UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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MEETING

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

SUBCOMMITTEE ON REACTOR FUELS

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WEDNESDAY

APRIL 21, 2004

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ROCKVILLE, MARYLAND

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The Subcommittee met in the Commissioners' Conference Room O-1G16 at One White Flint North, 11555 Rockville Pike, at 8:30 a.m., Dana A. Powers, Chair, presiding.

COMMITTEE MEMBERS:

DANA A. POWERS	Chair
MARIO V. BONACA	Member
THOMAS KRESS	Member
GRAHAM M. LEITCH	Member
VICTOR H. RANSOM	Member
SPYROS TRIAFOROS	Consultant
RALPH CARUSO	Designated Federal Official

NRC STAFF PRESENT:

Ralph Landry	NRR, Reactor Systems Branch
Steve LaVie	Probabalistic Safety Assessment Branch
Robert Martin	NRR
Ralph Meyer	RES
Undine Shoop	NRR, Reactor Systems Branch

ALSO PRESENT:

Patrick Blanpain	Framatome
Burt Dunn	Framatome
Jim Eller	Duke Power
Edwin Lyman	BREDL
George Meyer	Framatome
Steve Nesbit	Duke Power

I N D E X

P R O C E E D I N G S

Time: 9:29 a.m.

CHAIRMAN POWERS: The meeting will now come to order. This is a meeting of the Advisory Committee on Reactor Safeguards, Subcommittee on Reactor Fuels. I am Dana Powers, Chairman of the Subcommittee.

Subcommittee members in attendance are: Mario Bonaca, Tom Kress, Victor Ransom, Graham Leitch. Consultant in attendance is Spyros Triaforos.

The purpose of the meeting is to discuss the application by Duke Energy for authorization to load four mixed oxide fuel lead test assemblies into the reactor core of the Catawba Nuclear Station. The subcommittee will hear presentations by and hold discussions with representatives of the NRC staff, Duke Energy, Framatome and other interested parties regarding this matter.

The subcommittee will be gathering information, analyzing relevant issues and facts to formulate proposed positions and actions, as appropriate, for deliberation for the full Committee.

Ralph Caruso is the designated Federal official for this meeting.

The rules for participation in today's

meeting have been announced as part of the notice of this meeting previously published in the Federal Register on April 9, 2004. Portions of this meeting may be closed for the discussion of proprietary information.

A transcript of the meeting is being kept and will be made available as stated in the Federal Register notice. It is requested that speakers first identify themselves and speak with sufficient clarity and volume so they can be readily heard.

We received one request from a member of the public to make an oral statement. We have established an agenda for today's meeting that allows for members of the public to provide their comments early in the day, so the members can consider these questions throughout the day on issues that are of interest to the public.

Members of the public will also be afforded an opportunity to comment at the end of the day following the licensee and staff presentations.

The purpose of this meeting is limited. We are limited to the consideration of the reactor safety aspects of the application by Duke Energy to load four LTAs in the Catawba core.

We do not intend to discuss the MOX fuel fabrication facility that is planned to be built at Savannah River or the safety and security aspects of fuel transport and eventual disposal or the safety aspects of any plans to load batch quantities of MOX fuel into the Catawba reactors. Batch loading of the MOX fuel will be the subject of future licensing applications.

We have a fairly lengthy agenda that I hope we can move through expeditiously. Contrary to rumors that I know abound, all members of the subcommittee can, in fact, read the Vu-Graphs. So you can move expeditiously through it.

All members of the subcommittee are relatively aware of the background of this information. So you can truncate comments on the background and move to the heart of your presentation. I encourage you to emphasize the points you want to make clearly at the beginning, and then move on to your discussion for justification on those.

Do any of the members of the subcommittee care to make opening comments?

MR. CARUSO: I would just like to reiterate one point that I made before the meeting opened. No food or beverage is allowed in this room. It is the Commission's meeting room, and their rules are you are not allowed to eat, drink, smoke, do anything illegal in this room. So please honor that request.

CHAIRMAN POWERS: And, presumably, the craps game in the back corner will have to stop now.

With that, I will call upon Mr. Robert Martin of the Office of Nuclear Regulation to begin.

MR. MARTIN: Good morning. I am Bob Martin. I am the Project Manager in the Office of Nuclear Reactor Regulation for the review of MOX Lead Test Assemblies at Catawba Station.

We have members of the NRC technical staff with us today, which I will introduce them and their areas later in the agenda when we get to the NRC staff's presentation.

As you shall soon hear in more detail from the licensee, the license amendment application that we are discussing today is part of an ongoing program between the United States and the Russian Federation for the disposition of excess weapons grade plutonium.

That program in the United States has two major elements, one having to do with the fuel fabrication facility and one having to do with the irradiation of the material in commercial power reactors. As you mentioned, I believe, the fuel fabrication facility has been before the committee on previous occasions.

CHAIRMAN POWERS: Over and over again.

MR. MARTIN: The goal of the program is to dispose of excess weapons grade material by converting it into a MOX fuel and irradiating it in a commercial power reactor.

The application for amendment of the Catawba operating license was submitted on February 27, 2003, a little bit over a year ago. That application initially also included Duke's McGuire station, and that was subsequently withdrawn from the application.

Numerous supplements have been submitted since that time, which are identified at the end of the safety evaluation. The staff issued its safety evaluation on April 5th of this year.

The issuance of the safety evaluation does not constitute final agency approval of the application. Any NRC approval of the application will also require completion of other matters, including results of the staff's environmental and physical security reviews, etcetera.

CHAIRMAN POWERS: It seems to me, I received a letter that I probably cannot find that suggested that the actual core that is going to hold the LTAs will be different from the one addressed in the SER.

MR. MARTIN; I was just about to get into that.

CHAIRMAN POWERS: Okay.

MR. MARTIN: Okay. The staff's review, as reported in that safety evaluation, was conducted on the basis of what was in the application with its supplements, which are basically two fuel designs in that reactor core that would contain the MOX lead test assemblies.

Recently, the staff has learned of the licensee's plans that would include a third fuel design in that core. The licensee addressed this in its letter of April 16th. The staff and the licensee plan to meet to discuss this issue in further detail at the end of this week, two days from now.

At this time, the staff has not determined the extent to which this new information and the licensee's responses to it impacts the staff's conclusions reached in the SE.

I would say that the range of the impacts could range from -- As we learn more from the licensee about that additional fuel design, we could learn that the analyses that the licensee has already performed bound that fuel design.

The other extent of the range is that we could find that matters arise which require additional analyses or whatever from the licensee. We don't know that yet. That would be the purpose of Friday's meeting with them.

CHAIRMAN POWERS: One of the problems I face is I have to make a recommendation to Mr. Bonaca on whether to schedule anything for his May meeting. He gets irked with me if he finds that I am wasting his time of his committee. Are we going to be in a position to utilize the ACRS's time effectively on this in May?

MR. MARTIN: I don't know that yet. Our position at this time is that we are going to determine the impact of the new information on the conclusions that we presented in the safety evaluation, and we will issue a supplement to the SE as appropriate.

DR. KRESS: The audio is not working? We'll just have to speak up.

I was wondering what were the differences in this new design. so we might even have an opinion as to whether it will have a substantial effect on the various parts of the SE?

MR. MARTIN: We know some of the differences in not a deeply informed way. I think Duke's presentation will cover that in much more detail.

That concludes my comments, my opening comments.

DR. LEITCH: Just one general question: The scope has been narrowed to just Catawba, not McGuire, from what I understand?

MR. MARTIN: That is correct.

DR. LEITCH: And I think it said Catawba 1, and I'm a little confused if it is 1 and 2 or if it is just one unit that we are considering, or is it both units?

MR. MARTIN: Well, the application, in licensing space Duke has left it open such that their application applies to either unit. The LTAs would be put into one or the other.

DR. LEITCH: But not both, as far as this discussion is concerned?

MR. MARTIN: But not both. That's correct. Four lead test assemblies would go into one of the two units.

DR. LEITCH: All right. Thank you.

CHAIRMAN POWERS: Any other questions for Mr. Martin? Seeing none, I will now turn, I think, to Mr. Lyman, Dr. Lyman. Come sit with us, Ed. Welcome, sir.

DR. LYMAN: Good morning. It is always a pleasure to be here. Is this live? I'll speak up. How is this?

Well, as always, it is a pleasure to be here talking to the subcommittee on MOX. I've done it a few times now. I am going to give an overview of some of the issues that the Blue Ridge Environmental Defense League has raised in its intervention against Duke's LTA application.

The Union of Concerned Scientists is assisting BREDL in this effort, and I am just going to discuss some of the issues that we think are required to resolve before this amendment can be granted. Can I have the next slide, please?

The only thing to observe here is that the application really has two parts. One is the safety environmental application for the license amendment, the request for the license amendment to use the MOX LTAs at Catawba 1. The other part is a request for exemption from certain regulatory requirements having to do with the security of the stored MOX fuel. May

I have the next slide, please.

As I said, Union of Concerned Scientists is assisting BREDL, Blue Ridge Environmental Defense League, in challenging the LTA LAR and the security exemption request, and in this context we have entered both security related contentions which are being -- that proceeding is being conducted in a closed session, because there is safeguards information involved, and non-security related contentions which are the main subject of this meeting today. Next slide, please.

On March 5th the Atomic Safety Licensing Board in this case admitted three of BREDL's non-security related contentions after reframing the large number that BREDL had submitted and classifying them into three bins that are grouped by relevant issues.

In addition, there was an order on the security contentions, which was last week. I understand there will be a public version of that, but it is not out yet. So I am not going to say anything about that order.

This is an unusual proceeding, because Duke has asked the NRC to make a decision on this application by August 2004, and the timetable here is driven by a request from the Department of Energy, National Nuclear Security Administration, which actually wants the decision in hand before it ships plutonium from the United States to France for fabrication of the lead test assemblies at the Cadarache plant.

This is because the plant is already on borrowed time. France decided to shut it down last year because it is not seismically qualified, but it is limping along doing some clean-up work and waiting for this last mission.

As a result of the ASLB's attempt to accommodate this accelerated timetable, the adjudicatory proceeding schedule is in a highly compressed fashion. It seems to be proceeding twice as fast as other expedited proceedings before the NRC, and that is seriously compromising the ability of the intervenors to gather the evidence in an adequate fashion.

Now one question that BREDL has raised is what is the rush, because in every other aspect other than this proceeding the U.S.-Russian MOX program is proceeding at a glacial pace. For instance, there is a failure to reach agreement on --

CHAIRMAN POWERS: To be honest with you, we can't help you on that. I think we understand.

Let's move to the MOX.

DR. LYMAN: Okay. Well, I do think it is important to keep the context in mind in this proceeding, namely because the approval in this case is going to set precedents for the future batch loading, and also, like it or not, the U.S. is setting an example for its Russian counterpart, and the NRC is trying to instruct Russian regulators in how to actually conduct its own proceeding. So we do want to set a good example.

The ability of NNSA to ship plutonium to France is not affected by NRC's decision. It is simply a voluntary offer on the part of NRC to try to comply with the request. So we do think we need to take the time to do a thorough review. Next slide, please.

Now I do want to make a few comments on the security exemption request, because I think this is probably the only opportunity in an open session where we can get comments on the record. Nothing I say is going to have any safeguards information in it.

The cover letter for the security exemption as a rationale says that several requirements in 10 CFR 73.45 and .46 are, quote, "impractical and unnecessary to assure the security of any MOX fuel assemblies."

If you look in the regulations, these sections pertain to physical protection systems for protecting formula quantities -- that is, Category 1 quantities -- of strategic special nuclear material from the design basis threats of theft and sabotage, and the details of the request are provided in seven attachments, much of which NRC determined to have safeguards information in it. Next slide, please.

One of the only public statements about the substance of that comes from a Washington Post article from last month where it stated that Duke Power maintains that its security request is reasonable, given the difficulty of diverting plutonium contained the bulky fuel runs. Next slide, please.

There is also some hint of the thinking going on within NRC with regard to this application from a publicly released review plan, which is providing guidance to NRC staff who are reviewing license applications involving storage of MOX fuel at power reactors. This is a memo from Joe Shea to Gwen Tracy, January 29, 2004. Next slide, please.

Some of the key pints of that publicly released review plan is the staff's assessment that

MOX material is not attractive to potential adversaries from a proliferation standpoint, basically because it is big and bulky and dilute.

A large quantity of MOX fuel and elaborate extraction process would be required to accumulate enough material to fabricate and improvise a nuclear device or weapon. Finally, that review points to an exemption grant in 1989 from Category 1 security requirements for fresh fuel stored at the Fort St. Vrain gas cooled reactor. Next slide, please.

Some of the general observations I would make about their plan is that this approach is inconsistent with international standards and judgments associated with the threat or the attractiveness of plutonium contained in MOX fuel assemblies.

For instance, there is no distinction between plutonium in MOX fuel assemblies and separating plutonium with regard to security with regard to security, with regard to the international convention on physical protection which the U.S. is party, to the IAEA's guidance document on physical protection, INFCIRC/225 (Rev. 4), the U.S. plutonium disposition agreement which references INFCIRC/225 (Rev. 4) as a standard, and the National Academy of

Sciences original recommendation, which is that all weapons using plutonium should be treated and protected as if it were still in a nuclear weapon.

Also, the Fort St. Vrain security exemption has little relevance to today's MOX exemption request, putting aside the fact that it dates from 1989, long before September 11th and the security issues that have come forth since then. The SNM content is much lower in the gas cooled reactor elements than in the MOX fuel assemblies we are talking about here, and the process for extracting HEU from the gas cooled reactor fuel element is not nearly as straightforward as that for separating plutonium from MOX assembly.

So I just wanted to make those remarks. I would urge the committee to look into this, and I would be happy to come back and talk to you in a closed session, if you believe it is warranted to look more carefully at this other very important aspect of this application. Next slide, please.

Now to get into the non-security related contentions, the first reframed contention from the Board deals with the fact that BREDL alleges that Duke has failed to adequately account for differences in MOX and LEU fuel behavior with regard to loss of coolant accidents and other design basis accidents.

The issues that BREDL has pointed out involve fuel related phenomena -- that is, MOX fuel related phenomena that may affect compliance with the ECCS criteria in 50.46 -- and also M5 cladding related phenomena that may affect compliance with ECCS criteria for the MOX LTAs. In that context, we are thinking about any synergies between MOX fuel and M5 cladding that have not been adequately accounted for in experiment.

This does lead us to the fundamental problem that BREDL sees, which is that the uncertainties due to gaps in the experimental database from MOX under LOCA conditions is significant and affects the ability of the NRC to conduct an adequate review of this application.

I would point out the French safety organization, IRSN, has proposed out a test at the Phebus reactor, including a design basis LOCA test for MOX fuel which might help to settle some of these questions. Next slide, please.

CHAIRMAN POWERS: I am aware of the IRSN proposing those tests. I am not aware of anyone taking any action on that. I mean, people propose tests all the time.

DR. LYMAN: Right. That is right, and I don't know what -- Of course, I can't speak for NRC, but there was some reluctance at the last presentation I saw given by IRSN to NRC's supporting MOX fuel tests, but we think that that is shortsighted.

CHAIRMAN POWERS: When I look at the Board's discussion of the contentions, there seemed to be some confusion between issues raised by IRSN on high burn-up and issues raised on MOX.

DR. LYMAN: Well, there is no confusion in my mind. The issue is for a given burn-up, how does MOX and LEU compare. There may be a -- We know that, because of the MOX fuel microstructure and the limited experimental evidence there is, at relatively low burn-ups compared to LEU, MOX microstructure mimics that of higher burn-up LEU fuel.

So for a fixed burn-up, the concern is that MOX fuel may appear more like higher burn-up LEU fuel with regard to these effects.

Now whether -- how that translates to risk across the entire core, especially in this high burn-up fuel, is another issue. But our concern is really the substitution of a MOX assembly -- of an LEU assembly for a MOX assembly at the same burn-up.

DR. KRESS: My question was: Would that

concern be related to the future batch loadings or just to the lead test assemblies?

DR. LYMAN: No. All these concerns are amplified with regard to the batch loading, because if you are replacing more LEU fuel with more MOX fuel, to the extent that for a given burn-up MOX fuel is inferior with regard to LOCA performance to LEU, then that concern is going to be amplified.

Now one effect that we have identified that NRC has not adequately taken into account is the fuel relocation phenomena during a LOCA in which, as a result of the clad ballooning in a design basis LOCA and the fragmentation of high burn-up or MOX fuel, fuel fragments will collapse, therefore increasing the linear heat generation rate and potentially the ECCS related parameters like peak cladding temperature.

Fuel relocation is not considered in the Appendix K models at the present time, and we have introduced -- and we have located some correspondence of NRC that is questioning whether this was an appropriate decision, especially given more recent data and concern in Europe about fuel relocation and its impact on these parameters.

According to IRSN, fuel relocation for LEU fuel may increase peak cladding temperature by more than 100 degrees Celsius, which is 180 degrees Fahrenheit. The increase in peak cladding temperature also results in increase in LOCA clad oxidation by five to ten percent.

Our concern is that, by ignoring relocation, to the extent that MOX fuel may be more limiting than LEU in this case, that makes it even less -- or even more non-conservative, and I point to the lower margin for MOX generally because of the typically higher temperatures for a fixed power level and the fact that M5 cladding forms bigger balloons because of the greater ductility in the Zircaloy. So the synergy between M5 and MOX may be a problem that has not been studied in integral tests. Next slide, please.

CHAIRMAN POWERS: If I understand the IRSN issues that they were addressing, those were issues of relocation of fairly high burn-up material?

DR. LYMAN: Well, I think 48,000 megawatt days per ton was where they first saw the effect in LEU, but I'm not -- Lower than what was expected, in that is considered high burn-up today, you know, above 62,000.

Again, one problem BREDL has is it doesn't have access necessarily to a lot of the data generating these international fora, particularly if some of it is proprietary or concealed because of various memoranda of understanding. So we don't -- All we are getting is little hints of the data that is out there, and through the discovery process we are trying to get more.

This Vu-graph, just taking the results of Duke's large break LOCA calculation from the license amendment request, peak cladding temperature was 2018 degrees Fahrenheit for MOX, while it was 1981 degrees Fahrenheit for LEU for an assembly in the same position.

Clearly, an increase of 180 degrees Fahrenheit from relocation effects wouLD bring the PCT to just under the regulatory PCT limit of 2200 degrees Fahrenheit. So this, obviously, is a highly significant effect in either case, but to the extent that the margin for MOX is smaller, it is more of a concern.

CHAIRMAN POWERS: But isn't this margin -- this limit, isn't it a real cliff, that if I am 2199.9, I'm okay? Isn't there margin built into the whole concept there?

DR. LYMAN: Well, the 100 degrees Celsius actually was a nominal figure, and actually one of the documents that we obtained during discovery contains additional IRSN calculations and which looks like 150 degrees Celsius for higher packing fractions is also a possible increase due to this effect.

That's another uncertainty here, is any difference in the packing fraction which affects the peak cladding temperature increase due to relocation between LEU and MOX at a fixed burn-up.

I would guess, to the extent that MOX starts fragmenting at lower burn-ups than LEU and a greater part of the fuel pellet is affected and fragmented, that may mean the mean particle size -- or fragment size is lower for MOX. But I don't have any -- I haven't seen anything.

CHAIRMAN POWERS: What I am not aware of, and maybe you can help me there, is a tendency for MOX to fragment more extensively than LEU. In fact, one would think that MOX would have inherently a low fragmentation tendency, because crack tips get blunted.

DR. LYMAN: Well, I'm just going by the fact that the fission gas releases, you know, are greater and you have at lower burn-ups more -- The phenomena that were observed in the context of the reactivity insertion experiments where there did seem to be greater -- when MOX fuel rod failed at a lower burn-up than LEU, there seemed to be greater fragmentation and lower burn-up. But I don't -- There is definitely a difference in the particle size distribution. I don't know what it is, but it is certainly a difference that should be considered. It's possible that it is favorable.

CHAIRMAN POWERS: We can ask the staff to help us on this fragmentation issue, because seems to me that my limited experience with MOX is it shows less tendency to break up in normal operations at a given burn-up. But I can always be mistaken on those things.

The fission gas release I understand. It's not connected with fragmentation at all. It has to do with the microstructure.

DR. LYMAN: Well, again, these are uncertainties that need to be addressed. Next slide.

Now as far as the M5 cladding issues goes, in addition to any synergy between M5 and MOX, it hasn't been well studied. The issue of the tendency of the zirconium-niobium alloys to embrittlement appears to be, I think, a little less clear, and it may have been a couple of years ago, and the dependence on the initial surface treatment, polishing versus etching, I think, raises questions that need further examination.

I would point to the experiment that was done at Argonne that was disclosed in a letter dated May 5, 2003, which only appeared on ADAMS within the last few weeks, where it was remarked that an Argonne oxidation test on etched M5 samples showed a potential similarity to the oxide characteristics of alloy E-110. This is a letter from framatome to NRC that would seem to be quite concerned about the way the outcome of that test looked.

CHAIRMAN POWERS: Or words associated with those tests: One struggles to understand what a potential similarity means. Not your fault. It's the words the author used.

DR. LYMAN: Well, that is all we've got, but judging from the publicly available information, it seems that there is quite some concern on Framatome's part that this experiment was done. Since etching is not the initial surface treatment that is carried out for M5, it is not clear why there is that concern, but I think until this phenomenon is fully understood, there are going to be questions regarding the stability of M5 with regard to differences in production conditions, and especially the changes in irradiation, corrosion, hydrogen uptake that NRC hopes to address through tests on irradiated high burn-up M5 fuel at Argonne.

CHAIRMAN POWERS: In your researches on M5, have you been able to look at what the experiences are in Europe with the use of M5 ?

DR. LYMAN: Well, again the normal operation, which was what supported the original request for M5 cladding approval in this country, that is well documented. But what isn't so well documented is a full understanding of the relationship between surface condition and its behavior in embrittlement after oxidation.

The Framatome -- After the E-110 issue first arose, Framatome quickly provided the results of ductility testing that showed that M5 wasn't too different from Zircaloy and did not look like E-110, did not experience this nodular oxidation that seems to be the problem. But to the extent that, again, the phenomenon is not fully understood, I think we need to have tests on high burn-up irradiated M5 integral LOCA tests just to confirm that the surface changes during irradiation don't lead to any surprises.

You know, we don't want surprises to occur in the core of a reactor. You know, that is the last place you want a surprise to occur, and the fact that M5 turned out not to share E-110's propensity to embrittlement at a lower oxidation level seems to be a coincidence, as far as I can tell, and it's a lucky coincidence, but to the extent it is not understood, it is still a coincidence.

CHAIRMAN POWERS: I guess I am perplexed a little bit when you say it is coincidence. I mean it's the behavior of the material or you are you suggesting that there is a stochasticity here?

DR. LYMAN: Well, no. I'm just suggesting that the ring compression tests which showed that it revealed this behavior in E-110 which wasn't, I guess, seen through simple strength tests or quench tests, that those tests were not done in M5. M5 was originally qualified in this country based only on impact tests after Quench, and the ring compression tests were done after the issue associated with E-110 came up.

So to the extent that those tests sampled different material characteristics, that wasn't known before. If my recollection of the history of this is incorrect, I hope that staff will correct me. That is the way it looked on the outside. Next slide.

BREDL's contention 2, the reframed

contention by the Board, deals with differences in MOX and LEU with respect to the releases during what are called core disruptive accidents, which encompass any core melt from design basis LOCAs and beyond.

I don't see any need to sit here and tell you guys about the issues associated with uncertainties in experimental databases for core melts and severe accidents with regard to MOX fuel, but I would just remind you of the expert panel report which at least two of you sat on, which remarked that there may be a different degradation behavior of MOX during core melt that may lead to different release characteristics.

The few tests that have been conducted seem to indicate some radionuclides have enhanced release rates for MOX, and the current regulatory source term may underestimate release fractions of groups like tellurium and ruthenium, ruthenium in particular with regard to any air oxidation occurring late in vessel phase, and because both those isotopes are typically greater -- have greater inventories in MOX because of the different fission product spectrum for MOX fission that, to the extent those source terms aren't adequately taken into account, that is another potential non-conservatism with respect to MOX source terms.

Again, the -- Sorry.

CHAIRMAN POWERS: We are looking at LTAs here, so you could have relatively radically different source terms, and it wouldn't really affect things at all, would it?

DR. LYMAN: Well, it will affect things more than -- I mean, there will be a difference. Duke's own -- Duke referred to Department of Energy's original EIS to point out that there was a couple of percent at most difference in release in population dose or various dose related characteristics associated with that release, associated with the difference in source terms.

That didn't take into account -- That only accounted for differences in inventory. This just hasn't been -- The calculation hasn't been done yet with the uncertainties in the source terms that are important for the differences in MOX and LEU taken into account.

What is significant, what is insignificant is a judgment call. There is limited regulatory guidance, and that is not something we have taken a position on this time, but we think it has to be properly accounted for before you can make a determination whether it is significant or insignificant. And that simply hasn't been done yet.

To the extent that the uncertainties are large because the experimental database is sparse, that has to influence the ability of NRC to make a judgment call as to whether there is enough information to conclude that there is insignificant risk increase associated with this amendment.

Once again, I point out that IRSN has proposed doing source terms as to MOX fuel and severe accident conditions, and again we think that such experiments would be well worth the cost to resolve some of these issues. Next slide.

So to conclude, we believe that more research is needed to reduce the uncertainties in M5 cladding and MOX fuel performance to support this application. We note that there may be LOCA tests with a rated M5 clad fuel with LEU fuel.

We note there is fuel relocation tests going on at Halden, and I don't have the details how that's come out, and we note that IRSN has proposed an additional test series.

We think all of these are necessary to begin understanding and reducing the experimental uncertainties associated with MOX.

With that, I will conclude.

CHAIRMAN POWERS: I think you have raised three major technical issues here not associated with security that we need to consider as we go through here. By my count, you've raised the issue of relocation and fragmentation. You've raised the issue of the cladding, and you have raised the issue of source term here. Is that a roughly correct synopsis?

DR. LYMAN: Yes, that is. With regard to the MOX fragmentation issue, again it is something that just needs to be better studied, because there will be a difference. If it is beneficial, it is beneficial, but I think it needs to be taken into account.

CHAIRMAN POWERS: Now my understanding is that MOX fuel, at least in the pellet form, have been taken up to radiation levels as high as 100 gigawatt days per ton. Is that correct?

DR. LYMAN: In light water reactors?

CHAIRMAN POWERS: No, in test reactors.

DR. LYMAN: I'm not aware. Certainly, in the fast spectrum reactors those burn-ups were achieved, but there is no comparison. I think the neutron spectrum differences are significantly great and the production methods for those were considerably different from those today, but I don't how much relevance fast reactor fuel performs as of current -- on light water reactors.

CHAIRMAN POWERS: Any other questions you want to pose to Mr. Lyman?

DR. LEITCH: Yes. I had just a question of understanding. Both contentions really say that Duke has failed to adequately account for differences between the MOX and the LEU fuel. These differences are both known differences and recent information on possible differences, and one relates to primarily LOCA and the other primarily to releases.

Have you discussed the recent information on possible differences? Is that basically what -- I mean, I understand the known differences, but what is the issue about recent information on possible differences?

DR. LYMAN: Yes. I think what the Board was trying to get at -- what we have discussed now is largely in that category. Known differences, I think, would refer more to issues like in Duke's application and on the environmental report it simply referenced a Department of Energy calculation from a few years ago that was based on, let's say, an inventory generation radionuclide inventory that I don't think was correct, for example.

So I think the emphasis really is on differences that are -- Obviously, these are well known, but not well understood. So that's probably a better way to characterize, I think, where the heart of the matter lies.

DR. LEITCH: But my question basically: Is there other recent information on possible differences other than what you have discussed here or you've told us?

DR. LYMAN: No. These were the chief issues, stemming primarily from a presentation that IRSN gave to NRC in October, which crystallized in my mind how much isn't known about MOX fuel performance. Again, a lot of these are issues that have been kicked around a long time, but simply not taken seriously enough to call for an effort to resolve them fully in an experimental setting until now.

DR. KRESS: I am still hung up on whether BREDL was concerned with potential risk impacts of the two percent loading of MOX in Catawba or are they really concerned that this is a precedent for much higher loadings in a batch reactor later on.

DR. LYMAN: Well, in the context of this proceeding which deals specifically with the LTAs, our point is simply you can't begin to make a determination of what is significant or not until you have a good number, and we don't think they have a good number yet.

Then you can debate about whether that is a significant impact. Again, that is a relative term which has something to do with risk-benefit. So it's not directly comparable to other issues where there are risk increases associated with license amendment. I think each one has to be judged on its own.

Putting that aside, though, in the larger picture BREDL is, of course, concerned with batch loading and nailing down these uncertainties so that there is a proper counting of the additional risks associated with that application, which is coming or expected to come next year.

Of course, the sooner there is a commitment to resolving some of these issues, the better and the less potential delay there will be in a challenge to that amendment.

DR. KRESS: Thank you.

CHAIRMAN POWERS: Any other questions? Thank you, Ed. It is always useful to hear from you. You have raised some issues for us, and hopefully, we will get those clarified over the course of the day.

DR. LYMAN: Thank you.

CHAIRMAN POWERS: At this point I will turn to Mr. Nesbit to discuss the MOX fuel lead assembly program.

MR. NESBIT: Thank you. In the interest of time, I am going to dispense with the majority of my presentation and simply make a few points. I would ask you to go to Slide 6. That will be the first one that I will actually talk from.

These points that I make are not at the current time related directly to that slide, but I will point out the MOX fuel lead assembly program, which we are discussing today, is a critical part of the overall program to dispose of surplus weapons plutonium.

It needs to happen if the program is going to go forward. Due to factors, including the availability of a site for fabrication of weapons grade MOX fuel lead assemblies, it needs to happen now.

Duke and Framatome have engaged in a substantial dialogue with the NRC over the past years related to MOX fuel use, culminating in a number of topical reports and the license amendment request itself and responses to requests for additional information which grew out of the NRC review. And as Bob Martin pointed out, that culminated in the issuing of a safety evaluation earlier this month.

On Slide 6, which is up there on the screen, I summarize some of the technical work that has been presented to the NRC. Duke has provided NRC with topical reports related to the thermal-hydraulic performance of the MOX fuel and nuclear analysis.

AREVA or Framatome has provided topical reports related to fuel mechanical performance of MOX fuel. That is COPERNIC, the fuel assembly design that is going to be used, and a MOX fuel design topical report that addresses more specifically MOX fuel related issues. And of course, we have the license amendment request and associated exemption requests.

There is a security plan change and exemption request that has been provided to the NRC, and that is not the subject of this meeting, and I am not going to talk about that any further.

The DOE has requested -- applied to the NRC for an export license. That application is pending, as are requests for certification for transportation packages associated with plutonium oxide powder and MOX fuel lead assemblies.

There's a lot of things in front of the

NRC, but we are going to concentrate today on our lead assembly license amendment request.

You have heard from Ed Lyman about some of his concerns. We have addressed in our application and related materials the difference between mixed oxide fuel and low enriched uranium fuel to the extent that they could possibly affect the safety case for using the fuel at Catawba, and we are going to talk about that in subsequent presentations.

I would characterize the BREDL issues as threshing around between hearsay of presentations and letters here and letters there to try to come up with some issue that could be blown out of proportion, but in the context of this application for four MOX fuel lead assemblies, we have presented a robust safety case.

CHAIRMAN POWERS: If I look at look at the heart of the issues that Mr. Lyman has just addressed for us, it seems to me that one of his contentions -- the central contention he makes is that there is just not a lot of experimental data on the MOX fuel. I mean, is that a fair characterization?

MR. NESBIT: That is probably a fair characterization of his contention. I wouldn't agree with it.

CHAIRMAN POWERS: Okay. So you will help us to understand what we know here?

MR. NESBIT: And I'm not going to address that right now, but in subsequent presentations, we will.

CHAIRMAN POWERS: Sure.

MR. NESBIT: The presentations that follow will be by Patrick Blanpain of Framatome in France who will talk about the MOX fuel experience base, and also current and future plans for MOX fuel use. I think some of the members of this subcommittee have heard from Mr. Blanpain before.

George Meyer from Framatome in the U.S. will talk about the fuel assembly design. Jim Eller from Duke Power will talk about the nuclear design aspects and our plans for core loading of mixed oxide fuel in lead assemblies, and I will wrap up and talk about the safety and environmental analyses and evaluations that we have performed.

There is way too much information in what we have submitted to cover here, and we are not going to try. I will note --

CHAIRMAN POWERS: You have found a good occupation for my evenings and weekends. My wife thanks you. She hasn't seen her kitchen table now in several weeks.

MR. NESBIT: You're certainly welcome.

One thing that we are not going to talk about further except in a limited extent is the question of the Westinghouse NGF lead test assemblies that, based on current plans, would be co-resident with the MOX fuel as it would be loaded in the spring of 2005 at Catawba.

In Mr. Eller's presentation, he will show the locations of -- planned locations of the fuel. The details of the NGF fuel assembly are proprietary to Westinghouse, and I can't talk about them in this meeting, in this context. There will be a meeting on Friday that Bob alluded to at which some of that information can be shared, although that is also not a proprietary meeting.

I will characterize the Westinghouse NGF fuel assemblies as fundamentally similar to the current co-resident RFA fuel assemblies. It is a little different, but to the extent that there are differences, those differences, if they had any effect on mixed oxide fuel lead assemblies, which they would, the differences would actually be beneficial to the MOX lead assemblies.

We have some people in addition to the

presenters who are here. If you have questions that get into specific areas of technical detail, we will do our best to respond to them.

I'd like to make another point, and then I will turn it over to Patrick. What you have heard from BREDL is basically an argument that, in order to have a MOX fuel lead assembly program, you need perfect certainty about everything that is going to happen.

Well, I think that is inconsistent with the history of fuel development in the United States in nuclear power, and I think it would be fundamentally a chilling approach to take to say that, before we can run a lead assembly program in a reactor, we have to know everything.

That is not the NRC's regulatory charge. The standard NRC uses is reasonable assurance, and we feel we have met that standard.

So now I would like to turn it over to Patrick Blanpain, and he will discuss his experience and Framatome's experience with MOX fuel use in Europe.

MR. BLANPAIN: Thank you. The objective of my presentation is to show you the fabrication and the additional experiments of Framatome in Europe, and the main part of my presentation will be an overview of the reactor performance of the MOX fuel.

I start with some facts. Since the first commercial reloads in 1972 in Germany and in 1987 in France -- of course, it was in one reactor -- plutonium recycling as performed in the form of MOX fuel has reached an industrial maturity.

The production capacity today in Europe and used by Framatome ANP is about 150 Thm/year using the MIMAS process in the French Melox and in the Belgian Belgonucleaire plants.

More than 2400 fuel assemblies have been delivered by Framatome ANP/France to 20 French, two Belgian and four German pressurized water reactors, and more than 1300 fuel assemblies have been delivered by Framatome Germany to 11 German and three Swiss PWRs and BWRs. Next slide.

I will now recap about the MOX fuel fabrication. So we start with UO_2 and PuO_2 powder. That primary blend is mixed and micronized with some recycled scraps, and then that primary blend is sieved. We had lubricant of pore former, and feed UO_2 powder into that original to reach the final blend, and then that final blend is pressed, sintered, ground, inspected and loaded in the fuel rods like in the UO_2 process. Next slide.

Now the primary blend: We are starting with the UO_2 powder, with the PuO_2 lots, and they are mixed and then ground, micronized, which is the primary blend, and the principal compound in that primary blend is between 20 and 30 per the UO_2. I am showing that primary mixing. The recycled scraps from the fabrication are this. Next slide.

Then that primary blend after sieving is mixed with fresh UO_2 powder to reach the final plutonium content. Next slide.

Then at that final blend, the different lots of secondary blends with lubricant, then mixed, pressed, sintered and controlled as UO_2 fuel. Next slide.

Okay. That is the results. It's the microstructure of that MOX fuel using the MIMAS process. On the top of the slide it is electronic image obtained by a electronic probe microanalysis, but the top image is showing -- Showing white, the plutonium rich particles. That's in white, and in black is the UO_2 fuel matrix.

After image analysis, now going to the back of the image, we can see in red and yellow the plutonium, in blue the uranium that is on the right scale of the plutonium content from UO_2, 50 percent. So we can see that most of the concentration is around 20 percent in the master blend.

Then if we increase the contrast, we can see in red the plutonium rich particles with an enrichment -- plutonium enrichment higher than 20-25 percent of plutonium, a blue phase which is almost fuel UO_2, and a green phase called the coating phase between the UO_2 grains. We have a plutonium content between 2 and 5 or 10 percent. Next slide, please.

Then we can construct a calculated analysis if that plutonium distribution. So that graph on the y axis -- or x axis, sorry, there is a cumulative plutonium -- the plutonium content. And on the y axis the size of the plutonium rich particles.

We can see, for example, on the left that in the MIMAS MOX fuel we have only 25 percent of the total plutonium in the plutonium rich particles. For example, there is a 10 percent of the total plutonium is included in the large particle, larger than 100 microns. Also that means that 75 percent of the total plutonium is in the coating phases or in the UO_2 phase. Next slide, please.

That is another representation of the microanalysis. It is an analysis crossing a pellet diameter, and I have presented here a series of large plutonium agglomerates, but it is important to note that the maximum plutonium content in those agglomerates is around, in that case, 27 percent. It is the red line. It is the plutonium content of the primary blend.

It is important to note that there is no very high plutonium content in those particles, higher than the primary blend content. Next, please.

Now the different fuel designs used in Europe for the MOX fuel. So the MOX fuel is used in light water reactors up to power of 1300 electric, and it is different fuel assembly we can design, under 14 by 14, for example, to the 18 by 18.

In Europe we are using different design and type of fuel management. The plutonium content used is 75 percent with an average assembly. It just goes from U^{235} enrichment up to 4.3 percent, and really one-third and one-fourth core loadings can use it, usually NEL, but in some cases, for example -- Belgium, for example -- they are using up to 18 month cycles.

The core fraction is usually 30 percent, but 50 percent is licensed in Germany for boiling water reactors, and 38 percent are used.

For the UO_2 matrix, depleted uranium or -- uranium are used. It is normally depleted today. The current discharged boiler fuel assemblies is between 45 and 50 GWd/tHM and up to 60 for individual fuel assemblies.

It is important to know that the MOX can operate in load follow mode since more than 10 years in France, and also the failure rate -- We have the same failure rate as uranium fuel, an that no rod ever failed for MOX specific reasons. That means that, due to the correct fabrication of the MOX fabrications.

So typically, the failure rate is less than one rod there, 100 in 1,000 rods. Next, please.

CHAIRMAN POWERS: Just doing a quick calculation, that suggests you had a couple of rods fail? You've had two rods fail? Is that roughly correct?

MR. NESBIT: There is information in the MOX fuel design report about the MOX fuel failures in one of the appendices. It has been more than two rods.

DR. BONACA: On the recycle rate, license, what are the limits, the amount of MOX fuel you introduce in the core?

The question I had was on the recycle

rate. You said that 50 percent is licensed in Germany?

MR. BLANPAIN: Yes.

DR. BONACA: And 38 percent is used. What is the basis for the 50 percent?

MR. BLANPAIN: I do not exactly -- What is the basis of -- Maybe it is the basis of the parameter of the void fraction in the core that limits the use of MOX fuel or the fraction of MOX in the core.

DR. BONACA: Okay. So that is the basis there.

MR. BLANPAIN: Yes. But those are -- In France, for example, the 30 percent is also -- to go back to -- we have a UO_2 fuel. So that is to put two more MOX in one core to get a high visibility, easy.

So the next slide shows the irradiation experience in Europe by Framatome France. It is mainly -- It is in the pressurized water reactors at mainly 17 by 17, three reactors in France.

You can see mainly two peaks. The first one to the left is the French experience with the discharge burnup for up around 37. It is assembly burnup, and to the right is the discharge burnup in Belgium, in Germany, which are higher. The discharge burnup is around 55 to -- 45 to 50. Next one.

I have got the same representation of the fuel fabricated by Framatome Germany, ex-Siemens, and showing that the discharge burnup is quite higher in Germany and in Switzerland, and that also higher -- very high burnup fuel have been discharged from -- higher than 55. Next one, please.

So what about fuel design and then performance? So usually the mechanical design of the fuel assembly structure is identical for MOX and for UO_2 fuels. It is the materials used as well as the skeleton of the assembly.

As for UO_2, we need a reliable prediction of the thermo-mechanical behavior of the MOX fuel rods. That should be obtained through an adequate description of the MOX-specific properties as for UO_2, and that means the design models and the codes to be continuously verified by comparison with measurements to obtain finally the same level of accuracy as for uranium fuel, and show you that now. Next slide.

This is an example of the MOX properties. It's easy to see. That shows the thermal conductivity of the MOX fuel compared to UO_2, temperature. The graph shows a very small difference, a difference of the picture of MOX fuel with about 5 percent of PuO_2. It is around a 5 percent difference on thermal conductivity.

The next one shows the specific heat of the MOX fuel compared to UO_2. On the x axis here is the specific heat, on the y axis the temperature. The red star is specific heat of UO_2 with -- and the different points result from different experience coming from the literature, and but addressing MOX fuel with 20 percent of PuO_2.

You can see that sometime -- most of the time you have conservative, and for the MOX fuel with low plutonium content that we use in light water reactors, it is recommended to use the PuO_2 correlation.

CHAIRMAN POWERS: So you say that that is -- using the UO_2 correlation is conservative.

MR. BLANPAIN: This?

CHAIRMAN POWERS: Is conservative?

MR. BLANPAIN: It is mostly conservative, yes.

CHAIRMAN POWERS: But, I mean, the specifics of the heat seems higher, and it seems to me that the conservative position would be to use a lower specific heat.

MR. BLANPAIN: No. There is no -- We have data from MOX fuel is high plutonium content at 220, and they are using 5 percent plutonium content. Also, it is not seen here, but for very low plutonium content we have seen a decrease -- to some extent, a decrease in specific heat.

So it is the reason why we are using the UO_2 correlation, which is conservative compared with the UO_2 experiments.

CHAIRMAN POWERS: Well, I guess I am a little puzzled, because I mean, for a given heat input, the temperature you arrive is lower.

MR. NESBIT: I guess, if I can interject, it is sometimes dangerous to make a statement that so and so is conservative or not conservative, because it depends on the application that you are using it for.

CHAIRMAN POWERS: Absolutely.

MR. NESBIT: It may be conservative for one and not for another. I think the fundamental point that Patrick is trying to make is that, for the kind of plutonium concentrations we are looking at for our MOX fuel, which is less than five percent plutonium in the pins, the specific heat is virtually indistinguishable between MOX and LEU, and using the LEU value is appropriate.

CHAIRMAN POWERS: Let me also comment that, if I compare your specific heat curve here for

UO_2 against those in recent review articles on UO_2, I fail to see a discontinuity up above about 2600 Kelvin. It has been variously attributed to second order phase transitions and the like.

I wonder why that is not reflected there. You don't see it when you make the measurements?

MR. NESBIT: I don't think we've got an answer for you right here.

CHAIRMAN POWERS: Well, I think for some time, since at least 1980, there's been recognition that something funny happens in UO_2 in the vicinity of about 2600. Your initial reaction to it is we've had the onset of a disorder in the lattice, and it doesn't take long to figure out the discontinuity is way too big to be that, and it just didn't show up in these curves.

There has been a recent review article -- I probably cannot pull out the citation in my head, but of course, the most famous stuff is the stuff that was done at Argonne back in the Eighties in their review of uranium dioxide from a physical properties. But it has been reiterated.

I mean, people have made these measurements a lot, and they see this discontinuity, and you have a smooth transition looking like probably the onset of charge carrier effects here. But I don't know. Please go ahead.

DR. LEITCH: On that slide I don't understand what the X=2-O/M is. What is that? What is O and M in that?

MR. BLANPAIN: That is a deviation from stoichiometry. It is a deviation from the stoichiometry.

DR. LEITCH: Oh, okay.

CHAIRMAN POWERS: Yes, the plutonium tends to -- on these systems, and so you get hyperstoichiometric fuel pretty easily. Are you planning on touching on the oxygen potential changes as a function of temperature and plutonium content? It seems to me that that is the issue when you talk about internal oxidation of the clad, is what the change in the oxygen potential is for the hyperstoichiometric material here.

MR. BLANPAIN: The MOX fuel?

CHAIRMAN POWERS: Yes.

MR. BLANPAIN: I'm not sure -- the tendency is to become stoichiometric, because I think slightly hyperstoichiometric and going to stoichiometric.

CHAIRMAN POWERS: Yes. That is because of the fission release of oxygen. What I'm interested in is how the oxygen potential of the fuel changes over the course of the irradiation as a function of Pu content.

MR. BLANPAIN: We can make that later, because of all the cladding oxidation.

Okay, the next one, please. It's a graphic showing the thermal expansion of UO_2 compared to MOX fuel. It says there is no effect of the addition of PuO_2 thermal expansion.

Now some examples of the two graphs showing --

CHAIRMAN POWERS: On this slide you have -- In your legend here you have PuO2 with a bracket 1, UO_2 with a bracket 1, PuO_2 with a bracket 2. I'm just guessing that those are references?

MR. BLANPAIN: Yes, those are references.

CHAIRMAN POWERS: Are those references provided? I would appreciate it if you would provide those.

MR. BLANPAIN: Okay. So as an example that we are looking at the physical properties of the MOX fuel on the irradiated materials also during irradiation. That is an example of the measurement of the fuel central line temperature, first on the un-irradiated fuel and measured during irradiation with the central thermocouples on small -- in experimental reactors.

There is fuel central temperature on the x axis, and the heat generation on the y axis. That shows small difference between the temperature of the MOX and UO_2. Typically, the temperature of -- the temperature of the MOX fuel is 15 degrees -- 50 degrees higher than UO_2, 200 per centimeter. The next, please.

CHAIRMAN POWERS: Why?

MR. BLANPAIN: Thermal conductivity.

CHAIRMAN POWERS: The thermal conductivity is minusculely different. The density can't be wildly different. So it all has to be the -- It has to be in the specific heat. I mean, it has to be a difference in thermal diffusivity. Right?

MR. BLANPAIN: The thermal diffusivity.

CHAIRMAN POWERS: And it can't be the thermal conductivity. It can't be the density. So it must be the specific heat. I mean, that's the only thing left to you. Right? I mean, the thermal conductivity you showed us is --

MR. BLANPAIN: Diffusivity.

CHAIRMAN POWERS: Yes, it has to be a

difference in thermal diffusivity. So it has to be in the specific heat. That is the only thing open to us, unless the thermal conductivity changes wildly in the course of the irradiation, but these are modest irradiations.

MR. BLANPAIN: The conductivity curve, because this is what is used in the computer codes.

A feature to be predicted is the fission gas release, because it depends mainly on temperature. It is taken against measurements made by the COPERNIC code and the calculation on the x axis and the measurement on the y axis, and I represented here the database of -- and comparing the MOX fuel -- The MOX fuel is in red. No, in black. Okay, in black, compared to the UO_2 fuel in blue and red.

That is to show that there is no bias brought by the MOX fuel. The clarity of the prediction is very similar to the other fuels. That's the message.

DR. KRESS: Can we go back to the previous slide. I'm not sure I understood the discussion. These are temperature center line differences at steady state. Right? Stead state only involves thermal conductivity. I still don't understand this difference. These are steady state results.

MR. BLANPAIN: The steady state?

DR. KRESS: Yes. You are not in a transient.

MR. BLANPAIN: Yes, but --

DR. KRESS: The thermal conductivity is the only -- and the heat generated is the only difference in distribution, is the only things that enter into the steady state. So I don't understand the answers you got.

MR. BLANPAIN: Yes, but you need to know the -- to verify your calculation in your normal operation. That is the reason of that experiment. That is what normal operation causes in Phase 1 and Phase 2 situations, but the answer to check the -- accident condition is presented here.

DR. KRESS: Certainly. That leads me to my question. This must be due to the different power generation distribution. It can't be due to the diffusivity. It must be a power difference. Your thermal conductivity is almost the same. If I were to calculate this, the only thing that enters into that is power generation rate and the thermal conductivity.

MR. NESBIT: I think we would agree with you, that the two factors are the pellet radial power profile and the thermal conductivity.

DR. KRESS: I would buy that answer better. I didn't understand the previous one.

MR. BLANPAIN: Yes, but of course, those type of -- kind of experiments, because we have also done it for high burnup fuel and so on, and of course, the difference in temperature between MOX and UO_2 is not only due to the thermal conductivity, called the power radial profile difference. We know that it is taken into account in the computer codes.

DR. RANSOM: Were these cladded?

MR. BLANPAIN: Yes.

DR. RANSOM: So what about the gap conductance? It would enter into that, too.

MR. BLANPAIN: Well, we have performed experiments such as this one. We are using exactly the same pellet geometry and the same enrichments, the same cladding material, the same gaps and so on. We know that is not evident to compare that. We need such experiments to verify the predictions.

DR. RANSOM: Do you have any idea of the uncertainty in this measurement due to dimensional tolerances --

MR. BLANPAIN: Usually, the experiments -- People perform hundreds of experiments with central thermocouples to benchmark the temperature prediction of the fuel, but it is around -- what is used usually is between 5 and 10 percent on the final uncertainty, knowing that the highest uncertainty is brought by the uncertainty of the power during the experiment and not due to the instrumentation.

DR. RANSOM: Well, the 5 to 10 percent would almost account for the difference, if this is just one case.

MR. BLANPAIN: Yes, it is one case, but we have many cases. It is described in the COPERNIC manual, how we manage those datapoints. It is essentially --

DR. RANSOM: This difference may be insignificant. Is that correct?

MR. BLANPAIN: It is exactly the same for the other fuels, for UO_2, for example. There is no new bias brought by the MOX fuel.

So continuing with the feedback experience -- many come from the surveillance program where we build rods and examined those rods in hot cells, and also coming from analytical experiments.

So about more than 100 commercial fuel rods from German and French reactors were examined in hot cells up to high burnups, and also we participated in a lot of national and international programs for analytical irradiations. These are international programs like Halden, for example.

We have performed power ramping with the MOX fuel and with instrumentation to assess the PCI properties of the MOX fuel, the fission gas release, the in-pile densification and so on. Those experiments were largely published in the open literature.

And what was the main reasons of those programs? The next, please. We have very similar behavior of the MOX and UO_2 fuels concerning the rod growth. There is no effect on the MOX pellet, in connection of the MOX pellet and the cladding, model cladding diametral deformation, the same for MOX and UO_2 fuel; concerning the cladding waterside corrosion -- I shall show you an example; model pellet solid swelling -- I also show you an example; the zirconium internal layer. So the oxidation of the internal layer of the cladding is the same as on UO_2, and we have seen at high burnup that the mechanical interaction between the -- or the chemical interaction between the pellet and the cladding is similar.

As far as the fission product and activity, release of the failed rod are similar for MOX and the UO_2 fuel. So it is difficult to determine leakage from UO_2 fuel. That could be, but it is not so easy.

WE have seen a somewhat higher fission gas release in MOX fuel than for UO_2 fuel, mainly at high burnup. That could lead to higher fuel rod internal pressure. Also, the MOX fuel shows a better pellet-cladding mechanical interaction behavior due to the higher creep rate of the MOX pellets. Addressing mainly France, it shows the MOX is not limiting with respect to plant maneuverability.

Next slide, please. This slide shows the fuel rod growth is prediction against the measurement of the UO_2, again the MOX fuel and the gadolinium fuel. The MOX fuel is in blue in this representation. So there is no bias due to the MOX presence in the fuel rods.

DR. KRESS: Is the gadolinium mixed in with the UO_2 --

MR. BLANPAIN: The gadolinium -- but we don't use gadolinium fuel with MOX fuel.

CHAIRMAN POWERS: I notice on your slide that you distinguish UO_2 with what is called ZY4, which I'll guess is Zircaloy 4 cladding, and UO_2 with M5 cladding. But you only have MOX with Zircaloy 4.

MR. BLANPAIN: Here in this slide, we only have MOX with Zircaloy 4, yes. But we have MOX with M5 cladding, but not including the very recent data.

CHAIRMAN POWERS: Now what I was curious about is at the beginning of your presentation you listed some 2400 fuel assemblies -- 1300 fuel assemblies. I wonder, could you give me an estimate -- I don't need a particularly accurate number -- what fraction of those had M5 cladding and what -- Well, that's the only number I need, is what fraction had M5 cladding.

MR. BLANPAIN: Yes.

CHAIRMAN POWERS: Just a rough estimate.

MR. BLANPAIN: Today, usually in France we are using Zircaloy 4 cladding for our MOX assemblies. We have some M5 fuel rods with MOX for experimental purposes in France, but today the MOX product delivered by Framatome France in Germany is M5. The reference is M5.

So we have four reloads of MOX with M5 in Bergdorf and in four reactors today in Germany.

MR. NESBIT: I think that would maybe be tens of assemblies probably. It is relatively recent, if I'm not mistaken, after the transition to M5. So we are not talking, I don't think, about hundreds and hundreds of M5.

MR. MEYER: At Cadarache they have fabricated approximately 50,000 M5 fuel rods. Those are for the German market.

MR. BLANPAIN: To answer your question, it's the next one. You can see better about the clad outer surface corrosion. That is also the calculation against the measurement, and here the measurements all came from the reactor at Finisberg in Germany.

That graph is a result of a measurement of the rod of two assemblies, one three-cycle and another four-cycle MOX assembly, and compared to the L5 database.

In blue we have the M5 database. In the red points are the UO_2-M5 database on the KKP2 reactor, and the yellow points as well as the green ones are MOX rods compared to the world database showing that there is no MOX effect on cladding corrosion. It is a measurement of 100 percent of the rods in one assembly. Next.

That graph represents the pellets' density evolution with burnup. So it is a measurement of the pellets after irradiation, and it is also for the comparison of MOX fuel to UO_2 fuel. You can see there, there is no difference. The tendency is the same, because the surface -- is only due to the generation of fission products, and it is normally the same for UO_2 and MOX fuel. Next, please.

Here we have the fission gas release database. That is a database obtained from the examination of rods irradiated in EDF reactors. So we have here the UO_2 data in blue compared to the -- the MOX data in blue compared to the UO_2 data in red.

You can see quite an increase of fission gas release for the MOX. That corresponds to the end of the cert cycle, and it is not -- The ranking is not a certainty on the measurement, but is due to the different heat rate experienced by the different rods during the - mainly the -- of the irradiation cycle. So the fission gas release is a function of the temperature, and then to the -- experienced by the fuel.

If you notice also that there is no burnup enhancement due to the burnup when we see the blue points at 50 and then a 60 -- because there is normally a power decrease after the second -- cycle. The next one.

This slide, higher fission gas release sampled in MOX fuel. It explains we have neutronic properties of the MOX, leading to higher linear heat rates at medium and high burnup, but it is mainly in the -- in EDF reactors, and also different pellet radial power density distribution, leading to different temperature, at higher temperature for the MOX fuel at high burnup.

The physical properties: As I said before, is due to the slightly lower thermal conductivity of the MOX, leading to higher fuel temperature. But it could be also due to the oxide microstructure, to the presence of the plutonium rich particles due to the MIMAS process can change the mechanism of fission gas release due to the very high local burnup and leading to the formation of dense pore populations. But we think it is a very small effect, and there is no -- As I said at the beginning of the presentation, the contribution of the last plutonium rich particles is quite small.

What is important to note is that the Halden temperature threshold for fission gas release is the same as for UO_2 fuel. That has been measured several time sin the Halden reactor.

The next one, please. Here we have a radial cut of a high burnup MOX pellet, showing that there is no difference compared to the radial cut of a UO_2 fuel. It is showing the same cracking pattern.

CHAIRMAN POWERS: This hits to a point

that Dr. Lyman raised, a question that he raised in his presentation. I think it is fair to characterize it as a question.

Was there any inherent difference in the fragmentation of this MOX fuel relative to what we have experienced in uranium dioxide fuel a we go up to these burnups? I'll have to admit, had you not told me this was MOX fuel, I probably would not have known otherwise. I mean, it looks like pretty much the same kind of fraction pattern that I am used to, but that is not a statistical -- It's not a statistical set.

Do you have statistical data that suggests that this is about the same or is just a qualitative sentiment here?

MR. BLANPAIN: No. That is the cracking pattern of the fuel after normal operation as compared to UO_2 is the same. After the -- incident, UO_2 and MOX fuel is roughly the same also. But with -- roughly the same. But what -- Some differences were -- People have published that the MOX behaved differently during the AI situation, after the -- test. But what we have seen is not a difference in fracture of the MOX compared to UO_2.

If we can see the next figure, please. That is the evolution of the plutonium rich particles during irradiation. That is the period after three irradiation cycles.

The plutonium rich particles -- The evolution is like high burnup structure or the ring structure with the formation of small bubbles and then coalescing throughout the pellet center line due to the high temperature.

Okay. But after the Cabri tests, we have seen that, like for UO_2 fuel, we have the equation of the grains, the equation of the UO_2 grains, but what we have seen also -- and that is the plutonium rich particles remained intact after the Cabri test.

So today the explanation is not definitive, because we haven't seen the grain equation, but around the UO_2 -- the UO_2 grains, and not around the plutonium rich particles. Okay, next one.

Now I conclude with the short and medium term development. For economical reasons, MOX fuel has to perform as efficiently as UO_2 fuel with regard to the burnup and operational flexibility.

The burnup equivalence to uranium fuel assemblies has been demonstrated in Germany, Switzerland and Belgium. The discharge burnup is around 50 in those countries.

In France, that parity between UO_2 and MOX fuel is not rich today, but will be completed next year. The licensing process is almost completed for the 20 MOX licensed power plants, and that parity will be to discharge rod burnup of 60, as for UO_2 fuel.

That parity must be established on a medium or long term basis, of course. So we are to work and to demonstrate that the MOX product can follow the UO_2. Next slide.

Now to conclude: Extensive poolside and hot cell examinations have demonstrated the excellent behavior of the MOX fuel up to assembly burnup to around 60. The performance of the current MOX is equivalent to that of UO_2 in terms of discharge burnup without any penalty on core operating conditions and fuel reliability.

Now ongoing development are still underway, and to demonstrate equivalence of UO_2 and MOX fuel up to very high burnups. By that I mean that we are working on UO_2 for a very, very high burnup. For example, UO_2 -- to develop the UO_2 fuel with the chromium fuel with large range to go to very high burnup to increase the efficient retention.

We are working also on the MOX product in the same way, to increase the range. Thank you for your attention.

CHAIRMAN POWERS: Thank you for a most informative presentation. Do members have any questions they wish to pose to the speaker?

DR. BONACA: The MOX experience, the European MOX, which has a different kind of isotopic content of plutonium as well as different content of UO_2. Are you going to say about the effect of those differences on this experience or is somebody going to address that today?

MR. NESBIT: Let me take a short first. First of all, the issue that you bring up is addressed at substantial length in our application, in the topical report, and in some other materials which we have provided the NRC.

We didn't make that a point of emphasis today, because it is not one of those issues that is really highlighted in the current contentions that are before the licensing board. But let me make a couple of statements about that.

Fundamentally, the major impact of the isotopic differences is to lower the required plutonium concentration in the fuel pellet, and what that means is that the MOX we use is generally closer to LEU than the MOX that is used in the European experience.

The isotopic differences are addressed specifically in the nuclear design topical report and in the nuclear analyses by modeling the specific isotopics that are in play there.

Also one other thing. It is the same uranium we are using for our MOX.

CHAIRMAN POWERS: I thought their uranium drank more red wine.

MR. NESBIT: It may.

DR. BONACA: I simply -- Regarding some of the exhibits that we were shown, it would have been -- For example, there is an exhibit 24, fission gas release of BWR MOX. It would be interesting to have a comment that says, as to applicability to the specifics that we are going to insert in the U.S., this is the expected -- I mean, I realize that you have some information, but this is information that I hadn't seen before.

MR. BLANPAIN: Of course, it is difficult to do within half an hour.

DR. BONACA: I understand.

MR. BLANPAIN: And about the plutonium content, of course, we have in our database fuel rods with two percent of plutonium content up to 10 percent. From the EDF reactors it is from 2 at the low plutonium content in the assembly at the beginning to 7 today. It is higher content in our assemblies in France. But we have data from higher plutonium content from analytical experience from Halden or the BR3 in Belgium or from Germany.

If we -- Between the 2 and 3 and 7-8 percent, there is no effect of plutonium content on the BWR, but of course, there is no effect of the -- for us, the thermal mechanical point of view, there is no effect of the isotopic composition on the thermal mechanical behavior, because if you have a best isotopic composition, that means few fission content.

You have to increase the total plutonium content, and that is what we are doing today, because for the MOX parity project, for example, the highest plutonium content in the fuel rods will be around 10, due to the evolution of the isotopic composition of the reactor grade fuel. But the weapon grade fuel will be enveloped by the experience, because the average plutonium content will be 4.

DR. BONACA: I guess my questions are more directed at a full -- whatever a full load with MOX fuel will be rather than just the LTAs. The reason is that you presented to us some limits that are used in

Europe to the amount of MOX fuel that you can insert in a core, and they seem to be related to fission product releases or whatever. That is what you mentioned something there.

Now, you know, the plutonium composition, particularly the abundancy of 239, may cause some differences in this fission gas release in PWRs. In the long run, I am interested in knowing how that plays out in the amount of MOX fuel from weapon grade that you would be inserting in the U.S. reactors. Maybe that is not pertinent to the LTA presentation, but that is why I was interested in those figures.

MR. NESBIT: It might also be a little premature. Until we actually perform the safety analyses for batch implementation, which are ongoing at this time, and do the related analyses, we can't come back and say there is a hard and fast limit.

Based on everything we have seen right now, the 40 percent goal for batch implementation is achievable, but we haven't completed the case at this point.

DR. BONACA: That was the reason I was leaving those questions. Okay.

CHAIRMAN POWERS: Let me follow up that question. When you go to the weapons grade plutonium feed into the material, will you get the same distribution between particulate and matrix plutonium?

MR. BLANPAIN: So okay. In Europe Framatome is using primary blend. The plutonium content in the primary blend is around 30 percent. So it is for evident economical reasons and also the plant is designed for that.

As you know, we are using three different plutonium content for one assembly, and we dilute differently that primary blend to the secondary to redefine the plutonium content. Okay?

So the overall experience we get from those MOX fuels is a MOX fuel with 30 percent of plutonium in the primary blend with a fission density. It is a well known fission density, because it is the same -- different kind of fuel.

So the reason why we go to the different isotopic composition of the weapon grade fuel, we decided to reduce the plutonium content in the primary blend so to get the same fission density that we have in our ARPM fuel.

CHAIRMAN POWERS: So your primary blend will run more like 20 percent?

MR. BLANPAIN: Yes.

CHAIRMAN POWERS: And now when you come to the final fabricated material, you will blend up with a higher fraction of matrix plutonium or the same fraction?

MR. MEYER: We expect it to be about 20 percent.

CHAIRMAN POWERS: But 20 percent -- What we saw in the microstructural examination here is that you have about 25 percent of your plutonium in particulate, and the rest of it in matrix. I'm just wondering if you are going to get the same split there.

MR. NESBIT: I don't know the answer off the top of my head.

CHAIRMAN POWERS: Any other questions? Well, we very much enjoyed it. I can assure you, the committee very much enjoyed your presentation. They love to get into this.

I think we are now in a position to take a well earned break, and so I will recess us until -- what is that, 11:35? 11:40. Ralph is being generous today. We will recess until 11:40.

(Whereupon, the foregoing matter went off the record at 11:23 a.m. and went back on the record at 11:42 a.m.)

CHAIRMAN POWERS: Let me apologize for Mr.

Blanpain. I did not do my job of summarizing his essential points, which I took it to be that under normal operations, MOX and your AM dioxide fuels are about -- are essentially the same. There might be some differences for a reactivity initiated event, perhaps because there is an earlier onset to the equivalent of what we could call the rem effect.

He also made a point that the pellet clad mechanical interactions might be somewhat reduced because of a higher creep rate for the fuel. There might be other ramifications of a higher creep rate.

A quick and dirty summary of a very nice presentation. If you have no objections, then we will proceed on to discussing the fuel assembly design with George Meyer.

MR. MEYER: We expect My name is George Meyer. I'm the MOX Fuel Qualification Manager. I work for Framatome ANP in the U.S.

CHAIRMAN POWERS: You got to speak right into that thing and put your nose right up to it.

MR. MEYER: I've got three topics, and I'll try to keep it brief. I will tell you about the lead assembly design, a little bit about the design evaluation, and I will talk about the quality assurance programs that we have in place for the lead assembly fabrication.

CHAIRMAN POWERS: Let me just interject, George, that I did become confused over the exact design between the SER and the recent letter. It left me somewhat exactly what these rods will look like.

MR. MEYER: Recent letter?

CHAIRMAN POWERS: The April 16th letter, I think, last Friday. I'll have to look to see what the date is.

MR. NESBIT: That letter would not have impacted the design of the MOX fuel assembly itself. That letter, I think, if it is the same one, addressed the issue of the other Westinghouse lead test assemblies.

CHAIRMAN POWERS: I believe it contained a listing of what the various geometries of the rods are.

MR. NESBIT: Right.

CHAIRMAN POWERS: And that did not seem to square with the description I had in the SER. That may have been my poor reading.

MR. NESBIT: It came from -- The Framatome assembly information came from the MOX fuel design report.

MR. MEYER: Okay, next slide.

The MOX lead assembly is a U.S. fuel assembly design that integrates MOX pellets into the fuel rods. The assembly design is the same design as the advanced Mark-BW fuel assembly.

This design is presented in two topical reports, the first being one that presents the fuel assembly itself, the Advanced Mark-BW, and the second is the MOX fuel design report.

Can you hear all right? Good.

This design is an evolution of the Mark-BW. It is a 17 x 17 PWR fuel assembly which has operated successfully at McGuire, Catawba, Trojan and Sequoia, and as of March 2004 over 2800 Mark-BW fuel assemblies have been supplied.

The Advanced Mark-BW is the terminology for the latest evolution of that design, incorporating updates that have been made over the years since the first lead assemblies of this design were introduced in 1987.

The Advanced Mark-BW is also represented by four lead test assemblies which operated at North Anna Unit 1 for three cycles, completing radiation in 2002 with a fuel assembly burnup of about 52 gigawatt days per ton uranium, and one of those lead assemblies then was reinserted for a fourth burn, and it is completed its irradiation at North Anna Unit 2 now. It is completing its irradiation this month with a lead fuel rod burnup of 71.7 gigawatt days per ton peak rod. Next slide.

The fuel assembly design -- This is a picture of the -- or a schematic picture of the lead assembly design, and it is, as the note says, identical to the Advanced Mark-BW.

Design features listed: M5 structural tubing; M5 mixing grids; M5 fuel rod cladding; TRAPPER bottom nozzle. As I mentioned earlier, all of these features have been integrated into the design over the last several years.

The things that constitute the Advanced BW and are used in the MOX lead assemblies include M5 mixed-van mixing grids and the TRAPPER debris filter bottom nozzle.

The only MOX-specific features are, in fact, the fuel pellets, with a minor change to the fuel rods. One point that I want to make here is that the design, as I said earlier, is an integration of existing technologies and, therefore, the lead assemblies represent a demonstration of that integration rather than a test of a new design. Next slide.

DR. KRESS: Before you leave, a couple of questions. Has the Advanced Mark-BW -- Is that getting a separate approval from NRC or is it part of the MOX fuel?

MR. MEYER: It has a separate approval.

DR. KRESS: It already has?

MR. MEYER: Well, it was in parallel with the -- It has a draft SER, as does the MOX fuel design report, but it is addressed in a separate report, both reviewed at the same time.

DR. KRESS: Do the mixing vanes and the mixing grids have a significant effect on the LOCA analyses?

MR. MEYER: They are considered as a part of the LOCA analyses.

DR. KRESS: They are considered in there? And you have a database to back that up, or what its effect is?

MR. MEYER: The mixing grids are the same mixing grids that have been used in the Mark-BW fuel assembly design since 1987.

DR. KRESS: Oh, they are the same?

MR. MEYER: Yes.

DR. KRESS: Somehow I thought the --

MR. MEYER: What is added in the new

generation is the mix-van mixer, and that has the same vanes, and it is included and addressed in the LOCA analysis.

DR. TRIAFOROS: Excuse me. You wouldn't be discussing new generation here, is the Advanced design, and it is the Mark design? Right?

MR. MEYER: Wrong terminology.

DR. TRIAFOROS: Excuse me. Yes.

MR. MEYER: This is the Advanced Mark-BW. Yes.

DR. LEITCH: I am a little confused about the length of the fuel rods. The fuel rods are slightly longer in this design, but are the overall dimensions of the fuel assembly the same -- the overall dimensions?

MR. MEYER: Dimensions of the fuel assembly are the same. The next slide shows the fuel rod dimensions.

This slide shows some of the key design parameters for the fuel rod, and it compares the MOX lead assembly fuel rod to the Advanced Mark-BW fuel rod. In fact, that rod on the right is the same design that was used in the North Anna lead assemblies. So they are both M5 alloy. It is a .25 inch difference in overall rod length.

They have the same cladding diameter and thickness, the same pellet diameter, the same gap size.

DR. LEITCH: The actual fuel is the same? This .25 inches is just to give you more gap at the top?

MR. MEYER: .25 is to give you more quantum volume to accommodate the higher fission gas release.

MR. NESBIT: It is the same fuel stack height as the resident LEU fuel.

MR. MEYER: Stack height is not shown, but it is the same for the UO_2 and the MOX fuel and for the co-resident fuel in the Catawba core. It is 144 inches.

Okay. The design burnup is slightly different, and for the lead assemblies we are intending to take them to slightly under 60,000 peak rod burnup in three cycles. That is intended to support future batch operations to a burnup limit of about 50,000 megawatt days per ton.

CHAIRMAN POWERS: Here is where it is unknown, fuel rod length, why I got a little bit confused here; because in this April 16th letter we have fuel rod lengths for the RFA and the NGF listed at 152.8, whereas the Mark-BW/MOX1 is 152.4.

Yet the SER says that the MOX is longer.

MR. MEYER: That is compared to the Mark-BW and not to the RFA -- Advanced.

CHAIRMAN POWERS: Well, all this leads to confusion on exactly what is in this core.

MR. MEYER: Well, the comparisons made in the fuel design report are made to the Advanced Mark-BW, because that is the base design, the base Framatome fuel design for the MOX lead assemblies.

CHAIRMAN POWERS: I see.

MR. NESBIT: The Westinghouse RFA fuel is described in some Duke submittals and also in the safety analysis report.

MR. MEYER: I mentioned that the fuel rods shown on the righthand column are the same rods, same rod design as in the North Anna lead assemblies, which is being taken out to a burnup of approximately 71.7 gigawatt days per ton rod burnup in an ongoing irradiation. Next slide.

I don't plan to go into the details of the design evaluation unless you have questions about it. What I want to say is that the design evaluation is presented in the MOX fuel design report. It addresses the requirements of Standard Review Plan 4.2.

For the fuel rod, which is different than UO_2 in the sense that it uses MOX pellets, all of the evaluations use models that have been adjusted as necessary for MOX. It uses the COPERNIC fuel performance code which has benchmarked to the MOX data, and in particular the data that Mr. Blanpain showed you, and it uses models that have been shown to be appropriate for MOX and addresses the criteria, the same criteria that are addressed for UO_2 fuel.

Fuel assembly evaluations are, for the most part, not affected by the use of MOX fuel. Where they are affected -- for example, in evaluating the fuel assembly lift and hold-down analyses where the spring constants could be affected by the fluence -- that effect is incorporated into those analyses and those evaluations.

Those design evaluations are presented in the MOX fuel design report and, where appropriate, reference back to the Advanced Mark-BW design topical report. Next slide.

Framatome ANP is the fuel designer of the MOX fuel assemblies. The lead assemblies will be fabricated in Europe under the Framatome ANP quality assurance program. That program meets the requirements of 10 CFR 50, Appendix B.

It is a program that has been globally implemented within Framatome ANP.

CHAIRMAN POWERS: Is this, in fact, an ISO9000 --

MR. MEYER: It is, yes.

CHAIRMAN POWERS: There is enough equivalence?

MR. MEYER: ISO9000, NQA1, KTA1401. It addresses all of the requirements for the various regulators and organizations.

The suppliers -- Each of the suppliers that will be providing components for the lead assemblies are or will be qualified by Framatome under this QA program. That includes Los Alamos National Laboratory for the supply of the plutonium oxide powder. It includes COGEMA Cadarache facility for the fabrication of the pellets and fuel rods. it includes COGEMA MELOX facility for the fabrication of the lead assemblies.

CHAIRMAN POWERS: Whenever one thinks about plutonium dioxide derived from weapons components, one immediately punks in, gee, what is the gallium contamination of this. Contaminating plutonium dioxide with gallium, of course, is a difficult chore, since gallium is not favorably disposed to stand the oxide very well. Can you tell us what your spec is?

MR. MEYER: The spec is 300 ppb in the plutonium oxide powder, and that is achieved by polishing the powder by removing the contaminants, and that is done at Los Alamos.

The 300 ppb in the powder produces a component or an effect of about 15 ppb in the resulting pellets. That is consistent with what has been observed in irradiated UO_2 fuel pellets.

CHAIRMAN POWERS: Just because of the yield of gallium in the spectrum, in the fission spectrum?

MR. MEYER: No, because we have reduced the gallium to the 300 ppb level. The incoming feed materials have gallium on the order of one percent.

CHAIRMAN POWERS: Yes.

MR. MEYER: So we are reducing that by polishing -- by putting it through an ion extraction chamber, and to remove the contaminants and the resulting powder, the plutonium oxide powder, meets a spec which has a maximum gallium limit of 300 ppb.

MR. NESBIT: Dr. Powers, was your question about the gallium in uranium oxide fuel or MOX?

MR. MEYER: Oh, I'm sorry.

CHAIRMAN POWERS: Well, he said this is similar to what you get in the uranium dioxide fuel, and I asked if just because of the gallium yield --

MR. MEYER: Oh, no, no. This is un-irradiated uranium fuel.

CHAIRMAN POWERS: Just a normal feed.

MR. MEYER: I misunderstood the question.

CHAIRMAN POWERS: And do you -- This is a question on which an "I don't know" answer would not be surprising. Do you have any evidence of the gallium attempting to segregate itself out as a distinct phase?

MR. MEYER: In the pellets?

CHAIRMAN POWERS: In the pellets, yes.

MR. MEYER: There is no evidence. There is an experiment managed by Oak Ridge, an irradiation at the advanced test reactor that has irradiated MOX pellets that were fabricated at Los Alamos. I don't know if you are familiar with that experiment, the average power test.

Those pellets had gallium levels on the order of 1 to 3 ppm. That experiment just concluded the irradiation phase, 50,000 megawatt days per ton burnup. The PIE data is available for 20, 30 and 40 gigawatt days per ton irradiations. It shows no evidence of separation, shows no evidence of any detrimental effects from the gallium.

CHAIRMAN POWERS: Thank you.

MR. MEYER: Another comment on the quality assurance: All of the non-fuel components, the fuel assembly components, the cladding, the grids, etcetera, are supplied by Framatome ANP, and those are the same components that are supplied routinely for the Advanced Mark-BW fuel assembly design.

So to summarize, the lead assembly is an integration of European MOX technology into the U.S. fuel assembly design. It is not a new design. It is an existing design with different pellets.

The design evaluation uses approved methods that have been shown to apply to MOX, that have been submitted to the staff for evaluation, and approved.

The lead assembly activities, including fabrication, are performed in accordance with the Framatome ANP quality management system.

We consider the lead assemblies are a demonstration rather than a test, since they represent the integration of existing technologies.

Thank you. That's all I have.

CHAIRMAN POWERS: Any questions of the

speaker?

DR. RANSOM: Just a clarification on the notation. You refer to RFA, NGF and then now this Mark-BW MOX-1. In your presentation you've got the MOX lead assembly and the Advanced Mark-BW. What is this Advanced Mark-BW?

MR. MEYER: The Advanced Mark-BW is the latest evolution of the Mark BW, and physically it is -- External to the fuel rod, it is identical to the Mark-BW MOX-1 assembly.

The designation of Mark-BW/MOX-1 is the designation chosen for the MOX fuel design. The Advanced Mark-BW designation is the equivalent UO_2 fuel assembly design.

DR. RANSOM: And they are both slightly different than the NGF?

MR. MEYER: The NGF is a Westinghouse fuel design, as is the RFA. What I am speaking to here are the Framatome fuel designs.

MR. NESBIT: If I can maybe respond as well and answer in a slightly different way that might help clarify this.

The Framatome product, the Advanced Mark-BW, is their product that they market to customers such as North Anna. The Advanced Mark-BW MOX-1 is the modification of that product for our program, and that modification consists of lengthening the fuel rod and putting MOX pellets in instead of uranium oxide pellets.

The Westinghouse assemblies are, obviously, a very different design. Well, they are similar in that they are 17 x 17 fuel, and they have similar pressure drops and things like that, but they are Westinghouse components and grids.

DR. RANSOM: Well, have these Advanced Mark-BW UO_2 assemblies been used in U.S. reactors?

MR. MEYER: Yes. The Advanced Mark-BW, as shown there, has been used as an LTA. The Mark-BW with some of the features that we call -- that constitute the Advanced Mark-BW has operated in various reactors.

As I said earlier, we have delivered over 2800 Mark-BW fuel assemblies, and those have operated successfully. The features that constitute the Advanced Mark-BW have been integrated over the years. So the design has evolved, and what we did in the topical report for the Advanced Mark-BW was to put all of the features that had evolved over the years into one place for one licensing submittal that provides a description of that final -- called the final product, the Advanced Mark-BW.

Then that final product with all of those features together has operated as an LTA at North Anna, and the first complete batch of the product that has all of the features described as Advanced Mark-BW will be loaded into North Anna in the coming months.

DR. RANSOM: Are these Advanced Mark-BW assemblies made in like Richland, Framatome?

MR. MEYER: They are made at Framatome Lynchburg.

DR. RANSOM: And Richland also?

MR. MEYER: The pellets are made at Richland, but the fuel assemblies are made at Lynchburg. Framatome in the U.S. is making all of its PWR fuel in Lynchburg in the future and its BWR fuel in Richland.

DR. RANSOM: Thank you.

DR. LEITCH: Is it premature to talk about the post-irradiation test program? At 60,000 or so megawatt days per ton, I guess we are talking three cycles or so before we are to that point.

MR. MEYER: I can talk to that. We plan, first of all, a poolside PIE, another stroke of PIE after each cycle of operation. That would include the typical nondestructive evaluations. It would include visuals for overall appearance to assess the need for additional work, visuals of both fuel rod and fuel assembly.

It would include measurements on fuel assembly growth and fuel rod growth and the gap closure, and it would include evaluations of bow and distortion.

After the completion of the second cycle of irradiation, we expect to hold out two or three assemblies to reinsert one or two for the third burn. The assemblies that are held out, then we will do what we call an extended PIE, and the scope of that depends to some extent on what has been learned from the visuals and the other work. But we would expect to do measurements such as grid width and fuel rod oxide thicknesses and R-68 drag testing and rod to rod spacing, ITML profile.

We would also then at the end of the second cycle of irradiation where we will have achieved the burnup approaching 50,000 megawatt days per ton rod peak burnup -- we expect to take rods out to a hot cell for hot cell examinations. those examinations would include puncture, rod pressure, fission gas release, cladding metallography, cladding ductility, fuel pellet analyses, and fuel pellet ceramography as well as burnup analyses.

DR. LEITCH: While you are doing this then, if I recall you correctly, then a couple of LTAs will still be in the core?

MR. MEYER: Yes, for their third burn.

DR. LEITCH: When those come out, you will do similar tests?

MR. MEYER: When those come out, we have -- At this point we are considering it to be an option for additional hot cell work, and the decision to do that would be dependent on how the program is proceeding and what the experience is.

DR. LEITCH: Thanks.

MR. MEYER: You're welcome.

CHAIRMAN POWERS: Thank you very much. We will move on to Mr. Eller.

MR. ELLER: My name is Jim Eller. I work for Duke Power Company. Our role in the MOX fuel project is to provide irradiation services. Because Duke has been licensed to do their own reload design for many years, irradiation services means more than just putting fuel assembly in the core and irradiating it.

Some of the other services that are provided are the standard kinds of analytical services that we perform to support LEU cores. One area of those services is nuclear analysis, and that is the subject of my presentation.

The first slide is a general overview of the presentation. I have a couple of slides that describe the analytical models that we use in nuclear analysis. I have several slides that describe the benchmarking process that is used to define the fidelity of the codes, to show their ability to predict the physical world.

Then at the end of the presentation I have a couple of slides that give information about the core design that we have just completed that would include the four MOX LTA.

Duke Power uses computer models that come to us from Studsvik Scandpower Corporation. The name of their package of codes is called Core Management System or the CMS package. Studsvik Scandpower -- these codes are used by various organizations in Europe, North America and Asia, and we have been using them at Duke Power since the mid-1980s in one form or the other. Of course, the codes have evolved over the years.

They are currently used by about 55 organizations in 11 countries to support reactor core design and operation of many BWRs and many PWRs in, as I said, Europe and Asia and North America.

Those core designs perhaps number to as many as 2000 fuel cycles since the 1980s, and some of those fuel cycles have been fuel cycles in Europe that included MOX and LEU fuel. Next slide.

I have just a brief bit of information about each one of the major programs. The first is CASMO-4. It is a two-dimensional, multi-group transport theory model. We use it to analyze the detailed behavior of each unique fuel lattice in the core, and fuel lattice does not mean fuel assembly. A single fuel assembly may have two or three or four unique lattices along its axial high.

If you have a fuel assembly that has a blanket and a BP, there may be regions where the BP is present and regions where there's central enrichment and no BP. So there are more than one lattice in each assembly, and each one of those unique situations is modeled in two dimensions with CASMO-4.

CASMO-4 is executed for the range of temperatures and lattice configurations that would exist in the reactor or in the lattice itself, and the combination of all that, as you can imagine, produces a lot of information from CASMO, cross-sections and discontinuity factors that are used in the core model.

CMS-LINK gathers all that information together and tabulates it in a fashion that the core model can use, and it is in a single file now that the core model reads.

The core model is SIMULATE-3 MOX. It is a three-dimensional, two-energy group diffusion theory model. It includes enhancements to model cores containing LEU and MOX fuels. We generally run this model -- We can run this model in either full or partial core geometries, depending on what the symmetry of the analysis requires.

We usually run the model with four radial nodes per assembly and 24 axial nodes in the active fuel column. So the nodalization of the model would be -- The x-y would be half of an assembly pitch, and the z would be six inches. That is the normal nodalization of the model. We use the same nodalization, regardless of MOX or LEU in the core.

SIMULATE-3K MOX is an extension of the SIMULATE code that is used to model fast reactor transients, and in the work that we have done so far to support the licensing process and the benchmarking process, it was used to interpret the signals that we get when we do dynamic rod worth measurements at the beginning of each cycle. Next slide.

I am now moving on into the subject of benchmarking a little bit. Benchmarking is simply a comparison of the predictions that come from the models to measurements that we get when we operate the reactor or measurements that are made in laboratory experiments.

The type of measurements that are made during the operation of the reactor: Some of them are made at the beginning of the cycle during what we call zero power physics testing where we measure control rod worth, temperature coefficients, and we also take some careful boron concentration samples -- excess reactivity.

DR. BONACA: I had a question. You said that SIMULATE, you have four regular nodes per assembly. How do you get your pin to average peaks per assembly?

MR. ELLER: Say that again, please.

DR. BONACA: You said that the SIMULATE simulates four regular nodes per assembly.

MR. ELLER: Right.

DR. BONACA: And the question I had is how do you derive your pin to average for the assembly?

MR. ELLER: We can request many edits from the codes. When we requested an assembly that requires an assembly average to be performed, that is done internal to the code.

DR. BONACA: Okay.

MR. ELLER: So if we wanted hot pin to assembly average, there is an edit for that, and the averaging down to a full assembly and the normalization that is required for all that is taken care of internal to the code.

DR. BONACA: So you do have a fine mesh capability?

MR. ELLER: Yes.

MR. NESBIT: Well, the SIMULATE performs a 10-power reconstruction. It is kind of state of the practice, I would say, in the industry now, in order to get the detailed pin information out of a nodal code such as SIMULATE.

MR. ELLER: The primary solution in SIMULATE is a nodal solution, the fusion theory. SIMULATE uses information that comes from CASMO about the pin by pin power distribution and discontinuity factors at the boundary of the nodes and reconstructs the pin power distribution that would exist in the core if it had performed a pin by pin analysis.

DR. BONACA: So you perform comparisons,

and you have topical reports. You submit all that?

MR. ELLER: Yes, sir. The model and that function and capability has been used at Duke Power for LEU cores for many years, and it is in various topical reports, and we have repeated that work for cores that contain MOX fuel. I'll speak to that a little more as I go along.

DR. BONACA: Okay.

DR. RANSOM: With the nodalization you talked about using for SIMULATE, then does it include the entire core loading, assembly by assembly?

MR. ELLER: Yes, sir.

DR. RANSOM: So you have nodalized all assemblies?

MR. ELLER: Yes, sir.

DR. RANSOM: What, to a quarter of the core then at a time?

MR. ELLER: We can run it either way. For analyses where the core has symmetry about the core to core, we only model the core to core. But in an analysis like a rod ejection or a dropped rod where there is no symmetry in the core, we run the model in the full core, and when we expand the model to full core, the nodalization stays the same.

DR. RANSOM: What do you do for the

thermal hydraulics? Are you going to talk about that later?

MR. ELLER: The code does thermal hydraulics in a simplified fashion. Detailed channel thermal hydraulics are performed by a separate set of codes and a separate group of people, and the primary code there is VIPER. I am not prepared to speak very much about that today, but we have other people here that can talk about the detailed thermal analysis, if you have questions.

DR. RANSOM: Well, out of this calculation, do you get the rod temperature, especially if you have some a rod ejection or analysis?

MR. ELLER: In the SIMULATE model itself, we don't get rod temperature out of that model. The code calculates the simple rod temperature so that we can look up the appropriate cross-sections.

When we go to the SIMULATE-3K model, if we were to use that code for rapid transients where fuel temperature is very important, there is an explicit pin conduction model in that code, and it provides more detailed calculation of the pin fuel temperature.

MR. NESBIT: As a matter of clarification, what Jim is talking about is primarily the steady state core design calculations that are performed. What you are asking about are the safety analysis type applications, which are typically done using inputs derived from the core calculations, but with other codes.

DR. RANSOM: Okay.

MR. ELLER: Okay, back to the benchmarking slide. Much of the information, as I was saying, comes from measurements that are taken during power reactor operation, and I listed those previously.

Obviously, we have a lot of historical data from McGuire and Catawba, a lot of historical measurements, and we use those and compare them to predictions that come from these enhanced models. But all of those cores contained only LEU fuel, and in order to make a statement about the fidelity of the models for cores containing a mixture of MOX and LEU fuel, we obviously have to go somewhere else and find measure data.

So we went to data from the French reactor. St. Laurent has operated for many cycles with MOX fuel. It is very similar to a three loop Westinghouse plant like North Anna, for example, 17 x 17 fuel, etcetera.

They make the same types of physics

measurements at the beginning of cycle. They take the same type of flux maps during the depletion of the cycle. The in-core detectors are functionally very similar.

So we get very equivalent types of measurements from the St. Laurent reactor that we get from our own reactors. The total body of data -- of measurement data that we benchmark against there is approximately six fuel cycles for each reactor, McGuire, Catawba and St. Laurent, so a total of 18 fuel cycles, as I recall.

The data that comes from power reactor operations, the measurements that come from power reactor operations is not detailed enough for us to verify the models to the extent that is required to support reload design, and the specific example that I am talking about is the flux mapping and the power distributions that are measured during operations are assembly by assembly power distributions.

When we do reload design, we have many design criteria that are based on the fuel pin or on the channel. So the power reactor data doesn't provide enough information for us to characterize the fidelity of the codes on a pin by pin basis, as we have to.

In order to get at that fidelity inside the fuel lattice on a pin by pin basis, and in order to get at how well this pin power reconstruction process works, we have always had to go to critical experiments where the power distributions are measured on a pin by pin basis.

The experiments that we have used several times in the past are the B&W LEU experiments. They contain critical arrays of fuel pins that were LEU only, but there was some gadolinium in those. We don't use any of the gadolinium pins, but these experiments are used to qualify the performance of the code for LEU fuel. We have repeated that work with these codes.

Again because there is no MOX in those experiments, we had to go somewhere else to get additional data. So one place to go to is the SAXTON experiments which are multi-region LEU/MOX experiments performed in the Sixties and sponsored by Westinghouse, I think.

These experiments, while they are very old, they contained plutonium which had a Pu^{239} content of 90 percent, which approaches what is labeled weapons grade material.

We also got some additional measured data from experiments that had been performed in France. These EPICURE and ERASME experiments have the advantage of having a geometry in the center of the experiment that is geometrically similar to a 17 x 17 fuel assembly.

The EPICURE and ERASME experiments also have poison pins in the arrays that are like the types of poisons that we use in our PWRs, and this whole body of experimental data allows us to do fairly voluminous analysis of how well this pin power reconstruction works inside the core simulator.

So if you look at the benchmark analysis as a whole, we believe that it covers a wide range of reactor materials and operating conditions like you would find in the operation at McGuire and Catawba. Any one piece of it may have some deficiencies, may not be the perfect measurement to compare against, but the package as a whole, we think, provides a very robust analysis.

In general, the results from this benchmarking work show no significant trends or deficiencies related to MOX fuel. When you compare the results from St. Laurent to the McGuire and Catawba, you don't see any significant differences or trends there. The same for the experiments, the critical experiments that included MOX fuel or included only LEU fuel.

DR. RANSOM: By deficiencies here, you mean different designs. Is that right?

MR. ELLER: There are no biases that we see related to MOX. The comparison is never perfect. There are always differences between prediction and measurement.

DR. RANSOM: Well, I'm interested in whether there is any significant difference between what you would predict for LEU fuel and the MOX fuel.

MR. ELLER: No. There are differences, and I can't give the exact numbers in the meeting, because the French are very protective of their data, and we marked that proprietary in our topical report.

The topical report on all of this benchmarking process and the statistical treatment of the uncertainties and biases are provided in that report. A summary of that topical report was also presented and a paper at an ANS meeting last fall.

DR. RANSOM: You are referring to the Duke Power Company report?

MR. ELLER: Yes, sir.

DR. RANSOM: PPC 105?

MR. ELLER: Yes, sir.

MR. NESBIT: I think maybe a pertinent point that we can make in this meeting is that the results of the -- relative to the MOX fuel and the LEU fuel were so similar that we just ended up using the LEU fuel uncertainties as our overall uncertainties.

DR. RANSOM: Is all that in your submittal or the -- to the NRC?

MR. NESBIT: Our reference to the topical report. It's in the topical report.

MR. ELLER: Other questions? Next slide.

Moving on the subject of the core design that would contain the four MOX lead assemblies, we have some additional design criteria that we have placed on ourselves that are above and beyond the LEU fuel. One of them is that no control rods will be placed in the MOX fuel assembly in the first or second cycle of irradiation.

That's pretty much a given in the first cycle, because it is going to contain burnable poisons like all the other feed fuel does.

A second design criteria that is specific to MOX is that we made a commitment that at least two of the MOX fuel assemblies will be placed in locations that are instrumented. That is to say, locations that an in-core detector will pass through when flux maps are taken.

We have a strong preference to place all four of the assemblies in instrumented locations. Then as I will show in a minute, the current design puts the assemblies in core locations C-08 and its symmetrics which are fully instrumented around the core.

The MOX fuel assembly power peaking must not lead the core during nominal completion, and I will show a table that shows these power levels in a couple of slides.

The analysis -- The detailed analyses that take places through the summer and into the fall this year will check the power distributions, both for nominal and off-nominal operations, that occur in the MOX assembly, and verify that those power distributions have acceptable margins through all the MOX specific design limits that are generated by the mechanical analyses and the thermal hydraulic analyses.

The point being there is that, as the design work proceeds, we are doing very specific analysis for the MOX LTA core and the MOX fuel that is in that core. There aren't any fudge factors being applied for MOX.

In the next slide, which doesn't show up very well, I have shown a picture of the bottom right quadrant of the full core. Each cell represents an assembly location. The cell in the top left of the picture which is labeled H-8 would be the center assembly in the physical core.

The cells which are highlighted in yellow are assembly locations that are feed LEU feel, fresh feed LEU fuel. The locations that are magenta or darker shaded in core locations C-8 and H-13 are the locations of the MOX fuel assemblies. Those are on the major axis, and that represents a total of four assemblies in the full core.

DR. LEITCH: Say that again, Jim. That is what puzzled me. It looks like there's eight MOX assemblies.

MR. ELLER: We knew that was going to happen.

DR. LEITCH; But you are saying that this is the major --

MR. ELLER: Yes, that is the major axis.

DR. LEITCH: So one of these is reflected in the other quadrant?

MR. ELLER: Yes, sir. This picture shows two major axes, and there are two other major axes in the core. So there's two other assemblies that are not represented in this picture. If I had shown the full core, we would have not been able to see it either.

DR. LEITCH: Okay, I understand.

MR. ELLER: The information in each cell indicates the initial enrichment of the LEU fuel or the initial total plutonium concentration in the MOX fuel locations. The second number is an LBP identifier that tells me how much burnable poison is in the assembly. Obviously, the burnable poisons are in the fresh fuel only.

Then there is a batch ID number, and the last value in each cell is the hot pin average power in each fuel assembly at 4 EFPD nominal conditions. So you can look in this picture and see that the MOX hot pin was 1.37 at this burnup, and the hot pin in the core is over in location G-8, I think, which is 1.434. So that is an indication of how far behind the lead in the core the MOX fuel is running.

The MOX fuel power, though, is representative of much of the fresh fuel in the core. So we are not giving it a break either. It's just not the lead in the core.

DR. RANSOM: Jim, what were the slightly

shaded assemblies?

MR. ELLER: Okay. There are two locations in the core that I tried to shade in -- Oh, oh.

DR. RANSOM: Not the MOX but the other ones that are shaded.

MR. ELLER: Those are feed LEU.

DR. RANSOM: Feed?

MR. ELLER: Feed Westinghouse RFA, first cycle fuel, fresh fuel. The ones that are not shaded you see are fuel assemblies that are in their second burn or in their third burn in this cycle.

DR. LEITCH: And B-12 and D-10 are the locations for the NGF fuel?

MR. ELLER: Yes, sir, that's correct. Those fuel assemblies are neutronically -- In the work that I do, the nuclear analysis work, those assemblies are so similar neutronically to the dominant Westinghouse RFA fuel that we do not model them as unique assemblies.

That is based on analysis that was done at the beginning of cycle 15, the previous cycle, where it was shown that modeling the assembly, very exactly or not, made no difference in the nuclear analysis work. Next slide, please.

This graphic is an attempt to represent

the lattice of the pin by pin lattice inside the MOX assembly. I think it has been mentioned a time or two that radially it is zoned radially across the lattice. That means that the plutonium concentration in each in varies as you move across the assembly radially.

The dark, solid circles in the corner are MOX pins that have the lowest concentration of plutonium. The gray, solid circles around the face of the fuel assembly have an intermediate or middle concentration of plutonium, and the open circles in the center of the assembly are the highest concentration of pins in the assembly.

The assembly is zoned so that, when you place it in a core face adjacent to uranium fuel, you can maintain a flat power distribution across the assembly.

DR. LEITCH: So every rod in the assembly is a MOX rod?

MR. ELLER: Yes, sir. The next slide is a table of information where I have tried to show some of the major core, the key core characteristics as the cycle is depleted. The units of depletion here are shown in effective full power days. The anticipated cycle length is 515 days.

There is a column that shows the boron

letdown as the cycle depletes and the change in axial offset as the cycle depletes. Then the fourth, fifth and sixth columns give information about the power -- the peak power in the core on an assembly basis. That would be the hottest assembly power in the core, the hottest pin power in the core, and Fq is the hottest spot on a pin in the core.

The final three columns show that same type of power peaking information for the MOX fuel assembly, and you can see throughout the burnup that the MOX fuel assembly is held behind the leading LEU fuel assembly.

The last row at the bottom of the table shows the burnups on an assembly and pin basis, the maximum in the core and the maximum that the MOX would achieve in the first cycle of irradiation.

So in summary, the proposed core design that would contain the MOX lead assemblies is consistent with our current fuel management practices. It places all four MOX assemblies in instrumented core locations.

The MOX fuel duty is representative of feed LEU fuel, but is not leading the core during nominal depletion.

Based on preliminary analysis that we have done with this core, we believe that the normal operating limits and controls that we place on LEU cores will provide sufficient margin to all of the MOX specific design criteria. That is to say that the normal control rod limitations and the normal power maneuvering limitations and the normal axial offset windows that we place on our plants should provide adequate margin with the MOX fuel in the core. We don't anticipate having to do anything extra.

There will be a lot of additional detailed analysis that will occur in the summer and fall that will clarify that before we actually load the core.

That's all I have.

DR. LEITCH: Do you anticipate any additional operator training requirements or will this basically be transparent to the operators?

MR. ELLER: There are additional training that will go on. The design process at Duke involves plant personnel very early in the process. We are already engaged in that.

We are already providing data to the plant simulator people that train operators, and even though there's only four assemblies in the core, we want to make sure that those simulators will work and, if there's any impacts, there are -- the organization is in place to pass that information along to the plant folks.

DR. LEITCH: Do you have a position called the reactor engineer?

MR. ELLER: Yes, sir. Our reload design process has meetings all along the way. We have already had two, and at each meeting more of the work has taken place, and the reactor engineer is invited to all of those. He has been to one of them in person. He has been to the other one with a conference call.

So those people are involved in every step of the process.

DR. LEITCH: Is that an on-shift position or is it day shift?

MR. ELLER: You are beyond me now.

MR. NESBIT: It is not a shift position. It is in the engineering organization, but during start-up type time frame, there is somebody from -- there's actually a whole reactor engineering group. There's somebody there all the time from reactor engineering.

CHAIRMAN POWERS: Any other questions for the speaker? Thank you, sir.

Let me ask you if you can split your

presentation into two.

MR. NESBIT; Yes.

CHAIRMAN POWERS: Why don't we do that, because what I fear is that, if we don't break pretty exactly at one o'clock, this person won't get any lunch, and he's hungry. So the rest of you, I don't really care about.

MR. NESBIT: What I am going to suggest is I think I can cover everything through LOCA before lunch, and then we can finish up afterwards.

CHAIRMAN POWERS: Right, and I'm sure LOCA will -- We want to allow lots of time for LOCA discussion. Go ahead.

MR. NESBIT: I am Steve Nesbit, the MOX fuel project manager for Duke Power, and -- next slide -- I am not going to go through all the slides once again, because I think some of the issues that aren't controversial probably don't need to be treated. I take at his word Dr. Powers' statement that the ACRS can indeed read the slides themselves.

I've got my presentation broken into --

CHAIRMAN POWERS: Well, that may not be 100 percent true. We do have somebody from Tennessee.

MR. NESBIT: Maybe it is a 95 percent I've got six different sections. I am going to first talk about some of the lead assembly characteristics that impact the safety and environmental evaluations that we have done, talk about LOCA, talk about the non-LOCA safety evaluations.

Then I am going to talk about the radiological consequences or dose calculations from the design basis accidents. I will talk about our environmental evaluation, and then I will summarize. Next slide.

The key MOX fuel lead assembly characteristics that impact our evaluations are: To begin with, mixed oxide fuel pellets, as has been noted, are sintered ceramic oxide pellets similar to low enriched uranium fuel. They contain about five percent plutonium oxide and the remainder uranium oxide. They have similar physical characteristics as low enriched uranium.

George Meyer talked about the fuel assembly design. I won't into that any further.

One key factor here. The MOX fuel decay head is lower than the LEU fuel decay head, if you compare a MOX fuel assembly with the same burnup to an LEU fuel assembly with the same burnup during the time frame of interest for transient and accident analyses.

Finally, we have shown that four MOX fuel assemblies have very small impact on global core physics parameters, things like moderator temperature coefficients, etcetera, that are key factors in accident analysis, and also on the core-wide radionuclide inventories.

CHAIRMAN POWERS: That is something we have not explored quantitatively today yet, but it was brought up earlier today. If we radically alter, hypothetically, the fission product release characteristics of MOX for these four lead assemblies, say multiplying them by 10, it really makes no difference in an accident first term, does it?

MR. NESBIT: It doesn't, and we've got some work that, I think, demonstrates that.

CHAIRMAN POWERS: I mean, you are working with four out of roughly 200 assemblies now.

MR. NESBIT: Right.

CHAIRMAN POWERS: The four are located in regions more centrally in the core early. So they are more susceptible to damage, that it may not be -- making a four to 200 ratio is probably not quite right, but still you can radically alter the -- Let's say we used, say, a 1465 source term and made, say, 100 percent cesium release instead of 30 percent. It's still --

MR. NESBIT: You ar still dominated by the LEU fuel.

CHAIRMAN POWERS: Yes, everything is dominated there, and we don't have -- So for the purposes of lead test assembly, we are probably okay just using LEU kind of release patterns here.

MR. NESBIT: Yes, but we did, in doing our dose analyses, make some adjustments on the gap fractions.

CHAIRMAN POWERS: Sure, that's right.

MR. NESBIT: And for the LOCA, the design basis at Catawba is the TID 14844. So we just use that. I think that would generally be considered.

DR. KRESS: I wonder if you can say the same thing about the non-LOCA design basis accidents.

MR. NESBIT: We address those, and we can say that -- Even if you assume that all of the damaged assemblies in a non-LOCA event -- that all of the MOX fuel assemblies are damaged in a non-LOCA event, preferentially, which should not be the case since they are not the leading assemblies, but even if you assume that, the impact on the overall doses is negligible.

DR. KRESS: Did you look at the potential for reactivity insertion accidents that might fail the steam generator tubes?

MR. NESBIT: Well, we looked at the reactivity insertion accidents specifically, but we did not -- Say, an RIA that causes a tube failure would be a beyond design basis event that we don't analyze.

DR. KRESS: Yes, that's right, it would be. You're right.

MR. NESBIT: Next slide. Decay heat: This is a SCALE calculation, a side by side, apples to apples of MOX versus LEU, and it shows the ratio of MOX to LEU. The key point there is that the crossover of one at a value of one when MOX decay heat becomes greater than LEU is approximately at three days, 70 to 80 hours.

So for the time frame of interest for accidents, the MOX decay heat is lower. We just assumed it was the same as the LEU decay heat.

DR. KRESS: These are results obtained from ANS standards?

MR. NESBIT: This is from SCALE. This is a SCALE calculation, ORNL code package.

DR. KRESS: Oh, SCALE is the name of the code?

MR. NESBIT: Right. Well, it's the code

package. Now I think the ANS would give you -- and actually, I'm not sure we have a slide on that, but we did a similar evaluation for the ANS standard for the LOCA calculation. It showed that the LEU curve bounded the MOX decay heat curves, and we just used LEU.

DR. KRESS: You didn't do that for spent fuel pool cooling?

MR. NESBIT: For spent fuel pool cooling, the impact of the MOX is --

DR. KRESS: Well, it's like two percent.

MR. NESBIT; Yes.

DR. KRESS: Hardly can see it.

MR. NESBIT; It is inconceivable that it could impact the overall load on the spent fuel pool with 1,000 assemblies or so. Next slide, please. Thank you.

Global core physics parameters: When we submitted our application, actually well prior to that, we did a comparison of representative core containing all LEU fuel, and then we extracted four LEU assemblies and replaced them with MOX fuel assemblies to give us an apples to apples comparison of what that does to the physics parameters.

Now it's not the exact core that we are

currently planning on loading the MOX fuel in, but that core design was not available at that time, and the results would be similar.

As you can see, for the key parameters that are used of interest in accident analyses, some of which I have listed here, the changes are on the order of zero to four percent or so. The bottom line is that these are the same kind of changes we see from cycle to cycle anyway, just due to core design change. Next slide, please.

I didn't cover some of the other --

DR. BONACA: Just a second. So this, for example, says the moderated temperature coefficient would be slightly more negative.

MR. NESBIT: That's right. That's right, which, depending on what scenario you are looking at, is either good or bad. But the bottom line is it doesn't move enough to be significant, and typically we do bounding safety analyses with parameters that bound these.

I didn't talk about the physical characteristics like thermal connectivity, etcetera, because those were covered in an earlier presentation.

Next I want to talk about the LOCA analyses, and then I want to go to lunch. So let's see how quickly we can get through here.

The approach that we used for the loss of coolant accident analyses is that we did explicit analyses of the response of the mixed oxide fuel assemblies to a design basis LOCA. Now in many LTA programs, that's not done. It is just -- You just assume that the resident fuel analysis is bounding, given the fact that you are not going to operate them at the limiting power. But we went ahead and did an explicit calculation for the MOX fuel assemblies.

We used the Framatome Appendix K model as licensed, and then modified as necessary to address mixed oxide fuel. That is a RELAP5/MOD2 based model. We looked at the potential MOX fuel effects, incorporated them as appropriate.

Next we did an apples to apples MOX to LEU comparison calculation to see what difference things made, and then we finally finished up by doing a series of studies on burnup and axial peaking location, etcetera, to establish a comprehensive set of LOCA limits for the lead assemblies.

DR. RANSOM: That RELAP5/MOD2 -- that is not S-RELAP5, Framatome's?

MR. NESBIT: It is the Framatome version that is -- It's been around a while.

DR. RANSOM: It is the standard NRC RELAP5/MOD2?

MR. NESBIT; Well, it's got some B&W mods in there for Appendix K type calculations.

DR. RANSOM: Has that been approved for licensing?

MR. NESBIT: It's been approved for a long time. The next slide, please.

Here are some of the potential MOX fuel effects that we looked at, and thermal conductivity. We have talked about before. the impact is small. We use the MOX specific thermal conductivity parameters, volumetric heat capacity at the plutonium concentrations we are talking about. Essentially, there is not a difference relative to LEU. We used the LEU values.

Decay heat -- we used the standard decay heat curve after looking at it and verifying that the value would be less for MOX.

We looked at a couple of nuclear related parameters, void reactivity and delayed neutron fraction. In both cases, the impact of any MOX different, to the extent there would be one,would be beneficial. More void reactivity and smaller delayed neutron fraction would tend to quicken the decrease in power following the initiation of the LOCA, and we just used the LEU values for that as well.

CHAIRMAN POWERS: When you use the term volumetric heat capacity, you are speaking of the core as a whole or what does the term refer to?

MR. NESBIT: I think the term refers to the actual model of the fuel pellet, fuel rod itself in the hot channel. Is that --

Just come over here and sit. I imagine this won't be the last time. Just sit over here.

CHAIRMAN POWERS: Would you identify yourself also?

MR. DUNN: I'm sorry. My name is Burt Dunn. I am an advisory engineer in lost coolant accident safety analysis for Framatome, Areva now.

The term that Steve is referring to is simply the density of the material times the specific heat.

MR. NESBIT: I should probably clarify at this time and follow up on something that Jim said. At Duke we do mst of the core reload design ourselves through methods that we have licensed with the NRC. One thing that we don't do is the Appendix K LOCA analyses. We rely on our vendor for that.

In the case of the MOX fuel, the vendor is

Framatome. So what I am talking about is analyses that were performed by Framatome, and Burt is the knowledgeable person there about this work.

CHAIRMAN POWERS: You mentioned the volumetric heat capacity, and you said essentially none. Then you used LEU. When we go back to the slide, we see that indeed for the previous presentation, indeed for the temperature range we're talking about, that the LEU curve does form a lower bound on the actual measurements for MOX.

MR. DUNN: Well, I think Patrick addressed a little bit of that, sir, in terms of that curve was developed from 20 percent plutonium, whereas the analysis that we are doing here or the fuel that we are going to load is about 4.5-5 percent.

MR. ELLER: It's five percent max.

MR. DUNN: So there is some difference in there. The heat capacity itself, if it helps any at all, would not be a strong actor for this reactor in terms of lost coolant accident. The key item is the balance of the decay heat versus the reflect cooling mechanisms that occur.

So a few percent one way or the other would not worry us. The recommendation from France is to use -- There is a small adjustment for plutonium concentration, but it is not very much.

MR. NESBIT: The other thing that I will mention in terms of MOX effects, and I left it off the Vu-graphs inadvertently, is really the big change that we made -- that Framatome made to their evaluation model to account for MOX is to use COPERNIC, the fuel performance code that has recently been approved for MOX and previously for uranium oxide, as the source of the fuel temperature information, as opposed to -- I think, the currently licensed Appendix K methodology uses the TACO code, but COPERNIC is the code that has the MOX models in it that have been reviewed and approved now. So that was really the biggest change to the evaluation model itself. Next slide. Thank you.

The MOX fuel assembly radial zoning: As Jim has just showed in his presentation, there's actually three different plutonium concentrations in the MOX fuel assembly, depending on which pin you are talking about.

One of the things that Framatome did is they looked at, well, what happens if you specifically model each of the pins of the plutonium concentrations versus just an average concentration? What they found was that there really isn't an impact as you might have expected, because it is driven by the power. There is not an impact of the plutonium concentration on the peak cladding temperature. Next slide.

The next thing we did was what I am going to call a stylized MOX/LEU comparison. We said, okay, let's take this --

DR. BONACA: Just a second, if I could ask a question. We have a statement on page 9: No significant impact of Pu concentration, because as you said, the dominant effect is power. So, therefore, the peak clad temperature still is in one of the UO_2 assemblies.

MR. NESBIT: In the calculation that Framatome performed for us, they performed an explicit calculation of the MOX fuel assembly and the peak cladding temperature there. Then they modeled the LEU fuel as the balance of the core.

Now what we have done -- and this is kind of getting ahead, but what we have done at the end of the day is we have established LOCA limits that will ensure that the peak cladding temperature will be in the LEU fuel per the LEU fuel analysis, which is the analysis of record there being Westinghouse's SS LOCA calculation for that fuel.

DR. BONACA: Okay.

MR. NESBIT: But I think that is going to come out a little bit.

DR. BONACA: All right.

MR. NESBIT: Back on slide 10 here, the stylized MOX/LEU comparison: We asked ourselves the question, okay, let's just change one thing in this model that we have now come up with that can do not just LEU fuel but LEU fuel or MOX fuel. Let's just change the fuel pellet characteristics and run a case with the same conditions and see what difference it makes.

As you can see, side by side LEU/MOX comparison comes out to be within 40 degrees in terms of peak cladding temperature. In terms of a LOCA calculation, this is essentially the same answer. Furthermore, I will add that this does not take credit for some of the things that you could take credit for in MOX base, like lower decay heat, the increased void reactivity, etcetera. We are just trying to get a calculation that is conservative and shows that we meet our limits.

DR. BONACA: Those values probably are reached during the blowdown, the heatup, the decay heat.

MR. NESBIT; Let's go to the next slide.

That will show it.

There is the comparison in terms of peak cladding temperature. As you can see, the peaks -- I mean the traces are a virtual overlay, and the peaks occur between 100 and 150 seconds.

CHAIRMAN POWERS: This temperature plot is tracking the peak temperature of fuel rods in the MOX assembly?

MR. NESBIT: Yes, and then running the same calculation and changing that MOX assembly only to make it LEU instead of MOX.

CHAIRMAN POWERS: And if I looked at which rod was having the peak temperature in the MOX, does would that change a lot or is it typically one rod that's running hot?

MR. NESBIT: Well, I guess the question you are getting to kind of gets to the power profile across the MOX fuel assembly. It is fairly flat. The effect of the radials only is to provide for a reasonably flat power profile.

CHAIRMAN POWERS: So it might change quite frequently, but it doesn't -- there is not a vast difference.

MR. NESBIT; Yes. It might hop around from rod to rod during the cycle, but it's not a big difference between the rods.

CHAIRMAN POWERS: It's not one rod in the first 20 seconds and another complete rod during the next 20 seconds.

MR. NESBIT: Jim, do you think that is an accurate characterization of the power profile?

MR. ELLER: We have moved from the analyses that determined the peaking limits to LOCA into the analysis that examined all the possible power distributions that could exist in a core and verified that all of them are lower than the limits that are calculated by the LOCA evaluation model.

In the first burn in this fuel assembly, the pin that has the smallest margin to the LOCA limit is probably going to be either in the first row or second row of the assembly throughout the entire cycle.

I just summarized in a verbal fashion thousands of cases that will be examined. So I can't say with absolute certainty that it won't sneak to the third row in on the assembly, but it is not jumping all over the place.

This assembly will have burnable poison fingers in every guide tube. So to have it jump next to a guide tube or something like that, that's probably not going to happen in this cycle.

MR. DUNN; If I might add, the LOCA model is constructed purposefully to not try and position the hot pin within the assembly, because of what you said about it: Do these pins possibly moving around here or there, the thousands of cases that Jim has to study. The construction of the methodology is so that we don't have to know where that pin is.

DR. RANSOM: That is just a hot rod calculation that's done.

MR. DUNN; It is a hot rod calculation with a generic simulation of the assembly around the hot rod.

MR. NESBIT: Once we had satisfied ourselves that there weren't any major MOX impacts out there that would surprise us, we left the little stylized comparison and went -- Framatome went to a series of calculation that would determine the actual limits for the fuels, and they performed what I am calling sensitivity studies on things like the steam generator design.

Catawba 1 and Catawba 2 have a different design steam generator. So it does have an impact on the accident response to see which one is worse; time-in-life; location of axial peak; and also looked at the end-of-cycle coolant temperature reduction that we typically go through at our plants in order to squeeze out as much power as we can from the core.

The result of all this work was a set of LOCA limits for the core that is a function of axial elevation and a function of burnup, and would ensure that the peak cladding temperatures would remain below the regulatory criterion of 2200, and also below the resident fuel analysis temperature.

The actual values are provided in a letter that we submitted to the NRC in response to additional information, but I don't go through them here.

Next slide, Other Criteria and Evaluations: We looked at all the criteria, not just peak cladding temperature, maximum cladding oxidation, hydrogen generation, coolable geometry, long-term cooling.

Of course, they all met their limits well within the regulatory limits that are established. One thing that I will note was the maximum flow area reduction due to ballooning was calculated to be 54 percent at the ruptured location, which is well below 90 percent, because that is an issue that has been raised in conjunction with the accuracy of the LOCA calculations.

We also looked at the impact of the MOX fuel on the co-resident fuel, the analysis of record for the Westinghouse fuel. Basically, the analysis of record is still valid for the Westinghouse fuel, due in large part to the fact that the overall assembly pressure drop is very close between these two fuels.

Next I want to talk about some of the issues that have been raised in conjunction with our application by intervenors, and we have talked about some already.

Fuel relocation during LOCA has been identified as a Generic Safety Issue for LEU fuel back in the 1980s, and it was dropped in 1998 by the NRC. In october of last year IRSN, as you have already heard, made a presentation on the PHEBUS tests that they want to conduct, and they made some mention of fuel relocation in connection with lost coolant accidents and in connection with mixed oxide fuel.

One of the things they mentioned was higher MOX power at end-of-cycle, and another thing they mentioned was the so called filling ratio of a balloon, and asked a question apparently, whether there is a potential MOX agglomerates effect, and they talked about the potential for bigger balloons or blockage with modern alloys -- I put in "like M5."

I'm not sure they specifically mentioned that, but I guess other alloys would be applicable here as well.

The intervenors have asserted that the NRC should deny our application, because we haven't adequately addressed these issues. I am going to address them now, as we have also in our filings with the Board.

MOX fuel relocation during LOCA: First of all, there was confusion about -- in the IRSN presentation about what is a LOCA effect and what is a severe accident effect.

IRSN did note that there is a fuel relocation at a lower temperature in the VERCORS RT2 test, which was done with mixed oxide fuel, than in a similar VERCORS RT1 test that was performed with LEU fuel. However, the salient point here is that that relocation occurred at temperatures that are consistent with severe accidents, elevated temperatures on the order of 4000 to 4700 degrees Fahrenheit, which is much higher than the fuel temperatures that are experienced during a design basis LOCA.

Second, we talked about the -- Jim talked about the MOX fuel lead assembly power during the lead assembly operation, his analysis of what the cycle is like.

CHAIRMAN POWERS: Well, let's just be clear. The RT tests are single pellet or small numbers of pellets.

MR. NESBIT: Right.

CHAIRMAN POWERS: They are not talking about the relocation that you would have in a LOCA.

MR. NESBIT: It's a different relocation. It is a severe accident phenomenon, as I understand it, prior to the onset of melting, and it is potentially --

CHAIRMAN POWERS: Liquefaction, I think, is the term that is used.

MR. NESBIT: But there was a confusion of the phenomena that were at play there. There is a real phenomena associated with LOCA called relocation, and it is a real phenomenon. We acknowledge that.

MOX fuel lead assembly power is going to be lower than the co-resident LEU fuel for the whole cycle than the peak co-resident LEU fuels. So the statement that apparently was made by IRSN may have been applicable to some other kind of operation with MOX fuel, but not to the operations that we are planning for the Catawba cycles.

There is also some confusion there,

because there was no transcript made of that meeting, and the Vu-graphs have certain information, but only a certain amount of information. So it is difficult sometimes to tell what was being meant by them.

There is no quantification made of this postulated MOX agglomerate effect or, for that matter, the LEU RIM effect, on the filling size. There is really, as far as we have been able to tell, not any substantial basis for saying that, even if relocation is a phenomenon of concern, that MOX is going to be significantly different, better or worse, than LEU fuel.

Finally, I will note that the blockage due to ballooning of M5 cladding was evaluated in our LOCA model, and we evaluated it at the worst case conditions, which is un-irradiated.

The assertions that the intervenors appear to be making is that, because the M5 properties change less with irradiation than Zircaloy, that that all of a sudden is a bad thing and that we should be penalized for that. In fact, that is one of the attractive things about the M5 alloy, but we do evaluate it specifically at the most limiting condition, and evaluate the effects of ballooning and found them to be acceptable.

CHAIRMAN POWERS: Can you remind me of what our database is on M5 ballooning?

MR. NESBIT: I am going to have to ask Burt to respond to that one.

MR. DUNN: The database for M5 ballooning comes mostly from the EDGAR series of experiments conducted in France, which are single pin tests, pressurized, done in a steam atmosphere.

There are a total of -- I'm going to guess. It is documented in -- The database that the model has been constructed on is documented in the M5 topical report. I am going to guess that there is on the order of about 150 data points included in there.

That facility has also been used to do other cladding alloys, and so you can construct a comparison to other experiments that have been done on Zirc-4, for example.

CHAIRMAN POWERS: If I recall the arguments that IRSN has made, they show that single pin tests can either bound the amount of ballooning or underestimate the amount of ballooning that you would have in an array, depending on the particular conditions that you have. Is that correct?

MR. NESBIT: I think that is their basis for saying they want to look at this at Phebus where they have multiple rods and they can assess that impact.

MR. DUNN: They have talked about bundle effects. I didn't know that they had talked about individual pin strains, but rather maybe bundle blockage effects.

CHAIRMAN POWERS: You are asking me to speak from memory, and maybe I shouldn't. But my recollection is of a slide that says when we compare bundle tests to single pin tests, we find that in some cases single pin tests will bound the amount of ballooning. In other cases, they underestimate the amount of ballooning.

MR. DUNN: Well, but the amount of strain is quite -- I'll use the word stochastic in terms of an individual test. It's all over the place. So I would find that kind of an illegitimate statement. I will say I wasn't at that meeting, and I was invited and did not attend when some of that stuff -- In retrospect, I should have been there.

CHAIRMAN POWERS: I wouldn't trust my memory on the exact statement. Mr. Lyman would like to have a note.

DR. LYMAN: No, just to clarify. I think the issue was whether ballooning occurs at the same height or at different heights in a bundle, and that would lead to different blockage effects. If they all occur at the same height, then you would have constraints on ballooning from different rods, but if they occur at different heights, then you might not have those constraints.

MR. NESBIT; I think our fundamental position on the ballooning issue is this. There's a lot of interesting things you could look at with respect to ballooning during a LOCA, and they have proposed to look at some.

Fundamentally, the ballooning issue per se is not a MOX issue. It is a LOCA issue. Now the intervenors have attempted to tie it to MOX by hypothesizing that there is a different impact on -- now we are going away from blockage; now we are talking about fuel relocation, different filling size, filling ratio, etcetera. It is a pretty tenuous connection. But fundamentally, our position is that the Appendix K LOCA analyses that we perform are conservative, and there's a lot of reasons why they are conservative.

You can always hypothesize that you want to go look at something else, but we meet the current licensing basis. We are confident the fuel will perform adequately, even under the design basis loss of coolant accident, and we think we have demonstrated that with our evaluation models.

I guess, philosophically, I also have a fundamental disagreement with the intervenors on the point of their apparent belief that you must have perfect certainty before you can execute a lead test assembly program, and I mentioned this before. But I think it is worth repeating.

Such a position would basically preclude many of the fuel innovations the industry has put forward over the past few decades.

CHAIRMAN POWERS: Well, in their defense I haven't heard them actually say that.

MR. NESBIT: But that is the logical conclusion that you come to from the arguments that they have advanced, in my opinion.

One more slide on LOCA for now. To summarize, we did perform the specific evaluations with the models, as modified to address MOX fuel.

The analysis results were fundamentally similar to uranium fuel, as shown by the stylized side-by-side comparison. We did sensitivity studies to bound the range of plant operating conditions and establish the peaking criteria that MOX fuel remains within 10 CFR 50.46.

Earlier today, you heard Ed Lyman mention that MOX is inferior to LEU for LOCA. That statement is not true. What we did, we did a stylized comparison between MOX and LEU that showed that the performance was virtually the same. In that comparison, we didn't take credit for MOX benefits.

So there is no way you can draw the conclusion that MOX is inferior to LEU out of that comparison.

So that would be, I guess, where I would propose to stop at this point.

CHAIRMAN POWERS: Any questions on this? Otherwise, we are going to break for lunch, and we will come back and Mr. Nesbit will be in the barrel again.

MR. NESBIT: But not for long.

MR. CARUSO: I would like to make one observation. I would like to ask people to sign in. We have sign-in sheets that were not available this morning. They are available on the table over here. If you are present, please go and sign in. That is the normal ACRS practice. I like to keep track of who is attending the meetings. Thank you.

Actually, it is a requirement, but not all

Advisory Committee yet. So please sign in.

CHAIRMAN POWERS: With that, I will recess us until 2:15.

(Whereupon, the foregoing matter went off the record at 1:15 p.m.)

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A F T E R N O O N   S E S S I O N

Time: 2:15 p.m.

CHAIRMAN POWERS: Let's come back into session. I had an announcement to make, but it will have to wait until more of the members are here. So we will continue on with our Mr. Nesbit.

MR. NESBIT: When we last left our heroes, we were talking about LOCA, and now we are moving to the outline of the presentation of non-LOCA evaluations. Next slide, please.

I am going to go quickly through these in the interest of time. This slide talks about some of the things, characteristics of our MOX fuel program that render it benign with respect to non-LOCA events.

Generally, non-LOCA design basis events are driven by the global core physics parameters like MTC, etcetera, system thermal-hydraulic response, the stored energy in the core, and the decay heat.

As we have noted before, the MOX fuel impact on these parameters is typical of the kind of -- four MOX fuel assemblies on these parameters is typical of the kind of variations we see from cycle-to-cycle. So generally our safety analyses use bounding parameters that envelope the impacts of the MOX fuel lead assemblies.

System thermal-hydraulics aren't affected by mixed oxide fuel. Four mixed oxide fuel assemblies have no appreciable impact on stored energy, and as we noted before, the decay heat is lower in the time of interest. So that is also bounding.

Some effects, however -- some events do have effects that require some more specific evaluation. I am going to add another thing, and that is that Duke, obviously, does the cycle specific evaluation of each reload to ensure that it is either within the safety envelope or does reanalyses to ensure that the safety criteria are met, and we are very familiar for doing these for mixed core situations, because we routinely have mixed core situations at our plant.

We ran Mark-BW fuel from Framatome for years and years. We transitioned to Westinghouse RFA, and we still have a few Mark-BW assemblies -- At least we recently had some Mark-BW assemblies in the McGuire-Catawba plants. I'm not sure if we still do have any at this point or not. So mixed core analyses, transition core type things are not new to us.

CHAIRMAN POWERS: Mr. Nesbit, may I interrupt you to make one little announcement. Our member from Tennessee asked for a correction. He indicated it was not the written word he had troubles with. It was that all the Yankees have such thick accents, it's the spoken word he was having troubles with.

MR. NESBIT: Well, I endorse Dr. Kress' opinion there.

CHAIRMAN POWERS: Excuse me. I just felt an obligation to make that clarification.

MR. KRESS: I'm glad you made that.

MR. NESBIT: Control rod misoperation/steam line break: The bottom lien here is that, basically, the limiting assemblies are under control rods. I am going to again rely on the committee to read these, but by not loading the MOX assemblies under the control rods, it really precludes these from being a concern for MOX.

Control rod ejection is the next overhead, and again this is made a lot better by not loading the mixed oxide fuel under control rods. However, there is, obviously, the potential for a control rod ejection accident to affect not just the assembly under the control rod but other assemblies in the vicinity; and given the fact that the Cabri tests have indicated -- have looked at MOX fuel and particularly

Cabri Rep Na-7 tests experience a failure at about 125 calories per gram of an unusually energetic nature.

We felt like it was appropriate to do some specific analyses here. What we have done is use the SIMULATE-3K MOX code which Jim Eller spoke about earlier to do a three-dimensional transient simulation of the design basis rod ejection.

What we showed was that, even with very conservative assumptions for things like rod worth, etcetera, the maximum calorie per gram that we would see in the MOX assembly is less than 50 calories per gram, and well below any level at which you would expect to see any adverse effects.

CHAIRMAN POWERS: Now as I remember, when they look at the Cabri tests for sodium coolant and try to apply them to light water reactor situations, they make a correction in the energy at which you get the fuel dispersal and what not, like that.

MR. NESBIT: There was a recent paper that I read by Ralph Meyer in NRC Research that I think tried to do that on kind of a global nature for all of the rod ejection tests that had been performed, and I think the conclusion from that paper was a curve of energy versus oxidation -- yes, oxidation -- that below which, once you had made all those corrections, you could be sure that you are okay.

I think the values -- of course, it was a curve. So there was not one value, but I think values were on the order of 60 to 70 calories per gram there, if I'm not mistaken.

CHAIRMAN POWERS: And that is not such a huge margin above your 50 here, is it?

MR. NESBIT: It's not a huge margin, but I will make a couple of points there. First of all, I think that I would have to characterize that value that Research recently published as a conservative evaluation, trying to bound --

CHAIRMAN POWERS: So it is one of those things that, if the value is 60 and you are 59.9, you are in good shape. Is that what you're saying?

MR. NESBIT: I guess at this point, not having studied it in a lot of detail, I guess I would say I tend to agree with that.

CHAIRMAN POWERS: Well, maybe we'll just chat with Dr. Meyer here and see what he has to say. Why don't you just come over here and sit down. We have a hot seat just for you.

DR. MEYER: I am Ralph Meyer from the Office of Research at NRC.

The recent report was a research

information letter, and the adjustments we made with our transient fuel rod code gave us an estimate of the bias in the test data due to the atypical test conditions.

The number that we would apply to the M5 cladding, which is very low corrosion, from that study would have been 80 calories per gram, and this was an enthalpy change. So you get to add another 16 to 18 calories per gram to bring it back into the ballpark that you are talking about.

So I think there is a comfortable margin there.

MR. NESBIT: We proposed an acceptance criterion provisional for the lead assemblies of 100 total. So that does seem to be fairly consistent.

The other thing I will mention is that again our evaluation was very conservative. If you look at realistic rod worths for the McGuire-Catawba cores as they are configured now, you don't even go critical with these transients. So you don't really have a transient.

CHAIRMAN POWERS: Thank you, Dr. Meyer.

MR. NESBIT: Next one, fuel assembly misloading: The bottom line here is that the administrative measures that protect against this for

LEU fuel are equally, if not more, useful for mixed oxide fuel. So there is no special characteristic of MOX that would make this any worse.

CHAIRMAN POWERS: In fact, what you say is that a misload is far more readily detectable in the case of MOX than it is in the case of LEU?

MR. NESBIT; Yes, for a couple of reasons. One of the reasons we do our startup physics testing and core flux maps is to detect an instance where we might have misloaded the core.

In the instance of MOX, the actual thermal flux in the location of the fuel assembly is markedly lower than the thermal flux would be if there was an LEU assembly there at the same power level. So if there is a -- If you did switch out a MOX and an LEU assembly, it ought to be readily apparent just from the flux map.

The other, of course, is that we are going to preferentially instrument the MOX assemblies, as Jim Eller mentioned. So that further increases the probability you could catch it at that point.

In summary for the non-LOCA accidents, we have -- Most of them clearly have no MOX fuel impact for MOX fuel lead assemblies simply by inspection, and we evaluated the ones that would potentially have more of a chance of having an impact and addressed them on an accident specific basis.

Furthermore, as a part of the reload design process we will go look at each one of these accidents for all of the fuel that is proposed to be in the core, and evaluate it specifically for that cycle.

So next it brings me to radiological consequences. We have discussed before that, when you do radionuclide inventory analyses, generally you get about the fission product inventories between MOX and LEU fuel, but there are differences, in particular with respect to -- One of the important dose -- important isotopes for accident calculations is iodine 131, and in a case where you look -- a bounding type case, it would be as much as nine percent higher in a MOX fuel assembly. It's kind of a function of whether you look at it -- what burnup you look at it, etcetera.

With that impact in mind, we looked at the possible effect on thyroid doses and also total effective dose equivalent.

As we have discussed earlier today, accidents involving numerous fuel assemblies should see no significant impact. We looked specifically at

LOCA, which affects 100 percent of the fuel in the core, rod ejection which affects 50 percent per our analysis assumptions, locked rotor, 11 percent.

What we showed is that, even addressing just -- assuming that the MOX assemblies, all of them, are in the failed population, the impact on the overall dose that you would see is negligible and still much less than the acceptance criteria.

I don't present those numbers here. They are in the license amendment request and in the associated RAI responses.

On the next slide we get to the more interesting dose analyses, I guess I would say. Those accidents that involve just one or a few assemblies will see a bigger impact on a -- because the MOX is either all or most of the population of failed assemblies.

For our plant there's two accidents of concern there. A fuel handling accident affects one, and the weir gate drop affects seven fuel assemblies. So we performed explicit calculations for these using the approved alternate source term methodology for Catawba and using MOX fuel specific radionuclide inventories -- I would add, a very bounding calculation of those -- and we did a sensitivity study on the gap fraction; because one of the key aspects of this analysis is how much fission gas is in the gap that is available for release when the cladding is breached.

What we did is increase the gap fraction by 50 percent for the halogens and the noble gases over the Reg Guide 1.183 values. As you might expect, when you account for more iodine to begin with and increase the gap fraction by 50 percent, you get, lo and behold, an increase of about 60 percent for the calculated doses, which sounds like a big increase, but in reality in an absolute sense it is not, and it is still within the regulatory limits.

The values for these I actually do present on the next overhead.

Let's go to the summary for the radiological consequences. There is a potential for dose impacts even from just four lead assemblies, and that comes from the radionuclide inventory differences, the fission gas release.

The greatest impact is those accidents that involve just a few assemblies. We did explicit analyses of those. We put in place what we think is a conservative treatment of the differences, and we did indeed show higher consequences but well within the limits.

So next we come to the environmental evaluation, a subset of which is going to be a discussion of severe accidents.

In most of the license amendment requests we submit, we don't do an environmental evaluation, because there is a categorical exclusion. But in this one we felt like we should, and we did.

We provided an assessment of the potential impacts of using four MOX fuel lead assemblies on the environment. We looked at normal operations, showed that there should be no impact on effluents, just a very slight impact on occupational dose. That impact would derive from the fact that the fresh MOX fuel assemblies are slightly hotter than a fresh LEU fuel assembly.

We looked at the accident situations, which are already addressed in the safety analysis section and in the radiological consequences. Next slide, please.

So that left us with the potential impact of beyond design basis accidents or severe accidents, and we did also address these in the environmental report.

We based our evaluation on work the

Department of Energy had done for the surplus plutonium disposition and environmental impact statement in which they looked at the impact of using a 40 percent MOX core at Mcguire and Catawba.

They looked at four different beyond design basis event sequences. They used MOX specific radionuclide inventories, and what we did is we took those results for 40 percent MOX score and just scaled it back to a lead assembly core based on two percent of the fuel instead of 40 percent being MOX.

There are some assumptions that go into this calculation. First of all, we've got the MOX assemblies separated by 90 degrees in the core, widely dispersed, and we would assume that they would have no impact on a progression of a severe accident or a different progression of a severe accident. The DOE assumed the same release fractions for LEU fuel as for MOX fuel.

The results after you do the scaling approaching is that you get a change in consequences with four MOX fuel lead assemblies in a range of between -0.2 percent and +0.7 percent, depending on what figure of marriage you are looking at, early fatalities, latent cancers, which scenario, etcetera.

There is another analysis that has been

done and documented on this done by Ed Lyman where he did something that is similar in some aspects, and the results of his evaluations were a change in consequences once you scaled it from 40 percent MOX down to four lead assemblies of up to 1.3 percent more consequences, and that is early containment failure scenario.

He also did a sensitivity study in which he increased the actinide release fractions, and in that he got as much as a 1.6 percent change. Next slide.

CHAIRMAN POWERS: Your essential point here is that, if you have four LTAs in a core, it doesn't really matter what they do, inventory alone dictates that they are not going to make a very big impact.

MR. NESBIT: That is right.

CHAIRMAN POWERS: When we think of the phenomenology itself, the key assumption is these LEU release fractions are indicative of MOX release fractions, and we are getting increasing evidence that that is just not the case. Doesn't impact your argument here, because you don't care -- I mean, you can take wild, crazy numbers here, and you would have the same conclusion roughly. I mean, your argument is a one percent change is an indetectable change because of the general level of uncertainty here.

MR. NESBIT: Right. In the next slide -- Let me go through that real briefly, because I think I am going to address that point a little bit in the next couple.

So the basis again for us is we are saying there is not a significant change in core melt probability. That is related to failures that are not functions of the fuel, like equipment availability, can you cool the core, have you changed the fundamental design of the plant, etcetera.

We don't think there is a significant change in severe accident progression, how the core melts, so to speak, and what happens at that point on. That is based on the fact that the physical characteristics are similar, and the accident progression, we think, is going to be driven by the LEU fuel.

The next slide, radionuclide inventories: Those are specifically addressed in the DOE and the Lyman studies. Then that leaves you with source term release fractions, and those have been addressed or looked at by an expert panel, of which Dr. Powers and Dr. Kress were a part, a couple of years ago.

I am not going to sit here and characterize to you what you all said, because I think you all could probably do a better job of it than me. But there was a notation of potential for differences. Whether those differences would ultimately be significant or not, I don't think the work has been done to determine that.

Generally, of the elicitations that were performed, I think the results of the elicitations for the MOX were similar to the results of the elicitations for high burnup LEU fuel. In other words, there was nothing that was off the world here.

I think another thing that the expert panel did was they sort of brought to bear more recent information in this area that had not -- information that had been developed since the NUREG 1465 source term came about, in the first place.

I note the VERCORS tests. There hasn't been a lot of MOX specific tests in the severe accident situation. An exception to that is the VERCORS tests. It is our understanding there were two tests performed at VERCORS with mixed oxide fuel, RT2 in an oxidizing environment and RT7 in a reducing environment.

Both of them had somewhat analogous LEU

tests to compare to, and the information that we were able to get out of the IRSN representatives and from a later paper that was done on the RT2 tests is that RT2 showed an earlier cesium release and a lower fuel relocation temperature than the analogous LEU fuel tests, and we got indications from IRSN that RT7 was not -- didn't show the same trend and, in fact, RT7 apparently was more similar to LEU and may have had, in fact, a higher fuel relocation temperature than LEU.

I understand that NRC has that information now. We don't, but in terms of the lead assembly project I consider it to be interesting but not necessarily relevant for the reasons that Dr. Powers mentioned earlier.

In summary on severe accidents, the severe accident behavior is going to be driven by the LEU fuel, and any impact from the MOX lead assemblies is going to be negligible when you compare this to the overall uncertainties in light water reactor severe accident behavior.

Here I guess I would like to cite some examples in regulatory space. NRC authorizes uprates all the time. Every time you authorize a power uprate, you are authorizing a change in severe accident consequences, just by the nature of what you are doing and the change in radionuclide inventories.

I think that is -- The kind of power uprates that have been authorized over the last years are much greater in terms of potential for impact --

CHAIRMAN POWERS: Oh, you are a dirty guy. You are hitting below the belt here.

MR. NESBIT: Sorry.

CHAIRMAN POWERS: Good point. Touche.

MR. NESBIT: Another example is changes that most reactors have undergone in the last 10 to 20 years in which we have increased our cycle length from annual cycles to 18-month cycles, in some cases 24-month cycles. That would have a similar impact on increasing radionuclide inventories and, therefore, affecting severe accident consequences. But overall, I would say that's been a very beneficial exercise for the industry and for the performance of the plants.

I think I have said probably enough on this subject right now. I guess I'll -- if there are anymore questions on the severe accident, we can talk about them.

CHAIRMAN POWERS: The only question that comes promptly to my mind is -- suffers from the charge of irrelevance. That is, in the course of discussing MOX fuel in that expert panel you mentioned, we became acquainted with some efforts underway by IRSN to define an equivalent for NUREG 1465 source term. I wouldn't say equivalent -- a similar source term.

They were drawing heavily on the VERCORS and its antecedent tests, and they were, of course, much more familiar with those tests than we are. They were coming up with substantially higher release fractions of some of the more refractory radionuclides than we had ever seen anybody coming up with. Quite frankly, I think -- Dr. Kress can correct me if he thinks differently -- we were a little bit surprised at some of the release fractions that they were coming up with.

Though I think again we run into this four LTAs in an ocean of LEU, it doesn't really matter what you take as the fission part of release fraction, it is an interesting thing.

MR. NESBIT: Right.

DR. KRESS: And it may have implications for the 40 percent.

CHAIRMAN POWERS: Well, clearly, it does. Now I think that RES is looking at that on its own, and we will anxiously await what they come up with on that, but that's just an interesting probably headache that you have to confront.

MR. NESBIT: Yes, and we see that headache out there. We don't think it is an insolvable problem. I'm very interested to see what RES comes up with, because they have access to the data, and we don't. But I want to see what their work has to say on this.

The other thing, I guess, I would add is that, if you look at McGuire and Catawba as plants, they are very far below the NRC safety goals, and if you were to change -- Even if you were to change severe accident consequences dramatically as a result of release fractions for 40 percent MOX core, which I don't think is where we are going to be at the end of the day, but if we were to end up there, they would still be well below the NRC safety goal.

CHAIRMAN POWERS: Well, some of us are willing to challenge you on that, whether they are below the safety goals or not, but that again suffers from irrelevance here.

MR. NESBIT: That is all I have to say about the severe accident issue. Unless anyone else has any questions on that one, I'll wrap up.

CHAIRMAN POWERS: Charge ahead. Oh,

Graham?

DR. LEITCH: Not specifically on that, but just before you wrap up, I was curious. Apparently, you are not using the new fuel storage vaults in the MOX fuel.

MR. NESBIT: That is correct.

MR. LEITCH; I guess I was a little confused why that was the case, and what is the impact of that?

MR. NESBIT: There is no reason why we can't from a technical perspective. From a security perspective, it is our intent, once we receive the fuel, to put it in the pool underwater as quickly as possible.

DR. LEITCH: I guess the basis of my question was what about receipt inspection?

MR. NESBIT: We are going to do that. We won't use the -- but we won't leave the fuel in the new fuel storage vaults as a consequence of that receipt inspection.

DR. LEITCH: So there is no compromise to your receipt inspection?

MR. NESBIT: No. We don't see any reason why there should be. I mean, we could still -- The new fuel storage vaults are quite capable of handling the MOX fuel from a criticality perspective. We could still use them to lower the fuel into and then retrieve them out of for new fuel inspection, if we chose to do it that way, but we wouldn't unhook it from the crane and leave it there.

DR. LEITCH; Okay, thanks. I understand.

MR. NESBIT: I guess I am ready to wrap up. Actually, that was a while ago, but we've got a slide here that I call "The Big Picture," and again this is just to try to bring us back to what this particular application really is, and I think I'm telling --

CHAIRMAN POWERS: Before you go into your first point, I will give you an anecdote. Professor Apostolakis on this committee once suggested to Shirley Jackson your first point, and she beat him roundly around the head and the ears.

MR. NESBIT: She is not here now.

CHAIRMAN POWERS: And she is not here anymore. As a caution about your first point.

DR. KRESS: Some words like "don't hand me that old saw."

CHAIRMAN POWERS: Now did I help your presentation?

MR. NESBIT: But I am going to make that

point. Sorry. The fact is there is plutonium in our cores. There's plutonium in all power reactor cores, and the amount that we are adding with four lead assemblies is not an inordinate amount compared to what is already there.

The other point, second sub-bullet down there which I found kind of interesting when I actually confirmed it with Jim and his nuclear analyses, is that at the end of our cycles, 18-month cycles with increasing burnup on our fuel, we are getting 50 percent of our core power from plutonium, and on a fuel assembly basis a lot more from some of the twice and thrice burned assemblies.

A similar MOX fuel lead assembly program was executed at Ginna in the early 1980s. It is not the first time this has been done here. At Ginna, which is a very small core with 121 assemblies, they were actually 3.3 percent of the core with their four MOX fuel assemblies.

The point that Mr. Blanpain made earlier, I'll just reiterate. This has been going on for years and years in Europe. There's currently more than 30 reactors in four countries using substantial quantities of mixed oxide fuel. What we are talking about doing is four assemblies out of 193, about two percent.

So it is easy to get caught up in the interesting nuances of differences between MOX and LEU and what it might or might not mean, but fundamentally what we think we have shown in our application is that we can use mixed oxide fuel safely and ensure that the health and safety of the public is protected.

CHAIRMAN POWERS: Do members have any questions? Vic?

DR. RANSOM: I was wondering if some of the intervenor's troubles with this are a fear that in time you are going to go to much higher loadings.

MR. NESBIT: Higher than 40 percent or higher than two percent?

DR. RANSOM: Well, higher than two percent. I guess the Europeans have gone to 40 percent. Right?

MR. NESBIT: I'm not going to speak for them. I'm sure they will speak for themselves. My suspicion is that their position is that any percent is too much.

DR. RANSOM: What are the plans in the U.S.? Is that known?

MR. NESBIT: The plans in the U.S. are, once the mixed oxide fuel fabrication facility is constructed and is operating at the Savannah River site, to start loading mixed oxide fuel at both the McGuire and Catawba reactors.

We would gradually build up to, our current plans are, about 40 percent MOX fuel core fractions. Of course, that is contingent on a successful license amendment request for batch use of mixed oxide fuel and any conditions or agreements that we reach with the NRC concerning core loading limits there.

DR. RANSOM: Is that level driven by a desire to burn up the excess plutonium or --

MR. NESBIT: Yes. It is a combination of a desire to do it in an expeditious manner and a desire to keep the plant characteristics reasonably close to their current characteristics with LEU fuel, because when you go to those higher core fractions, the statements that I made earlier about negligible changes to global physics parameters don't hold anymore.

Our preliminary looks indicate that the kind of changes we are talking about are still within our safety envelope and don't pose a problem, but that doesn't mean you can just keep pulling the string indefinitely.

CHAIRMAN POWERS: Any other questions? Well, thank you, Mr. Nesbit.

At this point, Bob Martin, I guess you are back on. There you are. I will remind the members that we do have this subject on the agenda for our May ACRS meeting, and that we need to think about what should be presented to the full Committee, what we hear.

We have heard from the applicant, and we are going to hear now from the staff on this subject, and it has to be some mix of that, and the question is what mix to have. We have scheduled two hours for this presentation, or I should say Dr. Bonaca has graciously consented to give us a full two hours on this subject.

Well, Bob, you've got a powerful team here. We are ready.

MR. MARTIN: I am Bob Martin. I know we have an agenda item here on the review process itself. With respect to the safety evaluation, the radiological safety evaluation I would note two aspects about it.

It is a review of the application information submitted by the licensee for an amendment to the operating license. To that extent, the review process, the staff's review process, was much like it is for any other license amendment application.

We reviewed the licensee's proposal against the requirements of the regulations for an operating license amendment. We requested the licensee to provide information on, in addition to the original application, a lot of the supplementary information on design basis accidents and transients and their consequences.

The nature of the MOX review leads us to bring staff members to the table today in what I will call two functional areas. One is reactor systems areas, and the other one is radiological dose consequences areas.

We have Ralph Meyer who has already spoken today -- he has been introduced -- Undine Shoop, Reactor Systems Branch, Mr. Ralph Landry, and then for dose consequences area Mr. Steve LaVie.

With that, I would note administratively, we have two slide packages here. Mrs. Shoop will speak first. So that is one package, and then we will go into the other presentations, and that one is in -- Mr. Landry's and Mr. LaVie's will be in the second package.

CHAIRMAN POWERS: My understanding is you are requesting a letter from the ACRS on this.

MR. MARTIN; Yes. Yes, I believe that is the understanding. And to your earlier point about what we would do for the full Committee, we are, of course, meeting with the licensee on Friday. We will get as much information as we can there.

Of course, if we can, we will solve the problem there. If we can't, we will figure out what the next step is. We will have our eye on how we can communicate to the Committee where we are as soon after that as we can.

CHAIRMAN POWERS: I think it is still worthwhile -- Even if the Friday meeting does not yield your most optimistic outcome, it's still worthwhile to communicate to the Committee just to get the rest of the Committee up to speed on this issue. Most of the members have seen nothing.

I mean, all they have heard are rumblings in the background on this particular issue. A more optimistic outcome is that we simply have to defer a letter until you give us the -- we resolve whatever issues exist, that's feasible to do.

MR. MARTIN: Okay.

CHAIRMAN POWERS: Because I think 99 percent of what we've discussed here is still applicable in any case.

MR. MARTIN; I think that is a good plan.

DR. RANSOM: Is the licensee's application -- is this for two percent or does it include increasing in the future to like 40 percent?

MR. MARTIN: Two percent? I think, with respect to that, you meant that the lead test assemblies would constitute two percent of the core, 2.1 percent or thereabouts. This application is just for the lead test assemblies.

I understand that plans, prospective plans, are for the licensee to submit an application for batch perhaps sometime later in 2005. We have not seen that yet. That is just oral information of their possible future plans.

MS. SHOOP: If I could add onto that -- is this on? There was a request in the MOX fuel design report put forth by Framatome to have that approved both for LTAs and for batch loading. The staff reviewed it and approved it for the LTAs, but has deferred any opinion on batch for a future batch application.

Thank you very much. My name is Undine Shoop, and I am here to lead off the reactor systems review of the LTA application.

Our purpose here today is to discuss -- Well, I don't know. I seem to have them out of order, because actually I was going to say next that we are going to be going over the thermal mechanical design, the data collection, the nuclear design, the non-LOCA transients, and then Ralph Landry will be finishing up by discussing the LOCA transient analysis.

Actually, our administrative assistant helped me with this. We knew that that would challenge him.

Okay. I would like to start off by talking about the thermal mechanical design, and I broke the presentation up this way. That way it is clear when I am transitioning from one subject to another.

First of all, the lead test assembly: As with all fuel designs, we are using the licensing framework in SRP Section 4.2. Even though SRP Section 4.2 does not say what type of fuel it is applicable for, most of the analysis that is in there is applicable, and we would want to know the results of all the analysis that is in the SRP Section 4.2 for the MOX assemblies, in addition to the uranium assemblies.

Now this is where it gets a little bit

tricky. Whereas, Duke put in the application for using the LTAs, the actual design of the LTAs was provided by Framatome in Topical Report BAW-10238, which is the MOX fuel design report. That is where the specific thermal mechanical fuel design analysis was presented to the staff.

So I guess I am going to start off with, backing up a little bit, the purpose of an LTA. The purpose of an LTA is to gather data on fuel performance. We base it on a production design, in this case the Advanced Mark-BW, and before we put an LTA into the core, we make sure it is pre-characterized.

CHAIRMAN POWERS: Explain a little bit to me. What do you mean by based on production design? This particular LTA is being produced in an ad hoc fashion in France. I mean, this is not a routine production, day in and day out, going on in France.

MS. SHOOP: Okay. What I mean by "based on a production design" is that the fuel design itself, the number of grids it uses, where they are located, the mixing vanes in the grids, the top nozzle, bottom nozzle, all of that is the same as a production fuel design.

CHAIRMAN POWERS: Oh, okay. So it is the

microscopic -- It's not what is inside the fuel rod. It's what is outside the fuel rod.

MS. SHOOP: Exactly. So that it will perform based on characteristics that we have about known fuel assembly design.

When we pre-characterize it, we examine it between the irradiation cycles and after it is discharged. The information that we get from the LTA is the basis for our improved fuel design and analytical models.

In this case, you notice that sometimes when we have an LTA, we go beyond what is approved, especially in the coding area, because we don't approve a code for a certain burnup until you have data. How do you get data? You can only get data if you test, and that is the purpose of the LTA.

I know. I was told not to show that slide, and I forgot that I had it in my slide package. So I apologize for that, because it is not in the handout.

CHAIRMAN POWERS: Let me ask you a question. Is there someplace a list of the data that you would like to acquire from these LTAs?

MS. SHOOP: Absolutely, and that is in the data collection portion of this presentation. So if you could just hold your questions for a moment, I'll be glad to go over all of that.

CHAIRMAN POWERS: All right. If you want to be that way. I am patient.

MS. SHOOP: Can we have the next slide, please. The objectives of SRP Section 4.2: It outlines four objectives for fuel criteria.

One is that the fuel system is not damaged as a result of normal operation and anticipated operational occurrences.

Fuel system damage is never so severe as to prevent control rod insertion when it is required.

The number of fuel rod failures is not underestimated for postulated accidents, and coolability is always maintained.

I derived those directly from the SRP.

So that is the basis of anytime we review a fuel assembly thermal mechanical design. Now since our design is contained in the MOX fuel design report, if you go to the next slide, to give you a better flavor for what the MOX fuel design report included, it included MOX design consideration which went over the MOX fuel characteristics.

It included a discussion on weapons grade plutonium, which is both the isotopics and the impurities. Then it had the full thermal mechanical fuel assembly analysis.

As part of the review of the thermal mechanical part of the proposed topical report, the staff did go down to Framatome's offices and actually look at the calculations to perform an audit. In particular -- I know that this has been brought up -- there's a couple of places in the original topical where they only provided data to 50 and not to 60. I think it happened in about two instances.

So those, in particular, were reviewed, and they provided -- If you look through the RAI responses, they did provide that information as supplemental information. So that that is all on the record of what the fuel behaves out to 60.

They also provided in this document the experience database, which was predominantly the European experience. They also described their lead assembly test program, what Dr. Powers is alluding to. That describes all their PIEs.

Now because we say it is based on the production design, the Advanced Mark-BW, there are a couple of changes, though, to accommodate MOX fuel, and these are actually all things that -- are just things that you need to do in order to accommodate the

MOX.

One is that they have a longer fuel rod, and that is to increase your plenum volume for the fission gas. They are going to use the European dish and chamfer design. The reason is because, with this being built over in Europe, the pellet press machines are set up for those design specifications. That is not something that is going to change the behavior of the fuel.

CHAIRMAN POWERS: Can you tell us what the differences are? A European dish is one that you can only eat with, with a for in the left hand, is it?

MS. SHOOP: Yes. I believe that it is -- and I can be corrected by my Framatome colleagues if this is wrong. But it is in the depth and the flexing of the dish an in the -- what do I want to call it? -- the angle of the chamfer.

CHAIRMAN POWERS: Yes. Are they bigger or smaller?

MS. SHOOP: Actually, it wouldn't really change the characteristics.

DR. BONACA: Somewhere I seem to have read -- I think in the SER, wherever -- that the European dish and chamfer design is capable of preventing the hourglass of the -- of shaping of the pellet.

MS. SHOOP: Actually, that is the reason that we have the dish in all of the pellets, both uranium and MOX, and they will all include a dish. I think the difference between the MOX and the uranium is because you have different equipment that is already set up, what ends up happening is you get that dish from the pellet pressing.

DR. BONACA: That seems to be -- At least, I read a claim that that was a better design for the purpose of reducing the hourglass effect, and maybe I misread it.

MS. SHOOP: That is actually the case, and that the dish will help prevent the hourglass.

DR. BONACA: Okay.

MS. SHOOP: The other change was the 95 percent theoretical density. The Advanced Mark fuel design is approved for a 96 percent theoretical density, but the European database on MOX is a 95 percent theoretical density, and in order to be consistent with the database, Framatome opted to use the 95 percent theoretical density for the MOX fuel.

Then the obvious difference is that it is going to use mixed oxide for fissile material, because one of the things to note is, when we approved 10238, it is approved with the condition that it is only for

MOX LTAs, and the Advanced Mark-BW was approved with a condition that it is only good for uranium oxide fuel.

DR. KRESS: So you still have a loop to close then, using the two together?

MS. SHOOP: You would never use the two together. If you used -- In a core assembly that contained both the uranium and the MOX, you would have the Advanced Mark-BW fuel design for the uranium fuel, and you would have the Mark-BW/MOX1 assembly design for the MOX.

DR. KRESS: So you would never use the two together.

MS. SHOOP: You would never -- Yes, you would never use the design of the Mark-BW/MOX1 for the uranium, or vice versa.

Okay. Mixed oxide fuel: Well, I know we have already gone over this. So this is kind of a repeat, but basically the use of the depleted uranium matrix with weapons grade plutonium.

The significance of the weapons grade plutonium is that you have fewer absorber isotopes, and you have an increased fissile isotope. That just changes some of your characteristics. But overall, it also has -- Because of those two characteristics of the weapons grade plutonium, you have a lower enrichment requirement to have a comparable reactivity, because what they are doing with this fuel design is they are making it reactivity equivalent. That's what they are equivalencing.

CHAIRMAN POWERS: The previous speaker made a point that all LEU fuel quickly becomes mixed oxide fuel, and what you are saying here is, well, that is only kind of true, that because of the isotopes that this is different. Is that -- Am I reading this correctly?

MS. SHOOP: That is correct. That has been the staff's position since the beginning of this fuel review, that this is a new fuel design, is a new fuel type, and it is different because of the isotopic mixture. Even reactor grade mixed oxide fuel is reprocessed uranium spent fuel.

If you would like to see the reactivity requirement, in Figure 3.5 of the mox fuel design report they have a nice little chart that actually shows that the reactivity of weapons grade is between the reactivity of LEU and reactor grade, and that is how they can say that the database is adequate.

Go on to gallium. Gallium, one of the

favorite topics: Why is it here? Well, it is used to stabilize the plutonium when it is in the state that you need it for the bombs. But why is it a problem? Well, it has the potential to migrate to the cladding and embrittle the cladding material.

Because of that, we remove it through polishing, which they have already discussed as being performed out at Los Alamos to get it down to the appropriate levels.

Why the staff is okay with the reduced levels: There are some Oak Ridge tests on gallium migration. That test is actually testing two different fuel compositions, one of which has been treated and now has a 1.33 ppm gallium level. The other one is untreated and has a 2.97 ppm gallium level.

They put this material into the cladding, which actually they used zirc for, because they did not have access to M5, and they put it into the advanced test reactor. They have the reports out to 40 gigawatt days, and so far no migration of the gallium has been seen from the fuel to the cladding.

The staff will receive the 50 gigawatt day report before the LTA gets to 50 gigawatts, and if there is any migration that is shown in that report, then the staff will review the issue, and it will be reviewed well in advance of the LTA reaching that burnup.

Because of the current results, we are confident that a 30 ppb limit, which is much lower than what has been tested in those Argonne tests, will be appropriate for the plutonium feed material, and it will be incorporated into the fuel specification.

DR. RANSOM: When you refer to polishing, is that a chemical process for removing the gallium?

MS. SHOOP: They call it an aqueous polishing, and because that is considered to be part of the fabrication, I would have to actually ask my colleagues from Framatome or from Los Alamos, if anyone is here, to discuss that.

MR. MEYER: George Meyer. The material -- it goes through an aqueous polishing process, which means it is dissolved and run through an ion exchange column, and that removes the impurities or reduces them to a very low level.

MS. SHOOP: Now I would like to move on to the data collection portion of this presentation.

The purposes of the data collection program is to be able to get neutronic data through the startup physics testing and fuel behavior data through the post irradiation exams or PIE. This information is needed to support batch loading. So this is information that they will need prior to doing a batch submittal.

The reason why you need this information is because it is a code check for the CASMO-4/SIMULATE-3MOX and for the COPERNIC codes.

On the neutronic front, Duke has made a commitment that two of the LTAs will be located in core locations that are directly measured by moveable in-core detectors for the first and second irradiation cycles.

That will provide operating data so that we can actually compare the actual measured data to what CASMO-SIMULATE is predicting, which will give us confidence that the CASMO-SIMULATE code is predicting the appropriate information.

CHAIRMAN POWERS: You are talking about your confidence. You are going to get some data. You are going to have a code calculation. There is going to be some discrepancy between the data and the code calculation.

MS. SHOOP: That always happens.

CHAIRMAN POWERS: At what point do you say -- I mean, how do you decide this code is okay or this code is not okay? I mean, how accurate do these data have to be?

MS. SHOOP: Well, they have already done some benchmarking against data with the code. So what I would expect is that the difference between the measured and the predicted for the in-reactor should fall within the range that the St. Laurent benchmark did.

CHAIRMAN POWERS: You said here is the discrepancy between the code and the data. That means the data have to be a certain level of precision, preferably accuracy, but I don't think you can pull that. I mean, what do you do to tell them, oh, yeah, this data will, in fact, give us that required accuracy? Is that part of your responsibility or is that just part of theirs?

MS. SHOOP: Okay. If I'm answering what I think you are asking, and please tell me if I am not answering exactly what you are asking, they will get the data, and they will be able to compare it to the code predictions.

When they do that, they can look at what the uncertainty is and all the other things that are in that data. They can then compare it to what the code was able to predict for the St. Laurent data, and also for the criticals that it was benchmarked against.

I would expect that it should be -- the uncertainty should be within the range of that data. That way, they have a good correlation between saying that the database that they have used to benchmark, which was reactor grade MOX fuel, is appropriate for weapons grade MOX fuel.

CHAIRMAN POWERS: I guess what I am driving at is suppose they came in and said we are going to get this LTA data, and it is going to have an uncertainty twice as big as the discrepancy between the code calculation and the St. Laurent data.

Would that be a basis for you saying, no, no, you're not going to put this LTA in this reactor; it is not worthwhile, because the data is -- If the uncertainty in the data is bigger than the discrepancy you are looking for, you are not going to be able to say anything.

MS. SHOOP: Absolutely. If the discrepancy was that large, I think the staff would have to start a dialogue with Duke so that we could resolve the issue. Does that answer your question?

DR. TRIAFOROS: Are you planning on reducing the -- or defy them and use them for refabrication now or in the long run?

MS. SHOOP: For the LTAs, we are not. If they come in with a batch application, and that's a big "if," because we don't have one, so that is speculative at this point -- we would then do staff confirmation studies.

CHAIRMAN POWERS: One of the things that we noticed, and maybe, Bill, you are the one to bring this up, is that in most cases in the SER, especially in the neutronics area, you reviewed and did not seem to do independent calculations of the neutronics. Is that a fair characterization?

MS. SHOOP: That's a fair characterization. We did a data review of what other people have said, and we used our own engineering judgment to confirm that we believe that that data is accurate.

If I could go on to the neutronic: Duke has committed to the NRC to continue using the start-up physics test plan that they already committed to using previously. If this testing plan is consistent with the ANS 19.6 standard on PWR start-up physics testing, what it entails is critical boron concentration, isothermal temperature coefficient, bank worth measurements, low power flux map, intermediate flux map, and the high power flux map.

Duke has also committed that, when they take these measurements, especially like the intermediate flux map, they will take it at approximately the same power every single time. That way, you can actually correlate what you get from one cycle to the other cycle.

Then the other testing that we have is our poolside post irradiation exam. First, you have the examinations that are performed between cycles, between the second and the second, the second and the third.

You would do the visual inspection of both the fuel assembly and fuel rods, fuel assembly growth, fuel rod growth and the fuel assembly bow, because you want to confirm that all of that is good before you put it back in the reactor.

Then after you discharge the assembly, which would be after the second cycle and after the third cycle, you would have -- you would test your grid width, your fuel rod oxide thickness, grid oxide thickness, RCCA drag force, guide thimble plug gauge, and the water channels, which is a test for the fuel rod bowing.

CHAIRMAN POWERS: Would you define the

acronym -- the initials RCCA?

MS. SHOOP: Rod control cluster assembly. It is basically making sure that your rods can get in, in the time that they are supposed to get in.

Then after the fuel has cooled in the spent fuel pool and it is at a level that you can send it off to a hot cell, hot cell will be performed.

To correct the record, in our discussions previously with Framatome and Duke and in the RIA letter of March 1st on the BAW-10238, we had discussed that we agreed that if the third cycle was actually -- if the LTAs were used for a third cycle, they would complete the hot cell PIE for that fuel assembly, so that it would be done.

What they will be testing in the hot cell is the rod puncture which is for the fission gas releasing composition, the metallography and ceramography, which they would use eight clad samples and eight fuel samples to look at the oxidation in the hydrides, and the structure of the plutonium agglomerates.

They will also do the cladding mechanical tests, which is looking for ductility. They would do burnup analysis which is to confirm core power density, and they will use gamma scanning to do that.

They will also look at the burnup distribution, and they will use two transverse fuel sections to compare that to the prediction.

CHAIRMAN POWERS: The metallography and the ceramography, will they -- what kind of magnifications will they go to?

MS. SHOOP: That was not provided. I believe that they will use what Oak Ridge is capable of doing.

CHAIRMAN POWERS: Oak Ridge is capable of a lot of things. In fact, it is one of the more magnificent laboratories in the United States, is my understanding without personal experience.

No, my question really is whether we get information on intragranular fission bubbles or not.

MS. SHOOP: Since what they are looking for is the structure of the plutonium agglomerates, in order to be able to see the structure you have to get down to a level that you would also see the bubbles.

CHAIRMAN POWERS: You would go down to the level that you would see intergranular bubbles, but would you really go down to the intragranular bubble?

MS. SHOOP: Probably not.

CHAIRMAN POWERS: It would surprise me a little bit if you went that deep. Too bad. That's where all the fun is.

MS. SHOOP: We will suggest it to them.

DR. LEITCH: Can we consider this -- the completion of this post irradiation examination a prerequisite for batch loading or haven't we crossed that bridge yet?

MS. SHOOP: We would consider that they need this data in order to be able to support a batch application, and we have that in our SE.

Now I would like to continue and go on to the nuclear design properties. As we have stated a lot of times, for the four LTAs with 189 other assemblies it is going to have an insignificant impact on your core-wide neutronic behavior.

CHAIRMAN POWERS: Is that true, even with these other LTAs in there?

MS. SHOOP: I would like to make the disclaimer that what we reviewed was the application that was provided, and the application that was provided said it would be four MOX LTAs in a RFA core. No mention of the other LTAs was made. Therefore, what we are providing you today is the staff's evaluation of what we had.

MR. MARTIN: Orally, that is what Duke has told us. That would be one of the agenda items, obviously, we will get into on Friday.

CHAIRMAN POWERS: I mean, this is -- It says -- I mean, it is a pretty bold statement. So I wondered if we generalize here. She is not being led down any primrose path here, try as I might.

MS. SHOOP: Okay. Now to go into the core design. Duke is using a checkerboard pattern, which means that your once burned is next year's new fuel. So you don't have new fuel base adjacent to new fuel.

What they have promised is that the LTAs will be in symmetric core locations, and that during the first cycle the LTAs will not be in rodded locations. Therefore, they are not taking away from the rod worth.

The LTAs will also not be limiting, but they will be in prototypical locations, because the last thing we would want to do is to have the LTAs be in a place where they don't see a lot of flux, and then find out that there's problems later. That's why we encourage prototypical locations.

CHAIRMAN POWERS: You are not limiting the fact that you could have problems later, because you are not putting them in rodded locations, and you are not putting them in lead locations. Yet in a full load, they could be in lead locations.

MS. SHOOP: The purpose of the LTA is to be able to compare it to be able to determine whether or not the European experience database is applicable to weapons grade. Where they are going to be located right now will give us that, and it would give us that confidence, and the European database does have MOX fuel in rodded locations.

Duke as part of their application performed some core sensitivity studies. They performed studies in an all-LEU core and then they performed a study that had all LEU with four MOX assemblies.

They used the CASMO4/SIMULATE3/MOX code suite in order to be able to perform these, and they investigated the important core parameters. In particular, you can see the key core-wide physics parameters which are the critical boron concentration, the control rod worths, the moderator effect, and the fuel temperature coefficient.

They actually told me to use slides, and it would have been better if they had told me just to use, you know, the PowerPoint presentation, but they didn't let me know that in advance. So you are going to have to kind of spread out your handout, because charts, slides 6, 7 and 8 are actually the results of these sensitivity studies.

What you notice in each case on these handouts is that they have the Delta. If you actually look at the Delta for all of these important core-wide parameters, they are not changing very much, because you are not going to see a significant effect from four LTAs in a core this large on a core-wide basis, and that's what those studies show.

Now if we could go to Slide 9, which actually you also need your tables for, you can see the assembly physics parameters. The important ones are the reduced delayed neutrons which is on slide 8. However, the LTA will not be rodded. So it will not significantly reduce the rod worth of any rods in the core.

It also has an increased void reactivity effect, as the Duke people had already discussed. It provides a larger negative reactivity insertion during the LOCA event. So that is actually a positive.

The prompt neutron lifetime is also slightly decreased, and that is on Slide 8, there again not significantly.

So that's why I came up with my conclusion that adding four LTAs to a core this large will have an insignificant impact on the core parameters.

So now I would like to go to non-LOCA transients. I would like to start off by saying that this was a deterministic licensing application. So it only addresses Chapter 15 transients. We did not address any severe accidents beyond design basis. They are all out of the realm of this licensing application.

To perform the LOCA transients, Duke used their normal reload process. Part of that process is to design the core and then test it for all the Chapter 15 accidents, and they confirmed that all the physics parameters fall within the reference values previously calculated.

If you look at Table 30-1 of the November 3rd RIA response from Duke, what you will see is in that table they actually put for all their Chapter 15 analysis, what they actually use, because this is deterministic. So they actually have the bounding worst case parameters in there when they calculate it even for LEU fuel.

So what they did was they looked, and they said, okay, well, this is what we already use, is our worst case; where does MOX fall? And they found that actually MOX fell always within the envelope of what they were already calculating. They did, however, do look a little bit more at the transients that are most affected by the physics.

Mr. Nesbit has already gone over these, but it is control rod ejection, rod cluster control assembly misoperation, the steam system piping failure, and the fuel assembly misloading.

I also came up with little sheets on every one of these. The core loading pattern for their control rod ejection will preclude significant impact of RIA. It is because the LTAs are in unrodded locations, and the LTAs are also not close to fuel assemblies having significant ejected control rod worth.

When they actually did the core-wide basis, they found the peak LEU enthalpy of 54 calories per gram. They found the peak MOX enthalpy of 30 calories per gram. The maximum broad worth was 412 pcm.

As you can tell, the MOX is well below anything that any test has shown as being problematic.

CHAIRMAN POWERS: Are these particular results -- you just reviewed these. You did not use your own codes to go calculate these enthalpy inputs?

MS. SHOOP: That is correct.

CHAIRMAN POWERS: There has been

controversy within the research program about the calculation of these enthalpy inputs with various codes.

MS. SHOOP: Yes.

CHAIRMAN POWERS: Would Research come up with the same numbers or roughly the same numbers, if they did these calculations?

MS. SHOOP: Actually, Research doesn't have the capability, which is why we weren't able to perform an audit of this calculation. The worst control rod ejection, the worst LEU enthalpy is actually found at end of cycle, and the NRC does not have a depletion capability right now. Working on it.

DR. MEYER: Could I comment on that?

CHAIRMAN POWERS: Please.

DR. MEYER: WE actually are using the PARKS code coupled with RELAP to analyze typical rod ejection accidents. So we haven't analyzed the Catawba, but we have done a rather substantial generic study and looked at a -- looked at the relation between the worth of the ejected control rod and the peak fuel enthalpy change that you could cause by that.

For -- I hope I can remember these numbers. For control rod worth, it is around $1.50.

You get peak enthalpy changes on the order of 40 calories per gram. Let me look at Harold and see if that's the right number. So they nodded yes.

So this is work we have done at Brookhaven. It is summarized in the recent Research Information Letter, and that is very consistent with these numbers.

CHAIRMAN POWERS: I guess, should I send a note off to our Research Review Committee that says Research needs some enhanced capabilities here?

MS. SHOOP: Actually, the Office of NRR has sent over a user need letter, and the Office of Research is working on getting that capability.

CHAIRMAN POWERS: Would you be kind enough to send us a copy of that user need letter?

MS. SHOOP: Sure.

CHAIRMAN POWERS: I mean, it is not a criticism. It's just, if we don't have some capabilities that we need, we ought to set about getting them.

MS. SHOOP: Yes. That is a paper that we wrote back in, I believe, the '99 time frame, and we identified all the needs that -- all the information that we would need in order to be able to effectively review a MOX batch application.

CHAIRMAN POWERS: Yes. I mean, that's when we are going to need it.

DR. MEYER; We are -- In the research program, we are in fact working on that right now, and we will be in the next year participating in a couple of international MOX benchmark calculations.

CHAIRMAN POWERS: Very good. Very good.

MS. SHOOP: If we can go on to the rod cluster control assembly misoperation accident: Because the MOX LTAs are in unrodded locations during the first cycle and in non-limiting locations, they will not significantly impact this accident.

The reactivity of the MOX LTAs and the control rod worth for any rodded LTA during the second and third cycles will also be below the limiting values. That is because the reactivity of the MOX will decrease to such a level that it will not limit the accident.

For the steam system piping failure, the accident is performed with the most reactive rod stuck out. The LTAs are unrodded. So they are not going to impact the most reactive rods' worth. Duke has a criteria that for this accident they incur no loss of DNB margin. So there will be no fuel failure, including in the MOX fuel assemblies.

CHAIRMAN POWERS: So their DNB margin is always at least as great as it is in the LEU?

MS. SHOOP: Yes.

CHAIRMAN POWERS: And there is no -- it is always at least that big?

MS. SHOOP: Their criteria for this accident is that you have no loss of DNB margin.

CHAIRMAN POWERS: Tough people. Tough guys.

MS. SHOOP: And my last slide is a fuel assembly misloading accident. The administrative measures that Duke already has in place are equally effective for MOX as what they are for uranium fuel. In addition to that, the core distribution measurements -- When you look at the MOX fuel and you look at the LEU, when you actually run the in-cores, you are actually going to be able to detect if a MOX is misloaded, because the reactivity is -- or not the reactivity, but the parameters are different enough that you would be able to readily detect it.

CHAIRMAN POWERS: That is the claim that is made.

MS. SHOOP: Yes.

CHAIRMAN POWERS: And pretty good arguments were made in that. Did you look at that in any detail? You said, okay, here's the data I am going to have, here is the uncertainty I am going to have in that data, here is the noise I am going to have. Could I, Undine, looking at this come to that conclusion?

MS. SHOOP: We did not actually look at that data, because that data has not been generated.

CHAIRMAN POWERS: No, no. But I mean the hypothetical data.

MS. SHOOP: I believe that, based on the parameters that they will be able to get out from the in-cores that, yes, if we went down and did an audit after they loaded the core and ran the in-cores, that we would be able to detect that as well.

CHAIRMAN POWERS: And you are going to ask them to misload a core so that you can do that?

MS. SHOOP: Sorry, I'm not allowed to make that request.

Now I'd like to turn it over to Ralph Landry to go over the LOCA transient.

MR. MARTIN: We would go to the second package of slides and just flip past the first two, and you will begin with Mr. Landry's presentation.

MR. LANDRY: I'm waiting for Vic to get comfortable.

CHAIRMAN POWERS: No, don't wait for him to get comfortable. We don't let our members get comfortable. They are supposed to be on the edge of their seats, anxious to pounce at the slightest misstatement.

MR. LANDRY: My name is Ralph Landry -- I hope that is not a misstatement -- from Reactor Systems Branch in NRR. Today I would like to talk about the review that we performed of the Catawba MOX LTA LOCA.

To again give standard disclaimer number one, this review is based on the understanding that the core is going to be Westinghouse RFA fuel with four MOX LTAs inserted in the core. That was the analysis which we reviewed for the LOCA determination.

The discussion that I am going to present covers a couple of areas with regard to LOCA. We have to look at the analysis of record, the LOCA pertaining particularly to Catawba, the effect that the resident fuel has and the effect that the MOX LTA will have on that analysis of record.

When we look at the MOX LTA, we also want to look at LOCA effects specific to those bundles. You have heard some information already today, both from Ed Lyman and from Steve Nesbit, regarding LOCA calculations which have been performed, and I would like to go over what we reviewed on the staff and how we arrived at our conclusions.

The analysis of record for Catawba is a Westinghouse WCOBRA/TRAC/REALISTIC LOCA analysis. The resident fuel assumed in that analysis was all Westinghouse robust fuel assemblies.

There are going to be loaded into the core four MOX LTAs which are Framatome ANP -- or I guess we call it AREVA now -- Mark-BW/MOX1 or, as you heard this morning, hydraulically identical Advanced MARK-BW assembly design.

The analysis of record covers -- was done to cover the RFA fuel and the Mark-BW fuel which was resident in the core at the time that a transition was being performed from the Framatome Mark-BW fuel to Westinghouse RFA fuel. When Westinghouse performed their analysis of record, they performed a sensitivity study, one of which used a surrogate or a proxy assembly design with a pressure drop that was representative of the Mark-BW pressure drop.

That provided a sensitivity for the analysis of record, the licensing analysis, which said it would indeed cover the resident fuel, the Mark-BW fuel and the RFA fuel.

The Mark-BW/MOX1 assembly, or the Advanced Mark-BW assembly, has a pressure drop that is much closer to the Westinghouse RFA assembly's pressure drop than it is to the Mark-BW fuel assembly's pressure drop.

CHAIRMAN POWERS: I'm going to have to work on this one.

DR. KRESS: You are going to have to explain to me how you use analysis surrogate to give you the same pressure drop. Do you put a fake orifice on the end or do you distribute it all along by changing the Fl/D, the hot rod diameter, or what?

MR. LANDRY: The fuel vendors are very sensitive to the exact nature of the mixing vanes, etcetera, in their fuel assemblies, and they are loathe to share with one another a great deal of detail.

DR. KRESS: And I could envision a loss coefficient for each one of them.

MR. LANDRY: Right. Now what I'm getting to is when a core has only one fuel assembly in it, it is very easy to do a LOCA analysis, because you know the pressure drop, you know the flow characteristics, the hydraulic characteristics of every assembly in that core.

When a core contains fuel all from one vendor, you can perform a LOCA analysis, because you know the hydraulic characteristics of every assembly in that core very precisely. When a core does not contain fuel from only one vendor and the other vendors don't care to share details with one another, you have to find a way to represent the other person's fuel.

The way in which that is done is to do your hot rod calculation before the rest of the core, determine what is an average pressure drop, an average flow condition for the rest of the fuel where you have taken the other vendor's fuel, assumed a hydraulic condition, then imposed that on your own fuel so that you end up with an aggregate hydraulic condition for the remainder of the core.

This is the only way you can really do a calculation when you don't have the exact data on the other vendor's fuel.

What was done with the resident fuel by Westinghouse when they did the licensing calculation was to do a calculation for all of the RFA fuel, and then make assumptions about the pressure drop for the transitional fuel that was still in the core. That imposed an average pressure drop on the remainder of the core.

Now what Duke is coming in and saying is that the Advanced Mark-BW fuel characteristics lie between the characteristics of the RFA fuel and the Mark-BW fuel which was used in that licensing calculation of record.

So the calculation of record now is encompassing the effect of having an RFA core and now having four MOX assemblies in the core. Now that's the first piece of the puzzle.

DR. RANSOM: Are these multi-dimensional calculations like with COBRA/TRAC, so you have multiple passages through the core or are you talking about --

MR. LANDRY: Well, I am speaking in general terms of how with a 1-D code -- well, with a 3-D code you could take into account three-dimensional flow characteristics, but that information would definitely not be shared from vendor to vendor.

DR. RANSOM: So the calculations you are talking about are all 1-D representations of the core, possibly parallel channels of the hot rod?

MR. LANDRY: Yes, sir.

DR. TRIAFOROS: So, Ralph, at some point in time you anticipated that Catawba would be using --

or the study anticipated they would be using Mark-BW fuel assemblies for the whole core? I'm trying to find the rationale of having, if I understood it correctly, a study, a base study that has all, if I understood it correctly, Mark-BW fuel. What is the rationale, because we know that the fuel probably is not all Mark-BW?

MR. LANDRY: The understanding that we have -- and Duke may want to correct this, if I state this incorrectly. When the current LOCA analysis of record was performed, Duke was transitioning Catawba between Framatome fuel and all Westinghouse RFA fuel.

So there was at that point some RFA fuel in the core. That is why the analysis of record was performed for Westinghouse RFA fuel, to which Catawba was transitioning, but with a sensitivity study for the effect of the Mark-BW fuel which was already present in the core. Is that clear?

They are going from Mark-BW to RFA, but now they are going from RFA to include Advanced Mark-BW. So the study which was performed in reality going from Mark-BW to RFA encompasses the effect of going from RFA to RFA plus Advanced Mark-BW.

There was a time in the old days, the good old days, when cores were homogeneous in nature or manufacture, and we didn't have these issues to deal with. Now that we have heterogeneous designs in the core, we have to deal with how do you --

CHAIRMAN POWERS: It's a little bit like your telephone bill.

MR. LANDRY: How do you have an analysis that encompasses all the different types of fuel that you have in your core?

DR. RANSOM: Ralph, your second bullet, the Mark-BW/MOX1 -- that's a different geometry of the Mark-BW fuel?

MR. LANDRY: Yes. That is the fuel that was discussed this morning by Framatome, which is also called the Advanced Mark-BW assembly design. Framatome explained this morning that what they are doing is taking the Advanced Mark-BW assembly and putting the MOX pellets into that assembly. It is hydraulically identical to the Advanced Mark-BW assembly, but we are calling it Mark-BW/MOX1 to not confuse that issue any further.

The issue is perfectly clear right now.

DR. TRIAFOROS: Now you said, correctly so, that a new fuel vendor doesn't know what the pressure drop to the previous assembly's is, but you do, however, because you have access to that information. So have you done verification that all this squares away ultimately, all these assumptions, and is it a proper approach?

MR LANDRY: Yes. This has been done repeatedly for a number of plants, and we accept this approach.

DR. TRIAFOROS: In the SER there is the statement. There are four differences between the Advanced Mark-BW and Mark-BW/MOX1 fuel designs, and it enumerates what they are. I'm not quite sure if I understood you correctly. I understood that the Mark-BW and Mark-BW and MOX1 -- they are identical, which is not the case based on what we are reading here.

MR. LANDRY: Well, I am basing my statement on what Framatome has said, what they said this morning. I did not review that part of this submittal.

MR. NESBIT: Can I offer a clarification?

CHAIRMAN POWERS: Please.

MR. NESBIT: First of all, I apologize for the confusion engendered by these various fuel assembly names. But let me review three that we are talking about.

Mark-BW is the fuel that Duke began loading in its reactors in the late 1980s. We transitioned from Westinghouse fuel to the Mark-BW fuel assembly design. We still have a few Mark-BW assemblies around, a lot of them in the spent fuel pool.

The difference between that design and what we refer to as the Advanced Mark-BW design, which is what the North Anna LTAs are and the batches that's going into North Anna, is primarily the material of the cladding and the presence of intermediate flow mixing grids. There's other differences, because fuel assembly designs evolve, but that is the big -- what I'll say the big deal for the purpose of what we are talking about now.

The MOX1 assembly design, the Advanced Mark-BW/MOX1 that we are talking about using for the MOX fuel, is structurally the same as the Advanced Mark-Bw that's going in at North Anna, with the exception of the fuel rod length.

The pellet material is different. Obviously, it is MOX. But that is the evolution there, if you will.

DR. BONACA: Also if I remember, the springs are different, aren't they?

MR. NESBIT: The springs are different?

DR. TRIAFOROS: Springs. I thought that

the assembly shown here has lip springs on the nozzle.

MR. NESBIT: As far as the current Framatome products, the Advanced Mark-BW and the Mark-BW/MOX1 that will be the MOX lead assemblies, the spring design is the same, I believe. I'm looking at Framatome, but they are behind the pillar here.

DR. BONACA: Same as the old -- the original? Okay. Okay.

MR. NESBIT: And just to either further clarify or make it worse, I'm going to say something else. I probably should sit down.

The Westinghouse RFA design, which is the co-resident fuel, is very similar in terms of overall pressure drop to the MOX lead assembly design. We stated in our application it is within four percent.

The older Framatome design, the Mark-BW, plain old Mark-BW, is different, because it doesn't have the intermediate flow mixing grids.

MS. SHOOP: Actually, if I could specifically address your comment, I see in here where you are talking about the four differences on page 4 of the SER. Those four differences are the four differences that I had on my slide 6 from the presentation.

What Ralph is saying is that thermal

hydraulically the Advanced Mark-BW fuel design and the Mark-BW/MOX1 fuel design are the same, because most of these differences were just slight differences, and most of these are actually to the pellet.

DR. TRIAFOROS: It talks also about the difference in the dish and chamfer design.

MS. SHOOP: And those are both pellet parameters. That will not change the thermal hydraulics.

DR. TRIAFOROS: Yes, you are absolutely right. Thank you.

MR. LANDRY: That deals with the analysis of record, the licensing basis analysis.

Now you heard this morning from Framatome and from Duke a discussion of a LOCA analysis which was performed by Framatome, or AREVA, for the MOX LTA. That analysis used the Framatome ANP Appendix K code, RELAP5/MOD2-B&W, which is an approved Appendix K model.

I've got to keep this straight. We are now talking about a REALISTIC LOCA which is the analysis of record, and we are now talking about an Appendix K calculation which is the calculation for the LTAs.

That approved modeling also includes the

M5 cladding properties.

CHAIRMAN POWERS: And can you give us a thumbnail sketch of the cladding properties that are approved that are used for this calculation?

MR. LANDRY: No. All I deal with is the property tables.

CHAIRMAN POWERS: So we need to go look at the M5 SER.

MR. LANDRY: Right.

CHAIRMAN POWERS: Thanks, Ralph.

MR. LANDRY: Well, you need to have something to do tonight, Dana.

CHAIRMAN POWERS: That's right. Otherwise, I would be wandering the streets. Right?

MR. LANDRY: You would get in trouble. I'm doing this for your own good.

One of the things that we questioned, because they were going to MOX, they were using plutonium startup instead of LEU, was the decay heat model itself.

You heard some discussion this morning about the decay heat model. The model that has been used by Framatome for this calculation is the Framatome decay heat curve, which is approximately 1.2 times the 1994 ANS curve, which produces a majority of its energy from the fission of plutonium for highly burned fuel.

Now this curve also just happens to encompass 1.2 times the 1971 decay heat curve. If I can have the next figure -- I don't know if you are going to be able to see that.

CHAIRMAN POWERS: We got it here. We can see it.

MR. LANDRY: The upper curve is the Framatome decay heat curve. The lowest curve is the 1994 curve, and you see one in between that transitions back and forth between the two. That is the 1971 decay heat curve times 1.2.

So we looked at this and said, okay, for the LTA calculation -- this is an Appendix K calculation anyway -- that Framatome curve is definitely conservative. It bounds the '94 curve by 1.2. It bounds 1.2 times the '71 curve, and we agree that for this purpose it should bound any decay heat effects we see from a loading of plutonium.

In making that decision, I spent some time one day with Virgil Schrock and talked with Norm Lauben, our decay heat experts, and was assured that, for the purpose of the large break LOCA, that curve is adequate. It is going to bound the effect of plutonium.

The ANS Subcommittee is looking at, and I believe met this past January and discussed, the effect of plutonium decay heat curve. So at this point in time, this curve is a reasonably conservative curve to use. It meets the requirements of Appendix K. It is going to bound the effect of plutonium.

So we agree that, yes, indeed, they have done acceptable analysis. Let me have the next one.

Now this morning you heard comparison information being given of the MOX LTA predicted peak clad temperature being 2018 degrees and the LEU peak clad temperature being predicted at 1981 degrees.

As was stated, that is a calculation based on using the Framatome Appendix K model to calculate the LTA and then to substitute the properties of UO_2 in place of the plutonium to calculate an LEU number.

The licensing calculation of record states that the limiting case PCT is 2056 degrees Fahrenheit, meaning that the PCT now for the MOX using an Appendix K calculation is 38 degrees lower.

Sometimes when we look at Appendix K versus realistic, we say there should be a such and such a difference between the two. Well, we have to remember that we are looking at an Appendix K calculation for an assembly in a non-limiting location. We are now comparing that with a PCT for the PCT of the highest value for the 95th percentile realistic calculation.

Now the maximum LOCA oxidation predicted for the MOX LTA is 4.5 percent versus 10 percent for the resident fuel from the limiting case. As I said, the MOX LTA placement is in a non-limiting location. The next one, please.

The conclusion of the staff is that the MOX LTAs will comply with the requirements of 10 CFR 50.46 when inserted into a core of Westinghouse RFA LEU fuel.

CHAIRMAN POWERS: Any questions on this analysis? You're going to get away Scott-free?

DR. KRESS: Well, I'll ask him a question.

CHAIRMAN POWERS: All right.

MR. LANDRY: Tom has never let me move away from the table -- never.

DR. KRESS: It is my job. I was just mulling over how do you make a correction of peak clad temperature, which is a transient that involves heat transfer coefficients and specific heats and thermal conductivities and stored energies? What is it that goes into making a correction to an LEU calculation to get this difference in 38 degrees? Just exactly how did they arrive at that? I don't understand what corrections.

MR. LANDRY: I am not talking about corrections with this. What I was simply comparing was the predicted PCT for the MOX LTA when calculated by an Appendix K model in a non-limiting location versus a PCT predicted for the RFA assembly by its licensing basis calculation.

DR. KRESS: Oh.

MR. LANDRY: I was simply -- This morning, Tom, you were hearing the LEU versus LTA in the same location by Appendix K, and I am looking at the licensing limit at this point for Catawba Unit 1 is 2056 degrees Fahrenheit.

DR. KRESS: Thank you.

DR. RANSOM: You didn't do any independent calculations, I guess, for a LOCA transient?

MR. LANDRY: No, we did not. We do have the RELAP5/MOD2 B&W input model for Catawba. That has been supplied to us, but we have not attempted to run it. That is with the B&W modified version of RELAP5/MOD2 which meets full Appendix K requirements.

We have not attempted to convert that deck into a RELAP5/MOD3 form and try to run it yet. We may--

DR. RANSOM: Do you have plans to do that?

MR. LANDRY: We may when we start discussing batch loading at some point in the future. But, of course, we would have to determine what changes we were going to make to the decay heat model and so on to perform the calculation. But we have the deck --

DR. RANSOM: You plan on making Realistic calculations, I guess, right now?

MR. LANDRY: Right. But we have the deck, and we can run it, but we have to do some significant conversions from the B&W version of RELAP5 to the version that we have.

CHAIRMAN POWERS: Any other questions? Then I am going to recess us until 20 after the hour.

(Whereupon, the foregoing matter went off the record at 3:59 p.m. and went back on the record at 4:19 p.m.)

CHAIRMAN POWERS: Ralph, you left us feeling inadequate. We hadn't interrogated you close enough. So during the recess we got together and decided a few other questions, so that you felt fulfilled for the day.

MR. LANDRY: Well, I'm glad to hear that, Dana, because I gave the answers then, too. I was at the other end of the hall.

CHAIRMAN POWERS: Unfortunately, they didn't quite make the record. So we are going to have to have you repeat them.

The question came up. We have discussed throughout the day a little bit about this relocation during the LOCA. You didn't discuss that issue at all. I guess two questions came to mind.

One is that, gee, this used to be a GSI. How come it is not anymore, and if you knew why the decision had been dropped.

The second is: Do you find anything about the MOX fuel that would lead you think that any fuel relocation during a LOCA would be different than for LEU?

MR. LANDRY: First, I am not an expert on the fuel. Ralph Meyer is. I don't know why it was dropped. I defer those questions to him. But I would say at this point that relocation is not considered in Appendix K.

So this was an Appendix K calculation which was performed for the MOX LTAs, and since it is not required and not a part of Appendix K, one would not expect to see it there.

CHAIRMAN POWERS: I think we understand that. We just asked if you had given it any thought.

MR. LANDRY: Not at this stage, because of the nature of these calculations. It is a part of some vendors' models for Realistic LOCA, not all, though. So if this was a complete core Realistic model, then we would have to see how it was being accounted for. I would refer the rest of the comments over to Ralph Meyer.

DR. MEYER: I think that is essentially the right answer, since --

CHAIRMAN POWERS: His is a legal answer. The question we were asking that would go to you is a phenomenological question. Do we see anything about the fracturing during the operation of MOX fuel that would suggest to us that it is different than the fracturing of LEU fuel?

DR. MEYER: I can't answer the question, because I don't think that we have seen any MOX fuels being exposed to those conditions.

CHAIRMAN POWERS: These would be just normal operational conditions.

DR. MEYER: Yes. I don't -- I really don't think that is going to do it. I mean, we can look at cross-sections of the microstructures of stuff that's been through normal operation, but it seems to come unglued when you balloon the rod.

The thing that -- I would like to go back to Ralph's answer, though, because it is not a bad answer. Since Appendix K was formulated in 1973, we have recognized some really conservative features and some non-conservative features.

This is one of the non-conservative features that has been recognized and, in fact, one that has been mentioned many times and forms the basis, in fact, for resisting any changes to Appendix K, because you don't want to just cherrypick and take out the decay heat or the Baker-Just correlations that are giving you the known conservative margins, which are rather substantial.

So I think that you have an offsetting situation where it isn't well quantified. It is under study, and that is probably the best we can do right now. You can make estimates using packing fractions from rod studies and things like that, which have been done, and they are in the order of magnitude of the overconservatisms in some of these other features like Baker-Just which gives you big temperature differences.

CHAIRMAN POWERS: You say it is under

study. In what context is it under study?

DR. MEYER: Well, we have two experimental programs which have fuel relocation as major objectives. One of them is in the Halden reactor, and one is in our program up at Argonne.

So at Argonne we are testing rods under out of pile conditions with electrical furnace. It is a radiant heating furnace. The heat conditions aren't exactly right. So that gives you an incentive to go in-reactor and do some checking.

So these have been closely coordinated with four tests that are planned in the Halden reactor. I think they are called EFA-650. These will be about as close as we can ever come to a situation where you have ballooned and rupture and heat up and look for the relocation.

The interesting thing about those tests is the relocation has to come at a very specific time or it just doesn't matter. It has to come before quench. So that is not a very big window, and all of the rattling that goes on during quench, which might be the cause of some observations that have been seen, or the handling that takes place afterwards before you get it to a hot zone, might be responsible for the relocation.

I am not suggesting that this is a non-effect, but what I am suggesting is that it is tricky. At Halden, for example, we have two opportunities to observe this. One is that they are installing some fast responding neutron detectors through four axial locations in the region where the balloon is expected, to see if you can detect any change in the neutron flux right after the rupture occurs.

The other thing is that we will look very carefully at the balloon section to see if it is oxidized more than you would expect it to be, based on an analysis that did not assume relocation.

You know, if you can't detect that, then it probably doesn't matter. So I would say that we are on the verge of, first of all, trying to find out if this effect is real, if it has an impact, and now what you are talking about, would it be any different for UO_2 and MOX?

That is just a little hard to imagine. I mean, perhaps the packing fractions could be different if there is a different distribution of particle sizes, but this has got to be a second order effect, I would think.

CHAIRMAN POWERS: Thank you.

DR. BONACA; There was another question

that was raised this morning regarding the effect of surface treatment on embrittlement of Zircaloy, niobium alloy.

DR. MEYER: This is a completely separate question. This has to do with embrittlement, and the embrittlement phenomena don't -- I don't think they have any connection to what is inside the fuel. It could be MOX. It could be UO_2. It could be anything.

The embrittling stuff comes from steam on the outside. You oxidize. You absorb hydrogen, and you go through a phase change. You have some dimensional changes. You have oxygen diffusion. You have hydrogen absorption, hydrogen precipitation into hydrides, and then you cool down, which gives this all a chance to settle in an embrittled fashion.

The polishing of the surface is just one of three or four variables that affect this. It is a very fascinating and somewhat complicated situation, but the niobium alloy is different than the tin alloy.

So there was the question of why did this Russian alloy behave so differently from the French alloy. You know, we have uncovered, I think, the important reasons without putting too fine a point on it, and there is no -- Knowing what we do now after rather intensive study in the last 18 months on this, I don't think there is any reason to suspect a problem with the M5 cladding.

CHAIRMAN POWERS: At that point, I think we can turn to Steve. You're on.

MR. LaVIE: Well, thank you. My name is Stephen LaVie. I am with the Probabilistic Safety Assessment Branch. As I was waiting to get started here, I was reflecting on whether it was an advantage to be last on the schedule, and concluded that most of the people have already said a lot of my presentation. So that is an advantage -- or a disadvantage perhaps. Then perhaps you guys are all kind of tired out and have had all your questions already answered.

CHAIRMAN POWERS: No, we get even crankier as the day goes on. And you are not even near last on this schedule. This sucker goes on until midnight, I think.

MR. LaVIE: Okay. I am going to discuss the review of the design basis accident radiological consequences evaluation.

Normally, the staff does not assess the impact of LTAs on prior analyzed doses. This is generally because the pellets are not different isotopically.

There is no reason to assume there will be an impact on the dose concentrations. But there are indications on this review that MOX could increase the radiological consequences and, in fact, the licensee specified that in its submittal. So this forces the NRC's review.

A review focused on the impact of the four MOX LTAs on the previously analyzed radiological consequences of design basis accidents. My review did not look at severe accidents.

The Catawba units currently are transitioning from the traditional TID 1484 source term and the alternative source term. Presently, Catawba's licensing basis source term is TID 14844, with the exception of the two fuel handling accidents which are based on the alternative source term.

As a result, the acceptance criteria then was 10 CFR Part 100 for the off-site doses of everything except the fuel handling accidents, GDC-19 for the control room doses, and then 50.67 for the fuel handling accident in the first fuel drop.

We had several REIs. The review did focus on the licensee's submittal. Our approval is based on the licensee's submittal. However, the staff performed independent calculations of the licensee's work.

Our staff's confirming calculations included inputs from Sandia Lab, Pacific Northwest Labs, and source term and gap fractions. The conclusion of our review was that we agreed with reasonable assurance that the licensee's conclusion that the four MOX LTAs would have minimal impact on the prior analysis results was correct. All doses continue to meet the acceptance criteria.

I don't have to say too much about this, because Steve mentioned most of this. There were two groups of accidents analyzed, those which were large fraction and a small fraction. The small fraction we didn't spend an awful lot of time on, because as Dana very eloquently pointed out, it really doesn't make a whole lot of difference.

We did check the math and the scaling calculation, and we did detect an error that the licensee corrected. But the first group is more of interest to us, because we are dealing with a fuel handling accident design basis. It would be conservative to assume that the dropped assembly would be the one that was an LTA.

The other form that falls in this category is the weir gate being dropped in the spent fuel pool, which is postulated to damage seven assemblies. We assumed in this case that -- and the licensee assumed -- that four of those assemblies would be the LTAs. Obviously, you recognize the probability of this is probably pretty low.

Now for this particular category, the licensee recalculated the dose consequences of the accident using updated spent fuel inventory and an assumed 50 percent increase in the gap fractions.

Part of the second group involved the ones where the MOX LTAs were a small fraction. One of the points I would like to make regarding the LOCA is that there have been some comments made here and in other proceedings that I need to clarify, because I think they were misunderstood a lot, that in the design basis accident space the radiological analysis assumes there is core melt.

As a defense in depth measure, my colleagues in Reactor Systems go to great lengths to show that the fuel performance and ECCS performance will prevent that from happening. So the design basis accident space, we got to recognize, is the disconnect between the thermal hydraulic analysis which proved no fuel damage and the radiological analysis that start off assuming there was.

Since our assumption there was is rather arbitrary, the impact of MOX on that arbitrary decision is not all that strong. Add that to the fact that it is only two percent of the assemblies affected.

In addition -- It hasn't been mentioned earlier, but in addition to the accidents they already had in their licensing basis, the licensee performed an analysis of the consequences of a fresh fuel drop. I point out, none of these -- These accidents are typically not performed. I can't think of any other licensee that has this analysis in their license basis.

This is typically because the uranium assemblies have very low specific activity. For the low specific activity, they are not a big dose contributor. However, as the licensee pointed out and as we concur, is that the specific activity of the plutonium isotopes is significantly higher, and it was warranted to have a look at what the dose consequences would be of a dropped assembly.

Now the licensee's analysis methods were largely based on methods used by Sandia Labs for the Yucca Mountain calculations, and also those are methods used by the Office of Nuclear Materials and Safeguards for looking at fuel fabrication facilities.

Okay. When staff did its review, we were focused on three major aspects of what we believed was the MOX fuel use. First was the change in the fission product inventories. Second was the change in the gap fractions. The third one, which only applies to the fuel handling accident, was the change in the fuel rod pressurization.

The fuel rod pressurization is an impact that accident, because our assumptions regarding the decontamination of the iodine as it bubbles through the pooled water is dependent on the rod pressure.

For the fission product inventory, the licensee had used the scale suite from Oak Ridge, particularly the SAS2H/ORIGEN-S code, to generate the fuel inventory. The licensee determined the MOX LTA inventory to burnup at about 17 gigawatt days per metric ton uranium.

The reason it was done at this point is that the licensee had done a sensitivity analysis and found out that the iodine peaks at that point. So doing the calculation at this point maximized the amount of iodine.

The SAS2 -- I should say this slowly so you can catch it. The SAS2H code is particularly well suited for this application, since it calculates the cross-sectional libraries that are used by ORIGEN on a case by case basis and, as such, could be structured to address the MOX LTA fuel isotopics and configuration directly. We weren't using a generic off-the-shelf library.

In addition, the staff obtained some data files generated by Sandia Labs using ORIGEN-2.2 for purposes of comparison. In order to confirm that the licensee's basing its irradiation on the peak iodine inventory would not overlook a significant increase in another radionuclide, the staff evaluated the inventory at the end of the first, second and third cycles.

The staff used the SAS2H code for this purpose and took the maximum inventory of the three cycles, nuclide by nuclide, for its confirming calculations.

The observed increase in the iodine 131 inventory in the MOX LTA as compared to an LEU assembly was about nine percent, the value used by the licensee in its scaling calculations.

Now the licensee used iodine 131 in the thyroid dose for the purposes of scaling, concluding that this would be the most limiting isotope, the most limiting dose limit.

In order to confirm that, the staff looked at the impact of the increased noble gases. Some of the noble gases will increase over the cycles. We confirmed that the licensee's reliance on iodine 131 in the thyroid dose was bounding in the design basis space.

Another issue was the gap fractions. For assessing the gap fractions, the NRR staff requested the assistance of the research folks to perform a fission gas release analysis for the MOX LTAs. Research utilized the staff at Pacific Northwest Laboratories to perform this assessment with the FRAPCON-3.2 code.

Now the FRAPCON version 3.2 had been modified for use with MOX fuel as associated with its use in the review of the COPERNIC topical report. Changes to this code included adding thermal conductivity model for MOX fuel.

Adjustment was made to the fission gas release model diffusion constants to reflect the differences noted between predicted versus measured fission gas in MOX fuel assemblies. MOX fuel plutonium isotopics were addressed, and they made a change to the xenon-krypton ratio that used in the code.

Now the primary model in the FRAPCON code is a Massih model which can only provide predictions of the stable noble gas nuclides. We, of course, need to know the radiological ones.

To obtain the yields for the radionuclides, PNNL used the ANS-5.4 model, which is part of FRAPCON, but adjusted the inputs to obtain the same stable noble gas output from the ANS-5.4 model that they had obtained from the M-a-s-s-i-h model.

This is because the ANS-5.4 model is known to overpredict fission product release fractions. So in essence, they normalized the M-a-s-s-i-h model. With that change done, the ANS-5.4 model predicts the radionuclides.

The FRAPCON runs also showed that the end-of-life rod pressurization was less than 1200 psia. As such, the Safety Guide 25 assumptions regarding the spent fuel pool decontamination credit remain valid.

This table here -- next slide, please. This table here shows the gap fractions breakdown. I need to point out very carefully here that we are talking about non-LOCA gap fractions.

The numbers in Regulatory Guide 1.183

which address LEU are shown in the first column. The licensee assumed a 50 percent increase, and those numbers are shown in the second column. The staff's evaluation, based on the work done by PNNL, are shown in the third column.

You can notice that the numbers in the third column are bounded by the licensee's assumptions. There's a couple of ones that need to be addressed -- a couple of items that I didn't talk about here.

The reason there is a range in the staff ones is that PNNL had discovered a difference between the power history submitted by Duke in terms of burnup versus time and the F delta H values in the same table. So PNNL had done it using both sets of data. PNNL also tacked on a five percent margin to address uncertainties in the power history. So that's why you see a range for the staff's evaluation.

With regard to the alkali metals, the licensee is marked here as not applicable. The LTA gap fractions were used by the licensee only in the fuel handling accident.

The design basis fuel handling accident assumptions provide that particulate material will be retained by the pool. Hence, cesium was not an issue.

Maybe it will be an issue in the future if they pursue with the batch.

The reason it isn't addressed as a LOCA, the locked rotor accident or the rod ejection accident, is cesium is not part of the TID 1484 source term, which is the licensing basis for Catawba primarily.

The staff, of course, expects to get some input out of the post irradiation examinations as to find out where their numbers fall with measured data.

That is my comments.

CHAIRMAN POWERS: Any questions to pose to Steve? Thank you.

Steve, your post irradiation examinations of the gap inventories -- how accurate do you need those numbers?

MR. LaVIE: For the LTAs not very accurate at all, as we pointed out, with only two percent of the assemblies.

I do want to make a point about why that is significant. In doing the analysis, when they the scaling analysis for the LOCA, Duke applied the 50 percent increase to all the release fractions. Of course, this is based on TID 1484. There's only one release space.

The iodine assumption in the TID is 50 percent. So in reality, they did the calculation assuming that 75 percent of the iodine in those four LTAs was released. The dose went form 89.3 rem to 91 rem. Even if they had released 100 percent of the iodine in the LTAs, they still would have been well within acceptance criteria.

So how accurate we need those numbers is going to depend on -- For the it doesn't depend at all. In the future it may become important if they pursue a batch amendment.

They had plenty of margin in the fuel handling accident as well. The dose cited was down like at 1.2 rems or something like this nature. It was out of an allowable 25.

CHAIRMAN POWERS: Any other questions?

DR. TRIAFOROS: Yes, I have another question. This is on issues that we discussed, a little area, and it has to do with the fact that the subjects that are addressed in the safety evaluations are -- The evaluation is good up to a burnup of 60 gigawatt days per metric ton of heavy metal.

Now the safety evaluation -- the subjects are addressed in the safety evaluations. They refer to their COPERNIC code which, based on our review of the safety evaluation of the COPERNIC code, it is approved to 50 gigawatt days per metric ton heavy metal, and it is an apparent inconsistency there. I would appreciate if you can elaborate.

MS. SHOOP: Certainly. that is an issue that the intervenors have also raised. Basically, going back to what I started out with originally, the purpose of an LTA is to gather data.

Now how do you gather data? You can only gather data by burning it. How do you approve a code? Well, you can only approve a code if you have data. So you've got the chicken and the egg conundrum. I mean, you need data to support a code approval, but how do you get data if the code is not approved there?

So what we have done with LTAs and what has been reactor systems' common practice is we understand that the purpose of LTAs is to collect data and, therefore, we will extend the use of a code to an area where we believe that it is still good.

Framatome actually did provide some information or data between the 50 to the 64 gigawatt day range. However, that data was not statistically significant, and that is why we did not approve it up to 60.

CHAIRMAN POWERS: Any other questions to

pose? Well, good. What I wanted to do is just take a few minutes here to discuss presentations in front of the full Committee. I'm going to handle this one right now with Bob, and then we will go on to the rest of our agenda.

Bob, we have scheduled a two-hour block of time at the May meeting to discuss this, and the question before us is twofold.

One, do we need to have the applicant appear and redo his -- any portion of his presentation? Second, what fraction of the afternoon that you have presented here do we want to present in front of the full Committee?

I would invite the members to voice their opinions on that particular subject, those questions.

DR. KRESS: Well, I think if it is not too much of an imposition, we would like to have the applicant come back, partly because the Committee gets an impression from hearing the applicant.

My guess on that would be I thought the -- Well, in the first place, the extra data from Framatome was good, but I don't think -- I think we can just present that to them in slides or something.

I felt Nesbit's safety and environmental evaluation would be important to get in.

CHAIRMAN POWERS: My suggestion was that if Mr. Nesbit could factor into his presentation some of the slides on the fuel properties and microstructures --

DR. KRESS: That might be the way.

CHAIRMAN POWERS: Just to augment some of the points that he ordinarily makes, that that might be a particularly succinct way to make his points in front of the full Committee.

DR. KRESS: I also thought that -- You know, the Committee is going to be interested in the nuclear stuff, nuclear analysis. So if we get some abbreviated part of that -- I don't think the full thing.

CHAIRMAN POWERS: Maybe again just augment the points made.

DR. KRESS: Yes. And personally, I also think it was very useful to hear BREDL's concerns.

CHAIRMAN POWERS: Oh, yes.

DR. KRESS: And so if we can impose on them to more or less repeat those.

CHAIRMAN POWERS: Okay. I'll ask Mr. Lyman if he can bring those forward to us as well. I don't know. I haven't spoken to him about that, but we will. Mario?

DR. BONACA: Well, I agree with what Tom is suggesting. One thing that it is important. It would be valuable to have just a brief description on the slide of the isotopic composition of this MOX fuel versus the one that we saw in Europe, because otherwise there is an implicit assumption that they are similar, period.

Well, they are not that similar. The reason why it is important is that -- For four lead test assemblies I don't think it is important, but really when you talk about the future, it is important, and so I think it is beneficial to present it that way.

The other thing which is interesting is that, in licensing four lead test assemblies, you do have a fundamentally different philosophy in the justification that you will have for the full court, because what you are doing is strategically positioning your LTAs in certain locations, and for most advanced generalizing that is the way you say, well, you know, it's not -- the leading assemblies is not limiting.

It would be different when you go in and you insert 40 percent of assemblies or whatever you are going to insert. So even that would be an important point to make, I think. Since we have such little time there, these are important issues that you want to communicate.

I thought, within that context, the presentation we had on the French experience was very valuable, because it clearly tells me there is a lot of information out there that, taken in proper consideration of what differences it may be, says this is not a new venture. I mean, this is really something for which there is a solid basis.

So I also thought the radiological analysis was important, because it conveys some of those messages there.

CHAIRMAN POWERS: Professor Ransom, any comments?

DR. RANSOM: I think the same thing. I would like to see the French experience emphasized. I think that adds a lot of credibility to what is being done, or certainly minimizes the risk of the LTA question itself.

As far as I am concerned, it seems like very convincing argument, and I would guess you probably want to summarize what --

CHAIRMAN POWERS: We are not going to allow sufficient time for the full presentation, but if some of the salient points can be brought forward, it would be useful. Graham?

DR. LEITCH: I have nothing to add. Mario made the comment that I was going to make.

CHAIRMAN POWERS: Okay. What I would propose then is that we will kind of divide the time up equally between the staff and the applicant, and then ask Mr. Lyman what time he thinks he would need, if he can in fact be there, and do that.

So that brings us to the question of how to -- what to do with your time. Let me say at the beginning, I thought the staff's presentations were uniformly excellent.

MR. MARTIN: Thank you.

CHAIRMAN POWERS: And so it is very difficult for me to come in here and tell you what part to cut out. So I may just say figure out how to present all that material, but do it in a lot shorter time. This is not an unusual command from the ACRS, but yes, I think you are going to struggle on doing that, because I thought the presentations across the board were just excellent.

MR. MARTIN: Thank you, and I appreciate being able to go ahead with the May meeting. There are some considerable scheduler concerns related to hearing activities for any June time frame which make being able to go ahead with the May meeting very valuable.

CHAIRMAN POWERS: I think, regardless of the outcome of your meeting with the applicant that simply getting the rest of the ACRS on board and up to speed will just make it more efficient if we have to come back for some reason, and make it a little more efficient.

MR. MARTIN: Right.

CHAIRMAN POWERS: Without having the outcome from your meeting, I can't say for sure, but anything that is not heroic would not move me to have another subcommittee meeting.

MR. MARTIN: Okay.

CHAIRMAN POWERS: Let me turn -- I see Dr. Lyman is in the audience. You have a command performance here, sir. Would you be able to help us?

DR. LYMAN: Yes, indeed.

CHAIRMAN POWERS: Now you've got me.

MR. CARUSO: So the first week in May, the Thursday and Friday, the 6th and 7th.

DR. LYMAN: Sixth and seventh?

CHAIRMAN POWERS: Well, you guys will interact. I think sensibly, we are asking for your --

rehash -- not rehash but a repetition of your presentation in front of the full Committee.

DR. BONACA: And it would be interesting to know, again on the point I was making before of four lead assemblies versus 40 percent of the core -- It would be interesting if some of the issues that were raised and you addressed, you know, would be of more significance in consideration in the future, because you are doing something about that.

For example, location, you are telling me that you are still -- It would be interesting for the members to understand that in context. Again, it may not be an issue at all with four lead assemblies.

CHAIRMAN POWERS: I don't know that we can ask them to do everything. You are going to have to be more liberal with your agenda.

Okay, I think we've got a start on that. Mr. Caruso is here to help and facilitate these presentations, to the extent that they can be done.

At this point on the agenda, I have the item for additional public comment. Do we have any additional public comments based on what has been heard or otherwise? Be our guest.

Thank you all very much. I really did think your presentations were excellent. We will turn to Dr. Lyman here.

DR. LYMAN: I would defer to the staff if they have anything.

I appreciate the opportunity to clarify. During the day's discussion there was some question of what the intervenors are really seeking in this case. So I just wanted to clarify that BREDL isn't seeking absolute certainty. That is not the goal, and we understand that is unrealistic, and we also don't want to curtail scientific investigation by making conditions impossible to do any research on irradiation of LTAs.

In this case, we aren't talking about an incremental change in the type of fuel, but we are talking about a significantly new type of fuel in the U.S. experience, and to that extent, I think there have to be greater demands and reviews of the MOX LTA application than on the typical LTAs situations.

Now to some extent, that's occurred, but given the large body of data accumulated over decades with conventional LEU fuels, including under accident conditions, and comparing that to the relative paucity of data for MOX fuel under similar accident conditions and the fact that the few data points that have accumulated from MOX seem to suggest that there may be some issues that need greater observation, then we would say that would indicate a more thorough licensing basis before experimenting with these LTAs in U.S. reactors.

So that is the context, not that we demand absolute certainty, but just a greater -- There should be greater curiosity, I think, on the part of the NRC's staff review than there is, investigating some of these issues that have emerged.

In light of Dr. Bonaca's observation, to the extent that this application is approved largely on the basis of the relatively small number of assemblies that are affected and not on an understanding of the underlying physics and other properties of these assemblies, they will only come back to haunt everyone when the batch application comes in.

So it probably would make sense to start trying to nail them down at this point to avoid delays later.

The next point regarding source terms, I would just like to point out that, obviously, the deterministic -- the old deterministic TID source term doesn't include consideration of release of any of the refractory radionuclides, and considering that the largest inventory differences when you are talking about MOX fuel lie in plutonium and the higher actinides, not considering that is an oversight which I think at least needs to be addressed even though it is only going to affect a small number of fuel assemblies in this case, especially given if there are any indications from VERCORS tests that the potential release fractions of low volatiles are higher than what have been anticipated.

So none of that was considered in the application under review. Whole body doses were largely not considered, even though, as I mentioned before, two isotopes which the staff didn't mention which have higher -- substantially higher inventories in MOX fuel, including the ruthenium isotopes which are well over 50 percent greater in MOX, and the tellurium isotopes which would contribute to whole body doses were not considered, and looking only at the iodine source term and doses.

Finally, my last remark on post irradiation examination: If the staff is indeed requiring -- and I'm not sure this is a commitment, because I haven't double checked, but if they are requiring that the hot cell PIE be concluded before the batch application can be approved, I would suggest looking at that in the context of the current schedule, which doesn't seem to allow much time for that analysis to be completed before an approval would be required, since the fuel wouldn't be discharged after two cycles until spring 2008, and the current schedule is still that batch loading would start sometime in 2099. So factoring in the time for cooling, transport and analysis, it seems like that is cutting it awfully close.

That's all I have to say. Thank you.

CHAIRMAN POWERS: Thank you. Does the staff have any comments they would care to make?

MS. SHOOP: Thank you, Dana. Actually, I heard Dr. Bonaca, and I believe that Graham also believes the same, that for the full Committee meeting they would like us to discuss the difference between LTA and batch.

Batch application at this point is completely speculative. There is no application in front of us for batch loading of MOX fuel and, therefore, and because it is not part of this application, I think that that would be the wrong thing to put our time toward.

We have a very limited amount of time and, therefore, we should address what this application is, and this application is not for batch loading.

DR. BONACA: No, I want to make sure, I didn't say you have to address batch. I simply point out that the licensing of batch will be different, and therefore, you would be looking at different things. I'm just putting in context the fact that the LTA doesn't resolve all the issues to do with the differences. That would come later.

It will be a different kind of challenge, however, than purely for four LTAs.

MS. SHOOP: Okay. Thank you for clarifying that.

CHAIRMAN POWERS: I agree with you. We simply don't have time to delve into any kind of detail on this, but a cautionary note on that never hurts, and Mr. Nesbit, you, too, might want to introduce a cautionary note that the batch application clearly involves something different.

DR. KRESS: I wouldn't speculate on any of the outcomes.

CHAIRMAN POWERS: Dr. Meyer, you would like -- you have a word that you would like to pass on?

DR. MEYER: Yes. There is one thing that was said earlier in the meeting that has been on my mind, and I would like to come back to it.

It was something that Mr. Nesbit said about the reactivity accidents, the rod ejection accident somehow being worse for mixed oxide fuel than for UO_2 fuel.

What I wanted to point out about all of this is that, if you look at what is significant from a risk point of view, really, the only two events that you have talked about here that might come on the radar screen are the loss of coolant accident and the rod ejection accident.

The loss of coolant accident, the effect of MOX -- Well, first off, let me say that there clearly are neutron physics effects of MOX, and these can be and are being handled. But when you talk about the fuel part of that, for the loss of coolant accident any connection, any difference between MOX and LEU at this time is purely speculative, and I don't think there is any evidence that there is a difference, although we are, of course, interested in looking.

For the rod ejection accident, we know about these plutonium agglomerates, and they can have an effect on the fuel behavior during an accident. But as far as we can tell from quite a number of tests on MOX as well as on UO_2 fuel in pulse reactors, this effect is going to show up in the dispersal characteristics of the fuel rod after you have breached the cladding and you now have these little islands that are able to kind of pop open fuel rather than just a rem of extremely high burnup material.

So this will be different. But the criterion that is being used, both by the applicant in this case and by Research in its recent study, is a cladding failure threshold criterion.

We are using -- The numbers that we talked about today are numbers that are so low that you can't even crack the cladding open, and you can't get dispersal.

In our analysis of this Rep NA-7 test, which was the MOX test at Cabri which had a rather energetic dispersal of fuel material, when we analyzed that test in terms of its cladding failure threshold, it is no different than any of the LEU tests.

So if for this accident you are using the more conservative limit of the cladding failure threshold as your absolute limit, then I would say the evidence is that there is no difference in the fuel behavior in that case.

So just in summary, for these two risk

significant events at this point I think the effects of MOX fuel -- you can already see that they are going to be marginal at best.

CHAIRMAN POWERS: I have to admit that the conclusion I walked away from Mr. Nesbit's presentation was identical to this.

DR. MEYER: Okay.

CHAIRMAN POWERS: I mean, he can speak for himself here, but my conclusion was that that's what you said, is that the clad failure was about the same and that dispersal characteristics were different.

DR. MEYER: I must have missed that. I was having a hard time hearing.

CHAIRMAN POWERS: It could be, but there does seem to be consensus on that point.

Do members have any other comments they would care to make? I think we have a plan of attack on this. I will be chatting with you about draft summaries on some of the concepts that were put forward at this meeting and get your concurrence on that, but we will go to the full Committee meeting as planned.

I thank all the speakers. I complimented the staff on their presentations. Mr. Nesbit, I want to also congratulate you and your folks for excellent presentations to us as well. I appreciated Mr. Blanpain's presentation especially. We always like to see microstructures on fuels and properties and data. It always makes us feel like we really are scientists.

With that, I think I will adjourn the subcommittee meeting.

(Whereupon, the foregoing matter went off the record at 5:09 p.m.)

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