRIEPI. Dr. Ueda, please.

MR. UEDA: Good afternoon, ladies and gentlemen. I am Nobuyuki Ueda, Senior Research Scientist of Central Research Institute of Electric Power Industry, CRIEPI.

It gives us considerable pleasure to have an opportunity to present our 4S reactor.

First, I briefly introduced our institute. Central Research Institute of Electric Power Industry, CRIEPI, has been extensively contributing to the society through research and development for insuring the stable electricity supply since the establishment in 1951. CRIEPI is mainly supported by Japanese utilities and CRIEPI's similar institute is EPRI in United States.

We have two missions. As a first mission, we conduct fundamental pioneering research on the electric industry.

The second mission, we offer the results of this research to the public for the benefit of all.

And we set up the three research target: Cost reduction and greater reliability; harmonization of energy and environment; creation of comprehensive energy service.

We select research topics and consolidate resources based on five spheres. One of the spheres is nuclear technology. In the nuclear technology sphere, there are light water reactor technology, fuel cycle technology, back-end technology, spent fuel waste technology and fast reactor technology.

We started fast reactor research and development in early 1980s. We have carried out the conceptual design study of innovative fast reactors, the pilot process fuel recycling technology and the fundamental fast reactor design technology, high temperature performers over structural materials, seismic isolator and share backend performers over reactor vessel.

We have also developed several numercial codes as design tools. 4S has been developed by compiling these technology as well as the technologies of Toshiba and Westinghouse.

We would like to present the 4S to you with all our technologies.

Thank you for your attention.

MR. MATZIE: Again, my name's Regis Matzie. I'm the Chief Technology Officer of Westinghouse Electric Company.

On behalf of Westinghouse, I would like to thank the Nuclear Regulatory Commission for this opportunity to support Toshiba in their preapplication efforts to certified 4S.

As you are aware, Westinghouse is the world leader in advanced reactor design. We have a long history of innovation starting with the power plant for the U.S.S. Nautilus, the world's first nuclear submarine and continuing with the first U.S. commercial nuclear power plant at Shippingport, Pennsylvania. Over half of the world's operating reactor power plants are based on Westinghouse technology when the plants built by our licensees are included.

Our experience with liquid metal cooled reactors is also very significant. We were the prime contractors for both the Clinch River Breeder Reactor and the Fast Flux Test Reactor and we also participated in the advanced liquid metal reactor program called PRISM thereafter.

Unfortunately, the United States turned away from liquid metal cooled reactor technology in the 1980s and thus, there's been a gap in our intimate involvement in this technology.

Nevertheless, Westinghouse has not been standing still. We have developed three advance light water reactors over the past two decades and achieved design certification for all three of these. The last being the AP-1000 advanced passive plan.

In addition, we are involved with design development and the preapplication review of two other advanced plans, IRIS and PBMR.

Thus, Westinghouse's leadership in advanced reactor licensing continues. We are pleased to apply this experience and leadership in licensing advanced nuclear plants to the 4S reactor. We will work closely with our colleagues from Toshiba to insure that this licensing effort is successful and that it meets the expectations of the Commission.

Thank you very much for allowing me to make these brief remarks.

MR. MAURER: Good afternoon. My name is Brad Maurer. I'm with Westinghouse.

I would first like to add my thanks to the NRC staff for meeting with us afternoon and in particular, I would like to recognize Don Carlson in his efforts in helping us to arrange this meeting this afternoon.

We are here in what we hope is the first of series of preapplication meetings on the 4S reactor.

Before I go on, I would like to echo what Don mentioned at the breaks. Please feel free to go into the atrium and take a look at Toshiba's model. It's quite impression. There will be Toshiba and Westinghouse people nearby to answer your questions.

Also, Don, I would like to ask your help. If we run long and need a break, there are a number of natural breaks in the presentation. So, just let me know.

The goals of the 4S reactor are to provide clean, safe, reliable and grid-appropriate power. Grid appropriate means that the reactor is sized to fit the need and locations where large power plants or transmission of power from large power plants is not practical or available.

The second goal is to minimize security and proliferation risks and the third is to minimize the infrastructure operation and maintenance requirements of the plant.

Our meeting objectives today are to provide an overall introduction of the 4S, its basic operation, its main safety features, what we've done so far in the design and the verification of that design.

We wish to gain U.S. NRC guidance on the overall process to be followed from licensing the 4S in the U.S. We would like to offer our approach and request feedback from the staff on that approach. We would like to establish a continuing dialogue with the NRC staff. As you will see, we hope that the flow of information can also continue to the further submittal of technical reports and through further meetings.

Our goal is to submit the design approval application by 2009.

This is our presentation agenda today. We will go through a high-level overview, plan design parameters, regulatory conformance, main design features, metallic fuel experience and design verification testing, safety analysis and our conclusions. Well, this is not one-to-one with the published agenda in the public meeting notice, we will address all of those items and those included the 4S overview, current status of the design, testing and analysis that has been performed and that is planned to be performed and initial submittals of technical reports.

We want to give you a feel for what the 4S looks like, it's basic features and how it operates. Also, a status of the design and our proposed licensing approach.

The 4S is a sodium-cooled fast reactor. The size is 30 megawatts thermal which is about 10 megawatts electric. This is roughly 1 percent of the size of a typical reactor such as the AP-1000 which is about 3400 megawatts thermal.

The application for a small reactor like 4S is for areas of small concentrated need such as the city of Galena in Alaska. It also has the potential for oil sand processing seawater desalination. It is considered a candidate for the GNEP grid appropriate in small and medium reactor design. This means it's a candidate for funding for research and testing. We're hopeful currently for additional funding from CRIEPI in Japan through the GNEP process.

The main features of the 4S include passive safety both in the negative coolant temperature reactivity coefficient in the design of the core and also the decay heat removal systems.

There is no on-site refueling for 30 years with the 4S. There is low maintenance requirement with the 4S. This comes from the no refueling and the use of electromagnetic pumps and finally, a high inherent security.

As can be seen from the sketch, the bulk of the plant can be sighted below grade and there is no on-site storage of either new or spent fuel.

The core of the 4S is made up of metallic fuel, uranium and 10 percent zirconium. The activity control is by movable reflectors. There are no control rods. This utilized the high-leakage core design which allows the reflector control of the reactor.

The shutdown system is by the reflector which drop out of the core region if they need to shutdown the reactor and also by the shutdown road. So, we have not only diverse, but redundant shutdown systems.

The primary heat transport system includes the annular type electromagnetic pumps which have no moving parts and moves the liquid sodium through the reactor by induced magnetic fields. Also, the intermediate heat exchanger which is the IHX. This is an annular type inside the reactor. This is where the heat from the primary system, primary sodium system, inside the reactor is transferred to the intermediate sodium heat transfer system.

Also, I would point out that the containment system consists of a guard vessel around the reactor vessel which is somewhat difficult to see in this picture and also, the top dome which you can't see above the reactor in this shot -- in this picture. The top dome encloses the reflector drives, the shutdown drive and the other penetrations at the top of the reactor vessel.

There are two heat transport systems. The primary heat transport system as I mentioned is totally contained inside the reactor. The intermediate heat transport system is the steam generator, the EM pump, the air cooler and dump tank and then we have the water and steam system to the turbine generator.

The primary coolant is circulated up through the core in the primary coolant system to the top of the vertical baffle and then down through the intermediate heat exchanger around the annulus of the reactor vessel heating the intermediate sodium. The primary coolant is then drawn through the electromagnetic pumps through the shielding surrounding the core and back up through the core.

The immediate coolant flow is from the immediate heat exchanger in the reactor through the steam generator flow down through the tube region in the steam generator out of the steam generator through the air cooler which has no function during normal operation and back to the intermediate heat exchanger in the reactor.

Super-heated steam is generated inside the steam generator tubes and, of course, to the turbine generator.

This picture shows an elevation shot of what the plant would look like. As you can see, the reactor vessel and the steam generator are located in the reactor building and this includes both sodium-transfer systems, the primary system which, of course, is within the reactor and also the intermediate heat transfer system.

The reactor building is supported on seismic isolators which greatly reduces the earthquake response of the primary and intermediate heat transport systems. You can see in the corner of this picture is a sketch of what one of those isolators looks like.

Status of the 4S design, significant testing has been performed which supports the 4S. We've had decades of experience with sodium bonded metal fuel. Sodium bonded fuel means that there is sodium inside the fuel pins between the cladding and the fuel slug itself.

We've done integrated tests. The critical experiment which is verification of the physics design methodology for reflector control core and also heat transfer tests of the RVACS which is the Reactor Vessel Auxiliary Cooling System. We've also performed system and component tests applicable to the 4S in the area of fuel hydraulic test, test of reflector drive mechanism, sodium tests of the steam generator, sodium tests of the EM pump and tests of seismic isolators. These tests will be discussed in more detail later in the presentation.

As you can see, design work on the 4S has been in progress for many years. The conceptual design of the 4S was initiated 20 years ago. Preliminary design and progress over the last five years is now complete and detailed design work is underway. In addition, many international papers have been published relative to the 4S design work.

This slide shows a rather busy representation of the organizations involved in the 4S.

Toshiba is the overall project lead. It is involved in component and system engineering design and analysis, nuclear components, electrical system and the turbine generator. CRIEPI is involved with materials and component research, structural standards, fuel analysis and safety analysis.

Argonne National Laboratory has provided the technical support function in the development of the fuel and reactor safety.

Westinghouse is providing a licensing focus for the 4S in the U.S.

This is an overview of our proposed licensing approach. We are proposing a simplified schedule for achieving design approval.

In phase one, meetings with the NRC starting today with subsequent meetings to focus on more specific subject areas such as fuel design, safety design and seismic isolator issues will be discussed.

Phase two includes submittal of technical reports for NRC review.

Phase three is the submittal and review of design approval application culminating in the design approval FSER.

In the graphic, we have included the superimposition of a COL on this process. While this does not represent any predetermined or specific application, the city of Galena has expressed tremendous interest in the 4S and has, in fact, expended large amounts of effort in their early work in white paper generation in trying to go forward with the 4S at Galena.

This shows our proposed licensing approach and schedule for phase one. We are proposing this schedule for a series of subsequent meetings with the NRC. Obviously, the first one being today.

For example, we would suggest a second meeting that could include more detailed discussions on such subjects as fuel design and the safety and system designs. Other topics for subsequent meetings could include seismic isolation, regulatory conformance and the PIRT which is Phenomena Identification and Ranking Table which is currently in progress.

We understand that the NRC must work through its own process to evaluate budget and availability of resources. This is a schedule that we are proposing and which we would like the NRC to consider in their evaluation of what they can support.

Phase two would include the submittal of various technical report for NRC review. As you can see, the subjects are similar to phase one, seismic isolation, long-life metallic fuel, analysis methodology, safety analysis, PRA and the PIRT and test program.

We have provided this projection of phase two technical report submittals throughout 2008 for NRC review to support our proposed submittal for the application in 2009. There are discussions. Additional reports will likely be identified that would be necessary to support the NRC's review.

These next few slides show some of the basic physical and operational parameters of the 4S. Here we show the basic plan parameters, inlet and outlet temperatures for the primary and intermediate sodium heat transport systems and the mass flows.

The core has a relatively small diameter, less than 1 meter and is relatively a long slender core.

Fuel enrichment is in the range of 17 to 19 percent with a relatively low burn-up. Also, note that while the temperatures are high which allows high-quality steam to the turbine, the system pressure is quite low at about .3 megapascals.

The primary electromagnetic pumps are again of a single state or design, linear annular induction pump. Intermediate heat exchangers located in the reactor vessel are a shell-and-tube type with straight tubes. The steam generator design includes a double-wall design with helical coil type and also includes leak detection capability.

The containment once again consists of the guard vessel on the top dome. Okay. That's all for that one.

Now, we'd like to get into regulatory conformance. The current status of our code and regulatory conformance activities are the generation of principal design criteria for the 4S. That has been completed.

Regulatory guide applicability and exceptions, that has been completed.

Applicable codes and standards have been identified and applied.

For the principal design criteria, we started with the GDC Appendix A. We used the guidance from 52.47 and staff reviews of previous liquid metal reactor designs including Clinch River and PRISM. We have kept the criteria that was applicable to the 4S and we've replaced or modified criteria that are not applicable to the liquid metal reactors.

The results of this effort are that about 50 percent of the Appendix A criteria are applicable to the 4S. Approximately 45 percent of the criteria identified in the PDC came from the PRISM and the Clinch River staff reviews and approximately 5 percent of those criteria are specific to the 4S design.

Our approach with the regulatory guides was somewhat similar as for the primarily design criteria. We used the guidance in part 52 of SECY 94084 which is non-safety systems in passive plant designs and the SRP which expands on the application of reg guides.

We have characterized the applicability of the reg guides in terms of whether they are applicable, inapplicable or partially applicable in which case the intent of those reg guides has been applied to the 4S.

The results of that, we have identified exceptions to the reg guides and we have justified the following -- these exceptions using or applying the following reg guide divisions. Division one which is power reactors, four environmental and sighting, five materials and plant protection and division eight, occupational health.

Applicable codes and standards. The applicable codes and standards have been identified and applied. These are some of the major components: reactor vessels and internals, steam generator, reactor protection system and so forth.

The major codes that are used some of them are ASME, ANSI and ANS standards and IEEE codes.

New codes will need to be identified and have been identified. New code needs have been identified and are being or will be developed and one example for that is for the seismic isolators.

The QA program for the 4S will meet the requirements of 10 C.F.R. 50 Appendix B and ASME NQA-1.

The main design features of the 4S. First, we'd like to go through the safety features and then there are four key features of the 4S that we'd like to spend a bit more time on and those four key features are passive safety, no on-site refueling for 30 years, low maintenance requirement and high inherent security.

The safety features include low-pressure system with a pool design and guard vessel. There is no LOCA in this design. Negative coolant temperature coefficient promotes safe stable operation. There's a large margin to coolant boiling or cladding failure even when applying beyond design basis assumptions.

Reliable, redundant and diverse scram systems. These systems are the reflectors which move away from the core region by gravity and also the shutdown rod which inserts into the core also by gravity.

The design reduces the excess reactivity in metallic fuel core -- with the metallic fuel core design and limits the potential for reactivity insertion accidents.

Passive and reliable diverse shutdown heat removal systems. These include the Reactor Vessel Auxiliary Cooling System or RVACS and the Intermediate Reactor Auxiliary Cooling System, IRACS. Either system is sufficient to maintain cooling.

Continuing with the safety features, fuel sodium chemical compatibility which allows run beyond cladding breach. Double-walled steam generator tubes with leak detection protects against sodium water reaction. The tubes provide a double barrier as the leak detection system will detect a leak in either the inside tube or the outside tube.

Past experience has been applied to the design features to mitigate the effects of a secondary sodium leak. The addition of a sodium drain system has been added which drains the secondary -- the intermediate system sodium to the sodium dump tank in the event of a sodium leak.

And finally, the capability to survive beyond design basis events without core damage.

As mentioned previously, large margins are shown even in beyond design basis events.

Let me explain a bit what this slide is intending to show. The white boxes on the left are the target for the design, what we hope to achieve. The design feature in the blue box is what actually was implemented in the design and then the yellow box shows what the design feature achieved.

In this case for passive safety, the first target was passive reactor shutdown capability. The design feature which promotes this is a long cylindrical core with a small diameter and what this achieves is a negative coolant temperature reactivity coefficient and this promotes the passive reactor shutdown capability.

Secondly, passive decay heat removal. There are two redundant and diverse heat removal systems, the RVACS and the IRACS. Both of these systems utilize natural circulation of the sodium coolant and also natural circulation of air and either system is sufficient to cool the reactor.

This slide shows some results from analysis of a beyond design basis event demonstrating the passive shutdown. This particular event is the unprotected sudden loss of flow. The assumptions in this event were a loss of power -- loss of off-site power plus a loss of flow coast down plus a loss of both diverse scram systems.

As you can see from the chart on the left, large margins to coolant boiling and much larger margins to fuel melt are demonstrated.

On the right, you can see the makeup of the negative reactivity. The negative reactivity is very effective in shutting down the reactor in an un-scrammed event.

I would like to point out if you look at the lowest curve in the right-hand chart, you see the net reactivity and you follow that up and it appears to be continuing on an upward direction when the chart was cut off. Actually, the behavior of that curve asymptotically reaches zero if you carry that curve out far enough.

This shows you the core region of the 4S. Let me see if I can get this thing to work. This is going to be difficult.

If you look at the vertical section of the core on the left, where is it? There we go. Well, I can't get the pointer. Yes, if you would point to first the dark blue sections on the left.

Those are the reflectors in the reactor which move on the outside of the fuel up and down. As they move up into the core, the reflectors increase the reactivity in the core. The light blue segments above the reflectors are called cavities and these are basically empty sealed cavities filled with argon and what that does is allow maximum neutron leakage from the core. So, as the reflectors move up in the core, more and more neutrons are reflected back into the core following the core burn-up. So, throughout core life, these reflectors are moved up into the core and maintain the power level throughout the core life.

In the planned view in the center, the dark blue circle shows the reflectors. There are actually six segments there. It's hard to see the black lines.

In the center, you can see the 18 hexagonal fuel assemblies with two enrichments. The outer -- if you look, there's kind of like an outer row and an inner row. The outer row is the higher enrichment fuel which is 19 percent and the inner row is 17 percent enrichment. Again, there are a total of -- I believe it's 18 fuel assemblies. Okay.

The passive heat decay removal systems again as I mentioned utilize natural circulation of the sodium in both the primary and intermediate systems and also natural circulation of air.

The RVACS, which is the Reactor Vessel Auxiliary Cooling System, circulates air down around the guard vessel, around the vessel and then back up through exhaust stacks shown as the air outlet in the red air path on the sketch.

The IRACS involves an air cooler which is a part of the intermediate sodium coolant system and in the event that the IRACS is needed to cool the core, dampers are opened at the air cooler that allows natural circulation of coolant air through the air cooler. Natural circulation of the sodium then carries the heat from the intermediate heat exchanger in the reactor vessel through the intermediate system and up to the air cooler where the heat is exhausted.

I want to emphasize that the decay heat removal systems are redundant and diverse and either system is capable of removing heat from the reactor.

No on-site refueling for 30 years. Obviously, the target for this feature was a long-life core. The design features that contribute to this are high volume fraction metallic fuel core and a long cylindrical core with a small diameter. These two features achieve a small burn-up reactivity swing due to larger amounts of fuel.

And the third design feature is the

reflector controlled core which achieves neutron leakage control which is how the reactor power is controlled.

These design features contribute to the -- actually allow the design of the long-life core.

The fuel volume fraction for the 4S is about 50 percent. This compares to about 40 percent for the typical liquid metal reactor. Again, as I mentioned before, the sodium -- the fuel is sodium bonded metallic fuel which means there is sodium inside the cladding between the cladding and the fuel slug which gives the fuel a high thermal conductivity and the fuel integrity is maintained for 30 years.

The burn-up for this fuel design is consistent with our experience range with the EBR-II which is the experimental breeder reactor II.

Also, we have demonstrated that the fuel pins maintain their integrity against creep damage also for 30 years. This is done using the computer code developed by Argonne called, yes, life metal and we'll mention that a little later.

And you can see in the rest of this slide various of the fuel parameters.

The 30-year design life results in a low-fuel burn-up and also a low power density in the core.

Now, this demonstrates the operation of the reflector controlled core.

Rick, if you would point out again the reflector and the cavity on the left-hand side. The reflector again is the dark blue and the light blue are the cavities that sit about the reflectors. The fuel is on the inside. I guess that's kind of a dark red color on the picture. The reflectors move up on the outside of the core and their purpose is to reflect neutrons back into the core.

Now, the sketches on the right, if you look at the far right-hand sketch, this is the position of the core.

Would you indicate the reflector, Rick?

That's the reflector. You can see that its position is mostly below the core region.

At reactor start-up, the reflectors are drawn up into position until criticality is reached with the reactor and this is done using the high-speed drive mechanism shown in the sketch.

Once criticality is reached, the low-speed drive mechanism is used which maintains a steady upward movement of the reflector which compensates for the burn-up of the fuel throughout the fuel life.

When I say a slow movement, the movement of the reflector during normal operation is approximately 1 millimeter per week.

The low-maintenance requirement. The target here was the low maintenance -- was the maintenance-free reactor internals.

The design features were, first, the reflector. The use of the reflector eliminates the need for control rods and the replacement of control rods. The electromagnetic pump which eliminates any moving parts in the reactor -- in the reactor pumps and also the material selection which provides a compatibility between the coolant and the internal structures.

As I mentioned, the electromagnetic pumps have no moving parts. They develop head through use of a magnetic field which draws the liquid metal coolant through the pump annulus. The EM pump is totally emersed in the sodium coolant and is cooled by the sodium coolant. In this way, the heat generated by the EM pump is contained totally in the coolant and within the reactor. This results in a very efficient pump operation.

The fourth key feature is a high inherent security. The target is the ability for below-grade sighting to protect the reactor from external hazards and, of course, the design feature that helps promote this is the small size of the reactor.

Secondly, the target is a sealed reactor and no on-site fuel storage and the design feature that allows this to occur is that no refueling is needed for 30 years.

As you can see in the below-grade sighting, both the reactor and the steam generator including the primary sodium transport system and the intermediate sodium transport system are all located below grade.

Now, just a brief summary of the main design features. Passively achieved fundamental safety functions, this is performed using passive reactor shutdown and passive decay heat removal. It allows simplified operation and maintenance achieved by no on-site refueling. Reflector control which eliminates the need for control rods. The electromagnetic pumps and material capability. It enhances security and proliferation resistance with the possibility for below-grade sighting, a sealed reactor and no on-site fuel storage either for new or spent fuel.

Do you want to take a break now?

MR. CARLSON: Yes, we can take a break

now.

MR. MAURER: Okay.

MR. CARLSON: We're actually ahead of schedule and so, I would propose we break for 20 minutes and be back at 2:30.

(Whereupon, at 2:07 p.m., off the record until 2:31 p.m.)

MR. CARLSON: Okay, we can start back in again. I would suggest that there were some interesting discussions around the model. Questions were answered. I could volunteer one that was answered. The question what material are the reflectors made of. The answer I heard was ferritic steel. Is that correct?

MR. HANDA: The material is modified 9 CR-1 Mo steel is the material.

MR. CARLSON: Do you remember anything else that was discussed during the break that actually provided information that was not in the meeting itself so far?

MR. MAURER: I wasn't here. Rick, do you recall?

MR. WRIGHT: I think it was just general questions about how things operated and what pieces were where. Nothing was really new that I heard.

MR. CARLSON: Okay. Let's resume with the presentation. Thanks.

PARTICIPANT: If you could identify the page numbers if you recall from time to time so folks following by telephone can try and follow as well.

MR. MAURER: Okay. Sure. Thank you.

PARTICIPANT: Thanks.

MR. MAURER: Okay. We are going to continue on. The next section to talk about is metallic fuel experience and design verification testing. We are on slide No. 40.

Extensive experience has been gained in the EBR-II with uranium zirconium fuel. In particular, on this slide I would like to emphasize that over 16,000 uranium zirconium pins irradiated at the EBR-II reactor without failure during operating conditions.

Also, I would like to point out the longer length fuel that was tested in the FFTF, this longer fuel is more representative of the 4S fuel pins. I would also like to point out the large experimental database of the post-irradiation examination data.

We also have significant experience in transient fuel performance. The transient reactor test treat transient test, fuel behavior test apparatus out of pile test, whole-pin furnace test, and run beyond the cladding breach. Also, there are fuel cladding interaction diffusion couples testing on a lab scale.

Evaluation of the 4S fuel is performed using the LIFE-METAL code developed by Argon National Labs. This code has been validated against the extensive experience base that we have and is used to validate the performance of the 4S fuel. The results show that the 4S fuel retains its integrity over the 30-year lifetime at expected operating conditions.

I would like to point out in this figure. It's very light-colored so I don't think the pointer will work. If you look at the picture on the right in this slide you will see a white circle around the outside. The white represents the cladding. By the way, this is a cross section of an irradiated fuel pin. The while circle represents the cladding of the fuel and the shaded portion on the inside in this case represents an irradiated fuel slug.

In a new fuel pin you would see a gap between the fuel slug and the white cladding and that gap would be filled with sodium, what we call sodium bonded fuel. As the fuel is burned up and the fuel expands towards the cladding, the sodium is pushed or extruded up toward the top of the fuel pin which has a relatively large plenum which is typical of liquid metal fuel pins and allows enough room for the sodium to be extruded up into the pin and also room for the fission gases.

There have been many tests which support the 4S systems and components. We have tested the various design features, long cylindrical core with small diameter, reflector-controlled core, high-volume fraction metallic fuel core, the reflector, RVACS, electromagnetic pumps, steam generator with double-wall tubes, and the seismic isolation.

I won't go through all the information on this graph. I want to note that most of these tests have been completed. There are two tests that are currently planned, further testing on the electromagnetic pumps and also on the steam generator double-walled tube design.

We have some additional information on these tests in the following slides. We are now on slide 45. The critical experiment test was done to validate the design methodology for reflector controlled core. The criticalities predicted by analysis methodology which has been validated by testing. The results of this test showed good agreement. It meets the major neutronics characteristics within sufficient accuracies.

Next the heat transfer test of the RVACS. The test apparatus sketch is shown on the left side of your slide and the picture of the apparatus in the middle. The purpose of this test was to determine heat transfer co-efficient between the vessel and the air. Of course, the results of this test actually developed this correlation.

You will note in the sketch on the left of the test apparatus if you look close enough there is a small exhaust fan in the test setup. This is to generate a prototypic air flow during the test so that we could obtain the proper data. In the actual RVACS design in the 4S of course there is no fan. This is a completely passive design.

Also fuel hydraulic tests have been performed to verify the pressure drop for the high-volume fraction metallic fuel including the grid spacer. The results of these testing -- well, the test used a fuel design which is the same as the 4S design fuel. The results were the development of the empirical equation for pressure drop and also a selection of the grid spacer design itself designed for a low-pressure drop.

The reflector drive mechanism has also been tested on a scale model test. The purpose of this test was to evaluate being able to maintain the slow speed of the reflector as it is pulled up into the core or beside the core to maintain the criticality in the core during operation. The results of this test confirmed the target speed within an accuracy of plus or minus one micron per hour.

Sodium test of the steam generator tubes were performed at ETEC which is Energy Technology Engineering Center near Los Angeles. I might point out that the majority of these tests were performed in Japan. The ETC test and also the following electromagnetic pump tests were performed at the ETEC facility.

The sodium test of the steam generator included double-walled tests and the tubes were divided into two groups in the test. One had a wire mesh between the inner and outer tubes. The other had no mesh between the inner and outer tubes.

In each group of tubes there were also three tubes that were included in the leak detection system so it also detected or tested the leak detection system for inner tube failure. Sodium test of the electromagnetic pump was performed in sodium for a period of over 2,500 hours and from the integrity of the stator support and the coil insulation up to a temperature of 600 degrees C taken from the flow stability in the required operating range.

The comparison of the tested pump to the 4S pump, the tested pump diameter was 1.7 meters in diameter and the pump designed for the 4S reactor is 2.5 meters in diameter. The flow has shown on the figure here for the tested pump was 160 cubic meters per minute while the flow in the 4S reactor is only 10 meters per minute. This is also an area where additional testing will be done. We'll get to that in just a minute.

Test of seismic isolator has also been performed to confirm seismic isolator characteristics and confirm design assumptions. The results of these tests were to confirm linear critical strain, static shear modulus, and Youngs modulus of the isolator material.

Toshiba is now in the final process of building a new test facility including a sodium test loop in Japan. The test facility is scheduled to be complete by the end of this year with functional testing to begin early next year.

Two tests have now been identified to be run in this facility. One is a demonstration of a large-scale electromagnetic pump of the same diameter as the 4S pump and the second test would be a test of steam generator tubes also of double-wall design. They would include leak test of the outer tubes or leak test including leakage through the outer tubes.

This shows a summary of the 4S testing. 4S design has taken large advantage of a large experimental base both in Japan and internationally. Metallic fuel has been extensively tested. The 4S fuel design has been verified based on this testing base and performance simulations. Unique 4S features are supported by extensive testing.

In addition, we have more testing in the new sodium loop facility being built by Toshiba. The specific tests that are planned now again are the electromagnetic pump test and steam generator tube test.

The last subject area that we have --

MR. CARLSON: Would this be a good time to solicit questions from the staff?

MR. MAURER: If you want. We've got just one section to go if you want to wait until the end or we can do it now. It's just a few more slides, Don.

Okay. Our last section is the safety analysis section. We are currently on slide 55. Many design basis and beyond design basis events have been postulated for the 4S. A number of these events have been analyzed as well. This demonstrates a listing of the design basis events that have been analyzed. The underlying event, which is the loss of off-site power, was chosen as a typical design basis event that in a couple of slides we will show you the results for.

Also we have looked at the analysis of several beyond design basis events. Again, we have underlined in this case the unprotected loss of flow event. this is a representative event which we have chosen to provide some results for in this presentation. The unprotected loss of flow in this case was a loss of off-site power coincident with the loss of both diverse scram systems.

It's important to note that comprehensive analysis has been performed to identify accident sequences. Our preliminary results show that there are no sequences that have been identified that result in core damage.

This sketch shows the nodalization diagram of the ARGO-3 model which is used in these evaluations. The model includes point kinetics, spacial effects of reactivity feedback, multiple channels in the core, and one-dimensional flow network. We can discuss this methodology in future meetings. We had not planned on going into that at this time.

The next slide, and we are on slide 59, shows representative analysis results for a design basis event and also a beyond design basis event. The graph on the left shows the loss of off-site power but without the additional failure assumptions included in the beyond design basis analysis.

The graph on the right the analysis also includes the loss of both diverse shutdown systems. You can see in these graphs we show the boiling temperature of the sodium coolant and also the temperature at which the fuel would melt. You can see by the curves that both the fuel temperature and the coolant temperature were well below those values.

In all cases there continues to be a significant margin to critical performance parameters, coolant boiling and fuel melt. You can also see again on the right the negative reactivity and the various contributors to that reactivity. As I noted before the net curve on the bottom, which is the sum of the other contributors, it appears to keep going up at the right-hand side of the curve when, in fact, what this curve does is asymptotically approach zero if you carry it out far enough.

That's the last section we had, Don. Just conclusions are left. We'll just finish the rest of it.

Our conclusions then. The preliminary design is complete. Extensive testing needed for the design verification has also been complete and it is noted we have additional testing identified to further support our database of tests.

The 4S is considered to be a mature design and by this we mean that it's based on many years of design analysis and extensive testing that has been performed and that is specifically for or is applicable to the 4S reactor design. Toshiba and its partners are requesting continued dialogue with the NRC leading to 4S design approval.

We would like to leave you with this last slide which again shows the schedule for future meetings that we are suggesting. We are asking to work closely together with the NRC staff to try to come to an agreement on future meetings and document submittals.

This concludes the Toshiba section of the presentation. We thank you very much for your attention and would be happy to answer any questions you have.

MR. CARLSON: Are there any members of the technical staff of NRC who would like to ask any questions? If so, please use the microphone.

MR. RUBIN: Stu Rubin, Office of Research, Senior Technical Advisor for Advanced Reactors. Can you tell me what the role is of probabilistic risk assessment in developing your licensing basis? Is it simply what is called upon in Part 52 as it would be for light water reactors or do you use it to any greater extent in defining AOOs design basis accidents and beyond the design basis accidents?

MR. MAURER: The PRA is certainly used in defining the accident sequences and the design basis events and beyond design basis events. Does someone from Toshiba want to add to that?

MR. SAKASHITA: And also you have the criteria for the core damage for the analysis results and also the PRA results. Also the event selection and the PRA itself is provided for the license application.

MR. RUBIN: Okay. For light water reactor licensing traditionally kind of a deterministic engineering judgment is used for selecting design basis accidents. Are you planning to do that but also to utilize some PRA insights to either modify or to supplement that list?

MR. SAKASHITA: Yes. We also look at the deterministic analysis, the previous experience in the PRISM or the Clinch River. Also we look at the PRA. Both is considered in the event selection determination.

MR. CARLSON: Imtiaz, go ahead.

MR. MADNI: Imtiaz Madni, Office of Research. You mentioned that you have an analysis called ARGO-3 and I presume that is used for both design basis and beyond design basis calculations. How about for hypothetical core accidents even though you mentioned that none of the sequences that you have selected to analyze have led to any core damage?

When you consider design submittals and the NRC decides to review them and if they decide to look at analysis of hypothetical core accidents would you suggest that ARGO-3 be useful in that or you would have to design a different code or use NRC codes or what?

MR. SAKASHITA: At this moment ARGO-3 is only for the transient NRC code so the ARGO itself can't analyze this or evaluate the core damage sequence.

MR. MAURER: So a different code other than ARGO-3 would have to be used.

MR. MADNI: Okay. Thank you.

MR. CARLSON: Any other questions from the staff? Steve.

MR. BAJOREK: Just a couple of questions. This is Steve Bajorek for Office of Research. When you are talking about your test facility for future tests can you give some indication on the scale that you are talking about for the integral test facility? This is in reference to slide 52.

MR. SAKASHITA: I see. No, I can only answer the flow rate. The flow rate is about 10 cubic meters per minute. That is almost the same flow rate of the 4S reactor.

MR. BAJOREK: Okay. Are you saying that this test if more or less full scale?

MR. SAKASHITA: Flow rate almost full scale.

MR. BAJOREK: Okay.

MR. WRIGHT: Steve, I think in the sense of integral testing the test facility they are planning in Yokohama is a component test. They are talking about testing a steam generator tubes and also EM pump, but I don't think they are anticipating using that facility for an integral system test similar to what we did in LWR.

MR. BAJOREK: Okay. Okay. So your plan then would be to do the testing then on a component by component basis. You really would not have any plans to do a full integral facility. Is that correct?

MR. WRIGHT: Right.

MR. BAJOREK: Okay. One other question with regards to the testing on page 46. You are talking about your heat transfer test for the RVACS. I just want to be clear. Did you run that in a forced convection mode or was that run in natural convection to be more passive?

MR. SAKASHITA: Here we are using a fan. That means forced circulation. As for the number we adjusted the fan to simulate the number for the natural circulation.

MR. BAJOREK: Okay, but a natural circulation you would really characterize that more with the Grashoff or Rayleigh number and that is why I'm a little bit interested in how you would apply these forced convection tests to a natural circulation system where depending on the scale, I mean, this looks like it's fairly large, and the resistances throughout the flow circuit could give you something considerably different than the information that you have now.

MR. SAKASHITA: Your point is aggressive number could be significant in the natural circulation.

MR. WRIGHT: Steve, I wasn't involved in these tests but I was involved in the containment testing we did for AP-1000 and AP-600. We did a similar thing. I mean, we couldn't build a full-scale containment facility so in order to get prototypic velocities we put a fan at the top in order to draw the air through. We had similar arguments with the staff over whether this was going to be representative of a forced or free convection.

MR. BAJOREK: I remember the arguments at the ACRS for containment scaling, too. That's why I just wanted some clarification on what you had done there. That's all. Thank you very much.

MR. SAKASHITA: Thank you so much.

MR. RUBIN: Stu Rubin again. In your regulatory conformance analysis you went through and looked at existing regulations, reg guides, standard review plans. I was wondering if you could speak to the specific subject of control room and onsite staffing and what generalized kinds of conclusions you came up with and what you feel would be adequate for this design.

The other question I would ask was a statement made that Westinghouse was supporting licensing in the United States. The question there would be has there been any kind of interactions with regulatory authorities in other countries? That would be very helpful to know for NRC if there is any opportunity to dialogue with other country's regulatory authorities.

MR. SAKASHITA: At present we don't obtain any licensing from the Japanese government so this is the first try to obtain the license -- I mean, the discussion about the regulatory issues. Four operators is sufficient for the operation of 4S reactor.

MR. RUBIN: So you're saying there is a total of four on-site operational staffing with one or two licensed among those four?

MR. SAKASHITA: Yes. You're correct.

MR. DUBE: Done Dube, Office of New Reactors. Upon the double-walled steam generated tubes with leak detection is there a fill gas in between?

MR. SAKASHITA: Helium.

MR. DUBE: Helium. And is the intent that with the leak detection that if there were a leak up through one tube that it would eventually be plugged on a periodic basis immediately or that hasn't been thought of yet?

MR. SAKASHITA: Immediately shut down and plug it.

MR. DUBE: And is there an over design in terms of the number of tubes beyond capabilities of 10 percent margin, 20 percent margin?

MR. SAKASHITA: You are correct.

MR. DUBE: Some percent like that. Is it a possibility of steam generator replacement if it ever were necessary or you are not planning on that?

MR. SAKASHITA: Yes.

MR. DUBE: The answer is yes?

MR. SAKASHITA: Yes, you are right.

MR. DUBE: Thank you.

MR. MIZUNO: Gearry Mizuno, Office of General Counsel for NRC. I have primarily regulatory questions as opposed to technical questions as you can probably imagine. I guess the first thing is that you started out the presentation by saying you were going to seek design certification, but then your slides actually say design approval and there is a significant difference in terms of the regulatory process and its implications. I guess I would like to know which path Westinghouse is actually planning to take.

MR. MAURER: I think the path right now is toward design approval. I think since design approval is one step to design certification that path might change in the near future. Right now we are looking at design approval.

MR. MIZUNO: Just as a matter of clarification, the new Part 52 removes the need for design approval as a prerequisite for design certification so you do not necessarily have to get a design approval to achieve design certification just so that you know that. Okay.

Second question. Has Westinghouse and Toshiba done a review and are prepared to give the NRC a list of the regulations for which nonconformance with the regulations will be needed with respect to, I guess, in terms of an exemption or some other kind of regulatory dispensation with respect to the 4S design and whether you could give it to us in the near future?

MR. SAKASHITA: We have a discussion at the third meeting and then --

MR. WRIGHT: There is regulatory conformance here at the third meeting.

MR. MIZUNO: So I take it at the third meeting there will be a list available or prior to the third meeting?

MR. MAURER: Go ahead.

MR. SAKASHITA: After the third meeting you will see our report and we want to submit the report about regulatory conformance.

MR. MIZUNO: Okay. And that report will not only identify the regulations but indicate what would be the bases -- the alternative bases for evaluation of the 4S design?

MR. MAURER: Yes, that's correct.

MR. MIZUNO: Okay. And from a regulatory standpoint the industry in the past has opposed the use of what we call applicable regulations as a substitute for regulations which are not applicable to a particular design. I know we've had this problem or this issue come up with AP-600 and AP-1000.

Do you also plan with respect to the 4S design that the design approval and any subsequent use of that reactor in a particular combined license or construction permit would also not involve the explicit definition of these alternative design acceptance criteria or technical standards?

MR. MAURER: I'm not sure what your question is. Could you restate it, please?

MR. MIZUNO: Yes. For example, let's just assume that a particular GDC or a particular regulation like ECCS system involving fuel cladding interactions using 17 percent involving zirconium based cladding. I do not believe those would be applicable to the 4S design.

Presumably you would have to get an exemption from that particular set of regulations. In the past the staff has discussed with the industry the need to establish and have explicitly stated, at least in design certifications, alternative licensing or technical standards that would apply in lieu of the regulation which is not technically applicable.

Industry has opposed that by indicating that the design is what it is. Since we are just talking here about a design approval as opposed to a design certification and its ultimate use in a license without more, has Westinghouse thought about that issue and whether there is going to be additional regulatory time necessary in order to obtain closure on the development and use of these alternative licensing criteria or technical criteria?

MR. MAURER: I don't know about the additional time that would be required but as we indicated in the presentation, a good number of these requirements, the GDCs for example, about half of those we determined were applicable to the 4S.

MR. MIZUNO: I'm not talking about the ones that are applicable. I'm talking about --

MR. MAURER: I understand. I understand. Then nearly the other half we obtained from the Clinch River and the PRISM reviews which were the liquid metal reviews. We also identified some additional that were 4S specific. While we determined that some were not applicable, we identified almost an equal number in addition that are specific to liquid metal and to the 4S.

MR. MIZUNO: But you are not allowing for any additional time in your schedule to obtain NRC staff and ultimately Commission approval of in particular the ones that are most likely to result in some schedule impact are the ones which are 4S unique as opposed to ones which are drawn from our past experience, especially with Clinch River which I was involved with. Okay.

MR. MAURER: At this time it's difficult for us to foresee what that means in terms of the overall schedule but hopefully working with the staff we can better define what that is.

MR. MIZUNO: Okay. Assuming that we come to closure, at least with respect to the applicable technical requirements, we still need to deal with the issue about whether you are going to actually implement that alternative technical requirement in what the staff has previously called applicable regulations. I just raise that as an issue for you to think about. Maybe we can talk about that at the third meeting.

I guess my final question is with respect to your identification of certain accidents including the sodium water reactions as beyond design basis. From the standpoint of regulatory treatment what is your position, or what is the Westinghouse position with respect to the regulatory implications of treating an accident as beyond design basis?

MR. MAURER: I'm not sure of the answer to that.

MR. SAKASHITA: For the sodium water reaction our standard point is that we provide the double boundaries and provide also the continuous monitoring system for one boundary failure. That is our standard point for the sodium water reaction. If the question is specific for the sodium water reaction, this is my answer.

MR. MIZUNO: Well, I understand that you consider the design -- I mean, the likelihood of a sodium water reaction to be relatively low I assume, but now you are defining it as being beyond design basis.

I guess I would like to know from your standpoint what implications do you expect in terms of the NRC treatment of that whether in the review or the approval where it's in subsequent use with respect to its nature as a "beyond design basis accident?" Does that have any implications with respect to how the NRC is going to look at that or deal with that?

MR. SAKASHITA: We direct you review the comment.

MR. MIZUNO: Well, yes. I mean, okay.

MR. WRIGHT: You agree that there is at some point a limit to how low a probability is before you go from beyond design basis to the design basis. They are using that kind of criteria by making this a very unlikely event.

MR. MIZUNO: Okay. Let's assume that, for example, with respect to sodium water interactions that the NRC staff agrees with the Westinghouse assertion that it is an extremely low probability and, therefore, beyond design basis. The implications of that NRC decision would be what in your mind?

MR. WRIGHT: Well, it would be -- I'm not a liquid metal guy. I'm talking from --

MR. MIZUNO: I'm talking about from a regulatory standpoint because I presume they would still end up having to have features and treatment of whatever parts of the material and whatever systems that would ultimately result in the NRC -- what would form the basis for the NRC staff's determination that something is extremely low probability.

Usually that means they are going to be considered safety related and we are going to have the full set of Appendix B kinds of things applied to that and certain other what I would call special treatment requirements applied. Apart from that what would be the regulatory implications? Is there any reason to assign these things or to delineate these things as beyond design basis from your standpoint?

MR. MAURER: Well, I think they are identified as beyond design basis because of the number of assumed failures in the accident sequence.

As far as the consequences, I think we have shown that the consequences of these events still fall within the design basis of the plant.

MR. MIZUNO: Okay, but I'm trying to determine not consequences in terms of what a technical person would talk about but I'm talking about what would be its effect of treating something as beyond design basis upon the NRC's regulatory role, I guess, or its treatment of that particular design accident, I guess? Again, you can save this for the third meeting maybe.

MR. MAURER: I would like to ask Dr. El-Sheikh to respond to your question.

MR. EL-SHEIKH: My name is Kamal El-Sheikh from Cameron Consulting International. I have done some work for Toshiba in the regulatory conformance area. I suppose your questions are related to risk informed decision.

MR. MIZUNO: Could be.

MR. EL-SHEIKH: As to how the NRC would decide, it's the weight of the evidence that would be provided that in terms of the equipment, in terms of the lines of defense, in terms of the data, in terms of the level of confidence in the risk that would be done.

As far as the process actually in determining the design basis and the beyond design basis I supposed that there was a grouping based on the probability of a group of events, some class of events. There were determined that the damage, the cumulative damage, fraction or factor of the fuel would be below a certain limit so that it can survive with very high level of confidence for 30 years.

As you have more serious events, then you would confine, for example, the public consequences to 10 CFR 20 and so on. When you get to the beyond design basis it is something that would be in the range, I suppose, of less than 10 to the minus X. Let me put it this way at this point.

For that we try to look at the criterion for the release. You want to have the containment function. You have to have the exposure of the public minimized. These areas would be defended on the basis of evidence that would be shown there. In there would be data about sodium leaks and so on. I wonder if that answers your question.

MR. MIZUNO: That just goes to the defense of that particular accident being of relatively low probability. I mean, to help you out, to give you an example, if something is determined to be of a relatively low probability so that we call it beyond design basis, does that mean, for example, that Westinghouse is going to say that any additional mitigation measures whether they be procedural design would not be necessary as a matter of course?

MR. EL-SHEIKH: Of course there would be some that are like that and some you might say we don't have that much confidence or a decision is made that you want to put another level there. I think these issues have to be dealt on a case-by-case basis.

MR. MIZUNO: Okay. So you are going to be using a risk-informed approach which means you are going to look at defense-in-depth, diversity, those factors apart from -- calling it beyond design basis doesn't necessarily in your mind result in an explicit cutoff of regulatory control over those matters?

MR. EL-SHEIKH: Not necessarily. I think this is the area really where concurrence of NRC with the decision is very important and is critical to move on. I think these are the areas which the decision has to be made and it is in this area that interaction with NRC would be highly desired.

MR. MIZUNO: Okay. I guess that sort of leads on to another question which is we have been primarily focusing on design here. I can see that some of these determinations may result or have an impact upon the applicability or nonapplicability of operational requirements.

In particular I'm thinking about emergency preparedness and also operator licensing requirements. I'm just wondering again maybe just as a preview for the third meeting has Westinghouse thought about the implications of its design on operational requirements and even further on to decommissioning?

To put it one way, if the Westinghouse position is that there are never going to be under any possible circumstance any releases exceeding Part 100 at the site boundary, conceivably that could be a basis for a determination that no emergency preparedness or an extensively truncated set of emergency preparedness requirements would be appropriate for the 4S reactor even after considering a risk informed approach where you are looking at defense-in-depth.

I'm just wondering, again, have you gone through that process of looking just beyond the design to see how this thing might be utilized in a licensing scheme with an operating reactor.

MR. MAURER: I think at this point we will need to defer the answer to that question until a subsequent meeting.

MR. MIZUNO: Okay. Sorry. One more question. I forgot about this. Again, we are focusing on design from a safety standpoint. Has Westinghouse done a severe accident mitigation analysis, a SAMO/SAMDA analysis to assist the NRC in a NEPA evaluation of the design?

We will either do one -- well, ultimately they have to do one in any event as part of a license that actually will use this reactor. Depending upon whether you choose design approval or design certification, what you ask us, we might also have to do it at that stage.

PARTICIPANT: Not off hand.

MR. SAKASHITA: We have conducted a preliminary PRA. We have conducted preliminary PRA and the PRA results shows the probalistic or the probability of the core damage is very small so we didn't analyze it. We can show this kind of issue including the defense-in-depth at the second meeting.

MR. MIZUNO: Okay. I think I understand the theory now that Westinghouse may use at the NEPA for purposes of NEPA compliance.

MR. RUBIN: Stu Rubin again. Just to push a little more on the question of how the PRA is used, if you go back to the PRISM review the staff asked the Commission if it would be okay to take a certain approach in establishing the license bases for PRISM and to succinctly say it was to use PRA to supplement a basic deterministic approach. There was a limited application of PRA in PRISM.

More recently if you come forward into the current day, some of the non-light water reactors want to take even greater use of PRA in establishing the licensing basis. The HGGR community, for example, wants what we have come to call a fully risk informed approach to establishing licensing basis events, AOOs, EBAs, BDBAs.

In 2003 we asked the Commission if it would be okay to establish licensing basis making this fully a greater use of PRA and the Commission said that would be okay but you had to use engineering judgment to bound the uncertainties in the PRA. I guess where we are looking to get better understanding is where between those two approaches are you? Are you still back in the days of the PRISM concept of using PRA to supplement a basic deterministic approach to establishing the licensing basis or you really wanted to make much greater use of the PRA?

The further you go in using the PRA to

establishing the licensing basis events, the special treatment requirements, defense-in-depth and so forth, you will create a policy question on the part of the staff that we need to get decisions from the Commission and so these will take time to package these issues, to establish options, to make recommendations, and hopefully get approvals if you are moving further into making greater use of risk.

I would suggest as part of the Phase III we get a good look and understanding of how much you want to use the PRA. To the extent you get far into greater use, the whole quality and completeness of the PRA becomes a subject unto itself and how you use it is other subjects. Just a comment we like to understand that.

MR. SAKASHITA: Thank you for your suggestion.

MS. HULL: Amy Hull, Office of Research, Materials Engineering. I would like to have more understanding of your perception of the issues associated controlling the compatibility between coolant and structure. It has been my experience that it is not only the material selection. HT-9 and 9 CR-1 Mo there's a fair database on that but there are also issues associated with the purity of the liquid metal so I have questions concerning your sodium volume.

I have questions about how you are going to do the online monitoring if there have been major evolutionary changes since the last technology that was used. I have questions about your purification system. I have experience working with EM pumps and having some problems previously with clogging because of some nonmetallic issues.

With your new sodium test facility I have questions about the specifications there in terms of relative scale -- you know the scale compared to what you expect in the 4S, your flow velocity, your volume, your monitoring technology, your purification testing.

Can you either today or later provide more information about corrosion compatibility issues from a material sciences perspective. Also, my testing previously with liquid metal corrosion you can have variability almost within heat of HT-9 and modified 9 CR-1 Mo.

MR. SAKASHITA: Thank you for your comment.

MR. MADNI: Imtiaz Madni again. You did indicate in your design you have helical tube coils in the steam generator. They are doubled-walled with helium inside and this seems to have been the approach taken by PRISM as well.

In more recent discussions related to GNEP I think the tendency is to go towards single-wall tubes as opposed to double wall primarily because of efficiency. When you have a double-walled tube with helium in between, there's got to be some sacrifice in heat transfer. I was wondering if you looked at that and what would be the sacrifice in terms of efficiency that you would be willing to take or that you expect to take to get the better safety from the design.

MR. SAKASHITA: Actually, there is a tendency to make them extensive. Quantitively the deduction rate is about 20 percent or something like that because of the helium and the wire mesh between the outer tube and the inner tube so heat transfer reduction is not so much.

MR. MADNI: Twenty percent reduction?

MR. SAKASHITA: About. Almost. About 20 percent.

MR. MADNI: So you are losing 20 percent of your efficiency because of double wall.

MR. SAKASHITA: That is kind of a sacrifice but we selected the double tube.

MR. MADNI: Single tube versus double tube you would have a difference of 20 percent?

MR. SAKASHITA: Heat transfer efficiency at least by 20 percent.

MR. MADNI: I need to look at this again.

MR. MATZIE: It's not 20 percent reduction in efficiency of the plant. It's of a parameter.

MR. MADNI: This is a very key parameter in the efficiency of the plant because whatever heat you transfer from the primary to the -- I mean, from the intermediate transport system to the steam generator system, that's what is going to be used to generate electricity or generate power. If there is a 20 percent reduction in the heat transfer co-efficiency, you are going to get a substantial reduction in efficiency

MR. WRIGHT: Yes, but you would make up the 20 percent reduction. You would just have more tubes or increase length of the tube 20 percent to offset the difference in the heat transfer.

MR. MADNI: You're not really going to reduce the efficiency. You are just going to compensate for it by having more tubes or longer tubes.

MR. WRIGHT: That's right.

MR. DUBE: Don Dube, Office of New

Reactors again. On page 57 on beyond design basis events there is a statement that says, "Preliminary results show that no sequence has identified the results of core damage."

Up above under the list of beyond design basis events is listed IRACS failure and loss of partial function of RVACS. I'm presuming if I read between the lines that if one were to have a loss of normal cooling versus the steam generator cooling and loss of all IRACS and loss of complete function of RVACS so that might be a possible core damage sequence.

MR. EL-SHEIKH: Kamal El-Sheikh from Cameron Consulting International. Of course, if you track all the capability through the heat it would lead to core damage regardless of what reactor we're talking about. The likelihood of this to happen is extremely low.

The accident that was recognized was a partial blockage of RVACS and I don't recall how much blockage was there but one could assume you lose all heat remover capability. If that happens, yeah, it will heat up everything and that takes care of that in the structures and all of that. In fact, we try to find sequences but be reasonable in selecting these sequences.

I mean, one would have to look at the mechanisms by which one can read to these I suppose. The list that was selected that was shown here actually there is a lot of work that went into it in terms of variable effects analysis, full trees, event trees, and all the rest. Some of the assumptions actually are not mechanistically driven. It was assumed as reasonable.

MS. TENE: Hi. I'm Kimberly Tene with the Office of Research. I have probably some more basic questions. I noticed that in your overview you said that it's a 30-year recycling or before you need to replace the fuel is 30 years. Does that mean that is your plant life or are you trying to get a plant life that is 60 years or something like that?

MR. SAKASHITA: Plant life time in subyears.

MS. TENE: So once the -- so what you're saying is that you will never refuel these plants?

MR. SAKASHITA: You are correct.

MS. TENE: Okay. So at end of life for the plant, are you then assuming that the fuel stays at the plant or are you then looking at -- have you looked at the long-term storage? Do you know how that transport happens, that sort of stuff?

MR. SAKASHITA: Long-term storage.

MS. TENE: And you have -- it would follow the similar storage mechanisms for the light water reactor plants, the transport?

MR. SAKASHITA: Yes. After one year cooling the spent fuel will be transported to the long-term storage.

MS. TENE: And during that one-year cooling will the plant need to continue to cool? Will you still need the sodium pumps active, the liquid metal pumps, or will it be --

MR. SAKASHITA: Natural convection is sufficient for the spent fuel cooling.

MS. TENE: Okay. Thank you.

MR. RUBIN: A question that relates to the life time and life expectancy of the plant. The question is really the time or the periodicity of conducting any in-service inspections, in-service testing. One of the general generic concerns for plants that have online refueling are very long refueling periods is it stretches out the times and you can actually go in and look at components and degradation and so forth.

One of the ways you build confidence in

your PRA as those degradation mechanisms are understood is to have inspection programs that periodically take a look at testing.

Do you have anything you can say about what the periodicities will be, what kind of inspection or testing activities you will have over the life of the plant to be able to gain confidence that materials are performing as expected and structures are performing as expected and so forth and will that be a subject of the regulatory compliance phase?

MR. SAKASHITA: Thank you for your comment.

MR. EL-SHEIKH: Yes. I just want to add one remark on the situation where you choke the reactor completely. For that reactor if you look at the thermal capacity of the sodium it would take days at the level of great heat to evaporate the sodium.

It takes another two hours or so for the light water reactors but you have at least -- if you have a degradation, serious degradation of RVACS and the heat remover system, you would have quite a bit of time to recover from that. The thermal capacity is really an element of safety there that I just wanted to add in this remark.

MR. SAKASHITA: I just want to say here that we are looking at the ASME.

MR. BAJOREK: Maybe I missed it. I haven't seen anything on the containment. During an accident over a longer period of time you are dumping decay heat into the containment atmosphere. What is your heat sink beyond that? How do you remove that? What type of air temperatures do you achieve in the containment following one of these events?

MR. SAKASHITA: As for the cooling of the containment vessel is the RVACS using the natural draft and the containment vessel of the core vessel and it is cooled by the air as for the temperature.

MR. BAJOREK: So in your containment building regardless whether it's above grade or below grade you are still going to be relying on air cooling of the containment shell to achieve ultimate decay heat removal. Have you looked at the effects of above ground versus below grade?

I think that is one of the design features that you are looking at. Of course, if it's below grade it doesn't seem like it would have that much surface area to be able to remove that decay heat. In the preapplication are we going to see some of the containment parameters, containment performance?

MR. SAKASHITA: Yes, design verification.

MR. CARLSON: I don't believe we heard anything about operational monitoring, in-core detectors, out-of-core detectors. Have those been considered as part of the design?

MR. SAKASHITA: You mean the safety?

MR. CARLSON: Operational monitoring to confirm that the plant is operating within an expected set of conditions.

MR. SAKASHITA: We have a detection system for the reactor temperature and the voltage of the EM pumps.

PARTICIPANT: Excuse me. On the phone.

MR. CARLSON: Yes?

PARTICIPANT: I can't understand the answers that are being given.

MR. CARLSON: We should use the microphones a little better. Thank you for pointing that out.

Can you repeat your answer then?

MR. SAKASHITA: We have a system for the plant heat systems such as reactor outreach temperature, the mutual inspection, and the voltage of the EM pumps, those kind of things.

MR. CARLSON: Thank you. I had another

question on the 30-year life of the fuel. My understanding is that this fuel will operate or will be present in the core under core operating pressures and temperatures in particular, the high temperatures, for 30 years. Are you providing data to show that the fuel, the cladding will operate within a given range of conditions for that 30-year period?

MR. YACOUT: Abdellatif Yacout from Argonne. We don't have an operating data for 30 years. However, from the very low or the benign design of the fuel we are expecting that the fuel will last this long. This is really all about how much stress the cladding is exposed to and what are the heat properties of the fuel and what are the temperatures.

From our experience was the cladding materials EBR-II FTF. We are not expecting that -- we are expecting that the fuel will last for this period of time but there hasn't been specific like 30-year testing in the reactor.

MR. CARLSON: Thank you.

MR. YACOUT: That's what I was referring to that the burn-up is very low. I mean, we test it for 20 percent or 18 percent burn-up. This fuel is going to last only for 4 percent or 5 percent burn-up so burn-up is --

MR. CARLSON: So you feel that you have enveloped burn-up and you think you can make arguments based on testing and models that --

MR. YACOUT: Based on testing and models and existing database.

MR. CARLSON: That the 30-year performance of the materials is adequately understood?

MR. YACOUT: Yes.

MR. CARLSON: Thank you. Building on the earlier question that Kimberly raised, you indicated that at the end of the 30-year lifetime and when you are at least one-year cooling thereafter, the fuel will be removed from the reactor and transported to some interim storage facility?

MR. SAKASHITA: Yes.

MR. CARLSON: So that means you will be opening the vessel at that point and removing the fuel and putting those into a special transport package. Is that correct?

MR. SAKASHITA: Yes. You are right.

MR. CARLSON: In terms of getting the fuel to the site from the fuel fabricator, will you be shipping individual fuel bundles or a complete core? Have you analyzed how that will be done?

MR. SAKASHITA: We are using the cask for all of the 18 sub-assemblies apart from the vessel but all of the sub-assemblies in dry cask.

MR. CARLSON: And those will be loaded into the reactor vessel onsite?

MR. SAKASHITA: Onsite, yes.

MR. CARLSON: Thank you.

MR. SAKASHITA: For the loading we load each sub-assembly one by one.

MR. CARLSON: Thank you. Well, we are scheduled for another break. We could take it now and see if there are anymore questions afterward and then start moving toward wrap-up and the public comment period. Shall we break for 15 minutes and come back at 4:00.

(Whereupon, at 3:43 p.m. off the record until 4:03 p.m.)

MR. CARLSON: Okay. The next part of the agenda is discussion and next steps. As part of the discussion if there were any lingering questions from the staff, then we can resolve those and then move on.

MS. HARDIN: I had a follow-up question to your question, Don. I'm Kim Hardin. I'm in Spent Fuel Storage and Transportation. My question has to do with, of course, transportation. I think the answer to Don's question was that you plan to put the entire core in one transportation package?

MR. SAKASHITA: Put all fuel assemblies.

MS. HARDIN: And is that for both the fresh and irradiated fuel?

MR. SAKASHITA: Yes.

MS. HARDIN: And we don't currently have any certificates for that so are you planning to come in and apply for certification of a package with that design?

MR. SAKASHITA: Yes, I think so.

MS. HARDIN: Okay. I just wanted to give you a little bit of advice. 10 CFR Part 71 has very, very stringent and specific requirements for transportation in both contents and the packaging because you will be on public roadways.

I would pay close attention to 7155 criticality safety requirements and our guidance documents and pay very close attention to some of that before you come into apply for that certificate.

MR. SAKASHITA: Thank you.

MR. HARDIN: My name is Roy Hardin. I'm with New Reactors. I have a quick question. I didn't see anything really talking about instrumentation and control systems, control and operations, etc. I'm wondering when we might be seeing something about that in one of these meetings or is that something further down the road?

MR. SAKASHITA: The control room is located beside of the tibon buildings.

MR. HARDIN: We have a discussion about the instrumentation control, the software and the hardware systems, applicability to 603 standards, information like that. That's what I would be interested in seeing more information about regulatory compliance of your control room software and hardware. Will there be any discussion of that in any of the further meetings?

MR. SAKASHITA: I understand. Thank you.

MR. HARDIN: Thank you.

MR. CARLSON: Okay. If there are no more questions from the staff, we can move into the discussion of next steps. As you've indicated in your presentations, you proposed moving into a phase I which actually started with this meeting consisting of a series of meetings. You could put that up to help us discuss that.

Your proposed phase I activities the next step would be a meeting in, say, early December, late November, and a third meeting sometime in January and a fourth meeting sometime in February. I guess my first question on that is you would consider the staff effort in participating in this phase I. Would that consist of largely attending the meetings or would you expect additional homework, say, outside the meetings?

MR. MAURER: I think at this point, Don, the staff involvement is envisioned in those meetings as more of an attendance basis. We would provide, of course, the meeting materials beforehand but I think material for detailed NRC review would be provided in the form of technical reports.

MR. CARLSON: And that you indicated would be phase II.

MR. MAURER: Right. That will occur throughout 2008.

MR. CARLSON: So we will have to get back to you on whether we can accommodate the series of meetings that you propose.

MR. MAURER: Yes. Thank you.

MR. WRIGHT: Would you like us to request these in writing? Would that be good or --

COURT REPORTER: Speak into the microphone.

MR. WRIGHT: I'm sorry. I was just asking if NRC required us to request these subsequent meetings in writing. Is that usually the way it's done? We did this one, for instance.

MR. JOLICOEUR: I don't think it would be necessary to directly request them in writing that way. You have provided us information that you want to hold these meetings. We will need to get back to you in terms of whether or not we will be able to support the schedule you have laid out. The bigger challenge for us is going to be the schedule you have laid out for phase II.

MR. CARLSON: Okay. So we will take it on ourself to get back to you on the phase I proposal. Did anyone want to discuss anything more about phase I?

MR. WRIGHT: I'll say I know Toshiba and Westinghouse is aware of the constraints on the NRC as far as the ability to provide support and resources for this so we would like to work together to come up with something that is mutually for both of us. I think this is just something we put out as something we would like to do but we understand that you guys and us will work together to come up with the right dates for these kind of things.

MR. JOLICOEUR: I understand that we have a number of preapplications underway at this time and this is one of them and, frankly, we have one, two, three in-house in our branch right now. Only one of them is actually budgeted for this year because it's an ongoing event. The others are ones that were not anticipated so, therefore, weren't in budget planning except for in your case there is a small amount of personnel budgeted just to track what was going on because we had an ongoing preliminary discussion.

We have some internal discussions we have to go through concerning all of this including the DOE, the GNEP, and all the other impacts and figure out we can support and what level we can support all those activities. We'll have further discussions on that.

MR. RUBIN: Don, I would just like to make a suggestion. I know this is the project planning phase we're in here but it's always good to look forward to the actual topics that you would like to staff the review without any promises that we would have resources to meet the schedule but to know what those topics are that you want to review so we can see if your phase I activities seem to match up in a good way which we can give you some specific feedback and make sure we have the right people here in phase I because we know what is coming in phase II.

I don't know if you can kind of keep it in a box on phase II. That has to help inform us on what is going to be right in phase I and who are the right people to sit in on phase I. To the extent you can give us your list of topics just so we know and we can think about that without any promises, that would help us sort through phase I, I think, as well.

MR. CARLSON: Well, the next slide showed phase II. Is that what you are referring to, Stu? Slide 15.

MR. RUBIN: No. I think I have heard some things today that we have to consider that aren't listed in that list. Let me just put that up right now.

MR. CARLSON: Slide 15.

MR. WRIGHT: Okay. Thank you.

MR. CARLSON: Did you want to get back up on the mic? Get back to the mic, please.

MR. RUBIN: I guess my suggestion would be to revisit that and develop maybe what your final proposal would be of topics and then give us that and then we could see what makes sense for phase I given that is where you would like to go. We would have an opportunity to think about what we might want to add knowing what have been the regulatory issues for non-light water reactors in the last five years.

MR. CARLSON: That does bring in a more direct discussion of what we can do under phase II. That clearly is more than bringing a handful of staff to a meeting. You are going to be submitting actual reports and if we go by past practice there would be requests for additional information, perhaps a technical exchange meeting as part of that.

MR. RUBIN: We anticipate that.

MR. CARLSON: At some point some final product by the staff from the review of those documents. Have you envisioned what that final product would be for each of these technical reports?

MR. JOLICOEUR: Recognize that right now we do our preapplication reviews for advanced reactors and non-light water reactors in the Office of Research. The Office of Research doesn't issue SERs per se. In our current practice, for example, with EBMRs we have already gone through -- we have four White Papers in-house we have provided, a preliminary meeting, a set of RAIs. We are waiting for responses.

Once we get those responses we intend to provide them with a final conclusion on that, a document that gives them our final conclusions in that paper. I wouldn't be a safety evaluation report but would be our conclusions on those White Papers.

MR. CARLSON: In some regards the conclusions would be more like status reports, where we stand on the various issues but we would give you some final product.

MR. WRIGHT: Okay. Obviously we are looking for something that is going to allow us to do a submission of a DCD type of thing at a later date. That would be what we would hope to get, guidance from you as to how that would look.

MR. CARLSON: And building on what John said about some of the resource constraints that we will ultimately face and what Stu is saying about revisit some of your topics, perhaps it would be in your interest to prioritize these reports during phase II. Or if you were required to come up with a smaller set of reports initially what would those be. Yes, recognizing that the staff may have difficulty addressing everything you are asking for because of resource constraints and availability of key staff.

Anything else that anybody would like to bring into the next steps discussion?

MR. ROSENFELD: Don, are you taking questions from the public yet?

MR. CARLSON: You beat me to the punch but, yes, that is the next step now. We are now opening it up for public comment. We have people both on the phone and people in person so we will alternate phone and person.

MR. ROSENFELD: This is Rob Rosenfeld on the phone.

MR. CARLSON: Rob Rosenfeld wants to speak first. Is that okay? Okay. Rob, go ahead.

MR. ROSENFELD: Thank you very much. I really appreciate everybody coming together today and sharing all the important information that has been shared. My name is Rob Rosenfeld. I work with the Yukon River Inter-Tribal Watershed Council. The Yukon River Inter-Tribal Watershed Council was formed in 1997.

It's a treaty organization of 13 first nations, governments in Canada, and 53 tribes in Alaska. The 66 tribes and first nations come together once every two years for a full board meeting. I'll talk just -- I'll keep it brief. A couple of questions which at some stage would be very good to have answered. It doesn't have to be right now.

One question that is always lingering that is asked over and over is with regards to storage.

What storage solution will there be after the radioactive material -- after the reactor is finished operating after 30 years. The other question that comes up fairly regularly is the very low temperatures often exceeding minus 50 degrees below for extended periods of time.

We would like to make sure that whatever design is considered is suitable for a climate of this temperature. Also in the design we would like to bring up the fact that a fault line is very nearby where the reactor has been discussed as far as placement goes.

Lastly, as far as other concerns or questions to address at a future date is the fact that permafrost is melting in Alaska at a rapid rate so the integrity of the ground in many locations is not dependable so it is important to have permafrost experts that can determine the rate at which permafrost is melting, look at any sites that are being considered. Those are just a couple of questions.

I just want to make a couple of very quick comments. The full board of directors of the Yukon River Inter-Tribal Watershed Council has met in 2005 and 2007 and they have accomplished a consensus vote which opposes the transport of radioactive materials on the Yukon River Watershed, the storage of radioactive materials on the Yukon River Watershed, and the experimentation with radioactive materials on the Yukon River Watershed.

The resolution actually also states that the tribes are opposed to the 4S sodium cooled reactor. That is for the record. I wanted to have that stated.

Also, there is an organization called the Tanana Chiefs Conference which represents 42 tribal governments in the interior of Alaska, all Athapaskan tribes. They voted, I believe it was also in 2005, in opposition to this proposed reactor. I wanted to make sure Toshiba and Westinghouse was well aware of the local opposition.

We have received directives from the International Executive Committee that serves the board to go to Japan and speak with the Japanese DIA to talk about Toshiba's intent to put a reactor in a place where 66 indigenous governments from two countries are opposed so that trip is in the planning stages.

We have also been asked to put ads and opt-ad pieces in the New York Times and we've been asked to go the United Nations to bring life to the ongoing disregard for the wishes of the tribal governments on the Yukon River Watershed.

We have sent letters to Toshiba expressing the opposition and we've gotten no response. At this stage I would like to invite Toshiba to contact myself and I would be happy to organize dialogues with the tribes on the Yukon River Watershed. Again, I appreciate all your time and I do hope that in your exploration of trying to find energy solutions for tomorrow that you can find a remote area which doesn't sit on an international water body.

The Yukon River is an international water body. There are many treaties. I would draw your attention to the 1909 Transbounday Water Treaty and we have gotten many, many calls from Canadian government officials that are already expressing opposition. I do hope that you can find a remote place that won't be as volatile as far as opposition goes. I hope that you can come up with a solution that works towards coming up with energy solutions for the future. Thank you for your time.

MR. CARLSON: Okay. Does anybody want to provide another comment?

PARTICIPANT: Yes, on the phone.

MR. CARLSON: We're going to alternate between phone and people here in person if that's okay.

Paul.

MR. GUNTER: Hi. My name is Paul Gunter. I'm with Beyond Nuclear. A couple of questions. In going over the model outside could we get some clarification if there are valves on the hot leg to the steam generator in the current design? If so, what kind of valves they are.

MR. CARLSON: I believe that's a question addressed to Toshiba and Westinghouse.

MR. GUNTER: Is it possible to get some sense?

MR. WRIGHT: It's a pool reactor so there is no hot leg per se. You are talking about --

COURT REPORTER: Talk into the mic, please.

MR. WRIGHT: I'm sorry. It's a pool reactor so there is no hot leg per se. You are talking about the intermediate heat transfer loop. Is that right?

MR. GUNTER: Yes.

MR. WRIGHT: They want to know if there are valves on the piping coming out of the reactor vessel go to the steam generator.

MR. SAKASHITA: No.

MR. WRIGHT: I guess the answer there is no.

MR. GUNTER: Okay. Again, we are still educating ourselves about this design but it would seem to raise questions about isolation of the reactor.

MR. SAKASHITA: You mean containment isolation?

MR. GUNTER: Yes.

MR. SAKASHITA: Containment isolation valve is located at the water and steam systems. The containment boundary is expended to the isolation valve in the water and steam loop.

MR. GUNTER: It would be helpful to see that as a diagram. I haven't seen that yet.

MR. SAKASHITA: We can provide such kind of information at the second meeting.

MR. GUNTER: Okay. In earlier communications we saw from Toshiba it's my understanding that you intend to ship the high-level radioactive waste in the vessel whole back to Japan. Is that still your plan? We have documentation of communication that Toshiba -- the plan was to send the encapsulated waste back to Japan for interim storage or reprocessing, whatever your plan is at that point. Is that still your plan?

MR. MAEKAWA: We have not yet decided what to do with the spent fuel.

MR. GUNTER: Right, but it was in earlier communications. Correct?

MR. SAKASHITA: Toshiba never communicated about such kind of thing.

MR. GUNTER: Thank you. I guess this question is for the NRC. Given this novel -- let's call it a novel approach to 30 years of operation without maintenance, does the NRC consider that it has a database that it has confidence in for addressing cracking and corrosion mechanisms with this particular kind of operation and susceptible materials?

I mean, do you have confidence given that we have never really come across a proposal to operate a reactor for 30 years without maintenance of internal components and the issue of material susceptibility. Do you sense that you have a level of confidence to address potentially unknown or known cracking and corrosion mechanisms?

MR. CARLSON: I believe the question you are asking will, in fact, be a focus of our technical discussions moving forward.

MR. GUNTER: Is there a way to get -- I don't know if it's appropriate to ask for a gut reaction but I know from a public position this is -- it would seem to be an issue that is very startling and I'm wondering if I could get a response from the NRC.

MR. JOLICOEUR: Well, we obviously acknowledge this is a novel approach but, as you have already said, is something we are going to work on in the preapplication review, look at the data that is presented and we will be able to give you a value judgment at that point. Certainly based on what we have so far we wouldn't be able to render that judgment.

MR. RUBIN: I'm not a metallurgist by training but the NRC to approve an application of this sort will need to have an understanding of the failure mechanisms, data behind those failure mechanisms, or degradation mechanisms and we'll have to establish some value of potential failure.

We will have to conclude from the data and analysis that there is ample margins to failure or unacceptable degradation and boundaries, vision product boundaries to issue a design acceptance. On top of that for defense-in-depth purposes, and this was part of my questioning, was the whole subject of assurance that we truly did understand the degradation mechanisms and through in-service inspections, in-service testing monitoring programs. We will need both for defense-in-depth.

MR. GUNTER: Could I ask just one last quick question? Who is going to pay for the license approval certification submission at this point? It's not the town of Galena, right?

MR. CARLSON: I'm not going to speak on behalf of the town of Galena. That would be a question for them.

MR. JOLICOEUR: Let me clarify the question. Are you talking about the design certification or the design approval that we are doing now?

MR. GUNTER: Well, obviously going through this process is going to run up a bill and I'm just curious right now who is picking up the tab.

MR. JOLICOEUR: For the design approval

Toshiba is. We bill applicants for preapplication activities.

MR. GUNTER: Thank you.

MR. CARLSON: The applicant for siting related things would pay the bill for that.

MR. JOLICOEUR: To the phones.

MR. CARLSON: Okay. The next question from the telephone.

MR. BURKE: Richard Burke, ISO New England. Don, are you there?

MR. CARLSON: Yes. Yes, please go ahead, Richard.

MR. BURKE: I had a question. The question was there were seven White Papers produced as funded by the State of Alaska. Those White Papers, I guess, the Mayor of Galena back on September 26 in conjunction with this meeting asked that they be introduced concurrently with Toshiba's presentation. I personally have read those White Papers and have found them very informative and answered some of the questions that staff and others have asked today. I was hoping to hear more about that.

I realize that the time now wouldn't permit that but I'm wondering if they would be presented at a future date would it be part of a public meeting and who would present it. Would it be the town of Galena or whom? That's my question because I'm not sure who to direct that at.

The only comment I would have is that I kind of reviewed this and having been involved in nuclear power since the '06s the design is quite innovative and passive.

I've worked in light water reactors and have operated a nuclear plant commercially in this country for 20 years so I have some background in found this design to be, again, quite innovative and I would think that some of the things like reduce napping and worksheet planning zones and other things that are promised could, in fact, be done if people think out of the box because it's not a old light water reactor design.

MR. CARLSON: I think perhaps one way to address your question would be to Toshiba and Westinghouse in terms of whether they see a nexus between those White Papers and what the business of the day here is and that is design approval and the issues they want to pursue during preapplication review for design approval.

MR. WRIGHT: Well, I don't know but I think it's something that we need to further discuss.

I think this work was commissioned by the city of Galena. Is that correct?

MR. BURKE: State of Alaska, I think.

MR. WRIGHT: Yeah, it was. Okay. I think they are really the ones we need to talk to. Unlike you I haven't had a chance to read these White Papers yet so I don't know.

MR. BURKE: I mean, I found them quite informative. I think anybody that does read them will find there is a lot of information there. It would be a shame not to present them somehow on the docket.

MR. JOLICOEUR: Okay.

MR. CARLSON: Okay. The next question from the room. Edwin Lyman.

MR. LYMAN: Thanks. I have a couple of comments based on some of the things we've heard already today. One is one the issue of the HCDAs and whether or not they should be considered in the licensing process. We heard, and I just want to clarify, that Toshiba has not analyzed the core disassembly --

PARTICIPANT: Sir, could you identify yourself?

MR. LYMAN: Sorry. Ed Lyman from the Union of Concerned Scientists. Toshiba is not yet analyzed the core disassembly accident and I would like to point you to a presentation that Mr. Ueda gave in 2005 on core disassembly accident on the 4S. It was a discussion of evaluating them.

At that point it was said that no CDA evaluation had been done, but they also pointed out that a licensing body may require a CDA evaluation for the 4S as a quasi deterministic assumption. I would argue that this licensing should require that.

I would also point to the PRISM preapplication, the staff's view of this in the course of PRISM where it said that regarding accommodation of HCDAs there was not sufficient data at that time to confidently predict the size of an HCDA in a metal-fueled ALNR.

Therefore, the likelihood of an HDCA being accommodated when the PRISM vessel may remain an open issue at least until more analytical or test data become available. We are not talking about PRISM here but we are talking about a similarly hypothetical reactor. My question is have these data gaps been closed to the extent that you can continue to ignore HCDA.

In particular we are going to need to know whether or not the containment will be able to protect against the most energetic plausible HCDA. I think it's a little ridiculous after several decades that there is still argument over whether this accident should be considered or not. We think it should.

On the issue of fuel data I think there is some question whether EBR-II body of data is really representative or sufficient for making judgments about the performance of this fuel in a 30-year environment. I would also like to point out from the same presentation back in 2005 that Toshiba in their presentation, and I'm just reading from the overheads here, they said that the analytical code for fuel degradation 4S must be verified by safety tests up to failure with irradiated fuels.

This sounds to me like an additional test program on irradiated fuel and was part of the package, at least back in 2005. My question is it doesn't seem to be now. My question is what have you learned between 2005 and now that would allow you to no longer require any additional testing to validate your fuel performance codes.

My third point has to do with the plutonium versus uranium fuel in the reactor. My understanding is the reactor was originally designed with plutonium fuel and now it has been redesigned with LEU and we think that's a good thing but my question is what is the motivation for that and is this decision already written in stone? Is this the final fuel form that we would expect to see when you come in for design approval here or is the potential for going back to plutonium still there? Thank you.

MR. CARLSON: I believe those questions were addressed toward the preapplicant if they would like to say anything or take it upon themselves to get back to us on that.

MR. EL-SHEIKH: Yes, the hypothetical core disruption accident has been in the history of liquid metal reactors. Let's remember there has been a lot of development since this issue was faced and there has been a substantial effort to design against it.

One of the things that is true with nuclear reactors is that we have tried to learn from the problems or potential problems and tried to design against them. The negative feedbacks that are in that reactor indicate that it is very remote that you would have any CDA. That would be proven to the satisfaction of the regulators that there will be no core damage.

Nevertheless, because we will deal with a defense-in-depth concept we might say, all right, though we have enough confidence that would not happen, nevertheless we are going to do this just in case something happens. This is the kind of argument that was given in PRISM and that was given in other reactors.

There has been a development in terms of the design, in terms of that this is a unique design. Please look at it as a unique design because one of the objectives of it was not to have a duplicate. Let's not carry on the history of the past. We learn from things in the past and we try to improve the safety. Now we add to that that we want to improve the security of the nuclear reactor. Let's consider that. I guess that is my comment.

MR. CARLSON: Now do we have any questions from the public in the room?

MR. YODER: My name is Marvin Yoder. I am the consultant now to the City of Galena. I was a manager there for 10 years and was there when we started this process. I believe you got a letter from Alaska Electric Power indicating their support for us to continue on with this project. I also have a letter from the City of Nome which I will present to the Commission as well.

It was four years ago when representatives from Toshiba first visited Galena and presented information to the community regarding the 4S nuclear battery. Based on that presentation the city council decided to gather additional information about the Toshiba 4S.

We initiated two studies, the first one funded by DOE to compare the economic feasibility and environmental impact of generating power from the Toshiba 4S versus other generation alternatives. The study concluded that the 4S was preferred alternative based on lower cost of generated power and less impact on the environment.

The second study was a series of White Papers funded by the State of Alaska. The White Papers were prepared by Burns & Roe and Pillsbury Winthrop Shaw Pittman and dealt with various issues relating to the deployment of the 4S reactor in Galena. Galena would like an opportunity to discuss these papers with the NRC staff at an opportune time.

In February of 2005 Galena met with staff from the NRC to explain the difficulty that Galena and other rural or frontier communities have in supplying power to their residents. These communities are remote, off the grid, must purchase all their fuel in the summer, and are dependent on an ever escalating cost of diesel.

It is noteworthy that in 2005 we reported bulk fuel cost of $1.50 a gallon. Since then the cost has increased by more than 100 percent. There are communities near Galena that are now paying a dollar a kilowatt for electricity. With oil prices now hitting new records the hardships are going to be amplified.

In the past four years there have been many questions raised about the feasibility of operating a nuclear plant in a remote location such as Galena. Galena's interest generated opinions both pro and con and a lot of speculation regarding the advisability of placing a nuclear plant in our community.

The city council has continued to favor the consideration of the Toshiba 4S based on information from several sources. Primarily we have been impressed by the commitment of Toshiba and CRIEPI to fund the research and development needed to make this project viable.

Past experience in Idaho with EBR-II, the current JOYO experimental reactor in Japan also give us confidence. The President's GNEP initiative, which contain elements regarding burner reactors and small reactors with emphasis on a reactor core that lasted the life of the reactor.

Finally, United Nations nuclear advisor Mohammed ElBaredei recommended the installation of hundreds of small reactors around the world each of which would serve a single village and using reactors with long core life. Taken together these convinced us that this was something viable that we need to continue to look at.

My view is that the design certification process or the approval that we are talking about at the meeting today will lead to a decision point for Galena's consideration of this technology. Once the requirements and conditions of the nuclear design and the site permitting the licensing of the 4S are known, Galena will have the information needed to evaluate the feasibility and the practicality of using this technology in Galena.

Neither the financial feasibility nor the environmental impacts of such a reactor can be fully known until the NRC has proceeded with its review. Knowing the past record of thoroughness by the NRC we are confident that when a favorable design certification is made the Galena community can reasonably be assured that the safety of our people and our environment will be protected.

Thank you again for this opportunity and we will send a formal letter requesting a chance to present the White Papers. Thank you.

MR. CARLSON: Okay. Now is the time for anybody on the phone to provide a question or comment.

MR. BODONY: This is Tim Bodony from KIYU radio in Galena.

MR. CARLSON: Can you speak a little more directly into the phone? You are very faint here.

MR. BODONY: Okay. How is that?

MR. CARLSON: Bodony from the radio station in Galena.

MR. BODONY: Yes. Is anyone from Toshiba or Westinghouse willing to state how much money has already been spent on R&D for the 4S project and how much more will be needed to bring it to fruition.

MR. CARLSON: Did you understand the question?

MR. SAKASHITA: Toshiba has already spent huge money such as several million or something.

MR. CARLSON: What was the number you gave?

MR. SAKASHITA: Several million. Tens of millions.

MR. CARLSON: Okay. I think we had two questions in the room in a row so I'll do the same favor for those on the phone. Any other questions from the phone? I'll take that as a no and so anymore questions here in the room?

MR. JOHNSON: Hi. My name is Louis Tom Johnson. I'm here from the city of Galena. I felt it was important to say a few words today being the one elected official from the city of Galena here. I am here to stay involved and informed with the process that Toshiba and CRIEPI are embarking on.

I've lived in Galena for nearly 35 years and have raised my family there. My wife's grandfather, Edgar Nollner, was one of the original settlers to Galena and was the last surviving musher that took part in the infamous None Serum Run.

I think I understand the native elders' desires to see the small rural communities continue to be able to live the traditional lifestyles they grew up with. Many of our elders today are trying to make ends meet on a Social Security check. A few get a longevity bonus from the State of Alaska of a couple hundred dollars and an annual dividend check.

I'm here to tell you that with electricity in Galena at 40 cents a kilowatt and up to as high as one dollar a kilowatt in the surrounding communities, heating fuel at $4 a gallon, gas at $5.25 a gallon, it is impossible for many elders to stay in the communities they grew up in.

It may be unfortunate but the current generation of working adults is unable to live for that lifestyle and effectively raise and support a family. To raise their families in the community they grew up in they often have to seek jobs in oil fields or gold mines working two weeks on and two weeks off to earn enough to survive with the high cost of utilities in the rural areas.

Many find it just simpler to leave their communities and move to the urban elders which our elders do not want to see their children forced to do. There are many challenges facing the rural areas of the world with clean water, sanitary sewer conditions, proper trash collection, education of our children, and availability of reasonably priced groceries.

All of these things require a certain amount of hydrocarbon resources to provide, not to mention the heating of our homes and the driving of our vehicles which in our case includes boats, airplanes, snow machines, and 4-wheelers as there are no roads to Galena. It becomes clear that the single biggest item that impacts the lifestyle in the rural areas of the world is the cost of energy to support the required infrastructure.

Our options were evaluated by a DOE study and it was determined that nuclear power from a 4S style of reactor would provide a cheaper long-term source of energy. The city council of Galena has always said that they were interested in a potential source of predictable long-term reasonable energy.

We have also said until the design has been evaluated and reviewed and hopefully eventually approved by the NRC, it was too early to dismiss the idea based on assumptions and innuendos about potential environmental impact concerns. We always anticipated some groups would prematurely oppose any nuclear project before the specifics of the design could be fully evaluated and I guess that is just the nature of the nuclear world.

We are hopeful that with the heightened global concern with global warming and greenhouse gases that some of the groups that may have traditionally been anti-nuclear may realize that nuclear could be the best and maybe the only realistic hope to stem the ever-increasing amount of greenhouse gas emissions in our ever-growing rural society.

In summary, I would say that I believe that everything we have seen concerning the 4S project leads us to believe that it has a place in the world as one piece of the solution to the ever-rising cost of hydrocarbon fuels and the ever-increasing amount of greenhouse gases being released into our environment.

Galena continues to hope that the 4S may be a solution to these problems in our region of the world with NRC's eventual review and approval of this new small, simple, safe, and secure innovative design. Thank you.

MR. CARLSON: Are there anymore comments from the phone line? Okay. We'll go on to the next comment from the room.

Ed.

MR. LYMAN: I just think that the last question I had asked might have gotten lost and I'm really hoping to get an answer with regard to the final decision on the fuel composition. Can we take it for granted that the fuel is going to be LEU and not plutonium based?

MR. SAKASHITA: Yes, that's not plutonium.

MR. LYMAN: Is there another version of the 4S you may be interested in marketing elsewhere that would have plutonium as the fuel?

MR. SAKASHITA: At this moment the 4S is actually faster reactor but there is no commercial base to use plutonium so we have no plan to use plutonium.

MR. LYMAN: All right. Thanks.

MR. CARLSON: I think there were more questions in the room. Any questions or comments from the room? Any questions or comments from the phone line? Okay. Well, I think we finished on schedule, 10 minutes ahead actually. Thank you all for your participation and your interest. The meeting is adjourned.

(Whereupon, at 4:51 p.m. the meeting was adjourned.)